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SOVIET ATOMIC ENERGY

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SOVIET ATOMIC ENERGY

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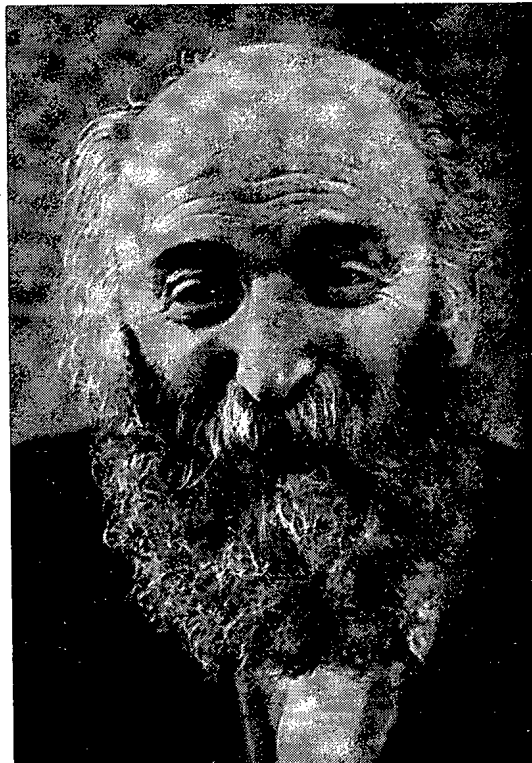
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Publication therefore did not occur prior to this date, but must be assumed
to have taken place reasonably soon thereafter.

OBITUARIES

ACADEMICIAN G. I. BUDKER

I. N. Golovin



May 1, 1978, would have been the 60th birthday of Academician Gersh Itskovich Budker, the leading Soviet physicist known throughout the world for his researches on charged-particle accelerators and controlled thermonuclear synthesis. He died on July 4, 1977, of major heart disease while still full of energy, both mental and physical.

G. I. Budker was born in the village of Muraf in the Shargorod area of Vinnitsa Oblast, and in 1941 he graduated from the Physics Faculty at Moscow State University; he served until the end of the war in the Army Air Services Branch in the Far East, and on demobilization in the summer of 1945 started work at the Institute of Atomic Energy, which was then called the No. 2 Laboratory of the Academy of Sciences of the USSR. I. V. Kurchatov directed him in the analysis of some urgent questions at that time. These particularly concerned the motion of charged particles in cyclotrons, followed by control theory for uranium-graphite reactors. However, he is particularly remembered for his outstanding work in the early days of controlled thermonuclear reactions. As early as 1952 he put forward almost simultaneously two outstanding ideas: magnetic traps for retaining thermonuclear plasma and stabilized electron beams, with the latter subsequently utilized in the construction of a new type of charged-particle accelerator.

A laboratory of new acceleration methods was set up in 1954 to implement his ideas on accelerators. In 1956 he presented his DSc, and in 1958 he was elected a Corresponding Member of the Academy of Sciences of the USSR, and in 1964 as a full Academician. From the end of 1957, with the support of I. V. Kurchatov, he participated in setting up the Institute of Nuclear Physics, Siberian Branch, Academy of Sciences of the USSR in Akademgorodok. He was the Director of this institute until his death, and he converted the institute to a

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major nuclear physics research center, which has become recognized throughout the world as performing unique scientific-engineering researches.

Throughout these years he participated in numerous conferences on high-energy physics and accelerators, plasma physics, and thermonuclear synthesis not only in the USSR but also abroad, and he had considerable influence on the development of researches in these areas throughout the world. His outstanding scientific researches were recognized in the award of Lenin and State Prizes, as well as medals and other orders.

From his early days he was accustomed to fundamental research into intricate problems in physics. He had a masterly way of resolving the paradoxes that frequently arise in careful examination of physical phenomena. He repeatedly emphasized that the physicist has no right to say that no result was obtained even if the result of an experiment was negative, since he is obliged to define the laws of nature that explain his negative results. In his own researches he encountered many different obstacles, for which he always invoked novel solutions, which, particularly in his early years, were sometimes dismissed as unfeasible fantasies. However, when it became clear that these ideas, were always feasible, these skeptical evaluations were replaced by international recognition.

Many of Budker's ideas were ahead of his time.

His concept of a stabilized electron beam provided access to a new area of high particle energies, and this attracted immediate attention and formed the basis of various researches here and abroad. However, Budker himself immediately recognized that the technology was not yet available to implement it fully, and therefore even in the early 1960s he concentrated the efforts of his institute on the colliding-beam concept for research on elementary-particle interactions at superhigh energies. Although the colliding-beam concept was not his own, he applied to its implementation some extremely original ideas, such as storage rings for examining electron-electron and electron-positron interactions ranging from the VÉP-1 to the recent VÉPP-4, which have considerably advanced this area of engineering. The VÉPP-2 storage rings and the high-current VÉPP-2M made a major contribution to elementary-particle physics, particularly to the determination of the exact mass and peak width in vector-meson decay, as well as the branching probability in such decay. The VÉPP-3 is now a powerful source of synchrotron radiation and is used by institutes in this country, in particular for research in molecular biology. The constructors of the VÉPP-4 had the foresight to provide facilities for raising the electron-beam intensity; the positron injector should be completed in the summer of 1978, and then the VÉPP-4 will constitute a world leader in systems of this type, since it will allow one to examine the interaction of electrons with positrons at collision energies up to 14 GeV. The plan for Soviet-American collaboration in this area includes utilization of the facility by a group of researchers from Stanford.

The VÉPP-4 required a high-frequency oscillator in the meter wavelength range working in the steady state at an output of several megaelectron volts; traditional tubes would have been difficult to employ, and Budker, as an experimental relativistic engineer, suggested a novel solution, namely a novel form of generator called a gyrocon, which uses a relativistic electron beam of energy 0.5 MeV to excite a cavity.

The VÉPP-4 ring was designed to store proton-antiproton beams; a store of this type requires a very small phase volume for the particles in a captured bunch. Various solutions to this problem are being examined in laboratories throughout the world. In 1967, Budker suggested the idea of electron cooling and confirmed its performance by experiment. It is now proposed that this method will be used in antiproton storage rings.

In the area of thermonuclear work, Budker developed at his institute only those lines that did not duplicate Soviet or foreign researches. He halted extension of researches on accelerators begun so successfully in the institute in the mid-1960s, since other laboratories were working in this area, and instead gave vigorous support to a mirror system in which a dense plasma was heated by a relativistic beam. The plasma pressure is contained by the wall in this system, while the magnetic field merely reduces the transverse thermal conductivity, and the multiple magnetic mirrors hinder the plasma from escaping along the tube. During his last months, he was working on an experimental test on plasma retention by an ambipolar potential formed in an open trap with two mirrors. This idea has given new life to open traps as major forms of thermonuclear reactor.

It is here in place to recall that in 1968 the International Atomic Energy Agency held an international conference in Novosibirsk on thermonuclear researches. Here Budker issued a challenge to initiate engineering developments on reactors on the basis of the available physical knowledge. The participants were thrown into some confusion, since they did not know how to respond to this call, but within a year the First International Conference on Synthesis Reactors was called at Culham (Britain), and now the number of engineering studies in this area is increasing continuously.

Budker was characterized not only by considerable scientific talent but also by organizational capacity. He clearly understood that the budget of the Academy could not possibly support the construction of all the systems he conceived, but he clearly appreciated the scope for practical use of accelerators in industry. He decided to initiate the construction of electron accelerators providing several megaelectron volts but below the photonuclear reaction threshold, in which the extracted beam should be used for radiation technology, e.g., for modifying polymers, sterilizing grain, and the like. Although the design principles of such accelerators were already familiar, they had to be made reliable to industrial standards and capable of operating without specialized staffs. He set up a large design office and production section to make accelerators for various users, and the facilities acquired by agreement in manufacturing the systems were then utilized in constructing the storage rings and the thermonuclear systems.

Budker attached very considerable importance to staff training; he always devoted much effort to any talented young worker. He welcomed visiting scientists and engineers personally and always discussed their topics in detail. An outstanding innovation of his was the regular round-table meeting, which will be familiar to anyone who has worked at the Institute of Nuclear Physics. This table was of diameter 5 m and readily accommodated over 30 staff members. Budker made it a rule to meet at this table every day with a cup of coffee at 12 noon along with the leaders of the scientific teams, with the particular object of becoming acquainted with any scientific news arising during the day. Guest workers were also welcomed. New ideas were exchanged here in a free and easy setting, and these regular meetings played a large part in consolidating the team, in which, as Budker constantly emphasized, there should be no internal rivalry on success of one at the expense of another. In this collaboration he saw the responsibility for the success of the Institute as a whole, since each of those gathered around the round table had considerable influence in the Institute and should support and extend this collaboration. He repeatedly said that the Institute is not a building or a set of hardware but a team that works in it in common.

Budker is unfortunately no longer with us, but his team lives on, in its unique form and with its considerable potential. The team at the Institute included many who have since had to move to other institutes or to industry; however, after working in close contact with him, they intend to advance his scientific and engineering achievements.

ARTICLES

THE GYROCON, A HIGHLY EFFICIENT CONVERTER OF ENERGY FROM POWERFUL RELATIVISTIC BEAMS FOR MICROWAVE SUPPLIES IN CHARGED-PARTICLE ACCELERATORS

G. I. Budker, M. M. Karliner,
I. G. Makarov, S. N. Morozov,
O. A. Nezhevenko, G. N. Ostreiko,
and I. A. Shekhtman

UDC 621.385.6

The problem of high-power rf supplies for new accelerators continues to be of constant interest. As a solution to this problem, in 1967 G. I. Budker proposed a microwave oscillator which employed bending of a relativistic particle beam, the gyrocon* [1].

The design of the simplest version of a gyrocon is sketched in Fig. 1. A continuous electron beam (ray) from a high-voltage accelerator is deflected by the rotating magnetic field of the sweep resonator. This resonator is supplied by an exciter to which an input rf signal is fed. The deflected electrons move along straight lines which form a conical surface and are located on it in the shape of a helical line whose end describes a circle. An electrostatic electrode system directs the electrons into an annular slit in the output resonator

*Gyros (Greek) = circle + continuum (Latin); referring to the circular deflection of a continuous (unbunched) beam.

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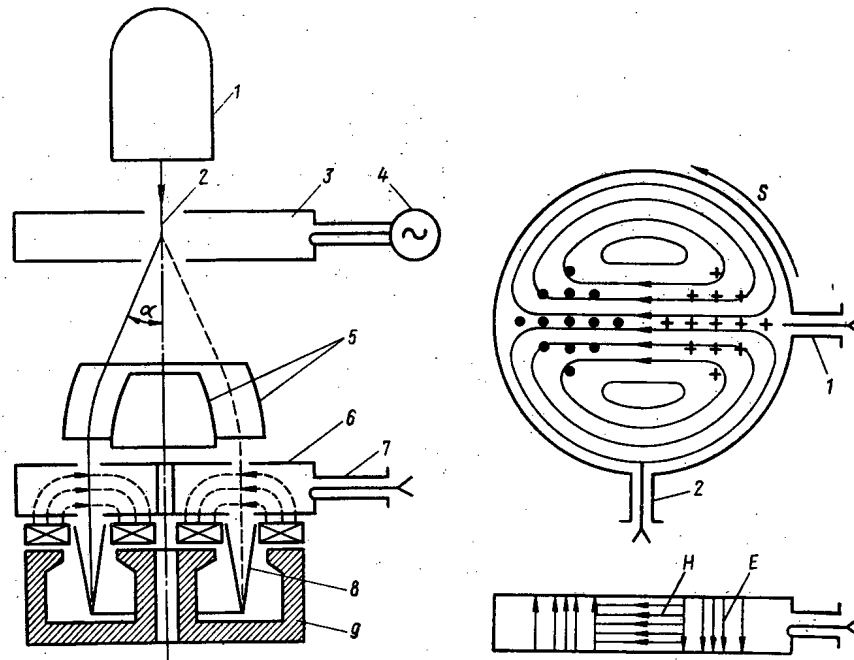


Fig. 1

Fig. 2

Fig. 1. A sketch of a gyrocon: 1) high-voltage accelerator; 2) electron ray; 3) sweep resonator; 4) rf exciter; 5) electrostatic deflection system; 6) output resonator; 7) energy outlets; 8) collector; 9) compensating electromagnet; α) the scan angle; dashed lines) static magnetic field lines.

Fig. 2. The sweep resonator: H) magnetic field lines; E) electric field lines; 1, 2) power inputs; S) the direction of rotation of the field.

which is tuned to the sweep frequency so that at resonance oscillations develop in it with a maximum electric field in the region of the annular slit through which the electron ray passes. The point at which this ray enters the resonator is continuously changing so the ray excites a traveling wave in the resonator. The electric field of the wave slows down the electrons and converts the power in the ray into microwave power. The remaining energy of the electrons is scattered in the collector.

The circular sweep of the gyrocon beam is done by the magnetic field of a cylindrical resonator with E_{110} oscillations [2] (Fig. 2). The two power inputs for this resonator are supplied with a phase shift of 90° which ensures circular polarization of the magnetic field in the near-axial region where the electrons pass as they are being deflected.

In the output resonator (see Fig. 1) the electrons excite a traveling electromagnetic field of the same configuration as in the sweep resonator but the electrons pass through in an antinode of the electric field. The output resonator is essentially a rectangular waveguide with an H_{10} wave bent into a ring. This resonator is tuned to the sweep frequency, but can be tuned to a multiple of that frequency. In that case the gyrocon operates as a frequency multiplier. The two energy outlets are located azimuthally at 90° with respect to one another, similarly to the power inputs on the sweep resonator (see Fig. 2). When they are loaded equally and the resonator is excited by the electron beam, they ensure a traveling wave regime in the resonator. This regime can be maintained with a larger number of energy outlets as well [1]. The electrostatic deflection system is part of a spherical condenser. An electromagnet produces a static transverse magnetic field in the drift gap of the output resonator which compensates the effect of the magnetic field of the traveling wave (which reduces the efficiency of slowing down the relativistic electrons).

The gyrocon is free of the fundamental limits on electronic efficiency typical of high-power klystrons and microwave devices with grid control since in it there is no density modulation of the electron flow. In fact, the transverse dimensions of the high-power relativistic electron beam are much smaller than the wavelength of the traveling wave in the output resonator so all the ray electrons are slowed down almost identically. Thus, it is possible to obtain an electron efficiency in a gyrocon of close to 100%.

TABLE 1. Maximum Power from a Gyrocon

u_0 , kV	P_{\max} , MW	u_0 , kV	P_{\max} , MW
500	8	2000	720
1000	70	3000	3000
1500	270		

Density modulation of the electron flow did not occur in the first microwave devices with beam deflection. One of these, a device with an annular traveling-wave resonator [3, 4], is a prototype gyrocon. However, the lack of means to obtain a relativistic electron beam and to efficiently slow it down limits this device to low powers.

In a gyrocon with relativistic electron energies it is possible to send a high-power electron beam through the apertures and slits of the resonators with a small current drop so that in it the forces of magnetostriction weaken the effect of the transverse space charge forces by γ^2 times (where γ is the relative energy of the electrons). This makes it possible to achieve microwave powers with a gyrocon which could not be reached with the devices described in [3-8] which operate in principle at nonrelativistic electron energies.

On going to relativistic energies in devices with an annular traveling wave resonator, the action of the magnetic field of the wave may significantly change the direction of motion of the electrons as they leave the resonator which has been excited by the beam. An electron directed along the lines of the slowing-down field gains a noticeable azimuthal velocity opposite to the direction of motion of the wave and that part of the energy associated with this velocity cannot be converted to energy of microwave oscillations. In this regard the electron efficiency (η_{el}) of a device with an annular traveling wave resonator can be substantially less than 100% (1):

$$\eta_{el} = \frac{1 - \sqrt{1 - \beta_0^2 [1 - (1/\nu^2)]}}{(1 - \sqrt{1 - \beta_0^2}) [1 - (1/\nu^2)]}, \quad (1)$$

where $\beta_0 = v_0/c$ is the reduced velocity of the electrons upon entering the resonator and $\nu = V/c$ (V is the phase velocity of the traveling wave and c is the speed of light). For $\nu = 1.84$ (resonator 6 of Fig. 1) and $\beta_0 \rightarrow 1$ the electron efficiency is 65%.

In the gyrocon this drop in efficiency is eliminated with the aid of a static magnetic field which is either produced in the output resonator (the compensating magnet 9 of Fig. 1) or is located so the electron beam passes through it in advance. In the latter case, the angle at which the electron enters the output resonator of the gyrocon in order to obtain 100% efficiency is

$$\psi = \arctg \frac{1 - \sqrt{1 - \beta_0^2}}{\sqrt{\nu^2 \beta_0^2 - (1 - \sqrt{1 - \beta_0^2})^2}} \quad (2)$$

and for $\nu = 1.84$ and $\beta_0 \rightarrow 1$ is no more than 33° .

The first working model of a gyrocon was built according to the design of Fig. 1. In 1971 a power of more than 600 kW into a load was obtained from it with an electron energy of 320 keV in a pulse lasting 20 μ sec and it was experimentally confirmed that an electron efficiency of over 90% could be obtained at a frequency of 430 MHz.

To illustrate the prospects for the gyrocon, we now evaluate its main parameters. For the calculations we shall assume that the transverse dimensions of the beam in the gyrocon do not exceed the dimensions of the free relativistic electron beam as it expands due to space charge forces [9]. We shall suppose that the electron beam at the outlet from the accelerator has been focused optimally [10] and has sharply defined boundaries. We shall consider only the azimuthal dimension of the beam cross section in the output resonator and assume that the other transverse dimension can be substantially reduced by means of focusing after the sweep. We shall assume furthermore that the deflection angle of the electron beam in the sweep resonator is small ($\leq 20^\circ$) and that the transit gap of this resonator has been chosen so that the losses at the walls will be roughly equal to the losses due to accelerating electrons from the swept beam at its peak power and with a minimum excitation power. We also assume that the transit gap of the output resonator has been chosen subject to the conditions for minimum electric field strength for complete slowing down of the electrons.

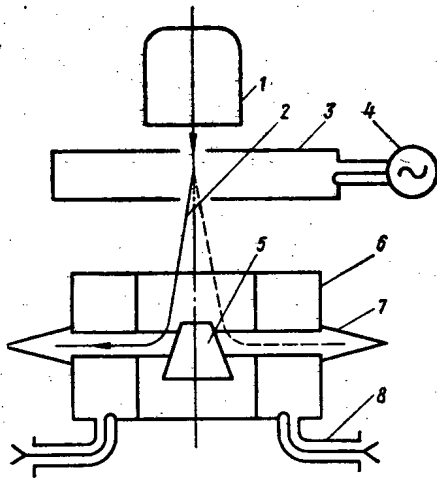


Fig. 3

Fig. 3. The layout of a radial gyrocon: 1) high-voltage accelerator; 2) electron beam; 3) sweep resonator; 4) rf exciter; 5) magnetostatic deflection system; 6) output resonator; 7) collector; 8) energy outlets.

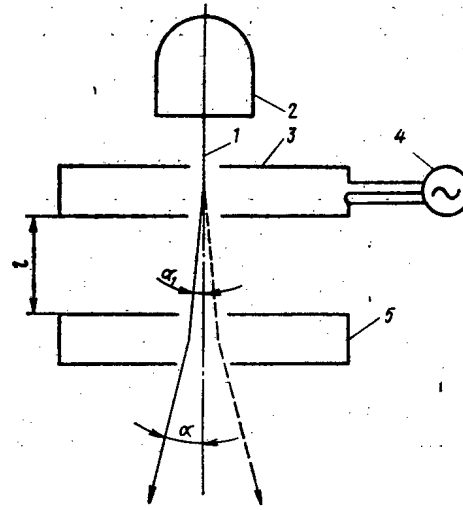


Fig. 4

Fig. 4. A sketch of a sweep system with a passive resonator: 1) electron beam; 2) high-voltage accelerator; 3) sweep resonator; 4) rf exciter; 5) passive resonator; α_1 and α are the sweep angles in the active and passive resonators ($\alpha/\alpha_1 \geq 10$ for $l \geq \lambda/2$).

An analysis of the operation of a gyrocon with these assumptions yields the following relationships for the principal parameters:

Output power

$$P_{\max} = 4 \cdot 10^{-5} (D/L)^2 U_0^{5/2} [(e/mc^2)(U_0/2) + 1]^{3/2}. \quad (3)$$

Electron efficiency

$$\eta_{el} = \frac{\sin \Delta\varphi/2}{\Delta\varphi/2} \left(1 - \frac{\pi D}{\lambda} \sqrt{\frac{\gamma_0 + 1}{\gamma_0 - 1}} \operatorname{tg} \alpha \right). \quad (4)$$

Gain coefficient

$$K_{\max} = \frac{5 \cdot 10^{-6}}{\operatorname{tg}^2 \alpha} \sqrt{\left(\frac{\lambda}{\delta} \right) \frac{P_{\max}}{\gamma_0 (\gamma_0 + 1)}}. \quad (5)$$

Minimum wavelength

$$\lambda_{\min} = (1.6 \cdot 10^6 \sqrt{\gamma_0^2 - 1}) / E_{\max}. \quad (6)$$

Here P_{\max} is the limiting beam power in the gyrocon, roughly equal to its output microwave power; η_{el} is the ratio of the power converted into microwave oscillations to the beam power; K_{\max} is the ratio of the power converted into microwave oscillations to the exciter power for the sweep resonator at λ ; λ_{\min} is the working wavelength of the gyrocon; D and L are the initial diameter and length of the beam; U_0 is the accelerating voltage of the electron source; e and m are the electron charge and mass; c is the speed of light ($mc^2/e = 5.11 \cdot 10^5$ V), $\Delta\varphi$ is the azimuthal dimension of the beam in the output resonator; $\gamma_0 = (eU_0/mc^2) + 1$ is the initial relative energy of the electrons; α is the deflection angle of the deflected beam; δ is the thickness of the skin layer of the sweep resonator wall material; and E_{\max} is the allowable electric field strength in the output resonator. (The coefficients in these equations are for SI units.)

Assuming that it is possible to draw high-power electron beams with $D \approx \lambda/20$ and $L \approx 2\lambda$ through openings in the resonator slits of the gyrocon, we find the maximum attainable gyrocon powers from Eq. (3) (see the table). Presently this power is limited by the capabilities of high-voltage accelerators whose steady state operating parameters correspond to the first column of the table and whose microsecond pulse operating regimes correspond to the third column. Evidently much higher powers are possible in the long term.

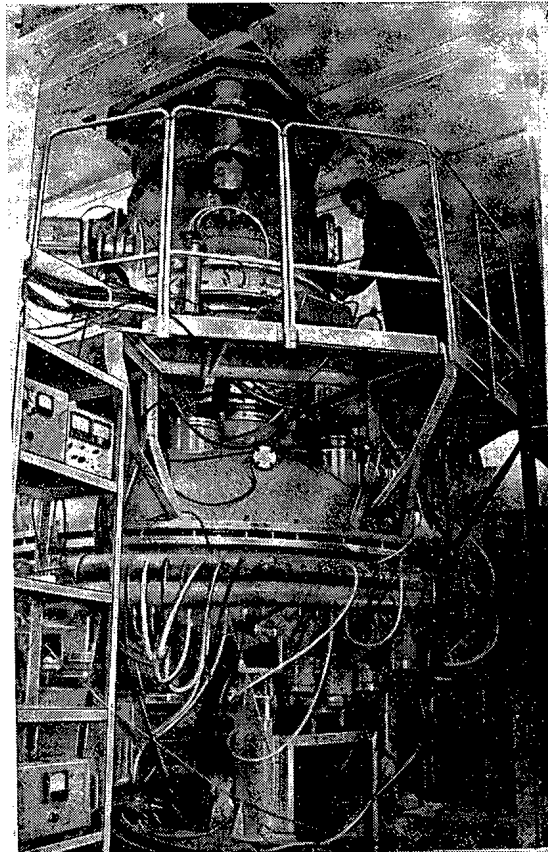


Fig. 5. A cw gyrocon with a design power of 5000 kW.

For electron beams with the parameters given in the table it is possible to obtain an azimuthal dimension $\Delta\varphi \leq 30^\circ$ for sweeping at an angle $\alpha \geq 5.7^\circ$ in the meter- and decimeter-wavelength range. Under these conditions $\eta_{e1} \geq 96\%$ (4). The efficiency of a gyrocon is limited in principle by the following factors: the finite transverse dimension of the beam in the output resonator and the resulting nonidentical slowing-down conditions for the electrons; the scatter in energy of the beam electrons due to the circular sweep; and, the incomplete slowing down of the electrons since they must be removed to the collector. Equation (4) takes only the first two factors into account. An analysis shows that the scatter in the energy during the sweep makes it possible to neglect the effect of the third factor.

There are other reasons for a reduction in the efficiency such as the initial scatter in the energy of the electrons, instability in the beam current, imprecise choice of load on the output resonator, and so on, whose effect may be eliminated by a reasonable choice of the parameters of the stabilizing systems and allowances for the structural elements of the gyrocon. An exception must be made for two such factors whose effect is difficult to remove in a gyrocon: for losses of beam current the level of which we shall assume can reach 1%, and for differences between the wave in the output resonator and the travelling wave. The latter is characterized by the ratio of the maximum and minimum values of the amplitude of the voltage on the transit gap of the output resonator around the annular slit. Assuming that this ratio can be held to a level of 1.03, we then assume a further loss in the efficiency of the gyrocon by 1.5%.

The overall efficiency of the gyrocon is reduced by losses at the walls of the output resonator (2%), as well as by losses in the electronic, vacuum, and other equipment in the gyrocon (3%). A further 5% can be due to losses in the stabilization system for the energy of the high-voltage electron accelerator. These estimates are for maximum power from a cw gyrocon in the meter-wavelength range ($U_0 = 500$ kV). As the accelerating voltage is increased ($U_0 \geq 1000$ kV) the first two loss mechanisms may be neglected. Thus, the limiting values of the electron efficiency (90-95%) and overall efficiency (85-90%) provide an idea of the energy performance of the gyrocon.

For gyrocons with the parameters listed in the table the gain coefficient (5) is $K_{\max} = 100-3000$ (20-35 dB) for $\tan \alpha = 0.1$. However, such a small sweep angle can be achieved in the simplest design (see Fig. 1) only

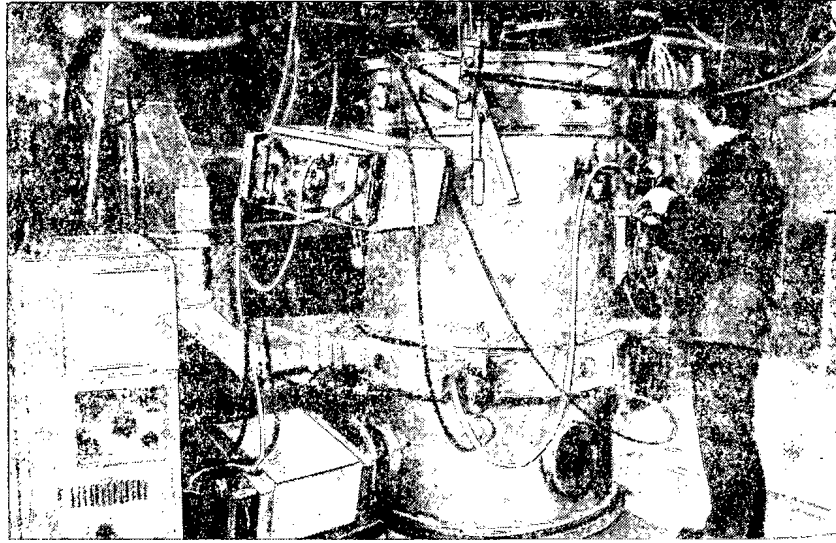


Fig. 6. A pulsed gyrocon with a design power of 200 MW.

by elongating the beam, that is, by reducing the peak power. The gain coefficient can be increased without reducing the power by further bending the beam with a static field which does not require expenditure of rf power. An alternate gyrocon design in which this approach is used with a minimum beam length is shown in Fig. 3 [11]. The additional deflecting system shifts the weakly deflected beam into a plane perpendicular to the axis of the gyrocon. In this variant the output resonator is a rectangular waveguide with an H_{10} wave bent into a ring in the E plane (and not in the H plane as in Fig. 1). The system that deflects the relativistic beam by almost 90° (see Fig. 3) is a magnetostatic system (e.g., in the form of a conical coil) since an electrostatic system does not have the required electrical breakdown strength.

As it moves in the radial magnetic field near the apex of the conical coil an electron describes a trajectory with a double curvature. It not only moves into a plane perpendicular to the axis of the gyrocon, but also acquires an azimuthal velocity component needed for getting into the output resonator at a given angle $\psi(2)$. The maximum magnetic induction in the deflection system does not exceed a few Tesla (for a gyrocon in the decimeter range with $\tan \alpha = 0.1$ and $U_0 = 3000$ kV). The azimuthal defocusing of the beam in such a system does not cause its size to exceed $\Delta\varphi = 30^\circ$.

Thus, a gain coefficient of 20-35 dB can be obtained in a radial gyrocon with $L \approx 2\lambda$, that is, without reducing the maximum power and electrical efficiency mentioned above.

The gain coefficient can be increased further by using a sweep with a passive resonator (Fig. 4) [12]. In such a device the sweep resonator is followed by a passive resonator which is identical to the sweep resonator but has no power inlets. The electron beam, blown up to a small angle by the field in the sweep resonator, excites a traveling wave in the passive resonator which further deflects the electrons. Part of the beam power is then lost to rf heating of the walls of the passive resonator. An analysis of the operation of such a system demonstrates the possibility in principle of increasing the gain coefficient of a gyrocon by a factor of 100 although this involves some reduction in power.

The formula for the minimum working wavelength of a gyrocon (6) takes into account the limitation due to the insufficient breakdown strength of the transit gap of the output resonator. For cw operation of the resonators $E_{\max} \leq 10$ MV/m is permitted, while for microsecond pulse operation $E_{\max} \leq 50$ MV/m is possible. Under these conditions and at the power levels listed in the table $0.3 \leq \lambda_{\min} \leq 1.1$ m for cw operation and $0.06 \leq \lambda_{\min} \leq 0.22$ m for pulsed operation. At wavelengths smaller than 0.1 m, however, the initial assumptions about the attainability of current deposition on the walls of less than 1% and small beam diameters ($D \approx \lambda/20$) at powers of tens and hundreds of megawatts may not be satisfied. This may place significant limitations on the efficiency and power of the gyrocon.

In this frequency range (10^8 - 10^9 Hz) the Q-factor of the sweep resonator is of order 10^4 . The narrow frequency bandwidth associated with this Q evidently limits the use of the gyrocon in information transmission systems. Any expansion of this bandwidth involves a corresponding reduction in the gain coefficient.

These calculations of the basic parameters show that in the meter- and decimeter-wavelength range the gyrocon is capable in principle of developing rf powers considerably higher than that of klystrons [7] and tubes

Realization of these parameters requires solution of a number of complicated technical problems which appear both in the development of any kind of very high power microwave devices and uniquely in the development of gyrocons. The latter problems include the construction of an electronic and optical circuit for the gyrocon such that the current deposition on the walls is less than 1% when the electron current cannot be tracked by a constant magnetic field. This problem involves building an rf system which maintains the high accuracy of the circular polarization of the magnetic field in the sweep resonator and constructing a system for stabilizing the energy of the electrons which come in from the high-voltage accelerator. Highly efficient operation requires a system for precisely maintaining the traveling wave regime in the output resonator.

This listing of problems shows that construction of a complicated device such as the gyrocon is justified only for such values of the main parameters as cannot be obtained with the aid of simpler microwave sources.

Electron-beam microwave devices similar to gyrocons but without their performance characteristics have been described in the literature several times [3-6]. The new components needed for high power, efficiency, and gain levels and which distinguish the gyrocon from these other devices are the following:

- a relativistic electron source;
- a compensating electromagnet or magnetic system for directing the electrons into the output resonator at a given angle which makes it possible to completely stop the relativistic electrons; and
- a magnetostatic deflection system which permits operation with a small sweep angle and a minimum beam length, as is necessary for high gain at high powers.

At the Institute of Nuclear Physics (Novosibirsk) two gyrocons have been built at the initiative of G. I. Budker. They have the following design characteristics.

1. a cw gyrocon for the rf supply of the resonators on the VEPP-4 electron-positron storage device:

Output power	5000 kW
Accelerating voltage	500 kV
Working wavelength	1.65 m
Gain coefficient	23 dB
Overall efficiency	80%

2. A pulsed gyrocon for the rf supply for the linear accelerator in the positron source for the VEPP-4:

Pulsed power	200 MW
Accelerating voltage	2000 kV
Pulse length	10 μ sec
Repetition rate	1 Hz
Working wavelength	0.7 m
Gain coefficient	25 dB

The first sets of bench tests have been made on these devices.

In the cw gyrocon (Fig. 5) [13] an electron beam with a power of more than 1000 kW has been obtained, swept with a frequency of 181 MHz, and passed through a detuned output resonator. The electron energy in this experiment was 220 keV. A 700-kW electron beam (with an electron energy of 210 keV) excited the output resonator of the gyrocon and the cw rf power was 500 kW. The goal of the tests is to develop the electronic and optical circuit of the gyrocon. The current deposition in these experiments was less than 1%.

The pulsed gyrocon (Fig. 6) [14] developed a pulsed power of 40 MW at 430 MHz with a pulse length (at the half power points) of 6 μ sec and a repetition rate of 0.5 Hz (the electron energy is 1300 keV). A gain coefficient of 23 dB was achieved. The purpose of these tests is to develop a scheme for producing positrons with a linear accelerator driven by the gyrocon.

The results obtained previously in tests with a pulsed gyrocon where an electron efficiency of over 90% was measured and the experimental data given here confirm the validity of our ideas on the performance of the gyrocon. The data show that this new microwave source can find application in accelerator and microwave

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INDUSTRIAL ELECTRON ACCELERATORS DESIGNED AT THE INSTITUTE OF NUCLEAR PHYSICS, SIBERIAN BRANCH, ACADEMY OF SCIENCES OF THE USSR

V. L. Auslender and R. A. Salimov

UDC 621.384.6

Budker [1] initiated work on electron accelerators for industrial purposes at the Institute of Nuclear Physics; until recently, this work was done under his direct supervision. The work had attained a large scale by 1966, and up to the end of 1977 the Institute had built and supplied over 45 accelerators.

Electron accelerators are used in many processes in radiation chemistry, in particular for irradiating polyethylene insulators, producing thermally shrinkable components, and killing insects in grain. No radioisotopes are produced on irradiation with electron beams of energy up to 2 MeV, while the dose rates are larger by many orders of magnitude than those available from radioisotope sources. An electron accelerator producing a current of 1 mA is roughly equivalent to a radioactive source of 10^5 Ci. There is no residual radioactivity, while the dose rate is high, and the shielding is comparatively simple, so electron accelerators are widely used in industry where substantial throughputs are required. For instance, a dose of 1 Mrad may be required with 60% use of the electron beam (one-sided irradiation of sheet materials or liquids), in which case an accelerator providing an output beam power of 20 kW can provide a throughput of 4 tons/h.

Various beam-extraction systems are used, which provide radiation fields of various shapes; several types of accelerator had been designed previously in the USSR [2, 3] for use in radiation chemistry, but the electron energies of these did not exceed 1 MeV, while the output at a single point did not exceed 10 kW, which restricted usage considerably.

Here we survey researches at the Institute of Nuclear Physics on the design of electron accelerators for industrial purposes, as well as major uses.

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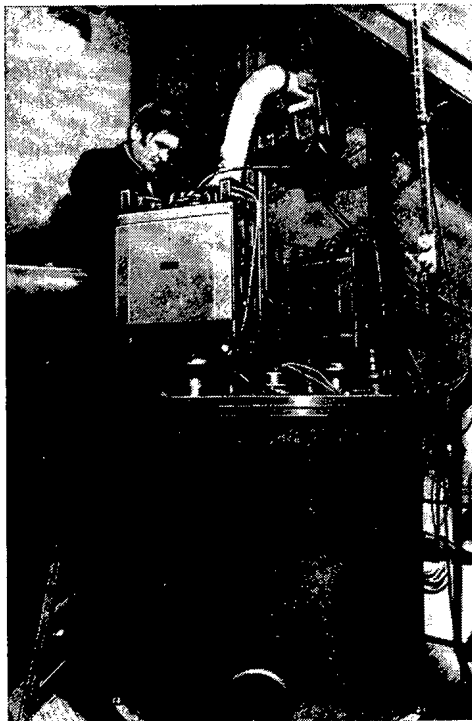


Fig. 1. General view of the ILU-6.

The Institute has designed the ÉLV and ILU* accelerator models, whose mean power output is considerably in excess of 10 kW at energies from 400 to 2000 keV, and which are intended for direct use in industrial processes. In addition, other accelerators of types ÉLIT and ÉSU have been designed, which are used for other purposes.

The ILU-6. This is a pulsed high-frequency accelerator used as a high-power radiation source.

The accelerator uses a toroidal cavity working at 100-127 MHz. The self-excited oscillator is mounted directly on the cavity; this uses a common-grid circuit. The accelerator gap determines the working frequency and the limiting energy, and this can be adjusted from 10 to 20 cm. If a 20-cm gap is used, the resonance frequency is 127.5 MHz, the quality factor is $2 \cdot 10^4$, and the shunt resistance is $4 \cdot 10^6 \Omega$. The cavity is contained in a steel vacuum chamber, which is evacuated by four magnetic-discharge pumps type NORD-250. All the main vacuum seals are made of indium. The working pressure is 10^{-6} - 10^{-7} mm Hg.

The electron beam is separated from the accelerating gap by a grid, which means that a bias can be applied to the cathode to regulate the current. The cathode cannot be mounted directly in the gap because the electrons entering the gap after a certain phase in the accelerating half-wave do not have time to pass across the gap and therefore return and may cause considerable damage to the cathode. This means that it is difficult to maintain a constant power output in the electron beam as the accelerating voltage is reduced. Constant power may be maintained throughout the working energy range (0.4-2 MeV) by increasing the pulse repetition frequency from 50 to 300 Hz while maintaining the same mean power input to the oscillator tube, while on the other hand the gap can be reduced from 20 to 7-10 cm. Both methods allow one to use the same accelerator design to produce a fixed mean power over a reasonably wide energy range.

There is a short magnetic lens immediately after the plate hole, which allows for on-line correction of the transverse beam direction at the exit.

*The following participated in the design of the ILU: V. L. Auslender, G. B. Glagolev, G. I. Kuznetsov, N. A. Livshits, R. M. Lapik, V. A. Polyakov, A. D. Panfilov, A. A. Tuvik, V. G. Chesnokov, and I. L. Chertok; and the following participated in the design of the ÉLV: A. A. Avdienko, V. A. Gaponov, B. M. Korabel'nikov, G. S. Krainov, S. A. Kuznetsov, N. K. Kuksanov, V. I. Kondrat'ev, R. A. Salimov, V. G. Cherepkov, and A. I. Sharapa.

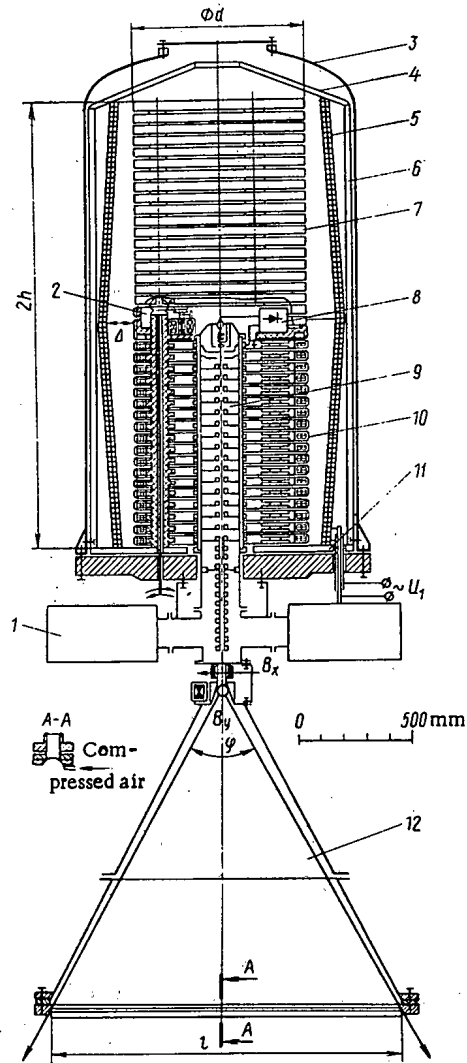


Fig. 2. Design of the ÉLV-3: 1) vacuum pump; 2) high-voltage electrode; 3) case; 4) conical magnet; 5) primary winding; 6) cylindrical magnet; 7, 10) rectifier sections; 8) control unit; 9) accelerator tube; 11) disk magnet; 12) output section.

The ILU-6 is a fairly compact system (Fig. 1) whose major parameters are as follows:

Electron-energy range, keV.....	400-2000
Electron beam power at all energies, kW... 20 (30-35 kW under certain conditions)	
Energy spread in beam, %.....	±10
Pulse length, μ sec	up to 700
Pulse repetition frequency, Hz.....	up to 300
Power supply, line frequency, V.....	3×380
Power drawn, kW.....	up to 100

The ÉLV accelerators [4] have the accelerating voltage provided by a rectifier whose design was initiated in the Institute in 1971; in 1974, the Interdepartmental Commission accepted the ÉLV-1, while the ÉLV-2 was accepted in 1975. Both of these have passed reliability tests over several thousand hours running and have been recommended for routine industrial use. The working principles of the ÉLV-1 have been described in detail elsewhere [4], and here we merely describe the differences in the subsequent models. The following are the parameters of these accelerators:

	ÉLV-1	ÉLV-2	ÉLV-3	ÉLV-4
Electron energy range, keV	400-1000	800-1500	400-700	800-1500
Electron beam power throughout energy range, kW	20	20	50	40
Ripple on accelerator voltage under load, %	2.5	2.5	2.5	2.5
Total dissipation in accelerator at maximum output, kW	3.5	4	5	5
Size of accelerator without output section, m:				
height	2.8	2.8	2.8	2.8
width	1.6	1.6	1.6	1.6

The increased output of the ÉLV-2 (by comparison with the ÉLV-1) has been attained by increasing the number of rectifying sections from 24 to 37 and extending the accelerator tube from 1200 to 1500 mm; the beam power in these first two models was restricted not only by the acceptable ripple at the filter capacitors (restriction on the rectified current) but also by the load characteristics of the high-voltage transformer. Subsequent developments have shown that the existing dimensions can be retained while the capacitance can be doubled in each section. The power restriction imposed by the high-voltage transformer is seen as a nonuniform voltage distribution over the secondary winding, since the diameter of the latter is comparable with the height of the primary winding. This effect has been eliminated in the ÉLV-4 by using a variable winding pitch for the primary. This design allows the rectifier to operate with loads up to 20 MΩ with the maximum voltages on the sections exceeding the mean by only 10%. An electron-beam power of 75 kW has been attained in short-term running with the ÉLV-4. Longer-term tests (over 100 h) have been performed at 40 kW.

The ÉLV-3 differs in design from the other accelerators (Fig. 2); here the two rectifiers are assembled in a common case and have a common magnetic system, and they work in parallel into a single accelerator tube. The upper and lower ends of the rectifier column are at ground potential, while the middle is at the high potential. The distribution of the magnetic flux is symmetrical on account of the disk systems used at the ends, so there is no nonuniformity in the voltage distribution over the secondary. The ÉLV-3 can work into a load of 3 MΩ. The accelerator has been tested at an output of 50 kW for over 100 h.

All the ÉLV series accelerators have high immunity to damage from the overvoltages that may arise from breakdown in the gas or vacuum insulation, which is a very considerable advance over existing Soviet accelerators.

The major elements have been maximally standardized for the various styles of accelerator; e.g., there have been virtually no changes in the size of the pressure vessel, vacuum system, or the control circuits, or in the measuring and gas-handling systems.

Pulsed and High-Power Electron Accelerators. The ILU and ÉLV are in direct industrial use, while the Institute has also developed the pulsed ÉLIT [6] and the high-power ÉSU [7].

The parameters of the ÉLIT are given below.

	ÉLIT-0.8A	ÉLIT-1B	ÉLIT-2
Mean electron energy in pulse, MeV.....	0.8	1.1	1.5
Mean beam power, kW.....	0.8	4.5	10
Energy spread in beam, %.....	10	10	15
Pulse length, μsec.....	1	2.5	3.5
Pulse power, MW.....	8	20	30
Window size (transmission through foil), mm.....	153×35	420×65	420×65
Input voltage to rectifier, kV.....	20	10	10
Line power drawn, kW.....	3	20	40
Dimensions of accelerator without output section, m:			
height.....	0.76	1.15	1.92
diameter.....	0.4	1.0	1.0

The ÉLIT-2 has means of outputting a focused beam into the atmosphere; this has operated stably during long-term tests.

The ÉSU accelerators have been designed to provide outputs in the megawatt range; the following are the design parameters of the ÉSU-2:

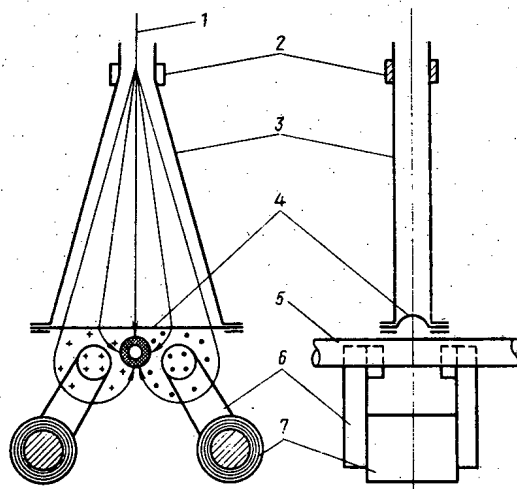


Fig. 3. Device for irradiating cylindrical components: 1) beam; 2) scan system; 3) tube; 4) foil; 5) specimen; 6) magnet; 7) winding.

Electron energy range, keV	30-500
Electron current, A	up to 10
Ripple on accelerating voltage, %	±2.5
Working mode	continuous
Dimensions of rectifier, m:	
height	3.5
diameter	2
Dimensions of accelerator tube with power and control unit, m:	
height	4
diameter	1.2

The ÉSU-1 is intended only for short-term operation; it can operate with an evacuated target for 10 sec to provide a power of 1100 kW at 250 keV. An air beam providing a power of 400 kW has also been extracted with this accelerator [6].

The ÉSU-2 is currently in use for testing the gyrocon high-frequency generator [9]; this has yielded a power of 1000 MW, but in this instance in continuous mode.

Beam Outlet Device. The ILU and ÉLV are equipped with various devices for outputting the electron beam into the atmosphere. The simplest device is one employing output of a linearly expanded beam through a titanium foil cooled by air. This can handle up to 0.15 mA/cm². The working life of the foil exceeds 1000 h. The standard exit window is 75 × 980 mm and can handle currents up to 80 mA. The beam deflection angle is 30°. Each current pulse is uniformly distributed over the foil in the ILU. A sawtooth scan of the beam along and across the window is used in the ÉLV. The transverse scanning frequency is 1075 Hz, while the longitudinal frequency is usually 50 Hz, but this can be raised to 400 Hz if there is some special requirement. The nonuniformity in the dose distribution along the window received by a component moving transverse to the window is not more than ±10%. The 500-keV electrons passing through the foil are scattered through about 45°, but this angle decreases linearly as the electron energy increases.

Cylinders (tubes and the like) may be irradiated with two types of device employing a ring scan; Fig. 3 shows a device of the first type. The beam emerges through a standard window and is directed by permanent magnets onto the cylindrical workpiece. This device has been tested at electron energies from 800 to 1500 keV with the ÉLV-2. The proportion of the current reaching the workpiece is 70% for a tube of diameter 6 cm or 50% at 4 cm, while the deviation from uniformity of the current distribution in azimuth at the tube is not more than ±10%. The same system has been used to irradiate three tubes of diameter 2 cm simultaneously. These receive 50% of the accelerator current. The deviation from uniformity is also not more than ±10%. The system can handle tubes up to 15 cm in diameter. The ILU and ÉLV can be fitted with similar devices.

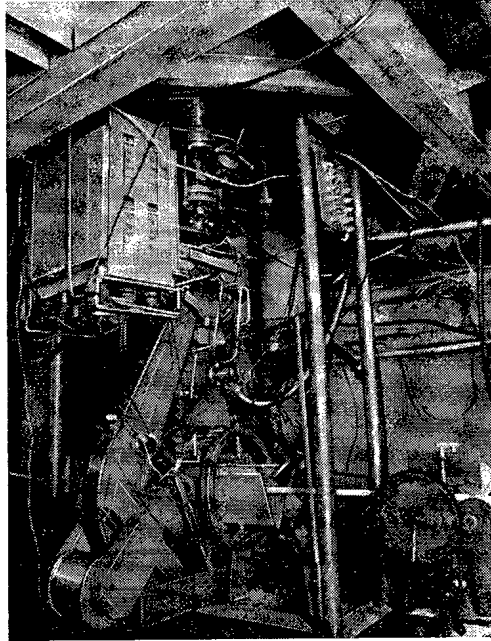


Fig. 4. General view of output section in ILU accelerator.

Another type of annular system can be used only with the ILU, since the pulsed operation allows one to deflect each pulse into a distinct channel and then direct it by means of magnets onto window foils. This means that a three-dimensional radiation field can be set up with a nonuniformity of not more than $\pm 15\%$ and with less loss. Figure 4 gives a general view of the device.

An output beam is also obtained in a device in which the beam escapes from the vacuum via a system of four holes [5]; the fall in pressure from atmospheric to 10^{-6} mm Hg in the accelerator tube is provided by five pumping stages, of which the last (magnetic-discharge pumps) is used with the accelerator in any mode of beam output. The parameters of the ÉLV beam are such that a hole of diameter 1 mm will transmit 20 kW with a divergence of $5 \cdot 10^{-2}$ rad. An output of 75 kW is available from a hole of size about 1.5 mm. The total power dissipated on all stops is not more than 3 kW.

Industrial Uses. Various processes in radiation technology have already been based on these accelerators built at the Institute of Nuclear Physics, Siberian Branch, Academy of Sciences of the USSR. Irradiation of polyethylene components is the most common use. Crosslinking gives improved chemical resistance to polyethylene, as well as mechanical strength and thermal stability, which means that components, e.g., tubes, insulated wires, and the like, can work for long periods at 135°C and for short periods up to 250°C .

Nine industrial lines have been built by the Ministry of the Electrical Engineering Industry to produce wires and cables with heat-resistant polyethylene insulation by means of ÉLV accelerators.

The ILU has been used in a system for producing heat-resistant polyethylene tubes for hot-water supply; 1 ton of these tubes saves 5 tons of metal tube. The economic gain from using a single line based on an ILU is given by the Khar'kov Institute of Physicochemical Research as more than 1 million rubles per year. A further use of irradiation is to produce shrinkable tubes based on polyethylene and fluorinated polymers. Processing lines with ILU-6 accelerators for the purpose are being built by the Plastik organization in Moscow and by the Plastpolymer organization in Leningrad.

Also, equipment being designed for producing regenerated butyl rubber from worn components and for radiation purification of industrial effluents containing large amounts of surfactants.

The ÉLV beams provide high energy densities; e.g., a beam power of 60 kW corresponds to a heat-flux density at the surface of the target under vacuum of $2 \cdot 10^6$ W/cm². The flux density at the surface of a target in helium at atmospheric pressure at 3-5 cm from the output device is $5 \cdot 10^5$ W/cm². These beam features mean that the beams can be used in electron-beam technology under vacuum or in gases. Preliminary tests show that the ÉLV-4 gives good performance in cutting and welding metals.

Accelerators have also many uses in agriculture; the ELV has been used in killing insects in grain, and an experimental system at Novosibirsk Branch of the All-Union Grain Research Institute has been used in processing about 1200 tons of grain. The tests proved very favorable, and a prototype industrial system of throughput 200 tons/h was built in 1978.

The construction of large pig-breeding stations (over 100,000 pigs each) has raised problems in processing effluents; in 1978, an ILU-6 was built into a prototype system for processing effluents with a throughput of 1000-2000 m³/day.

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GAS POROSITY ARISING ON ANNEALING IRRADIATED BERYLLIUM

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UDC 621.039.548

Neutron irradiation produces helium in beryllium; the solubility of the gas in the metal is negligible. Gas swelling then occurs if the temperature is high enough. It has proved extremely difficult to elucidate the behavior of materials of complex structure, since many processes act together. On existing views [1, 2], gas porosity is produced in the main by the displacement and fusion of pores. The rates of these processes are controlled by various factors, in particular the mechanism responsible for pore motion, the number of structural defects whose interactions affect pore mobility, and so on. Model calculations have to be compared with the available experimental evidence in order to elucidate how far these general concepts apply to gas swelling in beryllium.

To avoid complicating the model, we consider only the fusion of pores due to Brownian motion; the external forces acting on the pores and the interactions with other defects are neglected. Naturally, the results are to be compared with the porosity observed in some small region free from all other types of defect. Correspondingly, the dimensions of such pores must also be small.

Little evidence is available on the initial stages of the growth of gas porosity in beryllium; an electron micrograph has been published [3] of the porosity observed in single-crystal beryllium irradiated in the SM-2 reactor with a fluence of $2.6 \cdot 10^{21}$ neutrons/cm² ($E \geq 0.8$ MeV) at 60°C, which was followed by stepwise annealing over the range 100-800°C (temperature raised 50°C per hour). Although this photomicrograph does not meet the requirements fully (some of the pores appear to occur at dislocations), it may be compared with our calculations. The dislocations restrict the mobility of only a small proportion of the pores visible in the photomicrograph, so one assumes that the interaction with dislocations does not greatly retard the growth of the porosity, at least in the initial period. The comparison with experiment must show whether this assumption is correct.

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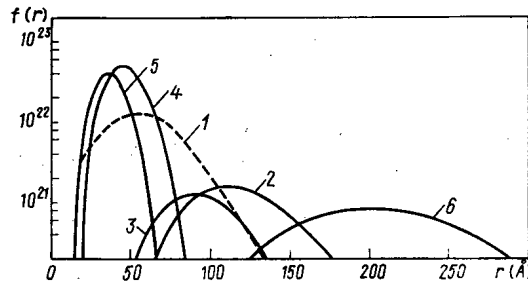


Fig. 1. Size distribution $f(r)$ in cm^{-4} ($f(r)dr$ is the number of pores of radius from r to $r + dr$ in cm^3): 1) observed (from photomicrograph [3]); 2-5) calculated for bulk diffusion; 2, 3) calculation with $DT = 0.19 \exp(-38.6/RT)$ (first series); 4, 5) calculated for $DT = 0.7 \exp(-50/RT)$ (second series); 2, 4) $C = 0.75 \cdot 10^{20}$ atom He/ cm^2 ; 3, 5) $C = 0.25 \cdot 10^{20}$ atom He/ cm^3 ; 6) calculation on 30-min annealing for surface diffusion.

The calculations on the kinetics of collision and fusion for randomly wandering gas bubbles were based on the following concepts.*

We assume that the gas in the pores is not ideal, while the pressure is balanced by surface-tension forces. There is then the following relationship between the radius r_i of pore size i (a pore of size i contains i times as much gas as a pore of the first size):

$$\frac{2\gamma}{r_i} \left(\frac{4\pi}{3} r_i^3 - imb \right) = imkT. \quad (1)$$

Here γ is the surface tension, b is the van der Waals constant, and m is the number of gas atoms in a pore of the first size. It is assumed that the pores grow only as a result of fusion occurring on collision by Brownian motion. Any pore containing two or more gas atoms is considered as stable with respect to gas loss.† Then the time course of the pore concentration for size i , viz., F_i , is described by

$$\frac{dF_i}{dt} = 4\pi \sum_{j=1}^J F_j(t) F_{i-j}(t) r_{j, i-j} D_{j, i-j} - 4\pi \sum_{j=1}^{\infty} F_i(t) F_j(t) r_{ij} D_{ij} (1 + \delta_{ij}), \quad (2)$$

where

$$J = \begin{cases} i/2 & \text{for even } i \\ \frac{i-1}{2} & \text{for odd } i. \end{cases}$$

The first sum on the right describes the increase in F_i due to fusion of pores whose size is less than i , while a second describes the fall in F_i due to fusion of pores of size i with pores of other sizes.

Here $r_{ij} = r_i + r_j$ and $D_{ij} = D_i + D_j$ are the sums of the radii and of the diffusion coefficients, respectively, for pores of sizes i and j ; the equations have been solved by finite-difference methods. The porosity observed in [3] developed on annealing beryllium saturated with helium, so the model calculation was based on a state corresponding to a solid solution of helium atoms. This was characterized as a distribution of pores identical in size each containing one gas atom (pores of the first size). Therefore, $F_i(0) = C\delta_{i1}$, where C is the initial concentration of the dissolved gas.

The calculations on the porosity employed various concepts of the mechanism of pore displacement; if this is due to surface diffusion, the diffusion coefficient for the pores [4] is

*See [4] for details.

†This corresponds to homogeneous.

$$D_b^s = \frac{3}{2\pi} D_{s0} \left(\frac{a}{r} \right)^4 e^{-\frac{2\gamma\Omega}{rkT}}. \quad (3)$$

Here D_{s0} is the observed surface-diffusion coefficient, γ is surface tension, Ω is the volume per atom in the lattice, $a = \Omega^{1/3}$, r is pore radius, k is Boltzmann's constant, and T is absolute temperature.

The diffusion coefficient for the pores responsible for the bulk diffusion is

$$D_b^p = \frac{3}{4\pi} D_v \left(\frac{a}{r} \right)^3, \quad (4)$$

where D_v is the bulk self-diffusion coefficient.

In the surface-tension case, a calculation representing annealing for only 30 min at 800°C gives a porosity far greater than that actually observed (curve 6 of Fig. 1 as compared with curve 1), particularly as the observations relate to prolonged stepwise annealing up to 800°C.*

A calculation that would reproduce the experiment completely should result in even greater enlargement of the pores; therefore, the porosity observed after stepwise annealing [3] can hardly be explained in terms of surface diffusion in its simple form (without introducing additional factors, in particular retardation by structural defects).

Two series of calculations were performed for bulk diffusion, in both cases simulating stepwise annealing [3]; these involved two different assumptions about the parameters that determine the temperature dependence of the self diffusion:

$$D(T) = D_0 e^{-\frac{Q}{RT}}. \quad (5)$$

One series was based on values derived from experiment: $D_0 = 0.19 \text{ cm}^2 \cdot \text{sec}^{-1}$ and $Q = 38.6 \text{ kcal/g-atom}$ [6]. However, the latter quantity is in poor agreement with the empirical relationship between Q in cal/g-atom and $32T_{mp}$, which expresses the activation energy in terms of the melting point, and which is known to apply for a large number of elements [6]. Therefore, the second series was based on $Q = 50 \text{ kcal/g-atom}$, which follows also from the empirical relationship and the value $D_0 = 0.7 \text{ cm}^2 \cdot \text{sec}^{-1}$ derived by calculation [6].

The values of Ω and γ were taken as in the calculation on surface diffusion; in [3] we find no data on the amount of gas formed in the specimen on irradiation in a fluence of $2.6 \cdot 10^{21}$ neutrons/cm² ($E > 0.8 \text{ MeV}$), so we performed two calculations in each series. In one of them we assumed that $C = 0.25 \cdot 10^{20} \text{ cm}^{-3}$, and in the other that $C = 0.75 \cdot 10^{20} \text{ cm}^{-3}$; estimates showed that the initial gas concentration must have fallen within these limits.

Curves 2-5 of Fig. 1 show the calculated distributions; curves 2 and 3 (series 1) go with curves 4 and 5 (series 2) to show that a threefold change in the initial concentration results in a relatively small change in the size distribution. The comparison with the photomicrograph shows that the theoretical distributions in both cases are fairly close to the observed one (i.e., curve 1).

As only approximate values are available for some of the parameters used in the calculations, while the model incorporates only fusion and neglects interaction with structural defects, we consider that the agreement with experiment is reasonably good.

This indicates the main mechanism for gas porosity is fusion of randomly wandering pores in this particular case.

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*The following values were used here: $b = 4 \cdot 10^{23} \text{ cm}^3$; $\Omega = 8.4 \cdot 10^{-24} \text{ cm}^3$; $C = 0.75 \cdot 10^{20} \text{ cm}^{-3}$; $t = 1 \text{ h}$; $D_{s0} = 10^{-6} \text{ cm}^2 \cdot \text{sec}^{-1}$ (this value was found from the observed curve relating $\log D_{s0}$ to T/T_{mp} , which itself was constructed from the available data on surface diffusion in various elements [5] and $\gamma = 1700 \text{ erg/cm}^2$ (this value was derived by extrapolation from $\gamma = 1200 \text{ erg/cm}^2$ near T_{mp} [6] on the basis of the $\gamma(T)$ applicable to copper).

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THE EFFECT OF A HORIZONTAL SHIFT OF THE REGULATOR

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The layers of construction materials or coolant surrounding an absorbing unit, fuel element, or regulator can exert a significant effect on the logarithmic derivative of the neutron flux at the surface [1]. It often turns out that the absorbing unit is positioned eccentrically with respect to the guide channel in which it is located. The motion of the coolant in the channel can cause vibrations of the unit, resulting in a redistribution of the thickness of the gap occupied by the coolant, and thereby in a change of the effective boundary conditions. This situation is one of the causes of variation in the reactivity of a reactor (the appearance of "neutron noise"). A number of such causes have been discussed in detail in [2-4].

In this paper the variation of the reactivity of a reactor is derived in the one-group diffusion approximation as a function of the eccentricity of the axis of the "black" absorbing unit (the regulator) with respect to the axis of the guide channel.

Let an infinitely long coolant channel of radius R_1 with a regulator of effective radius R_2 be placed in the active zone of a reactor with uniformly distributed neutron sources at its position. Let us assume that the conditions of applicability of the diffusion approximation are satisfied for the active zone ($i = 1$) and the gap with coolant ($i = 2$): $\Sigma_{a1} \ll \Sigma_{s1}$, $\bar{l} \equiv 2\delta \equiv 2(R_1 - R_2) \gg \Sigma_{s2}^{-1}$. Since a decrease of neutrons in this region is determined mainly by their leakage into the regulator, it is possible to neglect absorption in the surrounding medium by setting $\Sigma_{a1} = 0$ when formulating the effective boundary conditions (and only in this case). The neutron source is assumed to be located at infinity ($r_\infty \rightarrow \infty$) [5].

We will discuss within the framework of perturbation theory the variation $\delta k/k$ of the reactivity of the reactor caused by a small shift in the regulator axis by an amount $\Delta < \delta \ll R_1$. Since the unperturbed problem possesses azimuthal symmetry, one can expect that the variation in reactivity will be proportional to Δ^2 and consequently it is necessary also to take account of terms of the second order of smallness [1, 5, 6]. The variation of the principal eigenvalue λ of the one-group diffusion equation of the reactor caused by a variation in the logarithmic derivative at the surface of the channel R_1 can be determined from the relation

$$\int_V \varphi_0^+(r) (\lambda - \lambda_0) \hat{Q}\varphi(r) dr = -D_1 \int_F d\mathbf{r}_F \varphi_0^+(r_F) \varphi(r_F) [1/\gamma(r_F) - 1/\gamma_0(R_1)], \quad (1)$$

where V is the volume of the active zone, F is the lateral surface of the channel, \hat{Q} is the neutron production operator, D_1 is the diffusion coefficient of the active zone, λ_0 , $\varphi_0^+(r)$, $\varphi_0(r)$, and $1/\gamma_0(R_1)$ are the unperturbed values of the eigenvalue, the importance function, the neutron flux, and the logarithmic derivative, respectively; and λ , $\varphi^+(r)$, $\varphi(r)$, and $1/\gamma(r_F)$ are the perturbed values of the same quantities.

In the case of small perturbations, $\Delta/R_1 \ll \Delta/\delta \equiv \varepsilon \ll 1$, the logarithmic derivative can be expanded into a series whose terms are assumed to be proportional to the corresponding powers of the smallness parameter ε :

$$\frac{1}{\gamma(r_F)} = \frac{1}{\gamma_0} + \sum_{n=1}^{\infty} \frac{1}{\gamma_n(r_F)}. \quad (2)$$

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 (It follows from the explicit expression for $1/\gamma(r_F)$ given below that $1/\gamma(r_F)$ is expandable precisely into a power series with ϵ^n , where $n \geq 0$, and it does not contain terms of the type $(\epsilon \ln \epsilon)^n$ or terms like ϵ^n , where $n < 0$.) Then, following [6-8], λ and φ can also be expanded into series with respect to the corresponding order of the perturbation:

$$\lambda = \lambda_0 + \sum_{n=1}^{\infty} \lambda_n; \quad (3)$$

$$\varphi(r) = \varphi_0(r) + \sum_{n=1}^{\infty} \varphi_n(r). \quad (4)$$

Substituting these expansions into Eq. (1) and keeping terms out to the second order of the perturbation theory, we formally obtain:

$$(\lambda_1 + \lambda_2) \int_V \varphi_0^+ \hat{Q} \varphi_0 dr = -\lambda_1 \int_V \varphi_0^+ \hat{Q} \varphi_1 dr - D_1 \int_F \varphi_0^+ \varphi_0 \left(\frac{1}{\gamma_1} + \frac{1}{\gamma_2} \right) dr_F - D_1 \int_F \varphi_0^+ \varphi_1 \frac{1}{\gamma_1} dr_F. \quad (5)$$

From this we have for the correction of first order λ_1 [1, 5, 6]

$$\lambda_1 = -D_1 \int_F \varphi_0^+ \varphi_0 \frac{1}{\gamma_1} dr_F / \int_V \varphi_0^+ \hat{Q} \varphi_0 dr. \quad (6)$$

As will be shown, λ_1 turns out to be equal to zero, which is entirely natural, since in this problem it should not depend on the direction of deviation of the regulator from the channel axis. Then λ_2 is determined from Eq. (5) by the following expression:

$$\lambda_2 = \frac{-D_1 \int_F \varphi_0^+ (\varphi_1 \gamma_1^{-1} + \varphi_0 \gamma_2^{-1}) dr_F}{\int_V \varphi_0^+ \hat{Q} \varphi_0 dr}. \quad (7)$$

In order to obtain λ_2 from Eq. (7), it is necessary to find beforehand the first-order correction to the unperturbed neutron flux $\varphi_1(r_F)$. Using the method of [6-8], we obtain the same equation for the correction φ_1 as for the unperturbed flux φ_0 (the nonhomogeneous addition to its right-hand side, $\sim \lambda_2 \hat{Q} \varphi_0$, should be discarded as a quantity of higher order of smallness), however with a boundary condition at R_1 containing a nonhomogeneous right-hand side:

$$n \nabla \varphi_1 - \varphi_1 / \gamma_0 |_{r=R_1} = \varphi_0 / \gamma_1 |_{r=R_1}. \quad (8)$$

As a result of the integration of the right-hand side of (8) with Green's function $G(r_F, r_{F_0})$ of the operator of the unperturbed equation of a reactor with a surface source at the point r_{F_0} [9], $\varphi_1(r_F)$ can be found:

$$\varphi_1(r_F) = \int_{F_0} dr_{F_0} G(r_F, r_{F_0}) \varphi_0 / \gamma_1(r_{F_0}). \quad (9)$$

In the particular case of the one-group diffusion approximation for an infinitely long reactor with a control regulator, Green's function will satisfy the equation

$$\left[\frac{1}{r} \frac{\partial}{\partial r} \left(r \frac{\partial}{\partial r} \right) + \frac{1}{r^2} \frac{\partial^2}{\partial \varphi^2} + \kappa_0^2 \right] G(r, r_0) = -\frac{S_0 \delta(r-r_0) \delta(\varphi-\varphi_0)}{r} \quad (10)$$

with homogeneous boundary conditions at the channel surface R_1

$$\left(\frac{\partial}{\partial r} - \frac{1}{\gamma_0} \right) G(R_1, r_0) = 0 \quad (11)$$

and on the effective lateral surface of the active zone R_0 . Here κ_0^2 is a geometrical parameter of the unperturbed reactor, and S_0 is the power of a unit length of the surface source. If the size of the active zone of the reactor R_0 is much larger than R_1 , $\kappa_0 R_1 \ll 1$, and the radius of the regulator is much larger than the diffusion coefficient, $\gamma_0 / R_1 \approx (D_1 / D_2) (\delta / R_1) \ll 1$ [see Eq. (17)], then the perturbation of the neutron flux will be localized near the channel surface and will not depend on the boundary conditions at the outer boundary of the reactor and thereby on κ_0 (with an accuracy out to terms of the order $\kappa_0 R_1$). At the channel surface the main part of Green's function will be of the form $(S_0 \gamma_0 / R_1) \delta(\varphi - \varphi_0)$ with an accuracy out to terms of the order $S_0 (\gamma_0 / R_1)^2 \cos(\varphi - \varphi_0)$ (see the Appendix). Then, using (9), we obtain with the corresponding accuracy

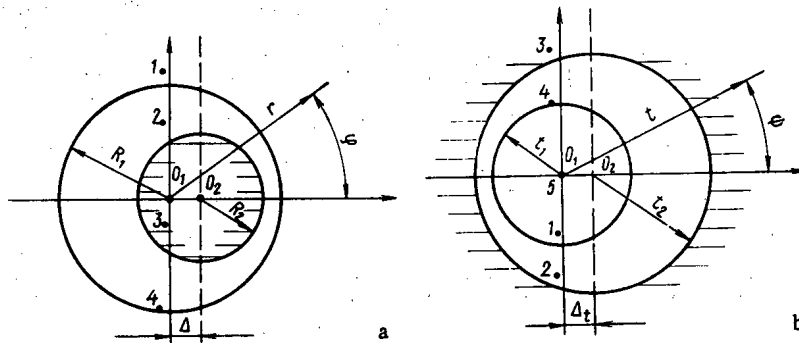


Fig. 1. Eccentric arrangement of the regulator in the channel on the (a) (r, φ) plane and (b) (t, ψ) plane: 1) active zone; 2) gap with coolant; 3) regulator; 4) guide channel; and 5) neutron source.

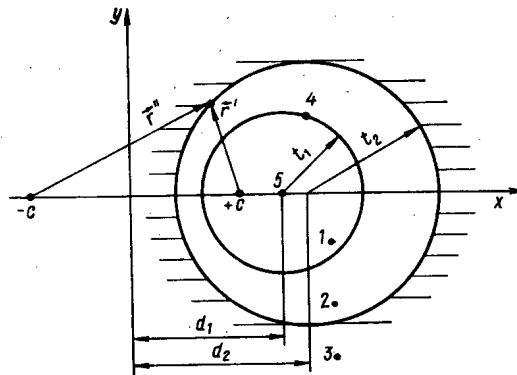


Fig. 2. Conversion to bipolar coordinate system (x, y) plane: $(-c, +c)$ are the two poles; r' and r'' are the radius vectors of an arbitrary point (x, y) .

$$\varphi_1(r_F) = \varphi_0(R_1) [\gamma_0/\gamma_1(r_F)], \tag{12}$$

from which, substituting (12) into (7), we find

$$\lambda_2 = \frac{-D_1 \int_F \varphi_0 \varphi_0^+ (\gamma_0/\gamma_1^2 + 1/\gamma_2) dr_F}{\int_V \varphi_0^+ \hat{Q} \varphi_0 dr}. \tag{13}$$

Converting from λ to $\delta k/k$, we obtain an expression for the variation of the reactivity of the reactor caused by a shift of the regulator axis by an amount Δ with an accuracy out to the second order of perturbation theory:

$$\frac{\delta k}{k} = \frac{-2\pi R_1 (D_1/\gamma_0) \psi_0^+(R_1) \varphi_0(R_1) \left[\frac{\varphi_0^+(R_1)}{\psi_0^+(R_1)} \right] I(\Delta)}{\int_S dS \varphi_0^+(r) \hat{Q} \varphi_0(r)} \equiv \left(\frac{\delta k}{k} \right)_0 \frac{\varphi_0^+(R_1)}{\psi_0^+(R_1)} I(\Delta), \tag{14}$$

where S is the area of the transverse cross section of the active zone, ψ_0^+ is the importance function of neutrons upon removal of the regulator, $(\delta k/k)_0$ is the reactivity of the regulator in the channel when $\Delta = 0$, and $I(\Delta)$ is a coefficient which takes account of the deviation of the regulator axis by an amount Δ :

$$I(\Delta) = -\frac{\gamma_0^2}{2\pi} \int_0^{2\pi} d\varphi [1/\gamma_0 \gamma_2(\varphi) + 1/\gamma_1^2(\varphi)]. \tag{15}$$

Equation (15) determines the total variation of the reactivity caused by two physical effects: the first term takes account of the total variation of the absorbing power of the regulator located in an azimuthally symmetric flux φ_0 ; the second term takes account of the effect of a variation in absorption due to azimuthal asymmetry arising in the flux φ_1 . In order to calculate $I(\Delta)$ it is necessary to find the correction to the unperturbed

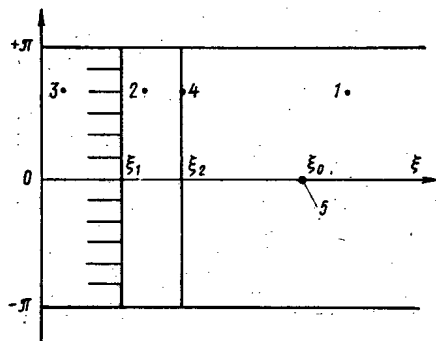


Fig. 3. Arrangement of the regulator in the bipolar coordinate system (the ξ, η plane).

boundary condition on the channel surface $1/\gamma_0$, which takes account of the shift of the regulator axis, i.e., the coefficients $1/\gamma_n$ of the series (2).

The effective boundary condition on the channel surface $1/\gamma$ (r_F) is obtained as a result of solving a system of one-group equations for the neutron fluxes Φ_1 in the active zone and Φ_2 in the coolant gap on the condition of continuity of the fluxes and flows on the channel boundary R_1 and for a null boundary condition on the effective surface of the regulator R_2 . When $\Delta = 0$, we will have, as a result of solving the system

$$\frac{1}{r} \frac{\partial}{\partial r} \left(r \frac{\partial \Phi_i^{(0)}}{\partial r} \right) = - \frac{S_0 \delta (r-r_\infty)}{2\pi r D_i} \delta_{i,1},$$

$$\delta = \begin{cases} 1 & i=1 \\ 0 & i \neq 1 \end{cases} \quad (16)$$

an expression for the logarithmic derivative at the boundary R_1 in the diffusion approximation [1]:

$$\frac{1}{\gamma_0} = - \frac{\partial \Phi_i^{(0)}}{\Phi_i^{(0)} \partial r} \Big|_{r=R_1} = \frac{D_2}{D_1 R_1 \ln(R_1/R_2)} \quad (17)$$

When $\Delta \neq 0$, the neutron fluxes $\Phi_i(r, \varphi)$ in the gap and the active zone will depend on the azimuthal angle φ (the axis of the polar coordinate system coincides with the channel axis) (Fig. 1a). Then the system of equations for $\Phi_i(r, \varphi)$ takes the form

$$\left[\frac{1}{r} \frac{\partial}{\partial r} \left(r \frac{\partial}{\partial r} \right) + \frac{1}{r^2} \frac{\partial^2}{\partial \varphi^2} \right] \Phi_i(r, \varphi) = - \frac{S_0 \delta (r-r_\infty)}{2\pi r D_i} \delta_{i,1} \quad (18)$$

with the boundary conditions

$$\Phi_1(R_1, \varphi) = \Phi_2(R_1, \varphi); \quad D_1 \frac{\partial \Phi_1(r, \varphi)}{\partial r} \Big|_{r=R_1} = D_2 \frac{\partial \Phi_2(r, \varphi)}{\partial r} \Big|_{r=R_1}; \quad (19a, b)$$

$$\frac{\partial \Phi_i(r, \varphi)}{\partial \varphi} \Big|_{\varphi=0, \pi} = 0 \text{ symmetry condition}; \quad (19c)$$

$$\Phi_2(R_2) = 0, \quad (19d)$$

where the regulator boundary $R_2(r_1, \varphi_2)$ is specified by the equation

$$r_2^2 - 2r_2 \Delta \cos \varphi_2 + \Delta^2 = R_2^2. \quad (19e)$$

Let us change in advance to the reference system ($t = 1/r, \psi = \varphi$) for the solution of system (18)-(19), thereby shifting the source from infinity to the origin of coordinates (Fig. 1b):

$$\left[\frac{1}{t} \frac{\partial}{\partial t} \left(t \frac{\partial}{\partial t} \right) + \frac{1}{t^2} \frac{\partial^2}{\partial \psi^2} \right] \Phi_i(t, \psi) = - \frac{S_0 \delta(t) \delta_{i,1}}{2\pi t D_i}; \quad (20)$$

$$\Phi_1(T_1, \psi) = \Phi_2(T_1, \psi); \quad D_1 \frac{\partial \Phi_1(t, \psi)}{\partial t} \Big|_{t=T_1} = D_2 \frac{\partial \Phi_2(t, \psi)}{\partial t} \Big|_{t=T_1}; \quad (21a, b)$$

$$\left. \frac{\partial \Phi_i(t, \psi)}{\partial \psi} \right|_{\psi=0, \pi} = 0; \quad \Phi_2(T_2) = 0, \quad (21c)$$

where the boundary $T_2(t_2, \psi_2)$ is specified by the equation

$$t_2^2 - 2t_2 \Delta_t \cos \psi_2 + \Delta_t^2 = T_2^2; \quad (22)$$

$$T_1 \equiv R_1^{-1}, \quad T_2 \equiv R_2 (R_2^2 - \Delta^2)^{-1};$$

$$\Delta_t \equiv \Delta (R_2^2 - \Delta^2)^{-1}. \quad (23)$$

The solution of system (20)-(21) can be found exactly by means of conversion to the bipolar coordinate system. According to [9-11], the bipolar coordinates (ξ, η) are introduced in the following way (Fig. 2):

$$\xi \equiv \ln \frac{r''}{r'} = \operatorname{arcth} \frac{2xc}{x^2 + y^2 + c^2}; \quad +\infty > \xi > -\infty; \quad (24a)$$

$$\equiv \theta' - \theta'' = \operatorname{arctg} \frac{2yc}{x^2 + y^2 - c^2}; \quad +\pi \geq \eta \geq -\pi. \quad (24b)$$

Here r'' and r' are the distances of an arbitrary point (x, y) from the two poles $\pm c$; θ and θ'' are the angles between the radius vectors r' and r'' and the abscissa. The coordinates of the poles are selected in such a way that the ratios $|r''|/|r'|$ of the absolute values of the vectors remain constant as their ends revolve around the circles T_2 and T_1 . It follows from this fact that

$$c^2 = d_{1,2}^2 - T_{1,2}^2, \quad (25)$$

and the equations

$$d_1^2 - T_1^2 = d_2^2 - T_2^2,$$

$$T_2 - T_1 = \Delta_t \quad (26)$$

yield the values $d_{1,2}$:

$$d_1 = \frac{R_1^2 - R_2^2 - \Delta^2}{2\Delta R_1}; \quad (27a)$$

$$d_2 = \frac{R_1^2 (R_2^2 + \Delta^2) - (R_2^2 - \Delta^2)^2}{2\Delta R_1 (R_2^2 - \Delta^2)}. \quad (27b)$$

The coordinates of the points of the circles in bipolar system $(\xi_{1,2})$, which are related to their radii $T_{1,2}$ by the relationship [9, 10]

$$T_{1,2} = \frac{c}{|\operatorname{sh} \xi_{1,2}|}, \quad (28)$$

are constant and equal to

$$\xi_{2,1} = \frac{1}{2} \ln \frac{d_{2,1} + c}{d_{2,1} - c} = \frac{1}{2} \ln \frac{R_1^2 (R_2^2 \pm \Delta^2) - (R_2^2 - \Delta^2) \pm}{R_1^2 (R_2^2 \pm \Delta^2) - (R_2^2 - \Delta^2) -}$$

$$\frac{+(R_2^2 - \Delta^2) \sqrt{(R_1^2 - R_2^2)^2 + \Delta^4 - 2\Delta^2 (R_1^2 + R_2^2)}}{-(R_2^2 - \Delta^2) \sqrt{(R_1^2 - R_2^2)^2 + \Delta^4 - 2\Delta^2 (R_1^2 + R_2^2)}}. \quad (29)$$

The source is situated at the point

$$\xi_0 = \ln \frac{d_1 + c}{d_1 - c} \equiv 2\xi_1; \quad \eta_0 = 0. \quad (30)$$

The system of Eqs. (20) along with the boundary conditions (21) will take the form

$$\left(\frac{\partial^2}{\partial \xi^2} + \frac{\partial^2}{\partial \eta^2} \right) \Phi_i(\xi, \eta) = -\frac{S_0 \delta(\xi - \xi_0) \delta(\eta)}{D_i} \delta_{i,1}; \quad (31)$$

$$\Phi_1(\xi_1, \eta) = \Phi_2(\xi_1, \eta); \quad (31a)$$

$$D_1 \frac{\partial \Phi_1(\xi, \eta)}{\partial \xi} \Big|_{\xi=\xi_1} = D_2 \frac{\partial \Phi_2(\xi, \eta)}{\partial \xi} \Big|_{\xi=\xi_1}; \quad (31b)$$

$$\frac{\partial \Phi_1(\xi, \eta)}{\partial \eta} \Big|_{\eta=0, \pm\pi} = 0; \quad \Phi_2(\xi_2, \eta) = 0 \quad (31c, d)$$

in bipolar coordinates (Fig. 3). Since the problem possesses symmetry with respect to η , the finite Fourier cosine transform [12] can be used for its solution:

$$\tilde{\Phi}_i(\xi, m) \equiv \int_0^\pi \Phi_i(\xi, \eta) \cos m\eta d\eta, \quad m=0, \pm 1, \pm 2, \dots \quad (32)$$

Then the solution will have the form

$$\Phi_i(\xi, \eta) = \frac{1}{\pi} \tilde{\Phi}_i(\xi, 0) + \frac{2}{\pi} \sum_{m=1}^{\infty} \cos m\eta \tilde{\Phi}_i(\xi, m). \quad (33)$$

Applying (32) to system (30)-(31), finding $\tilde{\Phi}_i(\xi, m)$ and substituting them into (33), we obtain expressions for the neutron fluxes in the active zone (Φ_1) and in the gap (Φ_2):

$$\begin{aligned} \Phi_1(\xi, \eta) = & \frac{S_0}{2\pi D_1} \left\{ (\xi_0 - \xi_1) + \frac{D_1}{D_2} (\xi_1 - \xi_2) + 2 \sum_{m=1}^{\infty} \frac{\cos m\eta}{mA_m} \left[\frac{D_2}{D_1} \operatorname{ch} m(\xi_1 - \xi_2) \operatorname{sh} m(\xi_0 - \xi_1) \right. \right. \\ & \left. \left. + \operatorname{sh} m(\xi_1 - \xi_2) \operatorname{ch} m(\xi_0 - \xi_1) \right] e^{-m(\xi - \xi_1)} \right\} \phi(\xi - \xi_0) + \frac{S_0}{2\pi D_1} \left\{ (\xi - \xi_1) + \frac{D_1}{D_2} (\xi_1 - \xi_2) + 2 \sum_{m=1}^{\infty} \frac{\cos m\eta}{mA_m} \right. \\ & \left. \times \left[\frac{D_2}{D_1} \operatorname{ch} m(\xi_1 - \xi_2) \operatorname{sh} m(\xi - \xi_1) + \operatorname{sh} m(\xi_1 - \xi_2) \operatorname{ch} m(\xi - \xi_1) \right] e^{-m(\xi_0 - \xi_1)} \right\} \phi(\xi_0 - \xi); \quad (34) \end{aligned}$$

$$\Phi_2(\xi, \eta) = \frac{S_0}{2\pi} \left[\frac{\xi - \xi_2}{D_2} + \frac{2}{D_1} \sum_{m=1}^{\infty} \frac{\cos m\eta}{mA_m} e^{-(\xi_0 - \xi_1)} \operatorname{sh} m(\xi - \xi_2) \right] \phi(\xi_1 - \xi) \phi(\xi - \xi_2), \quad (35)$$

where

$$\phi(\xi - \xi_0) = \begin{cases} 1 & \xi > \xi_0; \\ 0 & \xi < \xi_0; \end{cases} \quad (36)$$

$$A_m \equiv \frac{D_2}{D_1} \operatorname{ch} m(\xi_1 - \xi_2) + \operatorname{sh} m(\xi_1 - \xi_2). \quad (37)$$

Only the part of solution (34) for $\xi < \xi_0$, which can be identically rewritten in the form

$$\begin{aligned} \Phi_1(\xi - \eta) = & \frac{S_0}{2\pi D_1} \left\{ (\xi - \xi_2) + 2 \sum_{m=1}^{\infty} \frac{\cos m\eta}{m} e^{-m(\xi_0 - \xi_1)} \operatorname{sh} m(\xi - \xi_2) + (D_1 - D_2) \left[\frac{\xi_1 - \xi_2}{D_2} \right. \right. \\ & \left. \left. + \sum_{m=1}^{\infty} \frac{\cos m\eta \operatorname{sh} 2m(\xi_1 - \xi_2) e^{-m(\xi_0 + \xi - \xi_1 - \xi_2)}}{m [D_2 \operatorname{ch} m(\xi_1 - \xi_2) + D_1 \operatorname{ch} m(\xi_1 - \xi_2)]} \right] \right\} \phi(\xi_0 - \xi), \quad (38) \end{aligned}$$

is of interest for determination of the effective boundary condition at R_1 .

The first two terms in (38) describe the distribution of the neutron flux in the event of the complete absence of a gap between the active zone and the regulator surface ($D_2 \equiv D_1$), and the second two terms (in square brackets) describe the variation of the flux upon the replacement of part of the material of the active zone by the material of the gap.

Having derived from (38) an expression of the flow component normal to the channel surface, we find the value of the logarithmic derivatives $1/\gamma(\eta)$ on its surface $\xi = \xi_1$:

$$\begin{aligned} \frac{1}{\gamma(\eta)} \equiv & \frac{VD}{R_1^2 \Phi(\xi, \eta)} \frac{\partial \Phi(\xi, \eta)}{\partial \xi} \Big|_{\xi=\xi_1} = \frac{D_2}{D_1 R_1} \left(\frac{\operatorname{ch} \xi_1 - \cos \eta}{CR_1} \right) \times \\ & \times \left[1 + 2 \sum_{m=1}^{\infty} \frac{\cos m\eta e^{-m(\xi_0 - \xi_1)}}{1 + (D_1/D_2) \operatorname{th} m(\xi_1 - \xi_2)} \right] \left[(\xi_1 - \xi_2) + 2 \sum_{m=1}^{\infty} \frac{\cos m\eta e^{-m(\xi_0 - \xi_1)} \operatorname{th} m(\xi_1 - \xi_2)}{m [1 + (D_1/D_2) \operatorname{th} m(\xi_1 - \xi_2)]} \right]^{-1}, \quad (39) \end{aligned}$$

where, according to [10],

$$D \equiv \left(\frac{\partial \xi}{\partial x} \right)^2 + \left(\frac{\partial \xi}{\partial y} \right)^2 = \frac{(\operatorname{ch} \xi - \cos \eta)^2}{c^2}.$$

When $\Delta = 0$, Eq. (39) turns into Eq. (17) for $1/\gamma_0$ (since as $\Delta \rightarrow 0$, $\sqrt{D} \rightarrow R$, $\exp(-\xi_0) \rightarrow 0$, $(\xi_1 - \xi_2) \rightarrow \ln R_1/R_2$). Substituting $1/\gamma_1$ and $1/\gamma_2$ into (15) and performing the integration in the bipolar coordinate system over the variable η , we find the value of the coefficient $I(\Delta)$:

$$I(\Delta) = -\frac{\gamma_0^2 R_1}{2\pi} \int_{-\pi}^{+\pi} \frac{d\eta}{\sqrt{D}} \left[\frac{1}{\gamma_1^2(\eta)} + \frac{1}{\gamma_0 \gamma_2(\eta)} \right] \approx \left(-1 + \frac{1}{2\pi} \int_{-\pi}^{+\pi} d\eta \frac{R_1}{\sqrt{D}} \cdot \frac{\gamma_0}{\gamma} \right) + \left[1 + \frac{1}{2\pi} \int_{-\pi}^{+\pi} d\eta \frac{R_1}{\sqrt{D}} \left(\frac{\gamma_0^2}{\gamma^2} - \frac{2\gamma_0}{\gamma} \right) \right]. \quad (40)$$

Let us consider the case of a small gap thickness, $\delta \ll R_1$. Then, as follows from (29), $\xi_1 - \xi_2 \ll 1$:

$$(\xi_1 - \xi_2) \approx \frac{\sqrt{\delta^2 - \Delta^2}}{R_1} \left[1 + \frac{1}{2} \frac{\sqrt{\delta^2 - \Delta^2}}{R_1} + O\left(\frac{\delta^2 - \Delta^2}{R_1^2}\right) \right]. \quad (41)$$

The ratio γ_0/γ takes the form

$$\frac{\gamma_0}{\gamma} \approx \frac{\sqrt{D}}{R_1} \frac{\delta}{\sqrt{\delta^2 - \Delta^2}} \left[1 + \frac{\delta}{2R_1} - \frac{\sqrt{\delta^2 - \Delta^2}}{2R_1} + O\left(\frac{\gamma^2}{R_1^2}\right) \right] \approx \frac{\sqrt{D}}{R_1} \left[1 + \frac{1}{2} \frac{\Delta^2}{\delta^2} \left(1 + \frac{\delta}{2R_1} \right) + O\left(\frac{\delta^2}{R_1^2}\right) \right], \quad (42)$$

where the last expression is valid when $\Delta \ll \delta$.

Having substituted (42) into (40), we obtain the value of the coefficient $I(\Delta)$ in the second order of the perturbation theory; each term contributes an identical amount, to an accuracy of $(\delta/R_1)(\Delta^2/\delta^2)$, to the total effect:

$$I(\Delta) \approx -\frac{\Delta^2}{\delta^2} [1 + O(\delta/R_1)]. \quad (43)$$

As follows from Eqs. (14) and (40), the coefficient $I(\Delta)$ characterizes the relative variation of the reactivity in the case of an eccentrically inserted regulator, and therefore it does not depend on the characteristics of the medium in the case of small deviations ($\Delta \ll \delta$).

For this reason one can hope that the error from the contribution of nondiffusion corrections to the quantities γ_0/γ_1 and γ_0/γ_2 will be small in this case if the conditions of the diffusion approximation are satisfied properly enough. In the case in which $\delta \ll R_1$, but $\Delta \ll \delta$ as previously, and $(D_1/D_2) \text{ th } (\xi_1 - \xi_2) \ll 1$, an additional factor $2\delta[(R_1 + R_2) \ln R_1/R_2]^{-1}$ appears in Eq. (43). Having substituted the values of $I(\Delta)$ into (14), we obtain an estimate of the effect of the eccentricity of the regulator Δ on $\delta k/k$ of the reactor. It follows from the estimate given that vibrations of the regulating rods in the coolant flow at frequencies of the order of a few or tens of Hertz can serve as a source of neutron noise, producing a background in connection with physical measurements. Eq. (43) only determines the maximum value of $\delta k/k$. In order to obtain the true value it is necessary to take account of the interference of the regulators (if there are several of them), and knowledge must be had of the correlation function of the temporal variation of $\Delta = \Delta(t)$ and a number of other factors. For the VVR-M research reactor the maximum (without taking account of the factors enumerated above) relative variation of the reactivity, $\delta k/k$, may amount to $\sim 2 \cdot 10^{-2}$ of the total weight of the regulator, which can lead to oscillations of $\sim 2\%$ in the power (the regulator diameter along the boron is 26 mm, the inner diameter of the channel is 30 mm, the moderator is light water [13], and $\bar{L} \equiv 2\delta_{\text{eff}} \equiv 2(\delta + 0.7\lambda_{\text{tr}}^{\text{H}_2\text{O}}) \approx 2\lambda_{\text{tr}}^{\text{H}_2\text{O}}$).

APPENDIX

Let us find Green's function $G(r, r_0, \varphi, \varphi_0)$ in the one-group diffusion approximation for an infinite cylindrical reactor with central regulators which satisfies Eq. (10) with homogeneous boundary condition on the channel surface R_1 (11) and a null boundary condition on the equivalent inner radius R_0 . Using the cosine transform (32), one can represent the Green's function similarly to (33) in the form of an expansion in $\cos n(\varphi - \varphi_0)$.*

$$G(r, r_0, \varphi, \varphi_0) = \frac{1}{2\pi} \left[\tilde{G}_0(\rho, \rho_0) + 2 \sum_{n=1}^{\infty} \tilde{G}_n(\rho, \rho_0) \cos n(\varphi - \varphi_0) \right], \quad (44)$$

*Of course, Green's function may, according to [7, 9], be written immediately in the form of an expansion in a system of eigenfunctions of the operator of the equation; however, such a representation is inconvenient for investigation.

whose coefficients $\tilde{G}_n(\rho, \rho_0)$ satisfy the equation ($\rho \equiv \kappa_0 r$, $\rho_0 \equiv \kappa_0 r_0$):

$$\left[\frac{1}{\rho} \frac{\partial}{\partial \rho} \left(\rho \frac{\partial}{\partial \rho} \right) + \left(1 - \frac{\kappa^2}{\rho^2} \right) \right] \tilde{G}(\rho, \rho_0) = \frac{-s_0 \delta(\rho - \rho_0)}{\rho} + \beta_n^2 \psi_n(\rho) \psi_n^*(\rho_0) \delta_{n,0} \quad (45)$$

and the boundary conditions

$$\begin{aligned} \kappa_0 \gamma_0 \tilde{G}'_n(\kappa_0 R_1, \rho_0) &= \tilde{G}'_n(\kappa_0 R_1, \rho_0); \\ \tilde{G}_n(\kappa_0 R_0, \rho_0) &= 0, \end{aligned} \quad (46)$$

where $\psi_n(\rho) \equiv J_n(\rho) + C_n Y_n(\rho)$ are the eigenfunctions of the operator of Eq. (45), $\beta_n^{-2} \equiv \int_{\kappa_0 R_1}^{\kappa_0 R_0} \rho d\rho \psi_n^2(\rho)$ is the normalizing factor, and $\delta_{n,0}$ is the Kronecker delta. The constants C_n , $n \neq 0$, are determined from the boundary conditions on R_0 , $\psi_n(\kappa_0 R_0) = 0$, and the constant C_0 is determined from the condition $\kappa_0 \gamma_0 \psi_0'(\kappa_0 R_1) = \psi_0(\kappa_0 R_1)$ on R_1 .

Since κ_0 is an eigenvalue of the operator of Eq. (45) when $n = 0$ (the Laplacian of the unperturbed reactor), $\tilde{G}_0(\rho, \rho_0)$ should be the so-called generalized Green's function of Eq. (45) for $n = 0$ [14], and $\tilde{G}_n(\rho, \rho_0)$ are ordinary Green's function when $n \neq 0$. As a result, we obtain the solutions:

$$\begin{aligned} \tilde{G}_0(\rho, \rho_0) &= S_0 \beta_0^2 \{ \psi_0(\rho) \chi(\rho_0) + \psi_0(\rho_0) \chi(\rho) \\ &\quad - \frac{\chi(\kappa_0 R_0)}{f(\kappa_0 R_0)} [\psi_0(\rho) f(\rho_0) + \psi_0(\rho_0) f(\rho)] - \\ &\quad - C \psi_0(\rho) \psi_0(\rho_0) + \frac{\pi}{2} \beta_0^{-2} [\psi_0(\rho) f(\rho_0) \vartheta(\rho - \rho_0) + \psi_0(\rho_0) f(\rho) \vartheta(\rho_0 - \rho)] \}, \end{aligned} \quad (47)$$

where

$$\begin{aligned} C &= \beta_0^2 \{ \chi^2(\kappa_0 R_0) - \chi^2(\kappa_0 R_1) - \frac{(\kappa_0 R_0)^2}{2f(\kappa_0 R_0)} \chi(\kappa_0 R_0) \psi_0'(\kappa_0 R_0) f'(\kappa_0 R_0) \\ &\quad - \frac{(\kappa_0 R_0)^2}{2} \left[\frac{\pi}{2} \beta_0^{-2} - \frac{\chi(\kappa_0 R_0)}{f(\kappa_0 R_0)} \right] [\psi_0(\kappa_0 R_1) f(\kappa_0 R_1) + \psi_0'(\kappa_0 R_1) f'(\kappa_0 R_1)] \}, \end{aligned} \quad (48)$$

$f(\rho) \equiv Y_0(\rho)$; $\chi(\rho) \equiv \rho/2 [J_1(\rho) + C_0 Y_1(\rho)]$ is a particular solution of the nonhomogeneous Eq. (45) when $n = 0$ (equivalent to the solution "in resonance").

$$\tilde{G}_n(\rho, \rho_0) = \frac{S_0 \pi}{2(1 - B_n C_n)} [\psi_n(\rho_0) \zeta_n(\rho) \vartheta(\rho_0 - \rho) + \psi_n(\rho) \zeta_n(\rho_0) \vartheta(\rho - \rho_0)], \quad (49)$$

where $\zeta_n(\rho) \equiv B_n J_n(\rho) + Y_n(\rho)$, and the values B_n are determined by the boundary conditions on R_1 [$\kappa_0 \gamma_0 \zeta_n'(\kappa_0 R_1) = \zeta_n(\kappa_0 R_1)$]. Proceeding in Eqs. (47) and (49) to the limit as $r \rightarrow R_1$ and $r_0 \rightarrow R_1$, using the Bessel function representation $J_n(\kappa_0 R_1)$ and $Y_n(\kappa_0 R_1)$ for $\kappa_0 R_1 \ll 1$, and taking account of the inequality $\gamma_0/R_1 \ll 1$, we obtain, to an accuracy out to terms $\sim S_0(\gamma_0/R_1)^2$ and $\sim S_0(\gamma_0/R_1)(\kappa_0 R_1)$, an expression for Green's function (44);

$$G(R_1, R_1, \varphi - \varphi_0) \approx S_0(\gamma_0/R_1) \delta(\varphi - \varphi_0). \quad (50)$$

We neglect in Eq. (50) both the effect of the external boundary of the reactor (the absence of κ_0) and the azimuthal spreading out of neutrons along the surface of the unit. However, the correction which takes account of these effects (which can be done, since the exact expression for $G(r, r_0, \varphi - \varphi_0)$ is known), leads to quantities of the next order of smallness.

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SEPARATE DETERMINATION OF LOCAL LEVELS
OF BORON AND LITHIUM IN MINERALS,
ROCKS, AND ORES

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Various problems in mineralogy and geochemistry require examination of the spatial distributions of trace amounts of elements in minerals, rocks, and ores; this requires nondestructive local microanalysis methods of high sensitivity. It has been shown that (n, α) radiography can be used to advantage in determining boron [1-4], but it is applicable in the original form with high sensitivity and accuracy only if the specimen is either entirely free from lithium or contains the element at a level not exceeding local boron content in the specific area. Higher ratios of lithium to boron allow one to determine only the joint distribution or content of the two elements.

A method has been devised for specimens in which the boron level exceeds that of lithium by an order of magnitude or more [2-4].

Lithium may be present at much higher levels in rocks and in certain rock-forming minerals; further, the ratio of the boron and lithium concentrations is often not obvious. It is often difficult to determine the concentrations or concentration ratios if the levels of the elements are low, and attempts have therefore been made to devise very sensitive methods for separate determination of each of these elements in a single area; the spatial distribution is also of considerable interest.

Here we report a test of one possible method on some geological objects.

The reactions $^{11}\text{B}(p, \alpha)^8\text{Be}$ and $^7\text{Li}(p, \alpha)^4\text{He}$ were used to detect the boron and lithium, because these have high cross sections at low proton energies [5, 6]. The reactions are also exothermic, so it is possible to eliminate the effects of elastically scattered particles and measure the secondary radiation with almost no background.

The measurements were made with the cascade generator at the Nuclear Physics Research Institute, Moscow University, at a proton energy of 500 keV; Fig. 1 shows the system. The diameter of the proton beam at the specimen could be varied with stop system from 50 to 500 μ . The specimens were polished rock sections, and each were set up at the center of the scattering tube on a holder allowing of displacement by steps of 50 μ in two mutually perpendicular directions. A scan along a line or over an area indicated the spatial distribution of the element. The escaping α particles were recorded with a surface-barrier silicon detector having a solid angle of 0.3 st. The angular distribution of the escaping α particles is virtually isotropic for each of these reactions. For convenience, the silicon detector was set at 150° to the proton beam. The energies of the α particles from boron and lithium for a proton energy of 500 keV are, respectively, $E_\alpha \approx 3.7$ and $E_\alpha \approx 8.0$ MeV, so it is simple to distinguish the two species. The energy resolution of the detector was 30 keV. A

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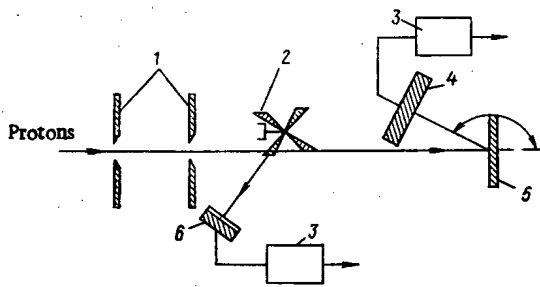


Fig. 1

Fig. 1. The system: 1) collimator; 2) rotating gold target; 3) spectrometer; 4) α -particle detector; 5) specimen; 6) monitor.

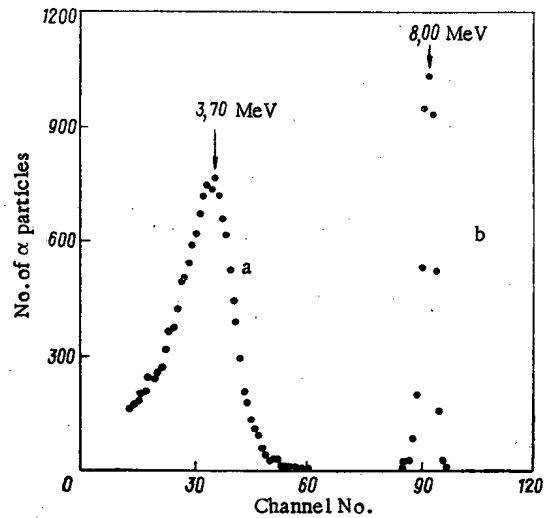


Fig. 2

Fig. 2. Energy distribution of α particles from standard specimens containing: a) 0.7% B_2O_3 ; b) 5.3% Li_2O .

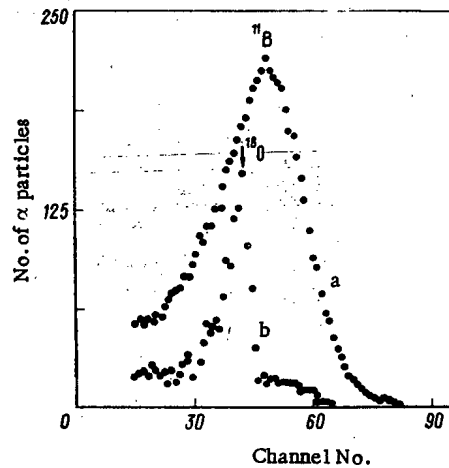


Fig. 3. Energy distribution of: a) α particles from borate glass containing 0.7% B_2O_3 ($\times 20$); b) fused silica containing less than $10^{-5}\%$ impurities.

plastic film of thickness 10μ was placed in front of the detector to eliminate elastically scattered protons. The detector pulses passed to ordinary amplifiers and a multichannel pulse-height analyzer. The proton beam was monitored by another silicon detector that recorded the elastically scattered protons from a thin layer of gold deposited on a beryllium substrate, which was periodically inserted into the beam.

The calibration runs were performed on standard specimens containing known amounts of boron and lithium. Figure 2 shows the energy distributions of the α particles. Here and subsequently, the spectra have been reduced to the same number of incident protons. Figure 2 shows that there are two groups of α particles that relate to boron and lithium, respectively. As $^{11}B(p, \alpha)^8Be$ involves excitation of the broad level of 8Be , the α particles have a broad energy spectrum. The cross sections of the reactions on boron and lithium increase rapidly with the proton energy, so the effective thickness of the reaction layer is substantially less than the penetration depth of the proton beam; it is about 5μ . The energy losses of the escaping α particles in this

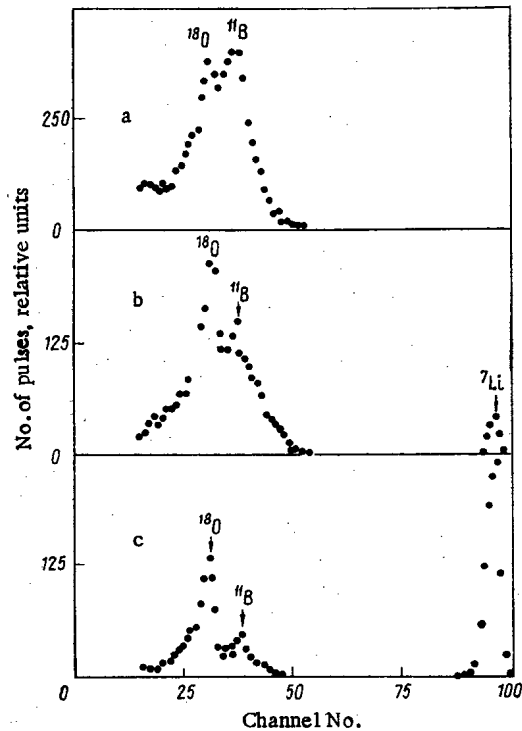


Fig. 4. Energy distributions of α particles from: a) epidote; b) muscovite; c) biotite.

layer do not exceed 1 MeV, so this factor does not make a substantial contribution to the shape of the α -particle spectrum. The numbers of pulses in each group are related to the boron and lithium concentrations in an area whose size is determined by the width of a proton beam. The exact relationship was examined by means of standard specimens; we used a borate glass containing 0.7% B_2O_3 and a lithium mica containing 5.3% Li_2O . Clearly, the limit of detection is dependent on the measurement time and the corresponding statistical accuracy. Under our working conditions, it was reasonable to work with a count rate of about 10^2 h^{-1} , and in that case the limit of detection for boron was about $10^{-7}\%$ and for lithium about $10^{-6}\%$, in both cases in the absence of interfering elements. The statistical error of the result is about 5% if about 10^3 counts are recorded.

There are interfering elements that can emit α particles of similar energy in response to protons with $E_p = 500 \text{ keV}$, viz., ^{18}O and ^{15}N . Oxygen is always present in a mineral or rock, so an estimate must be made of the effect, for which purpose we compared borate glass with fused quartz. Figure 3 shows the α -particle spectra of these specimens. Curve a is the spectrum of α particles for $^{11}B(p, \alpha)^8Be$ for a glass containing uniformly distributed boron, ($CB_2O_3 = 0.7\%$), while curve b is for fused quartz for which chemical analysis indicated that the impurity level did not exceed $10^{-5}\%$. The peak on this curve is due to $^{18}O(p, \alpha)^{15}N$, since (p, α) on silicon has a threshold energy substantially above 500 keV. The proportion of ^{18}O in the natural mixture of oxygen isotopes is 0.2% [7], but the cross section of the reaction is small at $E_p = 500 \text{ keV}$, so the peak corresponding to α particles with E of about 3.3 MeV [8, 9] makes itself felt at boron concentrations of $10^{-2}\%$ or less. In that case it is necessary to increase the measurement time in order to improve the accuracy. We found that it was possible to distinguish the α particles from $^{11}B(p, \alpha)^8Be$ and $^{18}O(p, \alpha)^{15}N$ if the resolution was high enough.

The energy of the α particles emitted in $^{15}N(p, \alpha)^{12}C$ at $E_p = 500 \text{ keV}$ is about 3.9 MeV; any nitrogen in the specimen could distort the α -particle curve for boron. We determined the contribution from the α particles due to ^{15}N by comparing the curves given by the borate glass and polyacrylonitrile C_3H_4N . It is clear that nitrogen makes an appreciable contribution to the total number of α -particle counts only if the concentration in the specimen is about 100 times that of the boron. Calibration experiments are required with specimens containing known amounts of nitrogen in order to correct for this effect.

Nitrates and minerals containing the NH_4 group are of limited occurrence in nature; further, nitrogen is characteristic of organic compounds, which can be identified under the microscope. If the specimen contains more than $C_{org} \geq 0.1\%$, it is necessary to apply a correction in the (p, α) result, e.g., by means of (n, α) radiography. Nitrogen-bearing minerals and organic compounds were virtually absent from our specimens, so the contribution from ^{15}N to the total number of α particles could be neglected.

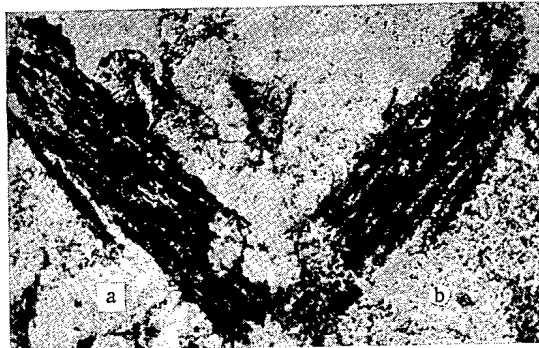


Fig. 5

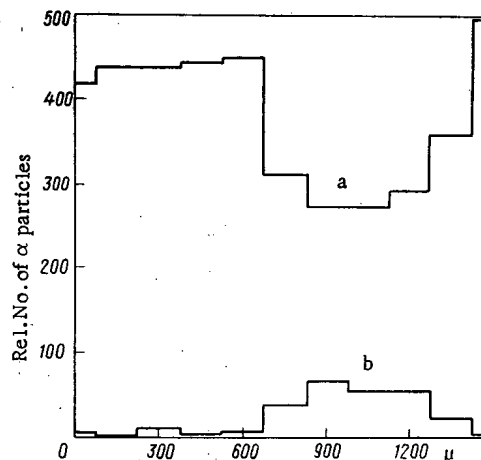


Fig. 6

Fig. 5. Biotite examined by (n, α) radiography: a) polished section; b) detector. The dots in the detector correspond to α tracks that cannot be assigned to lithium or boron on the basis of (n, α) radiography alone.

Fig. 6. Results of scan along a certain direction in a rock section (specimen 3302k) containing boron and lithium: a, b) numbers of α particles from $^{11}\text{B}(p, \alpha)^7\text{Li}$ and $^7\text{Li}(p, \alpha)^4\text{He}$, respectively; a grain of biotite lies between 675 μ and 1425 μ from the start.

TABLE 1. Levels of CB_2O_3 and CLi_2O in Specimens Indicated by (p, α)

Mineral	C, 10^{-2} %		Mineral	C, 10^{-2} %	
	B_2O_3	Li_2O		B_2O_3	Li_2O
Epidote	3,3	—	Stilpnomelane	5,1	1,7
Hedenbergite	3,2	—			
Muscovite	1,9	2,2	Biotite	—	7,3
Muscovite	0,7	1,4	Biotite	0,3	11,1
Muscovite	1,9	1,8	Biotite	0,7	8,1
Muscovite	2,9	1,0	Sericite	1,1	0,9

This method of determining boron and lithium separately has been tested on several specimens of minerals and rocks having various concentration ratios for these elements.

We examined minerals from skarns as polished sections. Parts a-c of Fig. 4 showed the results from separate determination of the two elements in rock-forming minerals: epidote, muscovite, and biotite. We found that epidote (Fig. 4a) contains boron ($\text{CB}_2\text{O}_3 \approx 3.3 \cdot 10^{-2}\%$) but no lithium (less than $10^{-6}\%$). Similar results were obtained for another rock-forming mineral from skarns, viz., hedenbergite. Figure 4b shows the α -particle spectrum from one of the four muscovites examined (Table 1), where the levels of boron and lithium were similar, with two of them having lithium predominant and two of them having boron. The elements were present in much the same relationship in the sericite. The stilpnomelane specimen also contained boron and lithium, but with the former somewhat predominant. All the biotite specimens had $\text{CLi} > \text{CB}$ (Table 1). Figure 4c shows the α -particle spectrum recorded for a biotite specimen. Figure 5 shows the spatial distribution of the α tracks produced in (n, α) radiography on the same biotite specimen, which indicates that the elevated track density reproduces the biotite grain, but it is impossible to determine which element (boron or lithium) is responsible for the elevated track density. The (p, α) results solve the problem completely.

All the α -particle spectra recorded with real specimens having $\text{CB} \leq 10^{-2}\%$ also showed a contribution from the α particles from $^{18}\text{O}(p, \alpha)^{15}\text{N}$ (Fig. 4a and c). Therefore, the total number of α particles due to $^{18}\text{O}(p, \alpha)^{15}\text{N}$ was subtracted from the total α count in order to determine the boron concentration.

The α -particle spectra, as corrected in this way, allow one to assign the boron and lithium to localized areas in the mineral and rock grains, as well as to determine the spatial distributions along selected lines.

Table 1 gives the boron and lithium content of certain minerals derived from the secondary α -particle spectra.

The spatial distributions of the boron and lithium were determined by scanning the proton beam over the surface; Fig. 6 shows a histogram for a particular direction passing through a biotite grain in a rock section (specimen 3302k). It is clear that the level of lithium in the biotite is higher than that in the rock, whereas the converse applies to the boron. Scans on other parts of the specimen showed that the balance between c_B and c_{Li} may vary widely from mineral to mineral. These variations are of considerable geochemical interest.

It is thus possible to determine the levels of boron and lithium localized in a specimen and to establish the spatial distributions; the method is applicable to geological, mineralogical, and geochemical specimens, particularly whenever the problem concerns the distribution of boron or lithium in rock-forming minerals. The technique can detect variations in composition, effects due to the behavior of both elements during mineral formation, and other features relevant to the geochemistry of boron and lithium in endogeneous and exogenous processes.

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SOLVENT-EXTRACTION EQUILIBRIA IN REPROCESSING FAST-REACTOR URANIUM - PLUTONIUM FUEL

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Exact information is required on the equilibrium distribution of major amounts of uranium and plutonium in fuel reprocessing by solvent extraction in order to ensure nuclear safety and to provide a basis for automating processes; this involves the system 30% TBP-diluent- HNO_3 (0.1-4.0 M) - U(VI) (0-300 g/liter) - Pu(IV) (0-100 g/liter) - Pu(VI) (0-30 g/liter).

There are many papers [1-5] on this system, but data on the equilibrium in extraction at high levels of Pu(IV) in the presence and absence of uranium are very scanty [6-10, 18].

A system containing levels of Pu(IV) up to 48 g/liter in the initial solution has been examined [6, 7], and empirical equations were given for the equilibrium as a relationship between the apparent extraction constant and the total ionic strength of the solution. However, the actual deviations of the real partition coefficients E for uranium and plutonium* were much larger than the $\pm 20-25\%$ claimed by the authors of [8]. Also, the mathematical model for the coextraction of uranium and plutonium and the models of [9, 10] have not been extended†

*Between the organic and aqueous phases.

†Probably on account of the restricted volume of data for this range.

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TABLE 1. Distributions of Pu(IV), U(VI), and HNO₃ on Extraction into 30% TBP from 2 M Aqueous HNO₃

[Pu], g/liter		E _{Pu}	[U], g/liter		E _U	[HNO ₃], M		E _{H+}	[Pu], g/liter		E _{Pu}	[U], g/liter		E _U	[HNO ₃], M		E _{H+}
A	B		A	B		A	B		A	B		A	B		A	B	
0,017	0,13	7,6	—	—	—	2,02	0,45	0,22	11,2	7,3	0,65	40	85	2,0	1,93	0,12	0,06
0,154	1,04	6,8	—	—	—	2,02	0,44	0,22	17,4	15,4	0,89	36	75	2,3	2,02	0,13	0,07
0,69	5,2	7,6	—	—	—	2,07	0,39	0,19	43,7	26,2	0,60	50	79	1,6	1,91	0,10	0,05
1,44	9,3	6,5	—	—	—	2,04	0,42	0,21	72,7	41,3	0,57	53	68	1,3	2,01	0,10	0,05
4,33	21,9	5,1	—	—	—	2,05	0,36	0,18	94	56	0,60	44	52	1,2	1,96	0,11*	0,05
5,9	30,5	5,2	—	—	—	2,09	0,32	0,15	—	—	—	—	—	—	—	—	—
7,1	32,2	4,5	—	—	—	2,04	0,30	0,15	—	—	—	—	—	—	—	—	—
8,5	35	4,1	—	—	—	2,00	0,29	0,14	—	—	—	82	106	1,3	2,02	0,10	0,05
9,2	38,7	4,2	—	—	—	2,02	0,28*	0,14	—	—	—	80	107	1,3	1,97	0,11	0,06
9,4	41,8	4,4	—	—	—	1,95	0,27*	0,14	—	—	—	84	106	1,3	2,00	0,11	0,05
10,8	43,4	4,0	—	—	—	1,99	0,26*	0,13	0,114	0,045	0,40	84	106	1,3	2,00	0,11	0,05
—	—	—	0,41	7,0	17	1,97	0,43	0,22	3,11	1,42	0,46	82	103	1,3	1,98	0,11	0,06
0,16	0,98	6,1	0,43	6,4	15	2,07	0,41	0,20	10,7	4,7	0,44	85	100	1,2	2,03	0,11	0,05
0,8	4,6	6,0	0,50	6,1	12	2,03	0,41	0,20	19,5	9,0	0,46	87	97	1,2	2,06	0,11	0,05
3,4	19,3	5,7	0,54	5,1	9,5	1,99	0,34	0,17	40,6	17,4	0,43	87	94	1,1	2,07	0,10	0,05
4,0	21,5	5,4	0,50	5,4	11	2,01	0,31	0,15	78,6	31,0	0,39	99	82	0,83	2,02	0,08	0,04
8,3	32,7	3,9	0,47	3,9	8,3	2,94	0,27	0,14	96,2	41,9	0,44	92	76	0,83	1,99	0,07	0,03
10,3	41,8	4,1	0,55	3,5	6,4	2,01	0,22	0,11	—	—	—	162	109	0,68	2,01	0,08	0,04
—	—	—	2,8	36	13	2,02	0,33	0,16	0,118	0,035	0,30	166	110	0,66	2,00	0,06	0,03
0,41	0,60	4,3	2,8	35	13	2,03	0,30	0,15	0,576	0,153	0,27	164	108	0,66	2,01	0,07	0,04
0,72	3,10	4,3	2,8	34	12	2,10	0,32	0,15	3,18	0,87	0,27	156	107	0,69	2,09	0,06	0,03
4,3	14,2	3,3	2,8	31	11	2,02	0,28	0,14	11,0	3,0	0,27	156	105	0,68	2,01	0,09	0,05
11,3	30,6	2,7	3,5	28	8	1,97	0,24	0,12	21,8	6,2	0,29	149	102	0,70	2,08	0,09	0,04
19,8	43,7	2,2	3,7	29	7,8	2,08	0,22*	0,10	43	13,5	0,31	144	100	0,69	2,09	0,09	0,04
—	—	—	11,4	72	6,3	2,02	0,20	0,10	74	20,7	0,28	165	96	0,58	2,02	0,09	0,05
0,160	0,252	1,6	12,3	66	5,4	2,03	0,21	0,10	90	28,7	0,32	161	90	0,58	2,02	0,09	0,05
0,68	1,20	1,8	10,9	63	5,8	2,07	0,21	0,10	—	—	—	316	117	0,37	2,02	0,07	0,03
3,36	5,39	1,6	12,8	60	4,7	2,00	0,19	0,09	0,103	0,020	0,19	314	117	0,37	1,93	0,09	0,05
11,5	15,6	1,4	14,1	58	4,2	2,00	0,19	0,09	1,17	0,102	0,20	301	118	0,39	2,04	0,10	0,05
18,2	26,8	1,5	16,9	58	3,6	2,02	0,18	0,09	3,30	0,241	0,21	332	117	0,35	2,02	0,08	0,04
40	52,2	1,3	19,8	52	2,6	1,85	0,11*	0,06	10,3	0,65	0,20	324	116	0,36	2,06	0,07	0,03
—	—	—	34,5	96	2,8	2,05	0,16	0,08	20,6	2,2	0,21	330	115	0,35	1,99	0,07	0,03
0,099	0,115	1,2	24,8	91	3,7	2,05	0,17	0,08	36,9	4,3	0,21	293	113	0,39	2,02	0,08	0,04
0,130	0,086	0,7	45,9	101	2,2	2,10	0,16	0,08	78	8,2	0,22	326	109	0,33	1,88	0,06	0,03
0,59	0,59	1,0	29,4	92	3,1	2,05	0,17	0,08	97,2	19,2	0,25	303	98	0,32	1,84	0,07	0,04
0,72	0,46	0,64	49,6	102	2,1	2,08	0,15	0,07	—	25	0,26	312	93	0,30	1,78	0,05	0,03
2,68	2,41	0,90	36	90	2,5	1,94	0,14	0,07	—	—	—	—	—	—	—	—	—

A — aqueous phase, B — organic phase.

to plutonium levels around 100 g/liter. Therefore, we need substantial qualitative improvement in the data on TBP containing macroscopic amounts of U(VI) and Pu(IV), as well as quantitative improvement, especially at high plutonium levels.

The equilibrium data given below for the partition of uranium and plutonium can be used in mathematical models for various forms of solvent-extraction reprocessing of uranium-plutonium fuel, as well as in calculations on nuclear safety, in distributing components, in designing automatic process controls, and so on.

Graphs have been presented [8] for the main trends in the coextraction of major amounts of Pu(IV) and U(VI), and formulas have been given for calculating E_{Pu} in the presence of uranium (15-300 g/liter in the equilibrium aqueous phase) and E_U in the range 25-300 g/liter in the presence of Pu(IV) (0-100 g/liter), i.e., in the range where the capacity of the extraction agent is essentially fully utilized.

We present here statistical data on the distribution of U(VI) (0-300 g/liter), Pu(IV) (0-100 g/liter), and HNO₃ (2-4 M) in 30% TBP in n-dodecane as well as mathematical formulas for the equilibrium extraction of Pu(IV) and U(VI) in this system.

TABLE 2. Distributions of Pu(IV), U(VI), and HNO₃ on Extraction into 30% TBP from 3 M Aqueous HNO₃

[Pu], g/liter			[U], g/liter			[HNO ₃], M			[Pu]g/liter			[U], g/liter			[HNO ₃], M		
A	B	E _{Pu}	A	B	E _U	A	B	E _{H+}	A	B	E _{Pu}	A	B	E _U	A	B	E _{H+}
0,011	0,140	13	—	—	—	2,97	0,59	0,20	—	—	—	40,6	105	2,6	3,01	0,14	0,05
0,124	1,35	11	—	—	—	2,99	0,59	0,20	0,101	0,108	1,1	42,1	99	2,4	3,00	0,18	0,06
0,50	6,2	12	—	—	—	2,98	0,55	0,18	0,48	0,53	1,1	36,5	96	2,6	2,95	0,18	0,06
0,90	12,0	13	—	—	—	2,95	0,55	0,19	3,13	3,37	1,1	42	99	2,4	3,00	0,16	0,05
2,29	28,8	13	—	—	—	3,01	0,48	0,16	10,6	9,5	0,9	35	94	2,7	3,02	0,12	0,04
2,99	27	9	—	—	—	3,02	0,47	0,16	21,0	18,5	0,9	48	89	1,9	3,02	0,14	0,05
4,2	35	8,3	—	—	—	3,02	0,43	0,14	41	33,1	0,8	47	73	1,6	2,99	0,11	0,04
4,9	44,9	9,2	—	—	—	3,01	0,32*	0,11	77,6	70,2	0,9	33,9	36	1,1	3,69	0,09	0,03
5,5	43	7,8	—	—	—	2,97	0,39*	0,13	—	—	—	87	113	1,3	3,07	0,09	0,03
6,5	48,4	7,4	—	—	—	3,00	0,38*	0,13	0,093	0,058	0,62	80	110	1,4	3,00	0,12	0,04
7,5	52,1	6,9	—	—	—	3,04	0,33*	0,11	0,56	0,32	0,57	78	108	1,4	2,97	0,15	0,05
9,3	54,7	5,9	—	—	—	2,97	0,38*	0,13	3,41	2,10	0,62	85	105	1,2	2,98	0,13	0,04
10	58	5,8	—	—	—	2,97	0,37*	0,13	11,3	6,6	0,58	82	103	1,3	3,00	0,11	0,04
14,3	71,8	5,0	—	—	—	3,01	0,27*	0,09	20,3	11,8	0,58	81	95	1,2	2,95	0,10	0,04
—	—	—	0,34	13,5	40	3,05	0,71	0,23	42,5	23,3	0,55	82	86	1,1	2,97	0,11	0,04
0,120	1,14	9,5	0,38	12	32	3,01	0,55	0,18	74,8	42,2	0,56	82	69	0,9	3,10	0,12	0,04
0,514	5,1	9,9	0,58	14	24	2,99	0,54	0,18	99,4	58,6	0,59	81	51	0,06	3,93	0,07	0,02
3,13	23,2	7,4	0,44	9,6	22	3,03	0,48	0,16	—	—	—	174	116	0,67	2,95	0,09	0,03
5,21	36,5	7,0	0,59	7,2	12	3,08	0,39	0,13	0,112	0,040	0,36	148	110	0,74	3,00	0,10	0,03
6,82	42,7	6,3	0,69	6,4	9,3	3,06	0,38	0,13	0,600	0,235	0,39	174	112	0,64	3,04	0,09	0,03
43	95	2,2	0,97	2,4	2,5	3,03	0,14*	0,05	3,39	1,12	0,33	176	112	0,64	3,09	0,11	0,04
—	—	—	3,31	49	15	2,95	0,40	0,13	10,9	4,26	0,39	161	107	0,67	3,00	0,11	0,04
0,113	0,60	5,3	3,62	53,9	15	2,98	0,38	0,13	19,5	7,9	0,41	147	104	0,71	2,95	0,08	0,03
0,50	2,58	5,2	4,7	55	12	2,97	0,34	0,12	39,3	13,7	0,35	159	108	0,68	3,06	0,09	0,03
3,09	14,2	4,6	4,6	50	11	2,97	0,30	0,10	69,9	20,1	0,29	190	98	0,52	3,02	0,08	0,03
9,5	32,8	3,5	5,4	43	8	3,04	0,24	0,08	80	29,9	0,37	169	90	0,53	3,09	0,11	0,03
—	—	—	—	—	—	—	—	—	100	40	0,40	164	73	0,45	3,39	0,07*	0,02
—	—	—	—	—	—	—	—	—	—	—	—	—	—	—	—	—	—
0,128	0,266	2,1	14,9	89	6	2,95	0,23	0,08	0,078	0,024	0,31	294	120	0,38	2,95	0,07	0,02
0,526	1,12	2,1	14,2	87	6,1	3,04	0,24	0,08	0,516	0,146	0,28	290	118	0,40	2,94	0,10	0,03
3,24	7,27	2,2	15,6	77	4,9	3,02	0,21	0,07	3,11	0,926	0,30	297	117	0,39	2,80	0,07	0,03
10,1	20,2	2,0	16,9	71	4,2	2,99	0,19	0,06	10,7	3,00	0,28	304	114	0,38	2,93	0,07	0,02
20,4	37,7	1,9	15,6	56,7	3,6	3,00	0,16	0,06	21,2	5,96	0,28	304	108	0,36	2,91	0,12	0,04
56,1	42,0	0,75	12	53	4,4	3,01	0,12*	0,04	42,7	12,1	0,28	307	105	0,34	2,86	0,09	0,03
—	—	—	—	—	—	—	—	—	76	23,0	0,30	296	94	0,32	2,67	0,07	0,03
—	—	—	—	—	—	—	—	—	96	40,5	0,32	284	89	0,31	2,94	0,06	0,02

Experimental

Reagents. The solution of plutonium nitrate (about 270 g/liter, acidity about 7 M) was made from the metal; 4.68% of the total α activity of the solution was due to ²⁴¹Am. The plutonium had a valency of four only. No accumulation of other valency forms was observed over a period of 1 year for this standard solution (checks were made by extraction and spectrophotometry). More concentrated solutions of Pu(IV) (up to 540 g/liter) were made by evaporating the standard solution (the acidity does not fall below 5.8 M). The uranyl nitrate, nitric acid, chemically pure n-dodecane, and TBP were purified by the methods of [11].

Methods. The extraction was performed with a phase-volume ratio of 1:1 in centrifuge tubes of volume 6-7 ml with mechanical stirring for 10 min, after which the phases were centrifuged, separated, and sampled (all operations were performed in an air thermostat at 26 ± 1°C).

Analysis. The plutonium was determined by α spectrometry, while uranium was determined by the vanadate method [12], with the nitric acid in both phases determined by potentiometric titration in oxalate buffer [13, 14], the error of measurement in each case being ±2-3% [15].

The distributions of U(VI) and Pu(IV) were determined on 2-4 parallel runs with two analyses for Pu and three of four analyses for U in each phase. The errors in determining Pu(IV) in the range 0.1-100 g/liter were ±2-3%, and the same applies for U(VI) at 0-100 g/liter, but this rose to ±4-5% at 150-300 g/liter. The reproducibility of E was always extremely satisfactory. The agreement on the component balance in each individual run was always better than ±4-5%.

Change in Phase Volumes on Extraction. The change in phase volume ΔV was determined for nitric acid levels of 2-4 M and U(VI) of 0-300 g/liter in the equilibrium water phase; the result was 2-5.5%. It was assumed

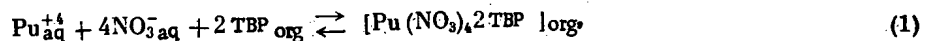
TABLE 3. Distributions of Pu(IV), U(VI), and HNO₃ on Extraction into 30% TBP from 4 M Aqueous HNO₃

[Pu], g/liter			[U], g/liter			[HNO ₃], M			[Pu], g/liter			[U], g/liter			[HNO ₃], M		
A	B	E _{Pu}	A	B	E _U	A	B	E _{H+}	A	B	E _{Pu}	A	B	E _U	A	B	E _{H+}
0,0145	0,342	24	—	—	—	3,89	0,77	0,20	—	—	—	84	114	1,4	3,96	0,15	0,04
0,129	2,27	18	—	—	—	3,98	0,75	0,19	—	—	—	79	114	1,4	4,05	0,11	0,03
0,450	8,6	19	—	—	—	3,94	0,72	0,18	0,120	0,096	0,80	77	110	1,4	4,07	0,17	0,04
0,98	16,7	17	—	—	—	3,93	0,65	0,17	0,50	0,39	0,78	86	110	1,3	4,04	0,16	0,04
1,77	25,0	14	—	—	—	4,01	0,67	0,17	2,91	2,35	0,81	86	110	1,3	4,04	0,17	0,04
2,60	33,0	13	—	—	—	4,00	0,59	0,15	10,14	8,12	0,80	74	101	1,4	4,08	0,15	0,04
3,26	37,0	11	—	—	—	3,99	0,59	0,14	21,8	15,5	0,71	86	96	1,2	4,09	0,14	0,03
3,65	42,1	12	—	—	—	4,03	0,42	0,10	41,8	28,8	0,69	82	79	0,96	4,14	0,12	0,03
4,30	42,0	10	—	—	—	4,05	0,42	0,11	82,1	50,4	0,61	78	60	0,69	4,28	0,09	0,02
4,82	50,0	10	—	—	—	4,03	0,39	0,10	99,4	58,6	0,59	81	51	0,63	3,93	0,07	0,02
5,9	55,6	9,4	—	—	—	4,07	0,44	0,11	—	—	—	—	—	—	—	—	—
7,37	61,7	8,4	—	—	—	4,06	0,39	0,10	—	—	—	—	—	—	—	—	—
—	—	—	0,74	20,8	28	3,99	0,64	0,16	—	—	—	160	117	0,73	4,03	0,12	0,03
0,17	2,27	13	0,85	20,6	24	4,00	0,63	0,16	0,115	0,052	0,45	168	116	0,69	3,99	0,12	0,03
—	—	—	3,5	51,4	15	4,04	0,46	0,12	0,52	0,239	0,46	168	115	0,69	3,97	0,14	0,03
0,124	0,87	7,0	4,5	58,3	13	3,95	0,43	0,11	3,12	1,44	0,46	150	114	0,76	4,06	0,14	0,03
0,468	3,33	7,1	3,5	48,2	14	3,99	0,46	0,12	3,21	1,35	0,42	174	115	0,66	3,96	0,09	0,02
2,42	20,6	8,5	2,4	33,2	14	3,94	0,46	0,12	10,0	4,8	0,48	160	106	0,66	4,01	0,13	0,03
8,01	47,9	6,0	2,4	22,1	9,4	4,05	0,32	0,08	10,3	4,1	0,40	171	107	0,63	3,98	0,12	0,03
—	—	—	15,9	93	5,9	4,03	0,24	0,06	18,3	8,5	0,46	156	103	0,66	4,00	0,11	0,03
0,144	0,30	2,1	15,9	92	5,8	4,02	0,24	0,06	33,0	11,2	0,34	176	98	0,56	3,80	0,12	0,03
0,69	1,57	2,3	16,9	87	5,2	4,05	0,24	0,06	40,2	19,8	0,49	159	94	0,59	3,95	0,09	0,02
3,7	10,1	2,7	19,5	80	4,1	4,05	0,22	0,05	69	30	0,44	195	84	0,43	3,95	0,11	0,03
8,4	21,5	2,6	13,6	64	4,7	4,03	0,26	0,06	75,4	37,9	0,50	160	77	0,48	4,11	0,08	0,02
17,0	25,5	1,5	19,9	62	3,1	3,96	0,20	0,05	100,1	47,1	0,47	156	67	0,43	4,25	0,10	0,02
17,9	47	2,6	9,5	44	4,6	4,15	0,22	0,05	—	—	—	296	119	0,40	3,91	0,12	0,03
—	—	—	43	108	2,5	3,98	0,20	0,05	0,097	0,036	0,37	294	118	0,40	3,94	0,13	0,03
0,113	0,169	1,5	42	107	2,6	4,03	0,19	0,05	0,493	0,175	0,36	294	117	0,40	3,82	0,12	0,03
0,55	0,70	1,3	40	106	2,7	4,02	0,20	0,05	3,15	1,07	0,34	299	113	0,38	3,89	0,13	0,03
3,22	4,15	1,3	40	97	2,4	3,99	0,20	0,05	10,0	3,6	0,36	297	108	0,36	4,00	0,14	0,04
12,6	13,5	1,1	46	87	1,9	3,99	0,17	0,04	19,2	7,0	0,37	297	106	0,36	3,89	0,13	0,03
21,3	24,8	1,2	36	77	2,2	4,14	0,17	0,04	39,4	14,4	0,37	306	103	0,34	3,88	0,10	0,03
39,2	44,5	1,1	35	59	1,7	4,20	0,17	0,04	73	29	0,40	302	86	0,29	3,96	0,10	0,03

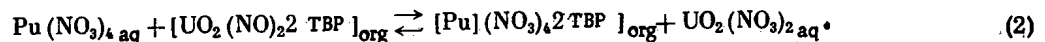
*Organic phase separated into two layers.

that ΔV for the organic phase was equal to that for the aqueous phase, since the total volume of the system was virtually unaltered. The data on ΔV in relation to nitric acid concentration in the aqueous phase and U(VI) in the organic phase were used in calculating the balances for each run and for correcting the radiometric data for ²⁴¹Am.

Discussion. Tables 1-3 give the equilibrium data on the distributions for 2-4 M aqueous solutions of nitric acid and 30% TBP in n-dodecane. A constant level of U(VI), which is always considered as applying to the equilibrium aqueous phase, leaves E_{Pu} unaltered within the error of measurement if the uranium level is over 15 g/liter, no matter what the plutonium level in the aqueous phase (the linear course of the extraction isotherm for plutonium with TBP saturated with uranium resembles the linear initial course of the isotherm for the extraction of U(VI) or Pu(IV) into TBP). This occurs because the concentration of the free extraction agent tends to zero when the levels of the metals in the equilibrium aqueous phase are high. It is likely that the plutonium is then not extracted by free extraction agent TBP₀ but in accordance with the following mechanism:



viz., by bound TBP, in particular reagent bound in accordance with



If the uranium level is high in the U-Pu mixture, the equilibrium in (2) tends to shift to the left, particularly if U(VI) is extracted preferentially into the TBP by comparison with the Pu(IV).

The activity coefficient for Pu^{4+} * should be such that the partition coefficient for plutonium is essentially constant although the Pu(VI) level in the system varies, as there are large and constant amounts of U(VI) , i.e., only low levels of plutonium in the extract, provided that the acidity is in the range 2-4 M, since the nitrate-ion concentration is essentially constant, as is the concentration of the extraction agent (bound TBP); naturally; a substantial rise in the level of Pu(IV) displaces uranium and causes the extraction agent to be saturated with Pu(IV) , so E_{Pu} should fall, so the extraction isotherm should be correspondingly curved.

Mathematical processing fails to yield a reasonably correct single formula for the uranium and plutonium extraction isotherms for the entire range covered by the studies.

For example, the isotherms for the extraction of plutonium take the form $y = kx^n$ in the absence of U(VI) and are described to within $\pm 2-3\%$ by

$$[\text{Pu}]_0 = (4.48 [\text{H}^+]_a - 2) [\text{Pu}]_a^{0.846 - 0.034[\text{H}^+]_a} \quad (3)$$

where $[\text{Pu}]_0$ and $[\text{Pu}]_a$ are the concentrations of plutonium in the organic and aqueous phases in g/liter, while $[\text{H}^+]_a$ is the nitric acid concentration in M. This describes the extraction of Pu(IV) from 2-4 M solutions of nitric acid at levels of 0.5-10 g/liter in the aqueous phase (up to about 50 g/liter in the organic phase), i.e., in the region where the two-phase system is stable ($26 \pm 1^\circ\text{C}$).

A moderate amount of U(VI) (0.5-40 g/liter) gives similar functions for the extraction of plutonium (error $\pm 5\%$):

$$\lg [\text{Pu}]_0 = \left(\frac{[\text{U}]_a}{0.15 + 1.08 [\text{U}]_a} \right) \lg [\text{Pu}] + 0.74 - 0.48 \lg [\text{U}]_a; \quad (4)$$

$$\lg [\text{Pu}]_0 = \left(\frac{[\text{U}]_a}{0.11 + 1.10 [\text{U}]_a} \right) \lg [\text{Pu}]_a + 0.90 - 0.34 \lg [\text{U}]_a; \quad (5)$$

$$\lg [\text{Pu}]_0 = \left(\frac{[\text{U}]_a}{0.59 + 0.98 [\text{U}]_a} \right) \lg [\text{Pu}]_a + 1.2 - 0.69 \lg [\text{U}]_a \quad (6)$$

for 2-4 M solutions of nitric acid, respectively. The isotherms for Pu(IV) in the presence of 15-300 g/liter of uranium are given satisfactorily by the formulas of [8].

In turn, the isotherms for the extraction of U(VI) without Pu(IV) resemble $y = x/(a + bx)$ in bilogarithmic terms for the range from 1 to 300 g/liter in the aqueous phase, the appropriate formulas with $\pm 3-5\%$ error being

$$\lg [\text{U}]_0 = \frac{\lg [\text{U}]_a}{0.445 \lg [\text{U}]_a + \frac{[\text{H}^+]_a}{20 [\text{H}^+]_a - 19.8}}, \quad (7)$$

or in the presence of 0-100 g/liter of Pu(IV) :

$$\lg [\text{U}]_0 = \frac{\lg [\text{U}]_a}{(0.445 - 0.001 [\text{H}^+]_a [\text{Pu}]_a) \lg [\text{U}]_a + 0.026 [\text{H}^+]_a [\text{Pu}]_a + \frac{[\text{H}^+]_a}{20 [\text{H}^+]_a - 19.8}}. \quad (8)$$

The extraction of nitric acid is described satisfactorily by the equations of [6, 7], apart from the region of high concentrations; this aspect will be considered in more detail in a forthcoming paper.

CONCLUSIONS

1. Equilibrium data have been obtained on the partition of Pu(IV) (0-100 g/liter), U(VI) (0-300 g/liter), and nitric acid (2-4 M in the equilibrium aqueous phases) for coextraction into 30% TBP in n-dodecane at $26 \pm 1^\circ\text{C}$.

2. Formulas have been derived for the extraction isotherms for Pu(IV) in 2-4 M nitric acid, which apply with an error of $\pm 2-3\%$, while new and improved concentration ranges have been defined for standard formulas that describe the equilibrium at levels of 1-300 g/liter of U and 0.5-100 g/liter of Pu, where the errors are $\pm 5-10\%$.

3. An extraction mechanism has been described that explains the constancy of the partition coefficient for Pu(IV) (0-100 g/liter) under conditions of high reagent saturation and constant uranium concentration (15-300 g/liter) in the aqueous phase.

*The activity coefficient of Pu^{4+} remains constant at 1.1-1.2 [17] at ionic strengths > 1 .

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ENERGY BALANCE IN A TOKAMAK REACTOR WITH
TURBULENT TRANSPORT COEFFICIENTS

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The optimistic situation which has built up around the problem of controlled thermonuclear fusion makes the preliminary analysis of the operation of the tokamak reactors of the future timely [1-3]. These investigations have exposed a number of problems in need of solution. In particular, it has become clear that the mechanisms of energy and particle losses observed in present equipment (neoclassical or pseudoclassical transport coefficients) will lead to exceedingly high values of the plasma temperature in larger equipment. In addition, waste removal has turned out to be a complicated problem. Since the limiting pressure of a plasma is restricted by the conditions for the existence of equilibrium, higher temperature values indicate a low plasma density, and consequently, a low specific generatable power, which is not advantageous for economic reasons. From this point of view the 10-20 keV temperature range is optimal. Such a plasma temperature cannot be provided in larger tokamaks with neoclassical transport coefficients. These coefficients should be appreciably higher.

The problem now occurring in the initial stage of the theoretical investigations, viz., the problem of increasing the ratio of the kinetic gas pressure of the plasma to the magnetic pressure by means of frozen-in magnetic fluxes [4], is not discussed in this article. In this case an increase in the plasma density, and consequently in the neoclassical transport coefficients, is possible.

Investigations based on the idea of reducing the temperature by means of the introduction of an impurity [2] have shown its ineffectiveness. The additional pressure caused by the impurity which appears turns out to be appreciable, which again requires a reduction in the fuel density.

Contemporary theoretical concepts [5] predict that the development of instability among the conducting particles is possible in the region of higher thermonuclear temperature than in the existing equipment. The

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turbulent transport coefficients corresponding to this instability will rapidly grow as $T^{1/2}$ with an increase in the temperature, whereas the nuclear generation is a more mildly sloping function in the temperature range under discussion. It is precisely this fact which permits us to hope that the temperature is stabilized at a sufficiently advantageous low level.

The present theory is of an evaluative nature, however, and the absence of experimental data does not permit refining the numerical coefficients. In view of this fact there evidently still does not seem to be much reason to investigate the energy balance on the basis of two-dimensional and even one-dimensional [6] equations in equipment with turbulent coefficients. One must assume that a zero-dimensional computational scheme based on averaging the equations over volume is more appropriate to the contemporary level of our knowledge about turbulent transport coefficients.

Such an averaged balance can be obtained by integration over the plasma volume of a system of local equations of the energy balance of electrons and ions, as well as the fuel and waste (α particles) balance:

$$\frac{3}{2} \frac{\partial}{\partial t} (nT_e) = -\operatorname{div} q_e + Q_e; \quad (1)$$

$$\frac{3}{2} \frac{\partial}{\partial t} (nT_i) = -\operatorname{div} q_i + Q_i; \quad (2)$$

$$\frac{\partial n}{\partial t} = -\operatorname{div} q_n + Q_n; \quad (3)$$

$$\frac{\partial n_\alpha}{\partial t} = -\operatorname{div} q_\alpha + Q_\alpha. \quad (4)$$

Here Q denotes the total sources of energy and particles; it is assumed that $n_e = n_i \equiv n$ ($n_\alpha \ll n$). It is assumed that the turbulent fluxes make a decisive contribution to the fluxes q , i.e.,

$$q_e = \frac{3}{2} D_T \nabla (nT_e); \quad (5)$$

$$q_i = \frac{3}{2} D_T \nabla (nT_i); \quad (6)$$

$$q_n = D_T \nabla n; \quad (7)$$

$$q_\alpha = D_T \nabla n_\alpha. \quad (8)$$

The possible discrepancy of the numerical factors in the transport coefficients for the various components is neglected in these expressions.

We will assume that all the quantities in Eqs. (1)-(4) depend only on r — the distance to the axis of the torus, whose cross section is assumed to be circular. Multiplying Eqs. (1)-(4) by $2r$ and integrating over r from 0 to a (minor radius), we have a system of balance equations averaged over the plasma volume:

$$\frac{3}{2} \frac{d}{dt} \langle nT_e \rangle = -\frac{q_e(a)}{a} + \langle Q_e \rangle; \quad (9)$$

$$\frac{3}{2} \frac{d}{dt} \langle nT_i \rangle = -\frac{q_i(a)}{a} + \langle Q_i \rangle; \quad (10)$$

$$\frac{d}{dt} \langle n \rangle = -\frac{q_n(a)}{a} + \langle Q_n \rangle; \quad (11)$$

$$\frac{d}{dt} \langle n_\alpha \rangle = -\frac{q_\alpha(a)}{a} + \langle Q_\alpha \rangle. \quad (12)$$

Here we introduce the following notation for average values:

$$\langle f \rangle = \frac{2}{a^2} \int_0^a r f(r) dr.$$

TABLE 1. Parameters of a Tokamak Reactor under Steady-State Conditions

Parameter	For R = 12 m, a = 4.5 m, and W = 5.0 GW	For $\beta_T = 2.5$		
		$q_0 = 300 \text{ W/cm}^2$	$q_0 = 400 \text{ W/cm}^2$	$q_0 = 500 \text{ W/cm}^2$
Radius of the torus, m:				
major	—	9,6	13	16
minor	—	3,2	4,2	5,2
Longitudinal magnetic field, kG	50	50	50	50
Stability reserve	2,5	2,5	2,5	2,5
Total power, GW	—	3,6	8,4	16
Specific power, W/cm ³	1,0	1,9	1,9	1,9
Plasma density, 10 ¹³ cm ⁻³				
at the wall	6,0	8,5	7,3	6,1
at the center	8,9	12	12	12
Helium density at the center, 10 ¹³ cm ⁻³	0,10	0,14	0,12	0,10
Temperature at center, keV:				
of electrons	19	20	20	20
of ions	17	17	17	17
Poloidal β_i	1,5	—	—	—
Energy flux along channel, W/cm ² :				
electron	19	25	33	42
ion	17	21	29	36
Bremsstrahlung flux, W/cm ²	2,9	3,7	4,8	5,9
Total thermal power of reactor per 1 cm ² of area of first wall, W/cm ²	230	—	—	—
Integral plasma losses, g/sec	0,17	0,10	0,40	1,2
Integral helium flux onto wall, 10 ⁻³ g/sec	9,9	7,2	17	32

In order to determine the average values and the fluxes $q(a)$ (the flux values at $r = a$) it is necessary to know the distributions of the densities and temperatures, which are selected from the condition of symmetry at $r = 0$ and the requirement that the energy and particle fluxes be finite at $r = a$. Of course, these conditions do not determine the distributions uniquely. This indeterminacy essentially denotes the indeterminacy in the choice of the numerical factor in the equation for the fluxes, which however, are also known only to within an accuracy of some factor.

The requirement of finiteness of the fluxes is not satisfied in this case in a trivial fashion, since the turbulent transport coefficients have a singularity when n and $T \rightarrow 0$:

$$D_T = \left| \frac{d \ln n}{dr} \right|^2 \left(\frac{r}{R} \right)^{5/2} \left(\frac{c T_e}{e H_0} \right)^2 (4 \nu_e)^{-1}; \quad \nu_e = \frac{4 \sqrt{2\pi} n e^4 \Lambda}{3 n_i^{1/2} T_e^{3/2}} \quad (13)$$

where H_0 is the longitudinal magnetic field, R is the major radius of the torus, and Λ is the Coulomb logarithm.

Denoting $1 - (r/a) = x$ and assuming $\Lambda = \text{const}$, we have near $x = 0$

$$D_T \approx \frac{1}{n^3} \left(\frac{dn}{dx} \right)^2 T_e^{7/2}$$

At the same time the energy and particle fluxes at the boundary ($r \rightarrow a$) are of the form

$$q_n \approx \left(D_T \frac{dn}{dx} \right)_{x \rightarrow 0}; \quad q_e \approx \left[D_T \frac{d(T_e n)}{dx} \right]_{x \rightarrow 0};$$

$$q_i \approx \left[D_T \frac{d(T_i n)}{dx} \right]_{x \rightarrow 0}; \quad q_\alpha \approx \left(D_T \frac{dn_\alpha}{dx} \right)_{x \rightarrow 0}$$

From the finiteness of the first two fluxes we find, assuming that $T_e \rightarrow 0$, $d \ln n_e / dT_e = \text{const}$, i.e., $n_e \sim \exp(AT_e)$. Assuming that $T_e \sim x^\beta$ near x , we obtain $n_e \approx n_1 (1 + Ax^\beta)$. Then we obtain q_e for the flux $q_e \sim x^{(13\beta/2)-3}$. Thus it is necessary that $\beta = 6/13$ for finiteness of q_e . In addition we have $dn_e/dn_\alpha = \text{const}$ and $dT_e/dT_i = \text{const}$. Assuming that n_α and $T_i \rightarrow 0$ as $x \rightarrow 0$, we obtain $n_\alpha \sim x^\beta$ and $T_i \sim x^\beta$.

The condition at $r = 0$ requires that all the distributions be parabolic. The simplest choice of distributions is the following, which satisfies both requirements:

$$n = n_1 + (n_0 - n_1) \nu; \quad n_\alpha = n_{\alpha 0} \nu; \quad (14)$$

$$T_e = T_{e0} \nu; \quad T_i = T_{i0} \nu; \quad \nu = [1 - (r^2/a^2)]^{6/13}$$

The subscript 0 refers to values at $r = 0$, and the subscript 1 refers to values at $r = a$.

We will assume that the fuel losses due to diffusion to the chamber walls (with subsequent pumping-off from it) are compensated by the stoking of fuel to the inside. At the same time $n = \text{const}(t)$, and Eq. (11) determines the amount of fuel $2\pi^2 R a^2 \langle Q_n \rangle$ which is necessary for stoking.

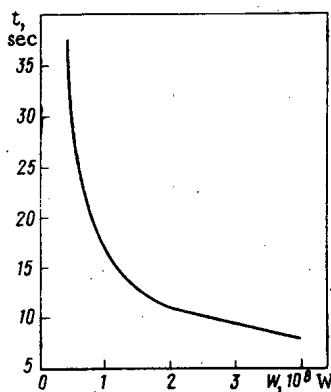


Fig. 1. Dependence of the time to establish a steady state on the injector power ($R = 12$ m, $a = 4.5$ m, and $W = 5$ GW).

One can calculate for the selected distributions (14) the quantities appearing in the equations:

$$\begin{aligned}
 \langle nT_e \rangle &= 0.52n_1T_{e0}(b + 25/19); \\
 \langle nT_i \rangle &= 0.52n_1T_{i0}(b + 25/19); \\
 \langle n \rangle &= \frac{13}{19}n_1(b + 19/13); \quad \langle n_\alpha \rangle = \frac{13}{19}n_{\alpha 0}; \\
 q_e &= 3T_{e0}I_0; \quad q_i = 3T_{i0}I_0; \\
 q_n &= bI_0; \quad q_\alpha = \frac{n_{\alpha 0}}{n_1}I_0.
 \end{aligned} \tag{15}$$

Here $b = n_0/n_1 - 1$; $I_0 = (12/13)^3 [3(a/R)^{5/2} b^2 / 16\sqrt{2}\pi\Lambda_0] (c/a^3) (m^2 c^4 / e^3 H_0)^2 (T_{e0} / mc^2)^{7/2}$, and Λ_0 is the Coulomb logarithm computed at $T = 1$ keV.

The sources which appear in the energy equations contain the power produced in D-T reactions of α particles, which is assumed to be completely absorbed by the plasma, this evidently being justified for large reactors [7]. We will use an interpolation formula (an equal-component mixture of deuterium and tritium) as the function determining the specific power:

$$Q_{\text{nucl.}} \left(\frac{\text{eV}}{\text{cm}^3 \cdot \text{sec}} \right) = 2.25 \cdot 10^{-4} n^2 \frac{1 + 7(T_i/93800)^{3/4}}{T_i^{2/3} [1 + 242(T_i/93800)^{13/4}]^{1/2}} \exp\left(-\frac{200}{T_i^{1/3}}\right).$$

Here T_i is in eV and the coefficient η is equal to [8]:

$$\begin{aligned}
 \eta &= \lambda \left\{ \frac{\pi}{3\sqrt{3}} + \frac{1}{3} \ln \frac{1 - \sqrt{\lambda} + \lambda}{(1 + \sqrt{\lambda})^2} + \frac{2}{\sqrt{3}} \text{arctg} \frac{2 - \sqrt{\lambda}}{\sqrt{3\lambda}} \right\}; \\
 \lambda &= 16.7T_e \left(\frac{3\pi}{4} \right)^{2/3} / \varepsilon_0; \quad \varepsilon_0 = 3.52 \cdot 10^6.
 \end{aligned}$$

A term is included in Eqs. (1) and (2) which takes into account the energy exchange among electrons, the power contributed to the plasma by the injector and distributed in an appropriate way between the ions and the electrons, and the losses to bremsstrahlung, in which the presence of impurities is neglected [in Eq. (1)]. The neglect of Joule heating and cyclotron radiation is justified for the values adopted for T_e and H_0 , $Q_\alpha = Q_{\text{nucl}} / \varepsilon_0$ in Eq. (3).

By virtue of Eqs. (5)-(8) $n_\alpha/n_1 \approx T/\varepsilon_0$ for the transport coefficients in the steady state, and we have $n_\alpha/n_1 \approx 10^{-2} \ll 1$ at $T \approx 10$ keV, which justifies the neglect of the presence of helium in the plasma.

An important characteristic of a plasma in a reactor is the ratio of the kinetic gas pressure of the plasma to the pressure of the magnetic field of the current [$\beta_I = \langle n(T_e + T_i) \rangle / (H_I^2 / 8\pi)$]. This ratio can be conveniently expressed in the following form:

$$\beta_I = 0.52n_1 \left(b + \frac{25}{19} \right) (T_{e0} + T_{i0}) (R/a)^2 q^2 (8\pi/H_0^2),$$

where $q = aH_0/RH_I$ is the stability reserve.

One can also calculate for the specified distributions the resistance and inductance of the plasma loop:

$$\rho = \frac{44}{13} \frac{R}{\sigma_0 a^2}; \quad \sigma_0 = \frac{1,96}{\gamma} \frac{3mc^3}{4 \sqrt{2\pi} e^2 \Lambda_0} \left(\frac{T_{e0}}{mc^2} \right)^{3/2};$$

$$L = 4\pi R [\ln(8R/a) - 2 + \delta];$$

$$\delta = \frac{1}{2} \int_0^1 \frac{[1 - (1-u)^{22/13}]^2}{u} du \approx 0.4,$$

where γ is the coefficient of anomalous resistance of the plasma. Eqs. (9)-(12) were integrated numerically by the Runge-Kutta method with the initial conditions: $T_{e0} = T_{i0} = 1$ keV and $n_{\alpha 0} = 0$. The injector of fast neutral particles was "switched on" from the instant $t = 0$ to the instant at which $T_{i0} = 10$ keV. The energy of the injected particles was 200 keV, and the injector power was varied. The values of the longitudinal magnetic field ($H = 50$ kG) and the stability reserve ($q(a) = 2.5$) were left constant in all the calculations.

As is evident from Eq. (13), the transport coefficients depend on the density gradient, which is reflected in the dependence of the fluxes q in Eq. (15) on n_0 and n_1 . Physically, the transport coefficients are specified by the values of the average plasma density $\langle n \rangle$ (or the total number of particles in the reactor, which is conserved by virtue of compensation of losses) and the amount of stoked (pumped-off) fuel. One should regulate the energy losses as well as the temperature and power to the extent that the possibility exists of varying the amount of stoked (pumped-off) fuel.

The possibility arises of a certain optimization of the reactor in the presence of free parameters (the average density and the flux of plasma or the values of n_0 and n_1 associated with them). Two groups of problems have been solved in this paper.

1. The parameters of the reactors have been specified, along with its total thermal power W (the total thermal power has been calculated with account taken of the energy of absorbed neutrons as well as of reactions in the blanket - they amount to 21 MeV per each fusion event in these calculations). Steady-state modes with smallest β_I have been found. This case has been calculated for a thermonuclear electric power station with a tokamak reactor [9].

2. The values of β_I and q_0 - the thermal flux through the wall of the reactor - which, as is well known, are limited in magnitude, have been specified. The dimensions and power of a reactor possessing the largest value of the specific generatable power were found.

Both optimization problems are based on the fact that the specific power of thermonuclear energy generation (with the condition $T_e = T_i$) is proportional to $\beta_I^2 F(T_{i0})$, where the function $F(T_{i0})$ has a maximum at some $T_{i0} = T^*$ (the value of T^* depends, in general, on the temperature distribution; for the distribution selected here $T^* = 17$ keV). Thus, with the specified dimensions and power (i.e., specific power) the value of T^* determines the minimum of β_I , and with the specified β_I the same value of T^* determines the maximum of the specific power. Specification of the maximum value of q_0 then determines the maximum dimensions and power of the reactor.

Consequently, the computational problem consists of determining those values of n_0 and n_1 for which the steady-state temperature in the reactor is equal to T^* . The reactor parameters obtained in the calculation for both groups of problems are given in Table 1 for four different cases.

The dependence of the time to achieve a steady state on the power of the neutral-particle injector is shown in Fig. 1 as obtained in the calculations for a reactor with the specified dimensions and power. The settling-down time was arbitrarily assumed to be the time from the onset of injector operation to the instant at which the reactor parameters began to differ from their own asymptotic values by less than 5%. As is evident, the injector power necessary for establishment of a steady state is minimal.

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TRITIUM IN ATMOSPHERIC PRECIPITATIONS, RIVERS,
AND THE SEAS IN AND AROUND THE
USSR TERRITORY

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The tritium concentration of natural media has been studied for many years, starting from the first thermonuclear weapons tests. The attention to this problem has not decreased in recent years because increasing amounts of tritium enter into the environment from nuclear energy sources [1-3], whereas the requirements to the cleanness of the environment have become more rigorous.

The International Atomic Energy Agency (IAEA) and the World Meteorological Organization (WMO) make continuous observations of the tritium concentration of atmospheric precipitates in ≈ 100 meteorological stations of 65 countries [4]. Similar observations have been systematically made in the USSR since the second half of 1966, first by 12, and since 1973 by 18 meteorological stations situated in the main climatic zones of the country. The results of these observations [5-8] help to establish some area- and time-dependent laws of the tritium distribution in precipitations.

In order to determine the laws governing the tritium migration in the natural waters on the USSR territory, the information on the tritium concentration of precipitations must be supplemented by information on the concentration of rivers and seas. For this purpose, in 1975-1976 not only samples of atmospheric precipitations were taken but also samples of river water [21] and water of the seas around the USSR territory were collected.

The concentration of tritium* in the water samples was measured with a "Mark-II" liquid-scintillator spectrometer after initial enrichment by electrolysis [9]. The sensitivity of the apparatus was $5.0 \cdot 10^{-9}$ Ci/liter when a ZhS-8 scintillator was employed; the relative error of the measurements was 30 and 5%, respectively, for concentrations of $5 \cdot 10^{-9}$ and $1.0 \cdot 10^{-7}$ Ci/liter at the 90% confidence level.

Figure 1 depicts the dynamics of the mean annual tritium concentration of atmospheric precipitations. The mean-square deviations from the average are indicated at each point. The drop of the tritium concentration between 1963 and 1967 (for the Northern hemisphere) can be adequately described by an exponential with a removal half-period $\tau = 1.2$ years [6]. When the change in the tritium concentration of the atmospheric precipitations on the USSR territory is approximated by a single exponential for 1970-1973, the values $\tau = 2.8 \pm 0.44$ years is obtained, which is about 2.3 times greater than the latter value. This increase in τ , as well as the increase in the tritium concentration in 1968-1970 and in 1973-1975, originate from injecting tritium into the atmosphere in the thermonuclear explosions set off in these time periods in the Southern and Northern hemispheres [10, 11].

*Present in bound form as tritium-containing water HTO.

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TABLE 1. Tritium Concentration in Rivers (fall-spring 1975-1976) and Precipitations

Region	Tritium concn. (TE) of the rivers	Point of precipitate sampling	Tritium concn. of precipitation sampling	
			annual av.	av. for Aug.-Sept.
Baltic region (Daugava, Neiman)	65 ± 15 (4)*	Riga	60	
Northern European Territory of the Soviet Union (Northern Dvina, Pechora)	98 ± 11 (10)	Arkhangelsk	67	74
Central European Territory of the Soviet Union (Volga, Kama, Vishera, Don, Dnepr, Ural)	94 ± 7 (25)	Moscow, Perm, Rostov-on-Don, Odessa	105	82
Northern Siberia (Ob, Enisei, Lena)	166 ± 18 (9)	Eniseisk, Yakutsk, Dudinka, Salekhard	159	163
Southern Siberia (Irtys, Uiba, Selenga, Ob, Angara)	150 ± 31 (4)	Novosibirsk, Omsk, Irkutsk, Skovorodino	152	141
Far East (Amur)	160 ± 21 (4)	Skovorodino, Khabarovsk	145	142
Central Asia (Amu-Darya, Syr-Darya)	127 ± 5 (3)	Tashkent	98	-

*The number of sampling points is indicated in parentheses.

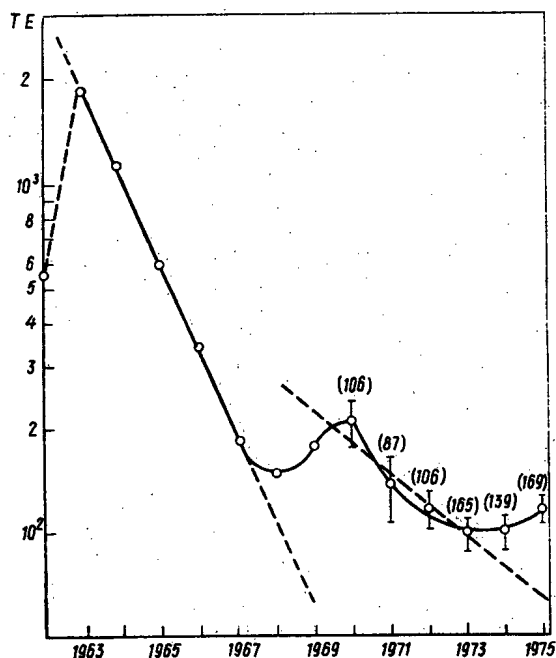


Fig. 1. Dynamics of the mean annual tritium concentration of atmospheric precipitations of the Northern hemisphere: 1963-1966, data of the WMO [4]; 1970-1975, USSR territory; the averaged mean monthly values are indicated in parentheses.

Figure 2 shows the seasonal dependence of the tritium concentration of precipitations averaged over the USSR territory (the mean monthly concentrations are normalized to the mean annual concentrations). A comparison of the concentration figures with the data on nuclear explosions [10, 11] shows that despite the tests, the overall seasonal dependence has a single form with a maximum in May-June and that only the 1973 explosion expanded the spring-summer maximum.

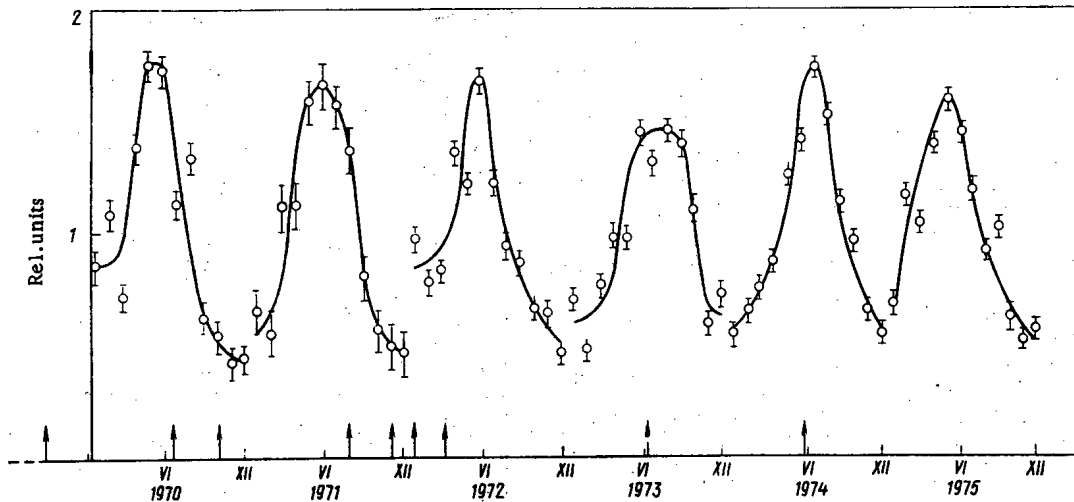


Fig. 2. Seasonal dependence of the tritium concentration of precipitations in 1970-1975 for the USSR territory; the arrows denote the dates of explosions [10, 11].

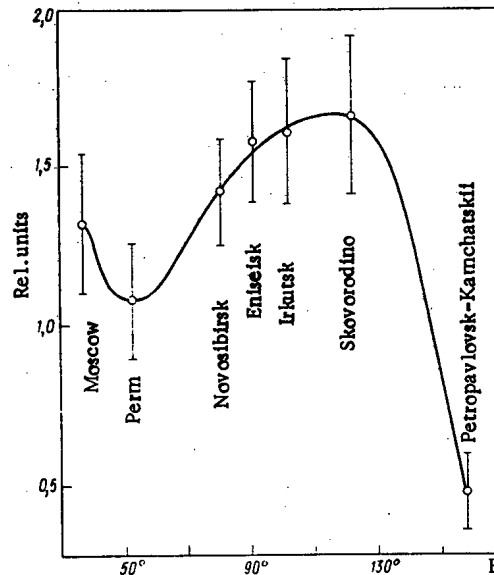


Fig. 3. Meridional distribution of the 1975 tritium concentration in the latitudinal zone 50-60°N.

The meridional distribution of the mean annual tritium concentration of 1975 precipitations is shown in Fig. 3 for the latitudinal belt 50-60°N (this latitude is rather completely represented by the sampling points of precipitations over the USSR territory); the continental effect is clearly visible on that figure [5, 12]. The maximum of the tritium concentration of precipitations is observed in the middle of the continental region (Irkutsk and Skovorodino), whereas the minimum occurs at Petropavlovsk-Kamchatskii, where the ocean strongly influences the concentration. A certain concentration decrease in the zone 50-70°E can be explained by the prevailing Northeast (Kara Sea) and Southwest (Caspian Sea and Black Sea) direction of motion of the moisture-bearing masses in 1975 within that zone; this caused a certain decrease in the tritium concentration of precipitations.

An analysis of the data referring to the entire USSR territory has shown a slight difference in the tritium concentration in 1975 over the latitudinal zones. Thus, at latitudes 70-60°N, the tritium concentration amounted to (125 ± 15) TE, at 60-50°N to (136 ± 16) TE, and at 50-40°N to (80 ± 10) TE, which is a basis for averaging the tritium fallout with the precipitation on the basins of rivers flowing in meridional direction, with the number of sampling points on the basin area being small.

TABLE 2. Annual Run-Off of Tritium from the USSR Territory into the Seas; 1975
Figures

Basin	Av. concn. (10^{-10} Ci/ liter) of rivers of a particular basin	Annual run- off of tritium (% of total annual run- off) into basin	Run-off coefficient		Tritium coefficient for sea water	
			tritium	water	year	10^{-10} Ci/liter
Baltic Sea	2,2	2,0	0,32	0,34	1974 1975	$2,6 \pm 0,5$ (11) $2,5 \pm 0,4$ (8)
White Sea	2,4	2,5	0,62	0,48	1974	$1,7 \pm 0,5$ (12)
Barents Sea and Pechorskoe Sea	3,5	6,0	0,54	0,36	1974	$1,2 \pm 0,4$ (14)
Kara Sea	5,2	40,0	0,55	0,49	—	—
Laptev Sea	4,1	15	0,71	0,42	—	—
Pacific (Northern part)	4,9	28,0	0,12	0,12	1974 1973	$0,26 \pm 0,09$ (48) $1,7 \pm 0,4$ (43)
Black sea	1,2	0,5	0,22	0,21	1975	$1,2 \pm 0,3$ (4)
Azov Sea	2,9	0,5	0,22	0,21	1973	$3,5 \pm 0,7$ (16)
Caspian Sea	3,2	5,5	0,43	0,37	1975	$1,4 \pm 0,5$ (43)
Entire USSR territory	—	100%	0,43	0,37	—	—

The mean annual tritium concentration of atmospheric 1975 precipitations, averaged over the USSR territory, was 115 ± 13 TE, which corresponds to $(3.7 \pm 0.4) \cdot 10^{-10}$ Ci/liter (1 TE = $3.25 \cdot 10^{-12}$ Ci/liter). The tritium fallout in precipitations amounted to 4 MCi per year for the entire USSR territory or to 180 ± 17 mCi/km².

The experimental values of the tritium concentration of rivers and precipitations, averaged over the geographic regions of the USSR, are listed in Table 1. River water samples were taken in the fall of 1975 and in the spring of 1976. The tritium concentration measured was averaged over the two periods. The average concentration figure is listed in the table, along with the mean-square deviations. Data on the concentrations in the precipitations of August-September 1975 are included in the table for comparison. Obviously, the concentration of the tritium in the rivers of a particular region differs only slightly from the concentration of the precipitates of August-September and from the concentration of the average annual concentrations. A sharp increase in the concentration is observed in the rivers and lakes from the Northwest to the Southeast and toward the middle of the continent. This made it possible to use a larger number of experimental data to confirm the previous conclusion [8] that the tritium concentration of the rivers under consideration is caused by the global fallout of tritium from atmospheric precipitations.

When the run-off of tritium from the USSR territory is evaluated, the tritium concentration in the delta region of rivers whose water drainage forms the major part of the drainage for an ocean basin was taken as the average tritium concentration for all rivers of that basin. The run-off of tritium into the ocean was determined by multiplying the average tritium concentration of the rivers of a particular ocean basin with the total volume of the run-off water taken as many-year average for the individual ocean basins. The seas were divided into ocean basins in accordance with the results of [13].

The concentration of the tritium of the rivers of a particular basin, the annual run-off of tritium into the basin (expressed in percent of the total run-off of tritium from the USSR territory), and the coefficient of tritium run-off in the basin are listed in Table 2. The tritium run-off coefficients obtained for 1975-1976 amount to 0.12-0.71 (depending upon the basin), whereas the average tritium run-off coefficient for the USSR territory is estimated at 0.43 which is about 1.2 times greater than the water run-off coefficient of 0.37 [14]. This result is correct, when one recalls that in the case of evaporation, the separation coefficient of the hydrogen isotopes tritium and protium is 1.1-1.2 at moderate temperatures [5]. It also follows from Table 2 that ~63% of the tritium which fell out with the precipitations on the USSR territory flow off with the rivers into the seas of the Northern Ice Ocean (owing to the increased tritium concentration in rivers carrying continental waters of large-area basins), 28% of the tritium flows into the seas of the Pacific, and only 9% of the tritium flows into the other seas and drainageless water reservoirs of Central Asia. Table 2 includes for comparison data on the average-ocean tritium concentration of surface waters with the mean square deviations and the number of averaged measurements (indicated in parentheses) for 1973-1975.

In the seas in which the water exchange between the surface water and the depth water is reduced but which have a limited water exchange with the oceans (Baltic Sea and Black Sea), the tritium concentration of the surface water is close to the tritium concentration of the rivers discharging into those seas. In the White Sea and in the Barents Sea, in which the water exchange with the ocean is strong, the tritium concentration is

lower than in the rivers discharging into those seas. In 1973 the tritium concentration of the Sea of Azov exceeded the tritium concentration in the other seas in 1974-1975 and practically coincided with the average value of the tritium concentration in the precipitations collected in the basin of the rivers of the Sea of Azov in 1970-1973 (the average value was 115 ± 20). This finding can be explained by the shallowness of the Sea of Azov, the relatively small total volume of the water mass which, on the average, is renewed every 8 years, and the large fraction of the river-discharge water in the balance of the waters of the Sea of Azov [15]. The tritium concentration of the surface waters of the Northern Pacific is lower than that of the seas around the continent by one order of magnitude.

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FIRE SAFETY AND EXPLOSION PROOFNESS OF
THE BITUMINIZATION PROCESS

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The use of organic binders to convert liquid radioactive waste of medium or low activity into a solid, water-resistant form may entail the danger of ignition or explosion both during the process of incorporating the radioactive waste into the binder and during storage of the final product. Bitumen is an example of such an organic binder. In the development of bituminization, much attention is paid to the safety of the process and the storage of the final product.

The main component of waste subjected to bituminization is sodium nitrate which can be a strong oxidant under certain conditions. The interaction of the sodium nitrate with organic products is accompanied, as a rule, by a high release of heat and gas formation. If such a reaction occurs in an apparatus the sharp increase in pressure may result in its destruction. In bituminization, therefore, it is primarily the thermal conditions which are determined and they are taken such that an exothermal process of an "explosive" is not possible. In view of the long-term storage of large volumes of bituminized radioactive waste a question arises about their detonative explosiveness and inflammability.

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TABLE 1. Combustibility of Bitumen Compounds

Filler	Bitumen-to-filler ratio, wt. %	Temperature, °C		
		flashes (in open crucible)	ignition (in open crucible)	self-ignition
Sodium nitrate	40/60	279	354	506
	60/40	293	356	548
Mixture imitating atomic power plant	40/60	293	346	532
	60/40	305	356	538
BNK-2	—	293	345	—

Choice of Safe Bituminization Conditions. Sodium nitrate and bitumen interact in the mixture at all ratios when a certain temperature is reached. Complex thermal analysis was used to determine the temperature at which interaction begins between sodium nitrate and the bitumens BN-4, BN-3, and BNK-2, and tar. The filler content in the mixture varied from 20 to 80 wt.%. For bitumens BN-4 and BN-3 and tar mixtures with 60 wt.% filler were characterized by the lowest interaction temperature (408, 385, and 420°C, respectively); for BNK-2 this temperature was 350°C at a filling of 70 wt.%.

The study of processes occurring when bitumens interact with sodium nitrate in the presence of other components (e.g., alkalis, iron and manganese hydroxides, boron compounds) showed that the temperature at which this interaction begins drops only in the presence of alkalis. As shown by observations, for bitumens BN-4 and BN-3 this decrease is proportional to the decrease in melting point of the nitrate-alkali mixture and exceeds it by 20-60°C. For BN-3, e.g., the temperature of the beginning of the interaction is 240°C for a 1:1 ratio of sodium nitrate to alkalis in the filler. An exception is bitumen BNK-2: when the mixture has this composition its interaction with sodium nitrate begins at ~190°C, i.e., below the melting point of the corresponding nitrate-alkali mixture. This is probably due to the specific nature of the composition of roofing bitumen BNK-2 which is characterized by a high energy release during interaction with alkali.

With the temperatures adopted (up to 200°C), therefore, bituminization of neutral or weakly alkaline waste should not lead to any complications. The danger of an exothermal reaction can arise only in the bituminization of waste with a high alkali content.

Explosion Proofness of Processing and Storage of Solidified Waste. The main characteristic of explosion proofness is the detonability of the materials studied. Theoretical calculations, as well as the few known experimental data, indicate that mixtures of nitrates (apart from NH_4NO_3) with organic substances rank among weakly detonable systems. Weakly detonable mixtures are characterized by large values of the critical diameters of charges, i.e., the diameters below which the mixtures are not capable of detonating [1].

The optimal conditions for stimulating detonation of weakly detonable mixtures comprise a large mass of mixture (respectively, large charge diameter), strong shell, and powerful initiator. Such conditions are observed most of all in field tests in strong steel tubes of large diameter (~100 mm) with initiation with a charge of a powerful explosive.

Field tests were carried out for a mixture of BN-3 with sodium nitrate in a 20:80 mass ratio. The steel cylinders with a sealed bottom (height 1200 mm, diameter 75 mm, and wall thickness 4 mm), used as a strong shell were filled with mixture from the top. An initiator of 400 g of TNT was inserted into the mixture before the test. The cylinders were taken to a testing ground where they were inserted into a hole in the ground only slightly larger in diameter than the cylinders. The mixture was then exploded.

Field tests, however, are extremely involved and cumbersome, require a large quantity of product (~8 kg), and special conditions for carrying them out. In most tests, therefore, a laboratory method was used to determine the detonability of weak explosive mixtures. To take account of the possible radiation heating of the mixtures in some tests the mixture was heated to 60-80°C.

Tests were made on bitumens BNK-2 and BN-3. The mixtures tested had a bitumen content of 7, 15, 20, and 30 wt.%. The fuel used was sodium nitrate as the main component of the waste. The aim of the tests was to establish the limit (of bitumen content) at which the mixture becomes incapable of detonating. The results showed that mixtures containing 15 wt.% or less bitumen are capable of detonating whereas mixtures containing more bitumen do not detonate. This ratio does not change when the mixture is heated to 60-80°C. The consistency of mixtures has a substantial effect on their detonability. Potentially the most powerful mixtures of

stoichiometric or near-stoichiometric composition (15% bitumen) which constitute dense viscous pastes proved to be either at the detonation limit or incapable of detonating. At the same time, a mixture of 7 wt.% bitumen and 93 wt. % sodium nitrate was considerably weaker in respect of energy index but, having a powdery consistency, detonated more easily. This dependence of detonability on consistency is evidently characteristic of mixtures with various aggregate states of fuel and oxidant. It may be assumed that when the mixtures have a dense consistency the propagation of a detonation wave by the shock compression mechanism is not realized owing to the low parameters of the detonation wave for these weak mixtures. At the same time, the propagation of the detonation wave by the mechanism of explosive combustion, which is characteristic of powdery explosives, is impossible because of the absence of free space between the particles of the explosive mixture.

In addition to these compositions, we studied mixtures of nitrates of sodium, potassium, and calcium with other organic substances (TBP, kerosene, toluene, naphthalene). The results showed that all systems are characterized by narrow limits of detonability: from 6-7 to 10-15 wt.% content of organic substances. The lower limit of detonation is evidently restricted by energy factors (specific content of latent energy in the system) and the upper limit, by the consistency of the system which hinders the propagation of the detonation wave. It may, therefore, be expected that variations in the character of the organic binder and the salt composition of nitrate wastes do not significantly affect this limit of detonability.

Determination of the Combustibility of Bitumen Mixtures. The combustibility was determined (see Table 1) for mixtures of BNK-2 with fillers of two types: sodium nitrate and a mixture imitating waste from atomic power plants (94.1% sodium nitrate, 0.2 sulfanol, and 5.7% other salts which are not oxides) [2]. The flash point and ignition temperature for compounds based on bitumen BNK-2 are close to the corresponding values for the pure bitumen. The self-ignition temperature for all samples $> 500^{\circ}\text{C}$.

The introduction of filler into the bitumen probably does not increase its fire hazard. If the adopted regime of preparation and storage is observed, the probability of fire and explosion is small. These aspects should, however, be taken into account in the development of any processes in which organic binders are used.

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DEACTIVATION OF STEAM GENERATORS OF
NOVOVORONEZHSKAYA ATOMIC
POWER PLANT

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In the operation of steam generators of VVER (water-moderated-water-cooled power) reactors, radionuclides build up on the surface of evaporator tubes and feeder collectors which are in contact with the primary loop coolant.

Information about the quantitative composition of these radionuclides can be obtained when steam generators are deactivated, which is done in order to improve the dosimetric environment. This information is of interest from the point of view of studying the dynamics of the formation and spread of radioactive fission products and corrosion products over the loop.

The Novovoronezhskaya Atomic Power Plant has developed a system for autonomous deactivation of steam generators with heating of deactivating solutions. The system ensures continuous circulation of the deactivating solution with a fixed temperature and yields considerably higher coefficients of purification than did static deactivation systems employed earlier.

The steam generators (SG) of the Novovoronezhskaya Atomic Power Plant subjected to deactivation operated on power for a long time: SG-3 with reactor VVER-440 for 27, 144 h, SG-9 with VVER-365 for 37, 149 h, and SG-10 with VVER-365 for 37, 478 h.

The Method and Technology of the Deactivation Process. The deactivation was carried out by an oxidation-reduction method with alternate use of an alkaline solution of potassium permanganate and a solution of oxalic and nitric acids at 90°C. The concentration of the deactivating solutions was, in g/liter: potassium hydroxide 30, potassium permanganate 3, oxalic acid 15, and nitric acid 1.

TABLE 1. Corrosion Products Extracted from Steam Generators during Deactivation

No. of steam generator	Operation	Corrosion products, kg		
		Fe ₂ O ₃	Cr ₂ O ₃	NiO
SG-3 of VVER-440	Alkaline treatment	0,074	0,34	0,015
	Acid treatment	45,76	0,292	0,084
	Washing with care	12,72	0,087	0,049
	Total	58,554	0,719	0,148
SG-9 of VVER-365	Alkaline treatment	3,775	0,784	—
	Acid treatment	17,303	0,562	0,181
	Total	21,078	1,346	0,181
SG-10 of VVER-365	Alkaline treatment	1,195	4,016	—
	Acid treatment	19,819	0,024	0,294
	Washing with care	4,278	0,264	0,127
	Total	25,392	4,304	0,421

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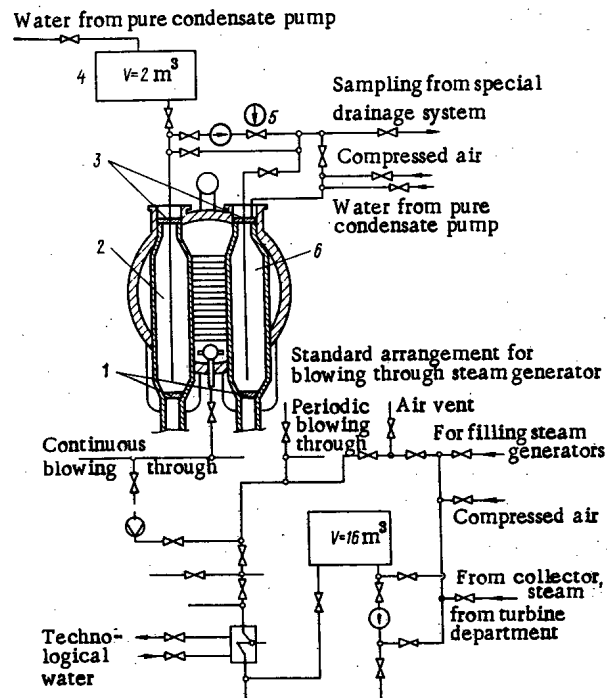


Fig. 1. System for deactivation of steam generator SG-3 of reactor VVER-440: 1) seals; 2) cold collector; 3) sealing flanges; 4) tank; 5) 3Kh-6K-1; 6) hot collector.

TABLE 2. Contribution of Radionuclides to Dosimetric Environment, %

Radio-nuclide	SG-3 of VVER-440	SG-9 of VVER-365	SG-10 of VVER-365
⁶⁰ Co	35,25	36,53	37,03
⁵⁴ Mn	12,18	19,41	26,70
⁵⁸ Co	19,73	9,51	5,21
⁹⁵ Nb	5,52	16,03	8,97
⁹⁵ Zr	4,93	6,63	4,05
⁵⁹ Fe	8,26	3,92	3,57
¹³¹ I	0,61	3,06	1,56
^{110m} Ag	3,21	—	—
¹⁰⁶ Ru	2,96	1,0	2,72
¹⁰³ Ru	3,66	1,03	1,18
¹³⁴ Cs	1,02	0,82	4,0
⁵¹ Cr	0,97	0,5	1,95
¹³⁷ Cs	0,12	0,67	2,58
¹⁴¹ Ce	1,47	0,64	0,29
¹⁴⁴ Ce	0,11	0,25	0,19

The following equipment was assembled for an autonomous deactivation loop for the steam generators (Fig. 1): seals in the Du-50 piping from the side of the main circulating pump, preventing the deactivating solutions from entering the piping; sealing flanges on the steam-generator collector for feeding and circulating deactivating solutions; a 2-m³ tank for preparation of the deactivating solutions; and a 3Kh-6K-1 pump with a capacity of 45 m³/h.

The deactivating solutions were heated from the secondary-loop side by feeding steam ($p = 8-12 \text{ kgf/cm}^2$) from the turbine department into the line for continuously flowing through the steam generators (see Fig. 1).

For periodic monitoring of the deactivation, samples were taken at the pump head and analyzed. The operation was carried out in the following sequence: circulation of the alkaline solution with a concentration of 30 g KOH/liter and 3 g KMnO₄/liter was established and maintained by the operation of the pump; steam was fed into the secondary loop of the steam generator which had been previously filled with water and the solution was heated to 90°C; the washing time with alkaline solution ranged from 4 to 25 h for different steam generators; the alkaline solution was drained by using compressed air and the steam generator was washed with clean

water for 30 min; the system was used for washing with acid solution with a concentration of 15 g oxalic acid/liter and 1 g nitric acid/liter. The washing time with the acid solution was 3-8 h; the final washing was with a weakly alkaline solution (2 g KOH/liter) followed by clear water.

At the temperatures and pressures chosen for the deactivating solutions the presence of usually detectable open defects of the heat-exchange bundle of steam generators does not result in increased leakage of the deactivating solution into the heating solution. Deactivation made it possible to appreciably decrease the level of γ -radiation. The deactivation coefficients were: 8.5 for SG-3 of VVER-440, 5.3 for SG-9 of VVER-365, and 5.6 for SG-10 of VVER-365.

Composition of Precipitates. Table 1 gives the total amount of oxide forms of the corrosion products of Fe, Cr, and Ni extracted from the steam generators during deactivation. The difference in the quantity of the hazardous forms of Fe, Ni, and Cr extracted during deactivation of the steam generators of VVER-440 and VVER-365 is explained by the fact that the area of the heat-exchange surface of the VVER-440 is 1.8 times that of the heat-exchange surface of the VVER-365.

The difference in the quantity of total activities extracted from the SP-9 and SP-10 steam generators of the VVER-365 during deactivation is explained by the different operating times of the steam generators during the run directly preceding the deactivation (6791 and 2087 h, respectively).

The total activity of all γ radiators extracted during deactivation was 80 for SG-3 for VVER-440, 111 for SG-9 of VVER-365, and 68 for SG-10 of VVER-365.

Below we give the quantities of radioactive isotopes washed out, in Ci:

	SG-3 of VVER-440	SG-9 of VVER-365	SG-10 of VVER-365
^{51}Cr	14.7	12.7	22.8
^{144}Ce	7.4	16.1	9.3
^{54}Mn	6.2	16.4	10.5
^{141}Ce	12.3	9.5	1.9
^{60}Co	6.8	11.7	5.5
^{95}Nb	3.2	15.4	4.0
^{58}Co	9.0	7.2	1.8
^{103}Ru	7.5	3.5	1.7
^{95}Zr	3.0	6.7	1.9
^{106}Ru	4.8	2.7	3.4

The contribution of various isotopes to the formation of the dosimetric environment of the steam generators is shown in Table 2.

A distinctive feature of this deactivation of steam generators is the use of an autonomous loop and heating of the deactivating solution from the side of the secondary loop, which appreciably increases the efficiency of deactivation.

The results of γ -ray spectrometric analysis of the deactivating solutions show that the activity in the steam generators is determined by the activated corrosion products (on the average - 36% ^{60}Co , 15% ^{54}Mn , and 10% ^{58}Co) and fission fragments (on the average - 10% ^{95}Nb , 5% ^{95}Zr , 2% ^{103}Ru , and 2% ^{106}Ru).

THE PRODUCTION OF TRITIUM IN FISSION
AND FUSION REACTORS

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The nuclear reaction of ${}^6\text{Li}$ with neutrons, i.e., the reaction ${}^6\text{Li}(n, \alpha)\text{T}$, which for the first time was studied by Chadwick and Goldhaber [1], is presently the main reaction for producing tritium. In the last decade the interest in tritium has increased, because a controlled thermonuclear fusion on the basis of the D-T reaction appears promising. This would make it possible to obtain exothermal energy and, at the same time, to use the neutrons generated for the ensuing production of tritium and other radionuclides. The tritium then could be used for a fusion reactor. In addition to the ${}^6\text{Li}(n, \alpha)\text{T}$ reaction, which takes place mainly with thermal neutrons, tritium can also be formed with fast neutrons in the ${}^7\text{Li}(n, \alpha n')\text{T}$ reaction.

These reactions are possible with lithium materials containing ${}^6\text{Li}$ and ${}^7\text{Li}$; the materials are irradiated in the breeding blanket of a thermonuclear reactor. The production of the tritium includes the preparation of

TABLE 1. Radiation-Chemical and Nuclear-Chemical Processes in Inorganic Lithium Compounds

Compound	$T_{\text{irr}}, ^\circ\text{C}$	Flux (neutrons/cm ²)	Radiolysis products	Chemical form of the tritium	Exit temp. of tritium, $^\circ\text{C}$	Ref.
LiH	From -150 to 80	To $8 \cdot 10^{15}$	—	T_2	200—750	[3]
	-193 † 25 †	—	F centers EPR centers, colloidal lithium	—	—	[4] [5, 6]
LiAlH ₄ LiF	~ 40	To 10^{17}	—	T_2	100—800	[6]
	~ 40	To 10^{17}	Metallic lithium (colloid)	T_2, TF	—	—
	~ 40	—	Molecular fluorine in gaseous pores	$\text{T}_2/\text{TF} = 11$	300—900	—
	~ 40	—	Metallic lithium and gaseous fluorine	—	—	[7—10]
LiCl	30—50	Accumulation of Tup to $4.2 \cdot 10^{18}$ atoms/g	—	T_2, TF	500—830	[11, 12]
	~ 40	10^{17}	—	Diffusion of T_2 T_2, TCl	400—750 100—900	[13, 14] [6]
LiBr	~ 40	10^{17}	—	$\text{T}_2/\text{TCl} = 5$ T_2, TBr	100—600	[6]
LiI	~ 40	10^{17}	—	$\text{T}_2/\text{TBr} = 2$ T_2, TI	100—500	[6]
LiNO ₃	~ 40	10^{17}	Release of oxygen, $\text{CO}_2 = 0.018$ mmole/mole	$\text{T}_2/\text{TI} = 2$ T_2O	200—450	[15—17]
LiNO ₂	270 †	—	$\text{NO}_2, \text{N}_2\text{O}_4$	—	—	[18, 19]
Li ₂ CO ₃	40 †	To 10^{17}	—	$\text{T}_2, \text{T}_2\text{O}$	100—900	[6]
Li ₂ CrO ₄	~ 40	To 10^{17}	—	$\text{T}_2/\text{T}_2\text{O} = 0,25$ $\text{T}_2, \text{T}_2\text{O}$	100—750	[6]
Li ₂ SO ₄	~ 40	—	Crystals of the salt are colored	$\text{T}_2/\text{T}_2\text{O} = 5,6$	—	[20]
LiAlO ₂	~ 30 †	To 10^{22}	EPR centers $\text{SO}_2^-, \text{O}_4^-, \text{O}^-$	—	—	[21]
Li ₂ O	40	—	—	T_2	200—1000	[22]
LiO + Ni	~ 40	To 10^{15}	—	T_2O (96—99%)	300—600	[23]
Li ₂ C ₂ O ₄	~ 40	$5 \cdot 10^{17}$	—	T_2O	100—800	[6]
Li ₂ C ₂ O ₄	~ 60	To 10^{17}	$\text{Li}_2\text{CO}_3, \text{CO};$ some $\text{O}_2, \text{H}_2, \text{CH}_4$	—	—	[24]

*All compounds are in the solid state of aggregation; the LiNO₃ was irradiated at 40 and 270°C.

†Gamma irradiation, in the other cases irradiation in a nuclear reactor.

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TABLE 2. Lithium Materials for the Production of Tritium by Irradiating the Materials in Fission Reactors

Material	Composition	Shell	Irradiation	Liberation of tritium	Ref.
Lithium-aluminum alloy	0.7 mass.% Li with 99% ⁶ Li	Aluminum 1 mm thick	Rods irradiated in reactor channel	Vacuum-thermal treatment of rods with melting	[25, 26]
	10.2% Li with 6.39% ⁶ Li		Material for control rods of reactor	As above	[27]
Lithium-magnesium alloy	With additions of Zr or Cd, Ad or Si, or rare-earth elements		Material for shielding from fast neutrons	• •	[28]
	18.3% Li with 73.04% ⁶ Li		Material for control rods of reactor	• •	[27]
Lithium aluminate	LiAlO ₂ fused tablets	Aluminium	Periodic irradiation of ampules in a reactor channel	Vacuum-thermal treatment of ampules with melting of aluminum shell	[29, 30]
High-temperature ceramics	Li ₂ SiO ₃ , Li ₄ SiO ₄ LiAlO ₂ , LiAlSiO ₄ LiAl(SiO ₃) ₂	As above	Irradiation in an ampule for extraction of tritium during reactor operation	Liberation of tritium in course of irradiation at 200-1000 °C	[31]
Lithium fluoride	LiF	Aluminum, salt in an Al ₂ O ₃ shell	As above	Liberation of tritium in the course of the irradiation	[32]

the materials for their irradiation, the irradiation, the accumulation of the isotope in the material, the liberation of the tritium, its purification, and its concentrating.

Production of Tritium in Fission Reactors. In the last 25 years, since the appearance of the atomic industry, the production of tritium has been undertaken in some highly developed countries. Metallic lithium, its alloys with other metals, and inorganic lithium compounds in the solid state or as melts can be employed for producing tritium [2].

To date a large number of radiation-chemical and nuclear-chemical processes, which occur upon irradiation of the materials and can be used to produce tritium, have been studied in dependence upon the conditions of irradiation and the problems to be solved. Table 1 lists several inorganic lithium compounds for which experimental data on the radiolysis and liberation of tritium have been obtained.

The main results of these investigations formed the basis for the production of tritium with fission reactors. When the lithium material to start from is selected, the problem of its irradiation without risk in a reactor must be dealt with, i.e., the material selected must have high radiation stability. The thermal stability of the material is another important point, because the nuclear reaction and the γ irradiation in the reactor cause considerable heating of the material, with the temperatures even exceeding the melting point. A third point is the tritium liberation process, which must allow complete removal of the tritium and reduce its concentration in the material to be converted, which thereafter is regenerated or stored underground.

In the case of metallic lithium and lithium base alloys, there are practically no problems of radiation-chemical stability encountered, because, except for the nuclear reaction products (helium and tritium), there exist no other gaseous radiolysis products. Radiation-induced damage to the material is possible as a consequence of prolonged irradiation, when tritium and helium are accumulated. In the case of inorganic lithium compounds, the problem of radiation-chemical stability is of great importance, because a large amount of gas and radiolysis products can be formed in dependence upon the irradiation conditions. The lithium materials considered have both advantages and disadvantages (see Table 2) and their utilization depends upon the actual problem. Lithium alloys with aluminum and magnesium and thermally stable inorganic lithium compounds must be considered among the lithium materials patented in various countries.

TABLE 3. Lithium Materials for Producing Tritium in the Breeding Blanket of a Thermonuclear Reactor

Material	Composition	T _{melt} , °C	Irradiation	Production of the tritium	Reference
Metallic lithium	Liquid lithium	162	Lithium surrounding the zone of the thermonuclear reaction	Vacuum-thermal technique	[33]
	Liquid lithium enriched by ⁶ Li; contains solid suspended LiH and LiD particles	162	Liquid lithium circulated with pumps within the zone	Lithium passed through a continuous extracting device	[34]
	Lithium melt with sodium	162	Circulation of lithium in lead tubing	—	[35]
	Liquid lithium				
Nitrate-nitrite mixture	Liquid lithium	162	Circulation of the metal	Extraction of the tritium by a melt of salts	[36, 37]
	LiNO ₃ -LiNO ₂ with Li ₂ O and BeO	200	Melt circulating in tubes surrounding the reactor	Extraction of the tritium in a device for removing gases	[38]
	LiF-BeF ₂ 50-50% mole 69-31% mole	355	Circulation of the melt in the zone of the breeding blanket	Extraction of the tritium by liquid lithium	[36]
		505			
Thermosable compounds	Li ₂ O	1600	Irradiation of solid ceramic materials, use of a gaseous coolant	—	[39, 40]
	LiAlO ₂	> 1620			
	Li ₂ C ₂	Dissociation at t > 700°C			
	Li ₃ N	845			
	LiD	~ 680			

TABLE 4. Some Characteristics of the Breeding Blanket of a "Hybrid" Reactor

Parameter	No. of the zone and its composition (%)					
	1	2*	3	4	5	6
	Nb 100	Nb 10,2 ²³⁸ U 63,0 Li 12,2 He 14,6	Li 100	C 100	Li 100	Nb 100
Thickness in cm of the zone	0,5	12,0	20,0	15,0	3,0	0,5
Tritium breeding	—	0,098	0,885	—	0,069	—

* Exit energy 100 MeV.

Depending upon the designation of the lithium material in the course of its irradiation (simple accumulation of tritium or accumulation of tritium in control rods), the method of liberating the tritium from the irradiated material is selected. Two such methods can be distinguished: extraction of the tritium from the material after unloading it from the reactor, the extraction being effected with the conventional vacuum-thermal technique; and extraction of the tritium from a material which is inside the reactor and in the course of the irradiation, the extraction taking place by using the heat liberated in the lithium material.

The first technique [25] comprises the following stages:

1. Removal of the adsorbed water from the surface of the aluminum shell at temperatures of up to 300°C; the water is retained in a trap cooled with liquid nitrogen.
2. Increasing the temperature of the muffle furnace to the complete melting of the rods and collecting all the gases liberated.
3. Purification of the hydrogen isotopes by absorption with pyrophoric uranium in the form of hydrides or passing all the gases through a palladium diaphragm.

TABLE 5. Rate of Tritium and Helium Formation in the Breeding Zone with Lithium*

Time		Burnup of the lithium (% of the mass)	Total amount of tritium, g	Tritium vol. (liters under standard conditions)	Helium vol. (liters under standard conditions)	Av. concn. of tritium	
days	hours					% mass	Ci/g
1	1	$8,6 \cdot 10^{-6}$	3	11,2	22,4	$4,3 \cdot 10^{-6}$	$4,3 \cdot 10^{-4}$
	10	$8,6 \cdot 10^{-5}$	30	112	224	$4,3 \cdot 10^{-5}$	$4,3 \cdot 10^{-3}$
4,15	24	$2,06 \cdot 10^{-4}$	72	269	548	$1,03 \cdot 10^{-4}$	$1,03 \cdot 10^{-2}$
	100	$8,6 \cdot 10^{-4}$	300	$1,12 \cdot 10^3$	$2,24 \cdot 10^3$	$4,3 \cdot 10^{-4}$	$4,3 \cdot 10^{-2}$
10	240	$2,06 \cdot 10^{-3}$	720	$2,69 \cdot 10^3$	$5,48 \cdot 10^3$	$1,03 \cdot 10^{-3}$	$1,03 \cdot 10^{-1}$
	41,5	$1 \cdot 10^3$	$3 \cdot 10^3$	$1,12 \cdot 10^4$	$2,24 \cdot 10^4$	$4,3 \cdot 10^{-3}$	$4,3 \cdot 10^{-1}$
100	$2,4 \cdot 10^3$	$2,06 \cdot 10^{-2}$	$7,2 \cdot 10^3$	$2,69 \cdot 10^4$	$5,48 \cdot 10^4$	$1,03 \cdot 10^{-2}$	1,03
	365	$8,76 \cdot 10^3$	$7,5 \cdot 10^{-2}$	$2,63 \cdot 10^4$	$9,82 \cdot 10^4$	$1,96 \cdot 10^5$	$3,76 \cdot 10^{-2}$

*Density of neutron flux $4 \cdot 10^{13}$ neutrons/cm²·sec; amount of lithium \approx 70 tons; surface area of the breeding zone \approx 400 m².

4. Separation of the hydrogen isotopes by thermal diffusion.

The second technique comprises the following steps:

1. Accumulation of the tritium in the material and subsequent tritium extraction by pumping off the gases from the channel space or the ampule (continuous extraction is possible during reactor operation).
2. Purification of the hydrogen isotopes.
3. Separation of the hydrogen isotopes.

Thus, at the present time the production of tritium in fission reactors is a problem solved from the technological viewpoint. The experience which so far has been obtained will probably be used to solve new problems related to the tritium-breeding zone in fusion reactors.

Production of Tritium in Fusion Reactors. The appearance of a new source of neutrons (from the D-T fusion reaction) has called for using the thermal energy and neutrons for producing tritium and other radioactive elements. The detailed development of a tritium-breeding zone, which was initiated at the end of the sixties, was related to the "clean" version of a thermonuclear reactor without materials undergoing fission. The active medium in the breeding zone is used as both a coolant and a material for producing tritium. Liquid lithium or melts of salts circulating in the blanket of a thermonuclear reactor and its external systems were suggested as lithium materials fulfilling the above functions (see Table 3).

Solid heat-resistant compounds were considered in addition to molten lithium materials; the solid compounds should be used in combination with a gaseous coolant, e.g., helium. In this case the tritium is separated from the gaseous coolant.

In addition to the clean version of a fusion reactor, recently the "hybrid" reactor version has been considered. Material undergoing fission is placed inside the blanket of the hybrid reactor [41]. When depleted or natural uranium is placed into the blanket of a hybrid reactor (Table 4), one obtains in it ²³⁹Pu, which increases the energy yield per fusion event [42].

In a hybrid reactor, most of the energy (more than 95%) is liberated in the uranium zone which can be used for winning energy. In this case the lithium zone has the function of a tritium producer. One can define for this reactor version both the tritium accumulation rate and the average tritium concentration in the lithium as a function of time (see Table 5).

Under given conditions, the lithium burnup after 1 year amounts to less than 1% and no substantial changes in the absorbing ability of the material are observed. After one year of reactor operation, the total amount of tritium and helium reaches 100 and 200 m³, respectively. The relatively low concentration of tritium in the lithium is of particular interest, as this complicates the separation of the tritium specifically in the initial stage of the reactor operation.

The functioning of the lithium zone in a hybrid reactor can be compared with the irradiation of lithium materials in fission reactors. In a thermonuclear reactor, the lithium materials are irradiated at a relatively high temperature as compared with the irradiation in a fission reactor which precludes losses of tritium by diffusion through the shield. When in the operation of a thermonuclear reactor the blanket temperature exceeds

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300°C, one can no longer ignore the losses of tritium by diffusion [25]. After tritium accumulation, the lithium zone is disassembled and treated to extract the tritium. This is basically a periodic technique of producing tritium; this technique is presently employed when lithium materials are irradiated in fission reactors. This form of irradiation and treatment of the irradiated material imply minimum risk in the work on the lithium zone and the greatest possible tritium extraction.

The disadvantage of this irradiation and conversion technique is the large volume of the initial lithium material to be used in the irradiation and the radioactive emissions after the extraction of the tritium. The slight burnup of the lithium necessitates regeneration and preparation of the irradiated material for its repeated utilization and prevents the permanent underground disposal of the lithium. The large volume of the irradiated material makes it necessary to build special regeneration plants. The duration of the lithium irradiation in the reactor causes tritium losses by radioactive decay (~5% per year).

The continuous liberation of tritium from the lithium zone eliminates some of the shortcomings, but other difficulties related to the reliability of the system as a whole and the quantitative, continuous extraction of the tritium from the lithium material are encountered. Since the lithium zone of the blanket shall be used in accordance with the working cycle of the uranium zone, one must develop a combined program of controlling and adjusting the temperature required for tritium extraction. Further, the time in which the lithium zone remains unloaded depends upon the working time of the uranium zone. The main advantage of the continuous tritium liberation results from the decrease in the tritium losses by tritium decay, the continuous fuel supply of the reactor, and the decreased amounts of tritium and lithium present in the breeding zone.

Obtaining the required temperature is the main difficulty in this process. The temperature in the lithium material present in the neutron flux depends mainly upon the neutron distribution, whose considerable non-uniformity heating of the material. One must therefore develop an automated system for adjusting and stabilizing the required temperature.

The data listed in Table 1 show that at tritium accumulations on the order of 10^{16} tritium atoms per gram (10^{-3} Ci/g), the tritium is separated from most inorganic compounds at a temperature of about 500-600°C. The extraction of the tritium from the lithium materials increases in proportion to the increase in the duration of the irradiation and reaches about 95-98% at an accumulation of about 10^{-20} mass. Therefore the conditions of operation of the lithium zone can be schematically represented as follows: accumulation of tritium at the lowest possible temperature at the lithium material and the shells (the accumulation time is chosen with proper regard for the actual conditions); and increase in the lithium material's temperature and pumping off the gases which contain tritium and are to be converted.

CONCLUSIONS

It follows from the published data on the selection of lithium materials for the breeding zone of a thermonuclear reactor [43-45] that the tritium production involves large-scale radiochemical installations for the conversion of the irradiated material and the extraction of the pure isotope which is present in extreme dilution. When such radiochemical installations are built, a set of basic technological problems must be solved (continuously maintaining the purity of the lithium materials, removing the radiolysis products, stabilization of the physicochemical properties of the material, removing the corrosion products of the materials used for the construction); one must also solve problems related to personnel safety and the protection of the environment from the emission of the radioactive isotope. Problems related to the construction of apparatus for the processes occurring at a high temperature must be solved.

When from this viewpoint lithium materials and their irradiation conditions are selected, one must recall that solid lithium materials (ceramics, alloys) and melts of salts have advantages over metallic lithium for the very reason that they eliminate operations involving a flammable material.

The use of solid lithium materials or of molten systems as a neutron-absorbing material in the breeding zone of the blanket depends upon the characteristics of the tritium-breeding zone in terms of neutron physics and upon the thermophysical parameters.

At the present time nuclear physics research, physicochemical research, and radiochemical research is done on systems with irradiated lithium materials. Methods are developed for separating the tritium which is present in various states at relatively low concentrations. The goal of this work is not only to obtain answers to the technological problems with the greatest reliability but also to effectively select a lithium material suitable for building the tritium cycle of a thermonuclear station as a whole.

In the ensuing stage of the research work, when the blanket zone will be drafted, one must evaluate the various conditions of operation of the breeding zone with metallic lithium, diluted fluorine systems, and solid ceramic materials. At the present time, the selection of the blanket design and the construction of the breeding zone, as well as the selection of the lithium material providing the highest tritium production coefficient, are the basic problems among the engineering problems of thermonuclear fusion and the technological and economical aspects of the forthcoming developments.

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CONTENT OF URANIUM ISOTOPES AND TRANSURANIUM
ELEMENTS IN THE SPENT FUEL OF A
VVÉR-365 REACTOR

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UDC 621.039.524.4-97:621.311.25:621.039

The study of the build-up of isotopes of transuranium elements as a function of the burn-up fraction of nuclear fuel is necessary for the implementation of the system of guarantees under the treaty on the non-proliferation of nuclear weapons, for the optimization of fuel cycles, and for the solution of a number of engineering and economical problems, e.g., establishing the correlations between the contents of various isotopes in spent fuel.

In our work we studied four specimens cut from fuel elements from three fuel assemblies of a VVER-365 water-moderated-water-cooled power reactor of the second block of the Novoronezhskaya Atomic Power Plant with an initial fuel enrichment of 3 at. % ^{235}U , average fuel burn-up in the fuel assemblies 29.0, 19.1, and 7.96 kg/ton uranium, and a cooling time of 2-3 years after irradiation.

The Experimental Method. Each specimen of irradiated fuel was dissolved in a solution of a mixture of 8 M nitric acid and 1 M hydrochloric acid and then the HCl was eliminated from the analyzed solution by three- or four-fold boiling down with 8 M HNO_3 . The solution was filtered and the filtrate was made up to 100 ml of 8 M HNO_3 . The radioactivity of the sediments was 0.01-0.1% of the radioactivity of the parent solution and did not differ qualitatively from it, which is indicative that there is no selective dissolution of specimens.

Determination of the Isotopic Composition of the Analyzed Samples. The ^{238}Pu content was determined by the radiometric method in the separated plutonium fraction and α -spectrometric measurements were carried out with a silicon semiconductor detector; absolute measurements of α -ray activity were performed with a 4π counter. The error of the final results on the ^{238}Pu determination was $\pm 10\%$. The content of ^{237}Np , previously separated by the anion-exchange method from the solution under analysis, was determined by controlled-potential coulometry. The relative rms deviation was $\pm 14\%$.

The content and isotopic composition of U, Pu, Am, Ce, and Nd were determined on an MI-1201 mass spectrometer and Cs, on an MI-1305 mass spectrometer. In the quantitative determination of elements by the isotopic dilution method the radionuclides ^{235}U , ^{242}Pu , ^{243}Am , ^{140}Ce , ^{133}Cs , and ^{142}Nd served as the tags. The elements were separated for mass spectrometric analysis by ion exchange and partition chromatography. The

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TABLE 1. Content of U, Np, and Am Radionuclides in Samples with Various Burn-Up Fractions, kg/ton U

Radionuclide	No. of sample			
	1	2	3	4
²³⁵ U	12,59±0,03	11,952±0,019	20,19±0,23	7,06±0,04
²³⁸ U	3,035±0,019	3,135±0,009	1,83±0,07	3,79±0,05
²³⁸ U	955,86±0,22	954,69±0,28	964,3±0,5	948,2±0,5
²³⁷ Np	0,225	0,265	0,100	0,360
²³⁸ Pu	0,057±0,007	0,054±0,006	0,0064±0,0006	0,132±0,009
²³⁹ Pu	4,92±0,05	5,03±0,06	3,40±0,04	5,04±0,06
²⁴⁰ Pu	1,447±0,018	1,596±0,019	0,518±0,006	2,115±0,023
²⁴¹ Pu	0,799±0,008	0,880±0,009	0,2056±0,0021	1,188±0,012
²⁴² Pu	0,1708±0,0021	0,207±0,003	0,0164±0,0004	0,470±0,006
ΣPu	7,39±0,09	7,77±0,09	4,15±0,05	8,96±0,10
²⁴³ Am	(20,8±1,1)·10 ⁻³	25,0±1,3·10 ⁻³	0,72±0,060·10 ⁻³	82,6±4,4·10 ⁻³
²⁴⁴ Am	(3,9±0,4)·10 ⁻³	5,1±0,5·10 ⁻³	0,080±0,008·10 ⁻³	21,3±2,1·10 ⁻³
Total burn-up	21,07±0,20	22,39±0,28	9,5±0,3	31,9±0,5

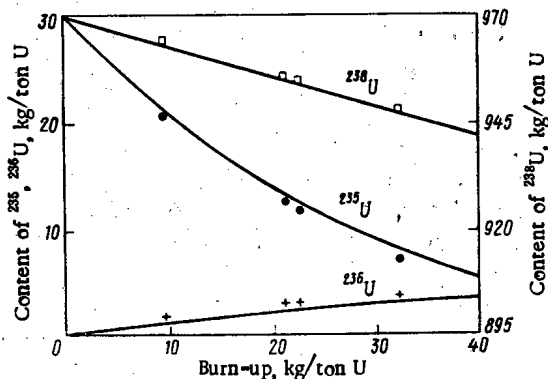


Fig. 1. Content of uranium isotopes vs fuel burn-up.

calculation of the content of isotopes of transuranium elements in the samples analyzed was similar to the method employed earlier [1].

Determination of the Burn-Up of Nuclear Fuel. The magnitude of the burn-up was defined as the ratio of the number of fissioned uranium and plutonium nuclei to the initial number of uranium nuclei. The fuel burn-up was measured in two ways: by the change in the isotopic composition of the nuclear fuel (the heavy-atom method) and by the fission-product build-up in the samples. Determination of burn-up by the heavy-atom method has been described in greatest detail in [2]. In this method the number of ²³⁵U fissions during irradiation with allowance for the diminution of ²³⁸U nuclei owing to fission and radiative capture of neutrons. The error of determination of the fuel burn-up by this method was ~5%.

In determining the burn-up from the fission-product build-up we chose ^{133,137}Cs, ^{140,142}Ce, and ^{143-146,148,150}Nd for the monitors; when using ¹⁴⁴Nd we corrected the measured quantity of this isotope for the ¹⁴⁴Ce constant in the samples. In the calculations of the burn-up from the isotope ¹³³Cs we introduced a correction for its diminution as a result of the n,γ reaction; the correction was determined from the measured ¹³⁴Cs constant with allowance for its decay during the cooling of the fuel after irradiation. In calculating the burn-up from the ¹³⁷Cs, we introduced a correction for the ¹³⁷Cs decay during irradiation and cooling of the fuel. Since the isotopes ^{143,145}Nd have a large neutron capture cross section, the burn-up was determined from the total build-up of the isotopes ¹⁴³⁺¹⁴⁴Nd and ¹⁴⁵⁺¹⁴⁶Nd.

In the present paper we used the yields of the selected burn-up monitors for the thermal spectrum of neutrons, as recommended by the Committee on Nuclear Data [3]. The mean value of the burn-up was defined as the weighted-mean value of the results of calculations of the burn-up by each method. The error of burn-up determination by this method was ~1-3%.

Data on the burn-up and the measured content of isotopes of transuranium elements in the samples analyzed are given in Table 1 which also gives the errors of measurements with a fiducial probability of 0.95.

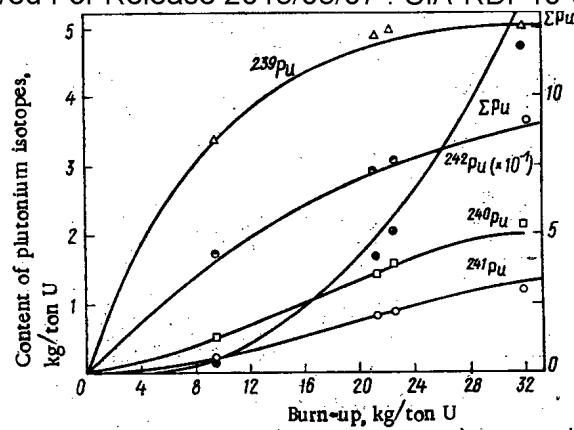


Fig. 2. Build-up of plutonium isotopes vs fuel burn-up.

Discussion of Results. It is seen from Table 1 that as the fuel burn-up increases the contribution of plutonium to the total energy release rises appreciably; with a total burn up of 31.9 kg/ton U the plutonium contribution is 34% of the number of atoms burned up. The values obtained for the burn-up of the sample by the heavy-atom method and the weighted means obtained from the fission products coincide within the limits of experimental error.

With a burn-up above 30 kg/ton U the number of fissioned plutonium isotopes tends to a constant value and in the case considered slightly exceeds 6 kg/ton U. The ^{237}Np increases practically linearly with the burn-up, reaching 360 g/ton (burn-up ~ 30 kg/ton U), whereas the quantity of such valuable elements as americium is only several tens of grams per ton of uranium.

Comparison of the experimental data with the results of calculation (solid lines in Figs. 1 and 2) carried out according to the ROR program [4] showed that the measured content of uranium isotopes coincides with the results of calculations within limits of 1-3%; for plutonium isotopes the difference between experimental and calculated data does not exceed 6%.

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BOOK REVIEWS

Yu. E. Kreindel'

PLASMA ELECTRON SOURCES*

Reviewed by E. G. Krastelev

A large number of plasma electron sources with different purposes, designs, and operating principles have been designed to date. The author of the book under review has attempted to generalize the accumulated material and to interpret it from one point of view, to systematize methods of producing and using plasma to obtain electron beams, and to analyze the design features of the sources developed and their applications.

Yu. E. Kreindel' classifies plasma electron sources by starting from the way in which the emitting plasma is obtained. This approach is fully justified from both the physical and engineering points of view since the mode of plasma production in great measure determines its parameters on which, in the final account, all the principal characteristics of electron sources depend.

A brief exposition of the physical fundamentals of the operation of plasma electron sources is given in the second chapter. In this chapter the author analyzes the emission properties of plasma and the distinctive features of the passage of current in systems with a plasma cathode. Without dwelling on the derivations of familiar relations, the author presents those relations which are necessary for understanding the processes which occur and for evaluating the effect of various factors on the parameters of sources.

Much space in the book is devoted to an analysis of the characteristics and peculiarities of the most numerous family of plasma electron sources based on gas discharges. In respect of operating principle the sources are subdivided into two groups. The first group comprises sources in which the region of discharge, usually a low-voltage discharge, acts as a localized emitter and is separated from the accelerating electrode whereas the second encompasses sources with a high-voltage self-maintained discharge in the anode-cathode gap of a diode, with the beam shaping and electron acceleration taking place in the region of the discharge itself.

The author pays much attention to plasma electron sources with nonstationary emitting plasma of comparatively high density. He focuses on systems in which dense plasma is formed as the result of the explosion of micropoints of the cathode in a strong electric field (high-current diodes with explosive emission) as well as because of a spark discharge on the surface of a solid dielectric, in gas or vacuum gaps. There is a discussion of the shaping of high-current electron beams as the result of the development of instabilities in the plasma of direct high-current discharges.

The last chapter of the book gives a short review of the practical applications of plasma electron sources: in industrial technological processes, electron accelerators, quantum electronics, and general experimental technique.

The relatively slim volume considers various aspects of the construction principles, designs, and characteristics of plasma electron sources of various types. We should point out the value of the systematization of the results of experimental research and practical developments of plasma electron sources.

Along with unquestionable qualities, however, the book does have some shortcomings. Thus, it does not clearly enough formulate the possibilities and ways of further developing various methods of obtaining electron beams. Because of the diversity of the conditions of experimental investigations, whose results are given in the book, it is quite difficult to make an appraisal of them. This is true of the characteristics of high-current diodes with explosive emission (Chap. 6), in particular, the operating time of such systems which are extremely important from the point of view of their practical applications. It is hardly desirable to employ the generic term "plasma electron sources" for all systems considered, let us say, instead of such terms of long standing as electron gun or diode with plasma emitter, especially when traditional schemes and designs are involved.

*Atomizdat, Moscow (1977), 144 pp, 1.43 rubles.

Translated from *Atomnaya Energiya*, Vol. 44, No. 5, p. 450, May, 1978.

Notwithstanding the shortcomings indicated, this book, the first monograph on plasma electron sources, is of great interest for specialists who are involved in one way or another in the development and practical application of such sources.

DEPOSITED PAPERS

CALCULATION OF ABSORBED DOSE IN SOME
PROBLEMS OF TWO-DIMENSIONAL
 γ -RADIATION TRANSPORTYu. I. Chernukhin, Yu. N. Lazarev,
A. I. Orlov, and N. I. Ivanova

UDC 539.122.173.001.2

The present paper analyzes a method developed in [1] for calculating the absorbed dose $F_G^D(z, r, E_0)$ from a collimated monochromatic point source [z and r are the coordinates of the point of detection in a cylindrical system with z axis in the direction of collimation and E_0 is the energy of the γ quanta of the source]; this method is based on the r expansion of the function $F_G^{(M)D}/F_G^{(1)D}$, where $F_G^{(1)D}$ and $F_G^{(M)D}$ are, respectively, the contributions to the dose from singly and multiply scattered quanta:

$$F_G^D(z, r, E_0) \approx A(z, E_0) F_G^{(1)D} [1 + B(z, E_0) r]. \quad (1)$$

The expressions obtained in [1] for the coefficients of the expansion of F_G^D were used to construct plots of $A(z, E_0)$ and $B(z, E_0)$ for water over a wide range of the arguments z, E_0 .

The approximated Green's function (1) was analyzed by comparison with the results of the numerical calculations by the Monte Carlo method. The analytic and numerical calculations were found to be in satisfactory agreement.

The approximate expression found for $F_G^D(z, r, E_0)$ was used to solve two problems of practical importance on the screening of γ radiation: in the determination of the dose from an isotropic γ -ray point source in a dense medium behind an absolutely opaque screen of radius R at a distance L from the source and in the calculation of the efficiency of a limited shield characterized by

$$\kappa = (B_{lim} - 1)/(B_{bar} - 1), \quad (2)$$

where B_{lim} and B_{bar} are the build-up factors behind a limited and a barrier shield, respectively [2].

The solutions of these problems were obtained in analytic form. Comparison of the results of calculations with the experimental data given in [2, 3] showed them to be in satisfactory agreement.

The studies demonstrated that the approximate method of calculation presented in [1] provides a quite reliable analytic estimate of the dose characteristics of the γ -ray field of collimated monochromatic point sources and make it possible to describe a number of functional relations in the problem of γ -radiation transport which had earlier been obtained experimentally.

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Translated from *Atomnaya Energiya*, Vol. 44, No. 5, pp. 451-454, May, 1978.

USING A POROUS METALLIC SILVER ELECTRODE TO
DETERMINE MICROCONCENTRATION OF CHLORINE
IONS IN WATER COOLANTS ON NUCLEAR
POWER PLANTS

L. N. Moskvina, N. Ya. Vilkov,
and V. M. Krasnoperov

UDC 621.039.531.22:543.31

At the present time there are two alternative directions in the search for solutions to the problem of potentiometric determination of chloride ions in high-purity water (including the water coolant of a nuclear power plant): overcoming the numerous technical difficulties which stand in the way of using Ag/AgCl electrodes in flow-type cells [1-3] or creating fundamentally new chloride-selective electrodes.

The paper discusses the possibility of direct potentiometric determination of the microconcentration of chloride ions in the water in atomic power plants by means of a flow-type porous metallic silver electrode (PMSE) and the formation of a reverse potential owing to spherical adsorption of halide ions on the surface of the metal [4]. As the concentration of chloride ions in the solution changes, the magnitude of the PMSE potential varies according to a logarithmic law with a gradient of 15-50 mV/pCl, depending on the conditions of the measurement. The studied operating range of the electrode lies within the limits pCl = 1-7 which facilitates calibration sensors with PSME according to standard chloride solutions.

The paper presents schemes for designs and shows the efficiency of sensors based on PMSE in water similar in chemical composition to the most common water coolants of atomic power plants. With a neutral correctionless regime it is possible to directly determine the concentration of chloride ions right in the current without correction to the composition of the sample. The value of the PSME potential is not affected by the oxygen (up to 12 mg/liter), and turbine oil dissolved in the water. The effect of ammonia, phosphates, and hydrazine over a wide range of concentration, characteristic of water in atomic power plants, can be suppressed by the stabilization of the pH of the analyzed samples in the weakly-acidic region (e.g., with an acetate buffer). The studied operating life of sensors of the proposed design, employing PMSE as the sensitive element, in high-purity water from the make-up system of the water-moderated-cooled-water reactor is no less than 7500 h.

Thus, sensors with a PMSE can be considered as a highly-universal means of measurement which are convenient for monitoring chloride ions in atomic power plants with water-cooled reactors.

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APPROXIMATION OF OPTIMAL CONTROL OF
XENON TRANSIENT PROCESSES

A. S. Gerasimov

UDC 621.039.514.25

A theory of the optimization of regimes of reducing power of reactors with allowance for xenon poisoning has been completed. For the point model and with the assumption that the reactor power can change instantaneously by a finite quantity, apart from phases of operation at maximum and zero power the optimal regime for reducing power can contain two other regimes in which the reactor power varies smoothly with time. These are the so-called singular or "classical" regime and a regime in which the xenon concentration has a maximum allowable value. In the operation of reactors realization of these regimes poses certain difficulties owing to the fact that the reactor power must be varied smoothly.

The present paper proposes a method of approximating the optimal regimes, the "classical" regime and the regimes with maximum allowable concentration of xenon, by means of relay regimes in which the power assumes alternately maximum and zero values. An approximation is carried out with the example of a problem on speed of response. The theory of sliding optimal regimes is employed.

The problem is posed as follows. Suppose that the optimal regime contains a segment of smoothly varying control $U(t)$ (where U is the reactor power), and the phase trajectory in the iodine-xenon phase plane at the beginning and end of this segment passes through points a and b . Find a relay control $U_p(t)$ with a given number of switchings between U_{\min} ($U_{\min} = 0$) and U_{\max} such that the corresponding phase trajectory passes through the same points a and b at the beginning and end of the relay segment; at the same time the xenon concentration should not exceed the maximum allowable value and the deflection of the time of passage over the relay trajectory from the time of motion over the optimal trajectory should be a minimum.

The results of calculations show that the "classical" control is approximated well by a relay control with a small number of switchings. The error of approximation is less than 1% in respect of time of passage with 2-4 switchings. The error of approximation of control with maximum allowable concentration of xenon is 16% in respect of the time of passage with 10 switchings. With 16 switchings the error drops to 8%.

APPLICATION OF Ge(Li) DETECTORS FOR γ -RAY SPECTROSCOPIC ANALYSIS OF ENVIRONMENTAL SAMPLES

É. G. Tertyshnik, L. P. Bochkov,
and S. M. Vakulovskii

UDC 539.1.074.55:539.1.074.3:550.379

The sensitivity of a spectrometric arrangement with a Ge(Li) semiconductor detector of the DGDK-60A type (relative efficiency 4.4%) is compared with the sensitivity of a scintillation spectrometer (BDÉG2-23 detector) based on a 63×63 mm NaI(Tl) single crystal. The energy resolution of the γ -ray spectrometer with semiconductor detector was 6.5 keV at the 1332-keV line and the scintillation unit chosen for comparison ensured a resolution of 59 keV at the 662-keV γ -ray line. To reduce the background, both detectors were placed in cast-iron protection cells with 200-mm walls. The γ -ray detection efficiency for the energy range 100 to 1600 keV was determined with reference radiation solutions and a set of standard spectrometric γ -ray sources.

It is shown that at an energy of 200-1600 keV a spectrometer with a semiconductor detector is not inferior to a scintillation spectrometer in respect of the quality $\epsilon(E)/\sqrt{h}$, where $\epsilon(E)$ is the efficiency of detection of γ -ray quanta with an energy of E at the total absorption peak and h is the background counting rate in the energy interval equal to twice the half-height width of the peak.

When detectors are compared by the value of $\epsilon(E)/\sqrt{h}$ the duration of the measurement of the preparation is not limited. Since the length of the exposure must be taken into account in mass measurements of samples, the minimum activity was calculated for each spectrometer, this being the activity measured in a given time (so that the relative statistical error $\delta = 0.5$ and a measuring time of 120 min showed that the minimum measurable activity for the semiconductor detector studied was, on average, 12% higher than when the BDÉG-2-23 scintillation unit was used. The certain advantage of the scintillation detector with respect to sensitivity can be attained only in γ radiometry when a preparation containing only one γ -ray emitter is measured. In the analysis of samples with several γ -ray emitters the superposition of the peaks in the instrumental spectrum of the scintillation spectrometer results in the errors which arise during the processing of these spectra being several times the statistical error.

The arrangement with the semiconductor detector was used to analyze samples of atmospheric aerosol, radioactive fallout, and bottom sediments taken in 1975. The level of radioactive contamination of the environment in 1975 was the lowest in 15 years. With an air sample of $3.6 \cdot 10^5$ m³ and a measuring time of 2 h the relative error of determination of the concentration of ¹⁴⁴Ce, ¹³⁷Cs, and ¹⁰⁶Ru did not exceed 6-10% and was 25% for ⁹⁵Zr.

With a DGDK-60A detector a ^{137}Cs concentration equal to $4 \cdot 10^{-11}$ Ci/kg in 1-liter samples of soil and bottom deposits can be measured in 120 min with 50% relative error in the presence of interfering radiation from the natural radioactive elements U, Th, and K.

PECULIARITIES OF THE VARIATIONS IN THE ELECTRICAL
CONDUCTIVITY OF ORGANIC DIELECTRICS UNDER PULSED
GAMMA-NEUTRON IRRADIATION

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and S. N. Makeev

UDC 537.311.22.539.12.04

It is well known that when dielectrics are irradiated with doses of less than 50 Mrad [1] their electro-physical properties (e.g., conductivity) vary inversely, i.e., the law of variation holds under repeated irradiation. This permits radiation effects which occur in irradiated dielectrics to be utilized for recording pulsed ionizing radiation.

In the present paper a relation is obtained between the variations in the voltage across the plates of a capacitor ($\Delta V = [V(0) - V(t)]$) and the dose rate of the incident radiation, $\dot{\mathcal{D}}(t)$. When $V(0) \gg 10^3$ V and $\dot{\mathcal{D}}(t) \gg 10^7$ R/sec, this relation for a capacitor with a dielectric of polyethylene and polystyrene is determined only by measuring the conductivity so that if the capacitance of the external circuit is neglected, then $\Delta V \approx \Delta\sigma(t) RV(0)$, where R is the resistance of the load; the change $\Delta\sigma(t)$ in the conductivity is subjected to the law $\Delta\sigma(t) = \gamma[\dot{\mathcal{D}}(t)]\Delta$ [2], where γ and Δ are constants depending on the material of the dielectric. At a dose rate less than 10^7 R/sec the recorded signal is more complex in character owing to the effect of the additional contribution from the radiation polarization which occurs in the dielectric.

On the basis of analysis of experimental studies [3, 4] it has been shown that the effect of radiation polarization cannot be explained as a change in the permittivity. It has been established that the vector of electrostatic induction is of the form $\mathbf{D} = \epsilon\epsilon_0\mathbf{E} + \mathbf{P}$, where \mathbf{P} is the radiation polarization, satisfies the equation

$$\frac{\partial \mathbf{P}}{\partial t} = a\dot{\mathcal{D}}(t)\mathbf{E} - b\dot{\mathcal{D}}(t)\mathbf{P}, \quad (1)$$

Here a and b are constants characterizing the processes of the build-up and decomposition of elementary dipoles in the dielectric owing to electron capture in traps with the formation of a bound pair and neutralization of the pair by free carriers. The equation obtained in this paper for the current $I(t)$ in the external circuit is of the form

$$\begin{aligned} I(t) &= -\gamma[\dot{\mathcal{D}}(t)]\Delta V(t) - \epsilon\epsilon_0 \frac{\partial V}{\partial t} - \frac{\partial \mathcal{P}}{\partial t}; \\ \frac{\partial \mathcal{P}}{\partial t} &= a\dot{\mathcal{D}}(t)V(t) - b\dot{\mathcal{D}}(t)\mathcal{P}(t), \\ V(0) &= \mathcal{E}, \quad \mathcal{P}(0) = \mathcal{P}_0. \end{aligned} \quad (2)$$

The validity of this approach to the interpretation of radiation effects in a dielectric has been confirmed experimentally under the conditions of pulsed γ -neutron irradiation. The paper gives the values obtained experimentally for the parameters b , γ , and $G_0 = a\mathcal{E} - b\mathcal{P}_0$ for polyethylene and polystyrene.

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BOOK REVIEWS

F. Ya. Ovchinnikov, L. I. Golybev,

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OPERATING CONDITIONS OF WATER-MODERATED - WATER-COOLED
POWER REACTORS*

Reviewed by V. I. Pushkarev

A considerable proportion of the program for the development of nuclear power in the USSR is based on atomic power plants with water-moderated-water-cooled power reactors (VVÉR). It thus becomes especially important to generalize the experience gained in operating these reactors. Although the book is entitled "Operating conditions . . .," the authors have extended this concept to the limit, presenting it as a complex of solutions (design and operational) which ensure the reliability, safety, and economy of atomic power plants. No aspect of VVÉR has been left untouched by the authors. In breadth of topics taken up, the book can easily be ranked among handbooks.

The first four of the 12 chapters in the book give a straightforward, readable presentation of the nuclear-physics foundations of the operation of VVÉR and their neutron-physics characteristics under start-up and normal operating conditions. Attention is drawn to the change in the physical characteristics of a reactor because of the properties of the control system and aspects of the maneuverability of the plant which are important for operation under the conditions of an energy system are discussed.

Chapters 5 and 6 analyze the thermal-hydraulic operating conditions of VVÉR at nominal power under transient conditions and during reactor shut-down.

Chapter 7 considers some aspects of in-reactor use of fuel: the reactor recharging conditions and the formation of the initial fuel charge. There is a detailed description of the computational programs used in the choice of fuel loads and for the prediction of fuel characteristics of the reactor in the process of operation.

Chapters 8 and 9 are devoted to the efficiency of fuel elements and structural elements of the reactor. The authors give a detailed description of the physical, mechanical, and corrosion properties of the material of fuel element cans, a zirconium alloy with 1% niobium. Methods are considered for evaluating the efficiency of fuel elements in the operating and shut-down reactor. The distinctive features of the water-chemical operating conditions of the primary loop are given.

Chapter 10 expounds the general conception of VVÉR safety and describes measures which ensure nuclear safety under normal and emergency reactor operating conditions.

Chapter 11 is devoted to the VVÉR-1000 which marks a new stage in the development and perfection of VVÉR. The comparative characteristics are given of VVÉR of three generations, showing the stage-by-stage improvement in the thermal-hydraulic and physical parameters of these reactors. The authors consider the changes made in the reactor control system because of the specifics of the VVÉR-1000 in comparison with previous reactors, especially ensuring inhibition of spatial xenon-induced power oscillations in the core volume. The last chapter discusses the economy of the operation of atomic power plants with VVÉR.

Shortcomings which might be mentioned but which do not reduce the overall high appraisal of the book are the use of the limit method in determining the allowable power level for the fuel elements, fuel-element assemblies, and reactor (Sec. 5.3) and some indefiniteness and imprecision in the formulations of the recommendations of the authors concerning the method of calculation of the fuel component of the electricity generated (Sec. 12.1).

*Atomizdat, Moscow (1977).

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The readability, consistency of the exposition of the material, references to the literature (123 countries), and maximum physical substantiation of the problems posed make the book under review extremely useful for the training of operating personnel of atomic power plants, with VVER. It should be noted that it would also be desirable to publish such books on power reactors of other types.

LETTERS

ERROR OF CALCULATION OF COMPOSITION IN
NONDESTRUCTIVE ANALYSIS OF
NUCLEAR FUEL

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UDC 621.039.548

The principal aim of nondestructive analysis of nuclear fuel unloaded from a reactor is to determine the content of ^{235}U and ^{239}Pu . To determine the composition of a substance containing two components it is necessary to use two measuring channels sensitive to only these components. The content of the components is found from two equations with two unknowns:

$$n_1 = am_5 + bm_9; \quad (1)$$

$$n_2 = cm_5 + dm_9. \quad (2)$$

where n_1 and n_2 are the results of measurements; a , b , c , and d are coefficients determined during calibration of the apparatus; and m_5 and m_9 are the contents of the components.

Usually, the error of the determination of the coefficients in the equation is smaller than the error of measurement. Then, for m_5 we get

$$m_5 = \frac{dn_1 - bn_2}{ad - bc}.$$

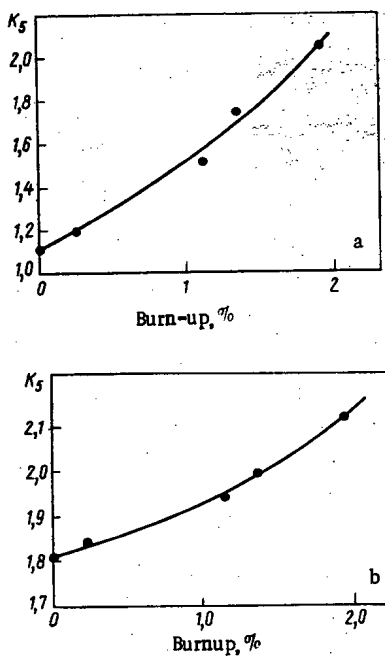


Fig. 1. Error of calculation of ^{235}U content in analysis by: a) γ rays and b) delayed neutrons.

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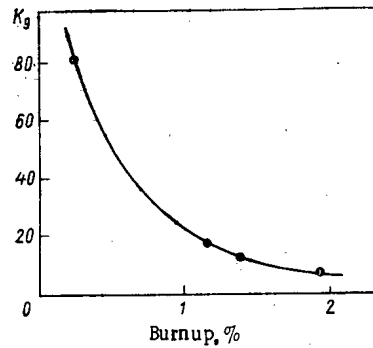


Fig. 2. Error of calculation of ^{239}Pu content in analysis by delayed neutrons.

In this case the error of determination of the ^{235}U content is

$$\frac{\Delta m_5}{m_5} = \frac{\sqrt{d^2 \Delta n_1^2 + b^2 \Delta n_2^2}}{dn_1 - bn_2}$$

If it is assumed that

$$\frac{\Delta n_1}{n_1} = \frac{\Delta n_2}{n_2} = \frac{\Delta n}{n},$$

then

$$\frac{\Delta m_5}{m_5} = \frac{\Delta n}{n} \frac{\sqrt{d^2 n_1^2 + b^2 n_2^2}}{dn_1 - bn_2} = \frac{\Delta n}{n} k_5,$$

where k_5 characterizes the error of calculation of the composition in the mode of analysis adopted.

Using Eqs. (1) and (2), we find

$$k_5 = \frac{\sqrt{(D+B_5)^2 + (B_5+1)^2}}{D-1}. \quad (3)$$

In this expression $D = ad/bc$ is the known value of the discrimination ratio characterizing the selectivity of the measuring channels with respect to the components [1]. Equations (1) and (2) are arranged in that D would be greater than one; $B_5 = dm_5/cm_5$ is the ratio of the terms of Eq. (2), For ^{239}Pu this value is equal to $B_5 = am_5/bm_5$.

Equation (3) enables the capabilities of the method of analysis to be evaluated. For example, in determining the ^{235}U burn-up from the γ radiation of the fission products [2], we get

$$\begin{aligned} n_1 &= 6.3m_5 + 6.5m_9; \\ n_2 &= 0.4m_5 + 4.3m_9, \end{aligned}$$

where the coefficients of the equations represent the yield of ^{137}Cs and ^{106}Ru in the fission of ^{235}U and ^{239}Pu by thermal neutrons [3]. Using data on the burn-up of ^{235}U and ^{239}Pu [4], we can find the error of calculation on the fuel burn-up (Fig. 1). With a low burn-up, the error of calculation of the ^{235}U burn-up tends to a value of $k_5 = \sqrt{D^2 + 1/D} + 1$.

With an increase in the burn-up the error of calculation of the ^{235}U burn-up increases. With a burn-up of 20 kg per ton of fuel and an 8% error of measurement of fission products [2], the error of determination of the burned ^{235}U is about 16%.

For a method of analysis based on the difference in the energy dependence of the cross section for fission with neutrons and the yield of delayed neutrons [2], the coefficients of Eqs. (1) and (2),

$$\begin{aligned} n_1 &= 4.8m_5 + 1.1m_9; \\ n_2 &= 1.8m_5 + 1.0m_9 \end{aligned}$$

are the products of the cross section for the fission of ^{235}U and ^{239}Pu by 10- and 800-keV neutrons [5] and the yield of the delayed neutrons [6].

Figure 1b and 2 give the error of calculation for ^{235}U and ^{239}Pu obtained on the basis of data on the uranium and plutonium content in fuel [4] at various burn-ups. The error of calculation for ^{239}Pu is large owing to the low content and lower sensitivity of the measuring channels to ^{239}Pu .

Thus, the expression obtained for the error of calculation can be used to assess the principal capabilities of the method of analysis.

In conclusion, the author expresses his gratitude to A. M. Kalashnikov and B. A. Bibeovich for their useful discussion.

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CALCULATION OF SHIELDING MADE OF CONCRETE WITH AN INCREASED HYDROGEN CONTENT

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UDC 621-039-78

It was shown in [1] that polymer-based concretes can have a comparatively easily adjusted hydrogen content (up to 60 kg/m^3). Their principal shielding characteristics in fields of neutron radiation with $E < 10 \text{ MeV}$ ($\lambda_{f,n}$ are the relaxation lengths of the fast-neutron fluxes and $B_{i,n}$ are the factors of intermediate neutrons build-up over the thickness of the shielding, equal to (3-4) $\lambda_{f,n}$, i.e., in the zone of the equilibrium spectrum) are nearly exponentially dependent on the hydrogen content in the concrete).

A nomogram (Fig. 1) has been developed for making approximate calculations of the thickness of technological shieldings of concrete with an increased hydrogen content; it shows the dependence of the shielding thickness on the hydrogen content in the composition for various forms of concrete with a density of $2000\text{-}2400 \text{ kg/m}^3$.

The nomogram data were calculated from the conditions of equality of the total doses of neutrons behind the shielding of the concrete being compared and concrete of a base composition with a known thickness (numbers next to the curves). A cement concrete with a hydrogen content of 6 kg/m^3 was taken for the base composition. Concretes with an inorganic (cement) and organic (polymer-cement and polymer) bases are presented along the abscissa axis in order of increasing hydrogen content; the quantity of granulated polyethylenes introduced into the composition is indicated in the case of polymer-cement concretes with an increased hydrogen content [2]. The curves establish the thickness of the shielding made of concrete under study in comparison with the thickness of the shielding made of the concrete of the base composition.

Analysis of the nomograms shows that the use of polymer-based concrete allows the thickness of the shielding to be reduced by 30-40% in comparison with that required with ordinary concrete; as the thickness of the shielding of base-composition concrete increases, the use of concretes with increased hydrogen content becomes less effective; the most pronounced decrease in the shielding size can be achieved at a thickness of 75-150 cm; it is inadvisable to raise the hydrogen content in the composition above 50 kg/m^3 .

The principle by which such a nomogram is drawn up is also applicable for concretes with a density of more than 2400 kg/m^3 , in calculations with conditions of equality of fluxes behind the shielding, as well as in

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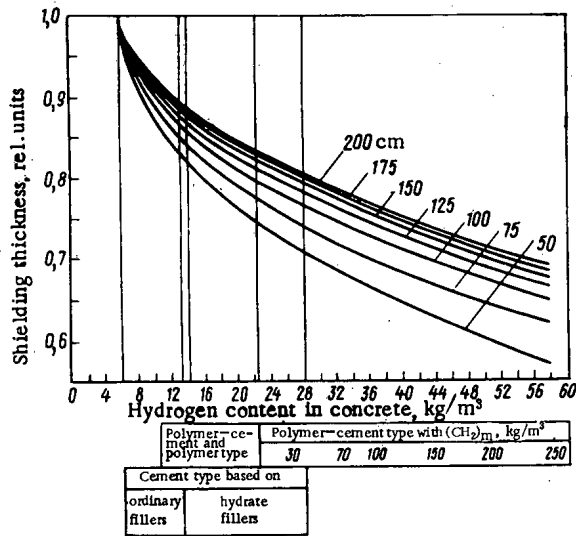


Fig. 1

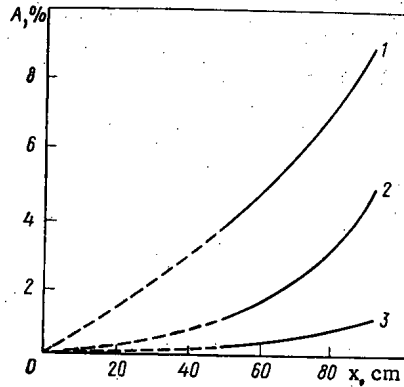


Fig. 2

Fig. 1. Nomogram for comparative calculation of thickness of technological shielding against radiations from reactor spectra.

Fig. 2. Contribution of capture γ radiation to the thickness (A) of biological shielding made of concretes with hydrogen contents of: 1) 30.5, 2) 22.1, and 3) 16.2 kg/m³.

calculations of biological shielding (with allowance for the contribution from capture γ radiation to the total dose rate behind the shielding). The following relations are used to draw up the nomograms:

under the condition of equality of total neutron dose rates behind the shielding

$$T_s = \frac{\lambda_s}{\lambda_o} T_o + \lambda_s \frac{K_f + K_i \cdot B_{i,n,s.}}{K_f + K_i \cdot B_{i,n,o.}} \quad (1)$$

under the condition of equality of neutron fluxes behind the shielding

$$T_s = \frac{\lambda_s}{\lambda_o} T_o + \lambda_s \frac{1 + B_{i,n,s.} + B_{t,n,s.}}{1 + B_{i,n,o.} + B_{t,n,o.}} \quad (2)$$

under the condition of equality of the total doses of neutrons and capture γ radiation behind the shielding

$$T_s = \frac{\lambda_s}{\lambda_o} T_o + \lambda_s \frac{K_f + K_i B_{i,n,s.}}{K_f + K_i B_{i,n,o.}} + \lambda_s \frac{1 + \kappa_s}{1 + \kappa_o} \quad (3)$$

where T_s and T_o are the thicknesses of shieldings made, respectively, of concretes with the composition under study with the base composition, cm; λ_s and λ_o are the relaxation lengths of neutron fluxes in concrete with the composition under study and with the base composition, cm; $B_{i,n,s.}$, $B_{i,n,o.}$, $B_{t,n,o.}$ are the build-up factors for intermediate and thermal neutrons in concrete with the composition under study and the base composition; κ_s and κ_o are the ratios of the dose rates of neutron (total) radiation and γ radiation in concrete with the composition under study and with the base composition; K_f and K_i are the coefficients of conversion of fast and intermediate-neutron fluxes into doses equal to, on average, $3.5 \cdot 10^{-2}$ and $1 \cdot 10^{-2}$, respectively [3].

The ROZ program was used to estimate the contribution of capture γ radiation to the thickness of the biological shielding for concretes of three compositions with various hydrogen contents. The results (Fig. 2) indicate that the contribution of capture γ radiation increases the biological shielding made of concrete with a density of up to 2400 kg/m³ by no more than 10% in comparison with a technological shielding (to a thickness of up to 100 cm). This makes it possible to use a nomogram (see Fig. 1) for the approximate calculation of the biological shielding against radiation from a reactor spectrum. If account is taken of the increase in the contribution of capture γ radiation to the total dose rate behind the shielding (see Fig. 1) as a function of the shielding thickness, it is expedient to include boron to the composition in the form of borax, boron carbide, or other boron-containing materials at the rate of 15-30 kg/m³ [4, 5]. In this case the thickness of the biological shielding can be found from a nomogram for the calculation of technological shieldings, i.e., without allowance for capture γ radiation.

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STUDY OF NEUTRON FIELDS IN CHANNELS OF IONIZATION
CHAMBERS OF WATER-MODERATED - WATER-COOLED
POWER REACTORS

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In recent years it has become practice to measure neutron fluxes and spectra in reactors of atomic power plants by means of activation detectors. The application of the detectors frequently involves calibrating them in a neutron field with a known space-energy distribution. In atomic power plants with water-moderated-water-cooled power reactors (VVÉR) detectors can be calibrated by using the channels of the ionization chambers which control the reactor power. The neutron field in the channels of ionization chambers must also be known for monitoring the operating life of the chambers and their commutation and, moreover, make it possible to verify calculations of radiation fields in a complex geometry of an ionization-chamber channel with a band.

The characteristics of fields of epithermal neutrons (temperature T , parameter $\sqrt{T/T_0}$, absolute values of flux density nv_0) were measured in the ionization-chamber channels of the Novovoronezhskaya and Kol'skaya Atomic Power Plants (NVAPP and KAPP) by the Westcott method [1] with activation detectors of lutecium, indium, gold, samarium, lanthanum, manganese, and copper. The neutron-physical properties of the detectors, apparatus, and methods of the measurements are described in [2].

The use of two-parametric representation of the neutron spectrum for ionization-chamber channels is justified because the spectrum of thermal neutrons in these channels is quite close to a Maxwellian distribution and that of epithermal neutrons, to a Fermi distribution. Thus, the measured values of the neutron temperature are in good agreement with data obtained at NVAPP by the filter method. Table 1 gives the results of measurements of the parameter $\sqrt{T/T_0}$ in the ionization-chamber channels IC-2 and IC-14 of block IV at NVAPP by various detectors (the detector parameters: E_r is the energy of the main resonance in the activation cross section $S_0 = (4/\pi)(I'/\sigma_0)$, σ_0 is the activation cross section for neutrons with an energy of $E_0 = 0.0253$ eV, I' is the excess resonance activation integral, τ is the detector thickness, and G_{epi} is the screening constant of epithermal neutrons).

It is seen from the data of Table 1 that the values of the parameter $\sqrt{T/T_0}$ measured by different detectors are roughly the same, i.e., the spectrum of epithermal neutrons in the ionization-chamber channels are close to the distribution $1/E$. The spectrum parameters and the values of the neutron flux density in channel IC-14 of block IV at NVAPP were also measured during the third run (Table 2). At different times similar measurements were also made in other VVÉR blocks. The results of these measurements are given in Table 3.

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TABLE 1. Characteristics of Detectors and Values of Parameter $r\sqrt{T/T_0}$

Detector	E_r, eV	s_0	$\tau, \text{mg/cm}^2$	G_{epi}	IC-2		IC-14	
					$r\sqrt{T/T_0}$	R_{Cd}	$r\sqrt{T/T_0}$	R_{Cd}
^{115}In	1,457	17,3	0,27	0,95	$0,073 \pm 0,004$	$1,81 \pm 0,05$	$0,073 \pm 0,04$	$1,82 \pm 0,05$
^{197}Au	4,906	2,08	1,27	0,935	$0,081 \pm 0,004$	$1,71 \pm 0,05$	$0,097 \pm 0,004$	$1,76 \pm 0,05$
^{140}La	73,5	0,92	2,43	1,00	$0,078 \pm 0,004$	$9,83 \pm 0,020$	$0,082 \pm 0,004$	$9,31 \pm 0,20$
^{151}Sm	8,01	18,6	1,35	0,82	$0,086 \pm 0,004$	$1,71 \pm 0,04$	$0,085 \pm 0,004$	$1,71 \pm 0,004$
^{63}Cu	580	0,79	27	0,885	$0,077 \pm 0,004$	$11,6 \pm 0,3$	$0,067 \pm 0,004$	$12,0 \pm 0,004$
^{55}Mn	337	0,8	0,27	1,00	$0,064 \pm 0,004$	$12,9 \pm 0,3$	$0,074 \pm 0,004$	$12,2 \pm 0,004$

TABLE 2. Characteristics of Neutron Field

Operating time eff. days	$\text{CH}_3\text{BO}_3, \text{g/kg}$	$I_{\text{Cu}}^{\text{Au}}$	$I_{\text{Cu}}^{\text{Lu}}$	$r\sqrt{T/T_0}$	$T, ^\circ\text{K}$	$n\nu_0 \cdot 10^{10}, \text{neutrons/cm}^2 \cdot \text{sec}$
3	6,5	$1,21 \pm 0,02$	$1,14 \pm 0,02$	$0,093 \pm 0,003$	412 ± 16	$2,02 \pm 0,08$
140	2,5	$1,10 \pm 0,02$	$1,14 \pm 0,02$	$0,077 \pm 0,002$	409 ± 16	$2,11 \pm 0,08$
233	0,8	$1,09 \pm 0,02$	$1,14 \pm 0,02$	$0,076 \pm 0,002$	407 ± 16	$2,03 \pm 0,08$
301	—	$1,05 \pm 0,02$	$1,12 \pm 0,02$	$0,070 \pm 0,002$	398 ± 16	$2,01 \pm 0,08$

*Maximum values in channel at thermal reactor power of 1375 MW.

TABLE 3. Characteristics of Neutron Field in Ionization-Chamber Channels of VVER

Parameter	IC-3 at KAPP I	IC-14 at NVAPP IV	IC-2 at NVAPP II	IC-2 at NVAPP IV	IC-2 at NVAPP IV	IC-19 at KAPP II
$I_{\text{Cu}}^{\text{Au}}$	$1,11 \pm 0,02$	$1,26 \pm 0,02$	$1,00 \pm 0,02$	$1,14 \pm 0,02$	$1,13 \pm 0,02$	$1,29 \pm 0,02$
$I_{\text{Cu}}^{\text{Lu}}$	$1,09 \pm 0,02$	$1,21 \pm 0,02$	$1,05 \pm 0,02$	$1,08 \pm 0,02$	$1,04 \pm 0,02$	$1,12 \pm 0,02$
$r\sqrt{T/T_0}$	$0,079 \pm 0,003$	$0,087 \pm 0,003$	$0,059 \pm 0,02$	$0,082 \pm 0,003$	$0,082 \pm 0,003$	$0,108 \pm 0,003$
T, K	409 ± 16	452 ± 16	380 ± 12	409 ± 16	385 ± 12	428 ± 16

It follows from the data of Table 2 that the neutron spectrum in the ionization-chamber channel does not vary significantly when the concentration of boric acid in the reactor decreases. The neutron flux density at nominal reactor power during a run is stable within the limits of error of measurements for the given channel. The difference in the spectral parameters is explained by the spread of the values of the temperature of the water in the annular tank (from 40 to 80°C) in which the ionization-chamber channels are situated and the spread of the boric acid concentration in the water of this tank (from 0 to 0.04 g/kg). The distributions of the neutron flux density with height, which were measured with copper wires during the run (IC-14; block IV at NVAPP), coincide. The distributions are symmetric about the center of the reactor core and obey a cosinusoidal law.

Thus, studies on the neutron fields in ionization-chamber channels at NVAPP and KAPP showed that the thermal-neutron spectrum in these channels corresponds to the Maxwellian distribution whereas the epithermal-neutron spectrum corresponds to the Fermi spectrum. The characteristics of the neutron fields differ as to stability. All of this indicates the possibility of ionization-chamber channels being used for the needs of the power plant as calibrated neutron sources.

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DEPTH DISTRIBUTION OF THE COLOR
CENTERS IN GLASSES IRRADIATED
WITH ELECTRONS

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Radiation-induced color centers are formed in glasses by trapping of free electrons and holes generated by irradiation, with the trapping taking place at various types of defects which were present before the irradiation. The concentration of the radiation-induced color centers and the extent of the induced optical absorption depend mainly upon the number of electron-hole pairs generated or upon the absorbed radiation energy. When the irradiation is rather weak, the dose dependence of the concentration of the radiation-induced color centers is almost linear and it seems that the spatial distribution of the color in an irradiated sample reflects the distribution of the absorbed energy. A similar correspondence may serve to evaluate the radiation-optical stability of components of electrooptical apparatus.

We consider in the present work the relation between the depth distribution of the absorbed electron dose and the induced optical absorption in industrial optical glass. The Monte Carlo method with a scheme of catastrophic collisions was used in calculating the depth distribution of the absorbed energy of the electrons in samples having a complicated composition and limited size. The algorithm of the calculations was previously explained in detail in [1]. Inelastic interactions taking place under formation of δ electrons and bremsstrahlung gamma quanta with an energy in excess of a given threshold W and events of elastic electron scattering at the atoms of the medium under an energy transfer exceeding 10 eV were considered catastrophic events. Losses of electron energy by ionization and radiation, multiple scattering of electrons, photoelectric absorption, Compton scattering of the γ quanta, and the generation of electron-positron pairs were included in the calculations.

Let us consider the case of normal incidence of monoenergetic electrons upon a glass sample of a given size and composition. The sample was split into several plane-parallel layers within each of which the energy losses, the number of thermalized (with an energy below W) electrons, and the excess charge induced were fixed. The number of particles passing through the surface, of particles reflected from the surface of incidence, and of particles leaving through a lateral sample surface and the energies of these particles were recorded.

TABLE 1. Dose Distribution and Color of the Glasses

Glass	Density (g/cm ³)	E, MeV	X _{max} (g/cm ²)		R, g/cm ²		10 ⁻⁸ (rad· cm ² /electron)	
			dose	color	dose	color	D _{max}	D _{av}
LF -105	3,23	5	0,84	0,74	2,42	2,40	3,8	2,5
		8,5	1,13	—	3,88	—	3,3	2,34
KF -104	2,57	5	0,95	0,90	2,70	2,88	3,5	2,28
		8,5	1,13	—	4,19	—	3,1	2,22
BF -101	2,67	5	0,75	0,67	2,38	2,42	3,6	2,58
		8,5	1,25	—	4,0	—	3,2	2,27
TK-102	3,20	5	0,70	0,64	2,40	2,47	3,6	2,40
		8,5	1,28	—	4,16	—	3,1	2,10
TK-8	3,64	5	0,79	0,72	1,94	1,99	3,8	2,46
		8,5	1,12	1,08	4,33	4,30	3,2	2,14
BK-6	2,86	5	0,80	—	2,58	—	3,6	2,28
		8,5	1,05	1,14	4,86	4,72	3,2	1,80

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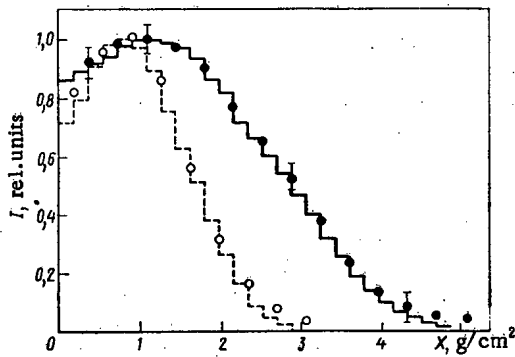


Fig. 1

Fig. 1. Comparison of the depth distribution of the dose (—, ---) and of the optical density (○, ●) in a TK-8 glass sample irradiated with 5-MeV electrons (---, ○) and 8.5-MeV electrons (—, ●).

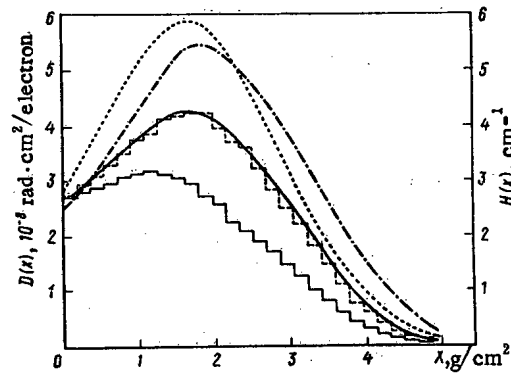


Fig. 2

Fig. 2. Depth distributions of the dose $D(x)$ and of the density $H(x)$ of thermalized electrons in TK-8 glass irradiated with 8.5-MeV electrons: —, ---- (histograms) dose distribution in a sample with the cross section $1 \times 1.5 \text{ cm}$ and dose distribution in a semi-infinite medium, respectively; —, ---- (curves) calculation of the dose distribution with the algorithm of [3]; -·-·-·, ---- density distribution of thermalized electrons with $W = 0.1$ and 0.01 MeV , respectively (for $W = 0.01$, tenfold reduction).

The depth distributions of the absorbed dose were obtained for each sample by following at least 10^4 histories on the computer. The W value was assumed as 0.1 MeV , and the range of the electrons did not exceed 0.1 mm at this energy and could distort the form of the depth-dose curves. When the cutoff energy was reduced to 0.01 MeV , no influence upon the distribution of the absorbed energy was observed. The concentration of the thermalized electrons and their depth distribution depend upon the W value adopted. For $W = 0.01 \text{ MeV}$, the depth distributions of both the dose and the thermalized electrons are almost identical. Calculations have shown that at electron energies of up to 10 MeV , the bremsstrahlung generated in the sample contributes to a lesser extent than the electrons to the dose by 2-3 orders of magnitude.

The calculated depth distributions of the dose were compared with the experimental profiles of the induced optical absorption in glasses having a size of $1 \times 1.5 \times 2.5 \text{ cm}$; the glasses had been irradiated with 5- and 8.5-MeV electrons. The electron transfer figures did not exceed 10^{12} - 10^{13} particles/ cm^2 in the irradiation. It was therefore possible to avoid a saturation of the color at the peak of the absorbed dose. The color profiles were measured with an MF-4 microphotometer.

A comparison of the calculated depth distribution of the dose with experimental color profiles is shown in Fig. 1 for TK-8 glass. The curves are normalized to the maximum. The observed agreement of the distributions indicates that the assumed correlation between the dose profiles and the color profiles must exist. The induced optical absorption by glasses irradiated with monoenergetic electrons increases with the depth and reaches its maximum at a depth corresponding to about 0.3 times the range of the electrons, whereupon the optical density drops off. An extrapolation of the linear sections of the descending parts of the dose curves and the color curves makes it possible to estimate the depth R of energy absorption and coloring of the glasses. The resulting R values are listed in Table 1 along with the depth figures of the maximum of energy liberation ("dose") and the peak of the optical absorption X_{max} (color), the dose value D_{max} at the peak, and the dose D_{av} averaged over the depth of the colored zone.

In [2] the relation between the depth distribution of the F centers in irradiated ionic crystals and the distribution of thermalized electrons was evaluated. For sufficiently small W , the calculated distribution of the energy losses resembles the distribution of the free electrons. The calculation made in the present work for 1.5 MeV electrons bombarding a KCl crystal has confirmed the good agreement between the depth dependence of the dose and the depth dependence of the density of thermalized electrons ($W = 0.01 \text{ MeV}$) with the experimental distribution of the F centers [2].

Figure 2 shows histograms of the depth distribution of the dose absorbed by TK-8 glass; the histograms were calculated for a semi-infinite medium and for a sample with the cross section $1 \times 1.5 \text{ cm}$. In the calculations the dose figures were represented per single energy transfer of 8.5 MeV electrons. The results for the

semi-infinite geometry are in very good agreement with the results of the calculations made with the semi-empirical formulas proposed in [3].

However, in the case of a sample with limited cross section, the depth distribution of the dose differs considerably from the case of semi-infinite geometry, because the release of radiation energy through the lateral sample surfaces is rather significant. For example, in the case of a TK-8 glass sample with a cross section of 1×1.5 cm and a thickness exceeding the range of the electrons, the leakage coefficient for the number of particles is 0.32 and 0.49 and the backscattering coefficient is 0.05 and 0.03 for electrons with an initial energy of 5 and 8.5 MeV, respectively. The leakage coefficient and the backscattering coefficient are defined as the number of electrons leaving through a lateral sample surface and through the surface of electron incidence, respectively, to the number of incident electrons. These electrons carry away an energy which, when calculated per single energy transfer, is 0.92 and 2.06 MeV for an initial energy of 5 and 8.5 MeV, respectively. Bremsstrahlung causes a leakage of 0.2 and 0.6 MeV, respectively. The boundaries influence both the total energy liberation and the depth distribution of the dose.

A comparison of the results obtained in the calculation of the depth distribution of thermalized electrons in a semi-infinite medium (TK-8 glass) with the distribution of the absorbed dose is shown in Fig. 2. It follows from Fig. 2 that at small cutoff energies W , the distribution of the density of thermalized electrons resembles the dose distribution.

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THE EFFECT OF ADDITIONAL MODERATION OF NEUTRONS IN A HETEROGENEOUS CELL WITH A SCATTERER

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UDC 621.039.519.4:621.039.512.4

In measurements on critical assemblies a real homogeneous reactor core is often simulated by an appropriate set of heterogeneously arranged components. In this case the results of the measurements must be interpreted in order to obtain the required physical characteristics of the homogeneous system. Heterogeneous effects are responsible for some features of the behavior of a number of physical characteristics of the reactor, e.g., the space-energy distribution of the neutron flux density.

The present paper gives results which show that not only heterogeneity itself but also its character may lead to certain singularities in the behavior of the neutron energy spectrum. In critical-assembly experiments on the PF-4 physical stand [1] it was found that the neutron spectrum in a heterogeneous cell consisting of blocks of fuel, moderator, and scatterer depend noticeably on their relative positions in the cell. If a scatterer is inserted between the blocks of moderator and fuel without altering the composition of the cell, the neutron spectrum is softer than when the blocks of fuel and moderator are directly adjacent to each other. The softening of the spectrum is evidently due to the fact that part of the moderated neutrons generated in the moderator are turned back into the moderator by the scatterer to undergo further moderated-neutron flux density in the cell owing to absorption in the fuel block. The effect of additional moderation of neutrons probably increases with absorption in the fuel.

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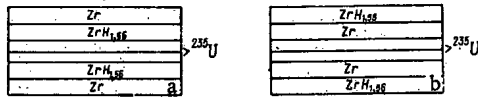


Fig. 1. Arrangement of components in cell.

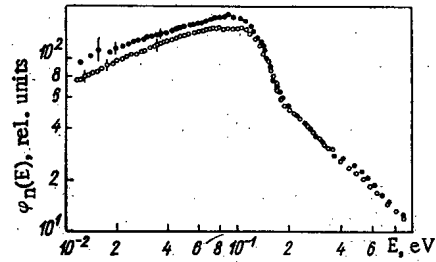


Fig. 2. Energy spectrum of neutrons for variants a (O) and b (●).

To verify this effect we measured the spectra of moderated neutrons in two subcritical assemblies of identical composition but different arrangements of the materials in the cell (Figs. 1a and b). The zero was built up out of disk elements of diameter 46.8 mm. The fuel disk of thickness 0.48 g/cm² consisted of uranium dioxide highly enriched with ²³⁵U. The moderator was a ZrH_{1.56} disk with a thickness of 6 mm and a density of 5.0 g/cm³. Zirconium disks 5 mm thick were used as the scatterer. The cell contained two disk elements of each component.

The neutron spectra were measured by the time-of-flight method under subcritical conditions by using the pulsed neutron generator of the PF-4 stand. The subcriticality of the assemblies was approximately equal to $10\beta_{\text{eff}}$. The neutron beam was brought out from the center of the core and directed at the detector; the luminous surface of the output channel contained two cells (Fig. 2) which made it possible to average the measured neutron spectrum over the height of the cell. The distribution of the neutron flux density $\varphi_n(E)$ was normalized over the energy interval 0.2-1 eV. The results show the neutron spectrum to be softened considerably for the case when the scatterer is between the moderator and fuel disks. A change in the neutron spectrum can lead to changes in the characteristics of the assemblies and this must be taken into account in the analysis of experiments.

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VARIATION OF THE RADIATION DIMENSIONAL STABILITY OF STRUCTURAL GRAPHITE

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Structural graphite displays a nonuniformity in its physical properties which arises at each of the technological stages in the production process. The nonuniformity of the material, in turn, has an appreciable effect on the efficiency of structures built out of it.

The variation of properties over the cross section of the blocks has been determined for some grades of graphite produced and the nonuniformity of these properties has been evaluated by statistical methods [1-4].

TABLE 1. Results of Statistical Processing
of Relative Radiation Variation of Lengths
of Samples*

Material	No. of samples	Cut of samples	$\Delta l/l, \%$			Irr. conditions		
			M	S_n	$\Delta x \dagger$	T, °C	$F, 10^{20}$ neutrons/cm ²	
GMZ	5		0,17	0,04	0,05	70	0,5	
	5	⊥	0,18	0,08	0,10			
	6		0,61	0,09	0,10			1,2
	5	⊥	0,84	0,04	0,05			
	20		1,06	0,12	0,06			1,5
	5		1,04	0,14	0,18			2,4
VPG	6	⊥	1,74	0,24	0,25			
	20		1,91	0,10	0,05		9,2	
GMZ	11		0,03	0,02	0,01	150	3	
	12		0,20	0,03	0,02			6
GMZ ‡	12		0,62	0,05	0,03		8	
	5		-1,14	0,20	0,25	450	12,6	
	5		-0,29	0,07	0,09	700-800	80	
	5	⊥	-0,19	0,04	0,05	700-800	80	
	12		-0,55	0,09	0,06	800	128	
	8	⊥	-0,18	0,09	0,08	800	128	
	16		-1,29	0,12	0,06	950	128	
	13	⊥	-0,01	0,20	0,12	950	128	
	12		-0,89	0,19	0,12	950	146	
	8	⊥	-0,88	0,14	0,12	950	146	
GMZ 2800	6		-1,12	0,22	0,23	450	126	
	5	⊥	0,15	0,18	0,22	450	126	
GMZ 2800 ‡	12		-0,32	0,21	0,13	700-800	126	
GMZ 2800	8		-0,36	0,05	0,04	800	128	
GMZ 2800 ‡	6	⊥	-0,17	0,10	0,11	800	128	
	6		-0,69	0,11	0,12	950	128	
GMZ 2800	5	⊥	0,42	0,21	0,26	950	128	
	5		-0,75	0,21	0,26	950	104	

*Samples of length 40 mm and diameter 4-6 mm.

†For a fiducial probability of 0.95.

‡Material compacted by impregnation with pitch in the intermediate product stage and graphitized at 2800°C.

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TABLE 2. Results of Statistical Processing of Relative Radiation Variation of Length (H) and Diameter (D) of Bulk Samples

Material	No.	Measurement	$\Delta l/l, \%$			Irradiation conditions	
			M	S_n	Δx^*	T, °C	$F, 10^{20}$ neutrons/cm ²
GMZ	27	H	1,38	0,16	0,06	100-180	27
		D	3,76	0,38	0,14		
GMZ	36	H	0,04	0,04	0,01	350-500	6
		D	0,13	0,08	0,03		
GMZ, 2800 compacted	84	H	-0,01	0,05	0,01	350-500	6
		D	-0,02	0,04	0,01		
	50	H	-0,10	0,03	0,01	350-500	6
		D	-0,22	0,05	0,02		
31	H	-0,22	0,03	0,01	350-500	12	
	D	-0,22	0,03	0,01			

*For a fiducial probability of 0.95.

The present paper considers the variation of an important property of graphite, viz., its radiation dimensional stability. This is necessary to do since it is usually assessed from the results of tests (especially tests of long duration) of an extremely limited number of samples. It is known that the radiation dimensional stability, according to [5], is determined primarily by the value of the coefficient of thermal expansion and the degree of perfection of the crystal lattice. However, it is not possible to calculate variation of the dimensional stability by starting from the variational coefficients of the given properties of the graphite; this is because both the properties themselves and their variational coefficients change during irradiation. The situation is also complicated by variations in the irradiation conditions, i.e., the temperature, density of the damaging flux, etc. The variation of the radiation dimensional stability, therefore, is evaluated in this paper by statistical treatment of the results obtained during irradiation of samples.

Graphite of the GMZ grade and its variants were studied. For this purpose, five or more samples 40 mm in length and 4-6 mm in diameter, cut from a single block or from different blocks, were irradiated simultaneously in a common ampul device at constant temperature and density of damaging flux. The change in the lengths of the irradiated samples was determined. In some cases the samples were irradiated repeatedly in order to obtain the dimensional variations as a function of the dose. The irradiation temperature ranges from 70 to 95°C and the fluence reached a value of 10^{22} neutrons/cm².*

Along with the small samples in some cases we also studied bulk samples with a length of 100 mm or more and a diameter of 50-55 mm which were stood upright in the irradiating arrangement. Reactor channels with identical irradiation conditions were used. Varying the height of the bulk samples, we measured their diameter.

Since the samples were chosen not only from the cross section of one block but also from the cross section of blocks of one production batch and different batches, the samples taken for study are quite representative of the given grade of material.

The results obtained from experimental measurements of the relative radiation variation of the lengths of the samples, $\Delta l/l$, of various graphites were processed using Student's coefficient. Table 1 gives the arithmetic mean M, rms S_n , and confidence interval Δx . The results of these calculations for bulk specimens with $l = 100$ mm and $d = 50-55$ mm are presented in Table 2.

For the graphite materials enumerated, the average value of the relative variation of dimensions under irradiation, obtained from tests of no fewer than six samples, characterizes the given material and can be used in calculations.

*Here and henceforth the fluence is given for neutrons with $E \geq 0.18$ MeV.

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TEMPERATURE DISTRIBUTION IN THE RADIATION HEAD
OF A GAMMA-TELE THERAPY UNIT

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Gamma teletherapy units are basic technical equipment in modern radiation therapy. The LUCH-1, AGAT-S, and AGAT-R teletherapy units widely used in the USSR have designs which are standardized to a considerable extent. Thus, all of these units contain a radiation source with the radioisotope ^{60}Co with a nominal activity of 4000 Ci; the source is housed in a radiation head with a standardized biological shield. For uniform attenuation of the radiation in all directions the biological shield is almost spherical in shape and consists of three layers: a tungsten holder for the radiation source, the main shielding jacket of slugs of depleted uranium, and an external jacket of cast steel. The uranium slugs are hermetically sealed in a sheath to protect them from oxidation which could result in the contamination of the construction elements. It is undesirable to cover the uranium slugs with relatively thick stainless steel sheet since this entails a weakening of the shield at the joints. Preference is therefore given to thin metal sheaths applied by electroplating; the strength of such sheaths depends significantly on the degree to which the uranium slugs are heated by the radioactive decay of ^{60}Co . The thermal power dissipated in the mass of the radiation head is determined by the radiant energy and the activity of the radioisotope, self-attenuation of the radiation in the source, and absorption of the radiation in the materials of the shield.

The thermal deformations and temperature gradients which arise in the process should not result in emergency situations caused, in particular, by the rupture of the air-tight sheaths of the uranium slugs.

In the course of operation the radiation head can be in the irradiation position when the exit window and the end of the source is cooled directly by air. The storage position when the exit window is closed is less favorable. Even with operation in two six-hour shifts, with allowance for the additional time for positioning the patient and adjusting the unit, the radiation head is in the storage position for no less than 80% of the entire operating time. Owing to the short duration of the irradiation (2-10 min) and the high inertia of the thermal processes, it may be assumed that the relevant thermal characteristics under storage and irradiation conditions differ very little from each other in practice.

For these considerations the experiment was conducted under conditions which simulated primarily the storage conditions of the source. The temperature of the radiation head was measured in one technically accessible radial direction since under storage conditions the temperature distribution in the head may be taken to be practically isotropic and the problem of determining the temperature distribution may be considered to be one-dimensional.

The thermal state of the standardized biological shield in the radiation heads of the domestic gamma teletherapy units LUCH-1, AGAT-S, and AGAT-R was studied on the identical radiation head of the RAD-1 unit which has the same biological shield and is designed to hold a ^{60}Co source with a nominal activity of 4000 Ci.

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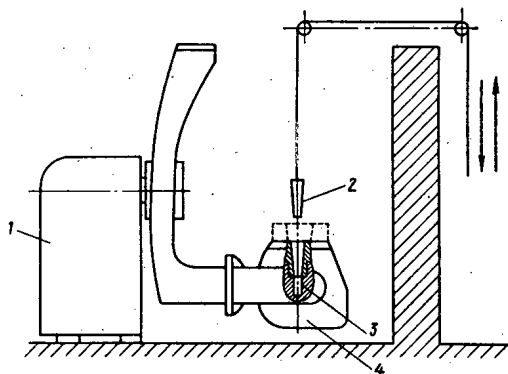


Fig. 1

Fig. 1. Experimental arrangement: 1) rotational gamma teletherapy unit; 2) cone with built-in thermocouples; 3) radiation sources; 4) radiation head.

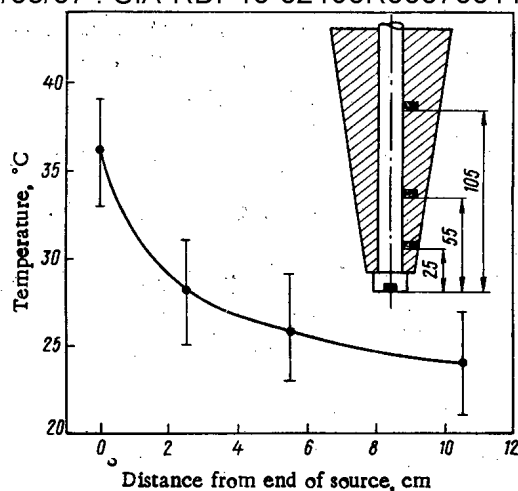


Fig. 2

Fig. 2. Temperature distribution over radius of shielding jacket of gamma teletherapy unit.

During measurements of the temperature the source activity was 2010 Ci and its thermal power was 30.9 W since the specific energy release of ^{60}Co is 0.0154 W/Ci [1].

The experimental arrangement is shown in Fig. 1. The radiation head was rotated on the stand of the unit with the exit window pointed upwards, the diaphragm was removed from it (without disturbing the electrical connections), the shutter was opened from the control panel so as to release the radiation beam, and then a cable and a system of control pulleys were used to insert the cone into the exit window; this cone, whose size corresponded to that of the window, had four (Chromel-Copel) thermocouples (Fig. 2) at various levels. One of the thermocouples was built into the end of a rod running through the center of the cone; the purpose of this thermocouple was to measure the temperature of the source ampul.

The cone fitted tightly into the window of the radiation head, thus ensuring reliable thermal contact between the cone and head. Stable temperature conditions were established in the course of one hour. The air temperature in the premises was 22°C during the measurements.

An EPR-09MZ potentiometer was the recording instrument. The error of measurement with allowance for the compensating coil error and the recording error did not exceed 3°C.

Solid cones of tungsten and of steel as well as a hollow conic vessel were tested. The results of measurements for the tungsten and steel cones practically coincided whereas lower temperatures were obtained for the hollow cone which simulated the conditions of long irradiation (practicable only for phantom measurements).

The results of temperature measurements in the radiation head (by using solid cones) are given in Fig. 2. The temperature of the radiation head where it comes in contact with the radiation source was 36°C, which was 14°C above room temperature. Proceeding from this, we can estimate the maximum temperature of the slugs in the shield for a source with a 4000-Ci ^{60}Co source. Although the relation between the maximum temperature of the slugs in the shield and source activity is nonlinear owing to the dependence of the heat-transfer coefficient on the temperature of the radiation head surface and the air in the premises, in the temperature range under consideration the nonlinearity has an insignificant effect [2]. Therefore, when the source activity is raised from 2010 to 4000 Ci the difference between the temperature of the shield and that of the surroundings may be assumed to increase in the same proportion, and then the maximum temperature of the shield at the place of contact with the ampul surface is 50°C. The error resulting from the assumption that the heat-transfer coefficient is constant in the temperature range considered does not exceed +15% and thus in fact the maximum shield temperature may prove to lie within the limits 48-50°C.

Bearing in mind that 1) according to current standards the activity of the radiation source at the time of installation may differ from the nominal value by up to +20% (which corresponds to roughly 6°C), that 2) according to the technical conditions the capability of gamma teletherapy units to operate should not be affected at an

ambient temperature of up to +40°C, and that 3) the error of measurement at an activity of 2010 Ci was $\pm 3^\circ\text{C}$, we may take $t_{\max} = 50 + \sqrt{6^2 + 18^2 + 6^2} \cong 70^\circ\text{C}$ to be the maximum temperature for the shield in the radiation head of domestic gamma teletherapy units.

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INTERNATIONAL COOPERATION**SOVIET - JAPANESE COOPERATION ON THE PEACEFUL
USES OF ATOMIC ENERGY**

B. A. Semenov

The Soviet Union cooperates with many countries on the peaceful uses of atomic energy. Under the terms of some 35 agreements signed with other countries the Soviet Union is engaged in extensive scientific contacts, exchanges of delegations, joint scientific research, the design and construction of various installations, nuclear reactors, etc. The use of atomic energy for peaceful purposes is an area of science, technology, and economics in which the Soviet Union and Japan have had great achievements.

Japan has built up a considerable atomic power industry (the total installed capacity of the atomic power plants is about 8000 MW) and atomic engineering industry. A fast experimental reactor (JOYO) and a heavy-water reactor (Fugen) have been developed and constructed. Work is under way on nuclear fuel processing and research is being conducted on controlled thermonuclear fusion. All of this provides prerequisites for the development of cooperation between the USSR and Japan on the peaceful uses of atomic energy.

Until recently, such cooperation did not have legal formulation but specialists had already been in contact. Thus, as the result of an agreement, a delegation of representatives of Japanese atomic science and technology, led by T. Doko, Vice-President of the Atomic Industrial Forum of Japan, visited the Soviet Union in June, 1973. The purpose of the visit was to allow the delegation to become acquainted with the scientific and industrial centers in the nuclear power field, the state of the art, and the prospects for future development. During the visit the delegation saw scientific research institutes engaged in research on nuclear reactors, atomic power plants, and power engineering works. The delegation had talks with A. M. Petros'yants, Chairman of the State Atomic Energy Committee of the Council of Ministers of the USSR (GKAE SSSR), and Soviet specialists.

During a return visit in January, 1974, a Soviet delegation led by I. D. Morokhov, First Deputy Chairman of the GKAE SSSR, familiarized itself with Japanese scientific centers and enterprises and had talks with eminent industrial and government figures.

At the request of the Atomic Forum a delegation visited the Soviet Union in January, 1975, to acquaint itself with problems of atomic power plant safety and radiation safety. The delegation visited the leading Soviet institutes working in this field, atomic power plants, and a station for radioactive waste burial, and also had talks in the GKAE and the Ministry of Power and Electrification of the USSR. Before its departure the delegation was received by the Environmental Protection Commission of the Supreme Soviet of the USSR.

These contacts and exchanges enabled the specialists of our countries to become more familiar with the successes achieved in both our countries and to discuss the possibility of concluding an agreement on cooperation between the GKAE and the Atomic Industrial Forum.

During a visit to Moscow in June, 1977, T. Doko, President of the Federation of Economic Organizations of Japan, and his talks with A. M. Petros'yants as well as in subsequent discussions with the Atomic Forum both sides drew up a draft agreement; and in November, 1977, a Soviet delegation headed by A. M. Petros'yants visited Japan to conclude the agreement on cooperation on the use of atomic energy for peaceful purposes. During the visit the delegation had discussions with officials of the Atomic Forum, had talks with the Ministers of Scientific Research and Technology, Foreign Affairs, and External Trade and Industry. The Soviet delegation also visited atomic research centers (Tokyo Research Center of the Japanese Atomic Energy Research Institute JAERI, the Oarai Engineering Center of the State Corporation for the Development of Power Reactors and Nuclear Fuel PNC); firms working in nuclear power, research centers, and factories producing atomic power plant equipment (Mitsubishi in Kobe, Sumitomo and Ikawajima Heavy Industries in Yokohama, and Hitachi in Hitachi); atomic power plant (Tsugura, Mikama, and Fugen) with boiling, water-moderated-water-cooled, and heavy-water reactors, as well as the JOYO fast reactor.

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Two meetings were held with prominent figures of the Federation of Economic Organizations and the Atomic Forum. During these meetings both sides informed each other about the development of the nuclear power industry in their countries, discussed the possibilities of economic cooperation, and also went into the subject matter which could be of greatest interest for mutual cooperation. The GKAE and the Atomic Forum agreed that a program of cooperation for a period of two to three years under the signed agreement would be drawn up in the near future and both countries would proceed to implement it.

In the course of the visit to Japan the leader of the Soviet delegation gave a lecture for the Japanese Scientific Society on the development of nuclear power in the USSR; a press conference was held and there was a working meeting with members of the Japanese parliamentary committee on science and technology.

The visit of the Soviet delegation was concluded with the signing of an agreement on cooperation in the domain of the peaceful use of atomic energy. On behalf of the Soviet side the agreement was signed by A. M. Petros'yants and on behalf of the Japanese side by H. Arisawa, President of the Atomic Forum. The signing ceremony was attended by the Soviet Ambassador to Japan, D. S. Polyanskii, and T. Doko.

The agreement will make it possible to organize scientific and technical cooperation in such major areas of atomic science and technology as power reactors and controlled thermonuclear fusion.

At its first session in Tokyo in January, 1978, the Soviet-Japanese Commission on Scientific and Technical Cooperation discussed various aspects of the cooperation between Soviet organizations and official Japanese organizations within the framework of the 1973 Agreement on Scientific and Technical Cooperation. The discussion dealt with a number of regions of mutual interest, including the peaceful use of atomic energy. The Commission set up a working group on atomic energy to discuss and prepare proposals on concrete subjects and forms of cooperation.

Soviet-Japanese cooperation in the realm of atomic science and technology will thus be implemented in accordance with the agreement with the Atomic Forum whose membership consists of some 700 firms and economic and research organizations of Japan as well as on a governmental level within the framework of the intergovernmental agreement on scientific and technical cooperation.

CONFERENCES, MEETINGS, AND SEMINARS

INTERNATIONAL SEMINAR ON THE TECHNOLOGY
OF SODIUM COOLANT

V. I. Kondrat'ev and Yu. V. Privalov

A seminar with the participation of some 30 scientists and specialists from the USSR, Belgium, and the Netherlands was held in Dimitrovgrad in October 1977. The successful development of fast reactors at the present time is determined in great measure by achievements in solving problems of sodium technology. The high cost and great complexity require the collaboration of scientists of various countries. This seminar marked one stage in such collaboration.

The seminar heard 18 papers on problems of sodium purification from impurities, indication of leaks from sodium-water steam generators, corrosion and mass transfer of steel into sodium, safety of sodium loops, and the behavior and analysis of impurities in sodium. In a review paper V. M. Arkhipov (USSR) generalized the experience from sodium technology of the BR-5, BOR-60, and BN-350 reactors, concerning the preparation of the coolant, its contamination during start-up and initial operation as well as normal operation, elimination of the consequences of breakdowns and washing off the equipment, sampling, and extinguishing sodium fires. Sodium fire extinction was discussed in greater detail in a paper by I. G. Kobzar (USSR).

M. Van Hasselt (Netherlands) told of experience of work with sodium on experimental stands, formulated requirements on the purity of sodium coolant which would be safe from the point of view of corrosion, and drew the conclusion that "cold" traps can ensure the required purity in respect of impurities (in this case, carbon as well) in addition to purification from products of the water-sodium reaction. F. Castilles (Belgium) presented the results of tests of structural materials in nonisothermal dynamic sodium systems using foils to determine the activity of carbon, noting that the activity of carbon in a nonisothermal loop is determined by the temperature of the "cold" trap. The plastic properties of stressed ferrite steel are improved by long holding (22,000 h) in a stream of moving sodium at 700°C. A paper by V. Kolster (Netherlands) was devoted to experimental substantiation of the required sodium purity in respect of oxygen.

Papers by M. Van Hasselt (Netherlands), M. Coenen and F. Livens (Belgium) devoted much attention to instrumental and chemical methods of impurity monitoring, methods which are at a high technical level in those countries. The extensive experience of Soviet researchers in the purification of sodium coolant by "cold" and "hot" traps, of inert gas, and of coolant and gas from radioactive impurities, and optimization of purification systems was presented in papers by F. A. Kozlov and V. F. Bargetsov.

F. Meyer and J. de Bries (Netherlands) reported on interesting developments of acoustical methods of monitoring water leaks in steam generators and monitoring sodium boiling in reactor core packets. In their opinion, which coincides with the point of view of Soviet specialists, these methods can be used to establish water leaks with a flow rate of more than 0.2 g/sec. The system for leak indication in industrial steam generators should, therefore, be a complex system using both concentration-measuring and acoustical methods. The advisability of using a leak indication method based on measurement of the increased hydrogen concentration in the gas cavity of the loop was shown in a paper by Yu. V. Privalov (USSR).

The kinetics of reactions of carbon-containing gases with sodium and the chemical equilibrium in the sodium-oxygen-hydrogen system were described in papers by Yu. I. Zagorul'ko and Yu. V. Privalov (USSR). Noteworthy attention was paid to the safety requirements during development of equipment for the SNR-300 for recharging and storing spent fuel elements in sodium (paper by M. Deboche, Belgium).

The final session summed up the work of the seminar and pointed out its usefulness. The participants exchanged views about the state of the art and further development of work in their countries on fast reactors and expressed a desire to hold joint seminars in the future.

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IAEA ACTIVITIES ON HIGH-TEMPERATURE REACTORS

V. N. Grebennik

A Technical Committee on High-Temperature Reactors met in Vienna in December 1977. The meeting, which was convened by the IAEA and was held within the framework of the nuclear power and reactor section, was attended by 23 specialists from Austria, Belgium, France, the Federal Republic of Germany, Japan, the Netherlands, Poland, Switzerland, the USSR, the United Kingdom, the USA, and two European organizations, the CEC and the OECD/NEA. Heeding the interest in the development of the concept and prospects of the industrial application of these reactors, the IAEA invited the countries most active in this area to participate in the work of the Technical Committee to draw up recommendations for future work on NTR and to set up an international working group. In the course of the work of the Agency over recent years a number of working groups on various subjects were set up. The function of these groups includes exchanging information about national program and concrete technical problems, working together with the Agency to organize conferences, symposia, and meetings of specialists on the most pressing problems, and to provide other assistance. The activities of the Working Group on HTR will be extended to helium-cooled thermal reactors for the production of electricity and technological heat as well as to CO₂-cooled reactors and fast helium reactors to the extent that they are not encompassed by other international organizations and working groups. It will be the task of the International Working Group to discuss the overall strategy program for HTR development, planning research, (considering problems of the design, construction, and safety and operational aspects of atomic power plants, the technology of the fuel cycle, and considering the areas and processes of application of thermal energy from HTR.

At its meeting, the Technical Committee considered national HTR programs and discussed papers on special topics of particular interest at this time (constructional and operational experience, promising fuel cycles, achievements in the application of thermal energy from HTR, etc.). The papers and communications presented by experts provided information about HTR already in existence, under construction, or on the drawing boards.

As before, the AVR (Federal Republic of Germany) with a power of 15 MW(E), which employs spherical fuel elements, is operating stably. Over the past 3.5 years AVR has been operating with helium at a temperature of 950°C with a power factor of 0.8-0.9. The construction of the prototype THTR-300, which will have a core of similar design, is to be completed in 1980. The delay in the completion of the reactor in comparison with previously set deadlines (1978-1979) has been due to the fact that additional safety requirements have been introduced into the project. Intensive testing is continuing on the reactor of the Fort Saint Wrane Atomic Power Plant (U.S.A.) which has a power of 330 MW(E) and operates on slug-type fuel elements. Over a period of 1 year (from mid-1976 to July 1977) the reactor periodically (~30% of the time) operated at 28-30% power. Once the difficulties of the start-up period were overcome and corrections were made for the malfunctions caused by the conventional steam power-generating equipment and specific malfunctions of the plant (internal leakages of helium in gas blowers, overheating of the reactor vessel at the places where the control and safety system passes through, the distribution of gas flows in the thermal insulation), the reactor reached 40% power in August 1977. It operated at this level for about two months. At the end of October, 1977, permission was obtained to increase the power to 70% and this level was attained. It is assumed that the reactor will reach nominal power in 1978.

In Japan work is under way on an experimental very-high-temperature reactor VHTR with a power of 50 MW(T) at a helium temperature of 1000°C at the outlet. Construction is to begin in 1978. As part of the project studies are to be made on processes of direct reduction of iron by using the heat from the VHTR. In the Soviet Union, work is being done on an experimental high-temperature energy-technological facility with a reactor power of 50 MW (electrical) and on a high-power prototype reactor operating on thermal and fast neutrons.

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The effects of the advanced countries in promoting the HTR concept in recent years have been characterized by the concentration of extension of work in the area of the most effective commercial application of this type of reactor. Some countries (the U.S.A., the Federal Republic of Germany, Switzerland, etc.) are working on the use of the HTR for the generation of electricity, preference being given in this case to the more promising gas-turbine cycle which ensures higher efficiency and makes effective use of air cooling. In the Federal Republic of Germany, Japan, the USSR, France, and the U.S.A. most intensive work is being done on the energy-technological application of such reactors in chemistry, metallurgy, and other branches of industry, for long-distance transmission of thermal energy, etc. The introduction of HTR in these areas is based on the production of hydrogen and synthetic fuel by gasification of coal and decomposition of water. At the present time intensive studies are being made primarily on topics which bear on the commercial introduction of the reactor: development of a fuel cycle, design of equipment and plant (vessels of prestressed concrete, gas flowlers, heat exchangers, conduits of the control and safety systems, etc.), substantiation of the safety and reliability of atomic power plants, licensing, etc.

In some countries work began in 1977 on changes in HTR fuel cycles in connection with the requirement of nonproliferation of atomic weapons. The fuel cycle envisaged for reactors in operation, under construction, or in the design stage is one which ensures a high coefficient of conversion with the use of highly enriched uranium (~93%) and thorium. As the result of new limitations on the enrichment of fuel for nuclear power it is proposed that the HTR operate with a uranium-thorium cycle in which the uranium enrichment is about 20% as well as with a cycle based on low-enriched uranium (5-10%). Work has begun on the technology for fabricating microparticles out of fuel with the new composition. A license for the industrial use of this fuel, especially in the reactor of the Fort Saint Wrane Atomic Power Plant, is to be obtained after 1980. The commercial introduction of large HTR in some of the most advanced countries in this area (Federal Republic of Germany, the U.S.A., etc.) is expected to begin in the late 1980s - early 1990s. It is proposed to use these reactors, which have higher efficiency and more economic fuel cycle, for electric power generation ($N_{e1} = 1200-1300$ MW) and especially for the production of industrial heat ($N_T \sim 3000$ MW). Large prototype plants for the energy-technological use of HTR operating with a gas-turbine cycle are to be built within the next decade. In the opinion of specialists, the process employing high-temperature heat from an HTR which is best mastered at the present time and which is proposed for immediate introduction evidently will be the process of obtaining synthetic gaseous or liquid fuel from coal. This is considered as one of the principal directions in the program to ensure future energy supplies in countries with considerable reserves of lignite and coal.

The meeting of the Technical Committee discussed a possible strategy for the development of nuclear power involving HTR and fast helium breeder-reactors with a short fuel doubling time. Work being done on fast reactors within European groups is devoted primarily to the study of safety, completion of the main design features, and reactor tests on the effect of high fluxes on the structural materials and fuel compositions.

The International Working Group on HTR drafted a program of work and recommended that meetings of specialists be held within the framework of the IAEA in 1978-1979 to consider the most urgent aspects of HTR; this will enable specialists of various countries to exchange views and to discuss concrete problems and ways of solving them. It was decided to invite specialists of Italy, Spain, and Sweden to participate in the Working Group on HTR in 1978.

The breadth of possible applications of HTR and the large volume of research entailed are inducing many countries to unite their efforts in order to concentrate the work and to reduce the time and expense of research and construction of facilities. International collaboration within the framework of the IAEA is an important condition for the successful development of the conception of HTR. The activities of the International Working Group on HTR set up by the IAEA will be aimed at providing all-round assistance in this field.

MEETING OF IAEA EXPERTS ON THERMONUCLEAR REACTORS

G. N. Popkov

The meeting, which was held in Madison, Wis., in October 1977, considered plans for thermonuclear reactors under development at the present time. These included plans for: 1) the TFTR, JT-60, JET, and T-10M facilities which will be built after T-10 and PLT. These facilities should yield reactor plasma parameters; 2) reactors for achieving ignition of a thermonuclear reaction and for studying the interaction of α particles with plasma. It is planned to build them after facilities of the TFTR and JET type; 3) reactors for studying tritium breeding, processes which occur in the blanket and point to the possibility of producing electrical energy for external consumers; 4) demonstration reactors, i. e., prototypes of thermonuclear power plants, demonstrating the economic feasibility of constructing competitive thermonuclear power plants. The participants in the meeting focussed their attention on the projects mentioned in points 2 and 3. Information papers about the state of development were presented on four projects from point 1; demonstration reactors (point 4) have been sufficiently developed.

At the plenary sessions 24 papers were devoted to plans for, and individual engineering and technological aspects of, reactors of the tokamak type, nine papers dealt with reactors with inertial confinement, four papers to reactors based on open traps, and six, to economic aspects and planning. The rest of the papers concerned engineering-physical problems and reactors of other types.

Following the U.S.A. and the Soviet Union, Japan and West European countries have started drawing up programs for the construction of thermonuclear power stations based on a tokamak. The Japanese program considers two possibilities of attaining the goal: quickly taking the first step on the basis of present-day technology without the use of superconductors or the simultaneous solution of physical and engineering problems (with the use of superconductors), but at a slower pace. The European program envisages parallel construction of facilities satisfying the requirements of points 2 and 3, using the same engineering designs in them wherever possible. Work on both programs is far from being completed.

Tokamaks are the most promising and have been developed with the greatest depth. The plasma parameters obtained in them have drawn considerably closer to thermonuclear parameters and optimization of the projects is proceeding along the lines of bringing the characteristics of future reactors close to those attained on existing facilities. Thus, the ratio of the process duration required for the reactor to that obtained experimentally in 1974 was reduced from 10,000 to 60. The ion temperature and the energy confinement parameter $n\tau$ is only one order of magnitude smaller than that required for the reactor.

Plans for tokamak thermonuclear reactors are being improved: Their dimensions are being reduced and their construction simplified. This can be traced most clearly in the work of the University of Wisconsin, Argonne National Laboratory, and Princeton Laboratory. The plans they presented contain no diverter and the control windings have been moved outside. The power cycle has been reduced to ~60 sec and the geometric dimensions have been reduced by increasing the toroidal magnetic field. It was noted that simplification of the design could make the tokamak competitive.

As the design studies are gone into more deeply, economic calculations begin to take up more space. Several papers were presented in computational codes which make it possible to solve the problem of obtaining the required plasma parameters at minimum cost. It should be noted that although all the codes yield accurate results, with such a method of calculation it has been possible to optimize the parameters and quite accurately predict the costs of TFTR.

The meeting had five sections: tokamak thermonuclear reactors, demonstration and commercial tokamak power reactors, thermonuclear reactors with magnetic confinement, reactors with inertial confinement, and the role and place of thermonuclear reactors.

Experimental data on many parameters permit them to be extrapolated quite reliably to reactor values. In this case acceptable results are obtained even for pessimistic versions of the prediction. Accurate design

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of power reactors requires further development of plasma physics; explanation of the dependence of the energy lifetime on the ion temperature; a study of the plasma-wall interaction and methods of protecting the plasma from impurities; a full-scale experiment on auxiliary plasma heating; study of how nuclear reaction products, especially α particles, behave in plasma.

The section on the role and place of thermonuclear reactors discussed the problems and prospects of both pure and hybrid thermonuclear reactors. Particular attention was devoted to safety, economic and social problems, the existence of fuel and material resources, and the economic characteristics of research programs.

The opinion that pure thermonuclear reactors are safer than other nuclear facilities found received further confirmation. The ecological aspects have been worked out much better.

Considerable success has been achieved in understanding the place and prospects of the hybrid thermonuclear reactor. The safety of such reactors differs little from the analogous problems of fission reactors. Hybrid reactors can be built upon the basis of tokamaks, systems with inertial confinement, and apparently on the basis of mirror traps. An appropriate program is thus far being drawn up only in the USSR.

The meeting of experts noted the desirability of: drawing up a program providing for a considerable fraction (e.g., 10%) of the world power requirements to be covered by thermonuclear reactors by the beginning of the 21st century; constructing facilities for technological tests; developing materials with an operating life of more than $10 \text{ MW} \cdot \text{yr}/\text{m}^2$; increasing the energy-intensity of reactor elements; and developing methods of replacing the first wall and the blanket. Note was taken of the need to accelerate the development of facilities of later generations so that ready facilities could be entered into the world energy balance by the beginning of the 21st century when the potential demand may exceed the power of the energy sources.

SEVENTH INTERNATIONAL CONFERENCE ON ATOMIC COLLISIONS IN SOLIDS

V. M. Chicherov

The conference was held at Moscow State University in September 1977. About 80 out of the 240 papers presented at the conference dealt with the interaction of plasma with the first wall of a thermonuclear reactor. In review papers M. Kaminsky (U.S.A.) and R. Berisch and A. Scherzer (Federal Republic of Germany) discussed the results of research done in various laboratories.

At the present time further progress in the most advanced area of thermonuclear research, tokamaks, depends on whether it will be possible to decrease the interaction of the plasma with the wall of the device. The problem includes protection of the plasma from the effect of the wall (reduction of the number of impurity atoms which cool the plasma) and protection of the wall from the destructive effect of the plasma (extension of the wall lifetime).

In devices of the present generation the most important thing is to reduce the flow of impurity atoms from the wall into the plasma. This flow is the result of elementary processes which in themselves have been studied for quite some time now. The problem now is to combine the research on elementary processes done in monokinetic beams of particles with research done in plasma devices. To do this it is necessary to have detailed data about the fluxes of particles bombarding the chamber walls in tokamaks.

In their review paper R. Berisch and A. Scherzer considered the conditions at the boundary of plasma in a toroidal device and gave the result of measurements of the fluxes of energy, hydrogen atoms, and impurity atoms in the shadow of the diaphragm in the DITE and TFR devices. It is noted that the measurements are difficult to interpret and the estimates obtained from them for the fluxes (10^{15} - 10^{16} atoms/cm²·sec for hydrogen and 10^{11} - 10^{13} atoms/cm²·sec for impurities) are quite rough. The fluxes of particles and their energy distributions depend on the conditions for the discharge.

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Approximately such fluxes of hydrogen ions on the wall are expected in future reactors. The energy of the particles will probably lie in the range from 1 eV to 100 keV. All possible angles of incidence of particles on the surface will be realized. Some fraction of the particles will be reflected continuously from the surface and will return into the plasma whereas the remainder will be captured by the material of the wall.

When plasma interacts with the wall the following occur: 1) desorption of atoms of the surface layers by ions, electrons, and electromagnetic radiation; 2) sputtering (physical and chemical); 3) back scattering of ions; 4) capture of ions by surfaces and reemission of gas atoms; 5) surface changes due to prolonged ion bombardment (e.g., blistering); 6) evaporation and breakdown of material as the result of surface overheating.

Desorption data for the greatest part concern electron and photon desorption from well-prepared, smooth surfaces, and very few studies deal with ion desorption. Back scattering of ions is studied by calculations and good agreement with experiment in the energy range 1-10 keV gives rise to confidence in the calculations in the region of 100 keV where measurements are very difficult to make. In the case of scattering of hydrogen ions by various metals the experimental values of the particle reflection coefficient and the energy reflection coefficient are about 10% with an incident particle energy of about 1 keV.

Insufficient reliable data are available on sputtering by hydrogen ions and there are no data at all on sputtering by tritium. The data that are available concern pure metals above all. It seems more promising, however, to search for a material with a low sputtering coefficient among compound substances: borides, carbides, and nitrides of light elements of their alloys. Moreover, the search is hindered by the fact that the angular and energy distribution of the bombarding particles from the plasma are now known. Meanwhile, the maxima of the calculated curves for the sputtering coefficient for some compound materials lie at different energies of bombarding ions.

Recent measurements on sputtering by neutrons yielded a coefficient of less than 10^{-4} atoms/neutron. "Small pits" will not be formed under irradiation of a polished annealed surface. In such a surface the stresses in the surface layer were removed and the microcracks eliminated.

Chemical sputtering is a quite strong process and poses great danger to low-Z materials: graphite, carbides, and insulators which have a high reactivity with hydrogen and oxygen ions. Such sputtering is temperature-sensitive as is any mechanism including chemical reactions. For example, sputtering of pyrolytic graphite by hydrogen ions is maximum at 400-800°C ($S \sim 0.08$ atom/ion for 2-keV H^+ ions). It may be possible to reduce surface erosion due to chemical sputtering by choosing and maintaining an optimal surface temperature. It is difficult to say at this time, however, to what extent it will be possible to utilize this effect.

Intensive studies are being conducted on methods of reducing surface erosion by blistering. For a long time, blistering was considered to be a serious mechanism of erosion in future reactors since it was assumed that this process can be repeated many times. These ideas were based on experiments in which irradiation was carried out with monoenergetic ions. It was recently established, however, that blistering is effectively suppressed when metals are irradiated with particles with a broad energy spectrum which apparently will be the case in the reactors. The suppression is due to the uniform distribution of vacancies over the thickness of the irradiated material and the formation of a porous structure allowing gas to escape from the depth, thus preventing further blistering. The emergence of a porous structure and the absence of blistering were also observed during irradiation of metals at a quite high temperature (up to 0.4-0.5 of the fusion point). It is not clear, however, whether such a high wall temperature will be maintained in reactors. Blistering is reduced appreciably in materials with small grains (and dispersed second phase), such as sintered beryllium and aluminum powders.

Hydrogen and impurity atoms and ions bombarding the first wall during discharge in turn causes desorption and sputtering of adsorbed layers and the metal of the wall, reproducing and increasing the quantity of impurities in the plasma. At the present stage the impurity concentration in the plasma is much higher than in the gas filling the chamber prior to the discharge. To create the conditions necessary for a self-sustaining reaction to be initiated, the impurity concentration should be reduced many times.

Contaminating oxygen and carbon atoms apparently enter the plasma as the result of desorption of H_2O , CO, CO_2 , and hydrocarbons from the chamber walls. It is, therefore, natural to try to have a chamber with as clear walls as possible prior to the discharge. A promising method of purification from impurities under consideration is that of irradiating the metal surface with streams of atomic hydrogen and forming volatile compounds which can then be pumped off. This method is implemented to one extent or another in plasma devices during conditioning discharges in hydrogen with low parameters: $T_e \sim 3$ eV, $n_e \sim 10^{12}$ cm^{-3} , $I_{dis} \sim 1-10$ kA. This method is used successfully to cleanse chambers of oxygen in the DITE and Alcator devices. It was

discovered, however, that as the oxygen concentration in the plasma decreases, the concentration of metallic impurities increases and this impairs the plasma parameters even more. The increase in the flux of metal atoms in the plasma can be explained in part by sputtering of the chamber wall stripped of its protective coating of foreign atoms. Research thus is being done on effects which may result from embrittlement of the surface of metals irradiated with streams of atomic hydrogen. It is now known, e. g., that embrittlement of metals in hydrogen may cause scaling and ejection of considerable numbers of very small metal particles. This example shows that the problem of impurities may become even more difficult when the wall becomes cleaner.

As before, processes which occur when various forms of particles and radiation act upon a metal surface at the same time, e.g., during simultaneous bombardment of the surface with hydrogen ions and heavier ions, remain unstudied.

The proceedings of the conference, which are to be published, will undoubtedly be of interest to specialists working on controlled thermonuclear fusion.

FIRST INTERNATIONAL SEMINAR ON USE OF PROTON BEAMS IN RADIATION THERAPY

The seminar was organized by the State Committee of the Council of Ministers of the USSR on Atomic Energy (GKAÉ) and the Academy of Medical Sciences of the USSR (AMN SSSR) and was held in the Oncological Scientific Center of the AMN SSSR (Moscow) in December 1977.

Physicists and clinicians have been working for more than 20 years now on the use of proton beams in medicine. By virtue of the characteristic of the interaction with matter, proton beams make it possible to form in the patient's body clearly delineated dose fields with a prescribed shape and a high boundary gradient. This opens up a unique capability for striking targets (tumors) situated in any part of the patient's body without damaging a number of nearby critical organs and structures or the organism as a whole. A great deal of radiobiological experience has been accumulated throughout the world to date from the use of protons and from the radiation treatment of more than 2500 patients. Research and clinical work in the Soviet Union are being done on the basis of the accelerators of the Joint Institute for Nuclear Research (JINR), the Institute of Theoretical and Experimental Physics (ITEF), and the B. P. Konstantinov Leningrad Institute of Nuclear Physics, Academy of Sciences of the USSR (LIYaF).

The use of proton beams in clinical diagnostics, in addition to radiation therapy, is extremely promising. Short-lived radionuclides (^{123}I , ^{11}C , ^{15}O , etc.) produced in proton accelerators can be used effectively for this purpose. In many cases, proton beams reveal fine topographical-anatomical structural differences in organs and soft tissue which are not identified by x-ray examination (proton radiography).

For the first time in world practice, extensive information on all of these topics was presented at the seminar. A. I. Ruderman spoke of the experience from proton therapy in the USSR. A review of the scientific and technical aspects of the application of protons in therapy was presented by B. Larrson (Sweden). In addition to Soviet clinicians (E. I. Minakova et al., B. A. Konnov et al., E. E. Marienbakh et al., V. N. Kiseleva et al., G. D. Zarubel et al., B. V. Astrakhan), who reported on approaches to the effect of radiation on variously located tumors, clinical experience with the use of proton beams was described by American researchers (R. Kelberg, H. Suit et al., J. Castro et al.).

Some papers were devoted to proton radiography (I. P. Kalshnikov et al., USSR, R. Martin, U.S.A.; J. Saudinoce, France) and the production of radionuclides.

The distinctive characteristics of proton beams and the accelerators that generate them require fundamentally new approaches to physicotchnical provision for their clinical use. The cardinal problems in this area are those of shaping and measuring dose fields and creating special accelerators and special-purpose medical equipment (radiation stands). In addition to the review by M. F. Lomanov, 12 papers were presented by Soviet

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and foreign scientists on the first topic. An interesting paper on the principles underlying the choice of parameters and type of a special method accelerator was presented by L. L. Gol'din.

A separate section of the seminar was devoted to the radiobiological aspects of the application of protons (papers by L. Reves, Sweden; S. P. Yarmonenko, USSR; M. Raju, U.S.A., etc.). Interesting microdosimetric approaches to the radiobiological problems were presented in a review paper by V. I. Ivanov.

The successes achieved in the application of proton beams in medicine confirm the need to extend the use of this form of radiation in clinical practice. It is not surprising, therefore, that part of the papers dealt with the construction of special-purpose complexes on the basis of existing accelerators in the USSR, the U.S.A., and Sweden (V. P. Dzheleпов et al., I. V. Chivolo et al., B. A. Konnov et al., B. Larrson et al., C. Liman et al.) and construction of a complex with a special medical accelerator in the USSR (I. V. Chivolo et al.).

The seminar was the first representative forum devoted to this subject, was extremely interesting, and enabled scientists of various countries to share the experience gained in research in the most humane area of modern applied physics.

The proceedings of the seminar are to be published by Atomizdat in 1978.

IAEA MEETING ON THE USE OF PHYSICAL STANDARDS

P. I. Fedotov

At a meeting in Vienna, Austria, in August 1977, IAEA experts discussed a report and recommendation on the use of physical standards during inspections and measurements of nuclear materials. They discussed the physical standards employed in nondestructive methods of analysis by IAEA inspectorates and operators of enterprises in order to determine the quantity of materials. The meeting was attended by experts and observers from Great Britain, Belgium, Bulgaria, the Netherlands, the Federal Republic of Germany, France, the USSR, the U.S.A., and Euratom.

The report prepared consists of four sections. The first considers the types of nuclear materials inspected and the nondestructive methods of analysis appropriate for them and discusses the criteria which determine the priority in the preparation of standards. Tables have been drawn up with data on the expected error of measurement for various nuclear materials and methods of analysis. The diversity of materials analyzed and methods of analysis require a large number of physical standards to attain the necessary accuracy of measurement. In view of the high costs and difficulties owing to transportation, preparation and storage, criteria are presented for determining the priority in the preparation of standards: 1) the strategic importance, number and frequency of inspections of the nuclear material; 2) the maximum indeterminacy in striking the nuclear balance, this being due to error; 3) increased accuracy of measurement which can be attained by using a new standard; 4) the reduction which can be attained in the time and cost of inspection by application of the standard; 5) the universality of the standard. On the basis of these criteria, data on the specific properties of the nuclear materials and the character of the production IAEA will establish the priority in the preparation and provision of physical standards.

The second section is devoted to the preparation and certification of primary, secondary, and working standards. The standards should encompass a large area of the isotopic and chemical compositions of materials widely used in the nuclear fuel cycle (UF_6 , metallic uranium or plutonium, mixtures of uranium and plutonium oxides, etc.). It is preferable, in the opinion of the experts, that the efforts of enterprises, IAEA member-states, and the IAEA in preparing secondary and working standards should be combined. The enterprise provides material of appropriate quality and prepares a standard on the basis of this material, the state certifies the standard and issues a certificate, and the IAEA measures and verifies the characteristics of the standard.

The third section discusses the goals and strategy of the use of standards, factors which affect the strategy, and the use of the standards in concrete situations. The principal factors determining the strategy of the

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use of physical standards are: 1) the necessary IAEA staff participating in the work on the given problem; 2) the cost of preparation, storage, and transportation of the standard; 3) transportation; 4) verification and preservation of its authenticity; 5) the effect of the use of standards on the degree of reliability in the results of measurements. Depending on the situation, the effect of each factor on the choice of strategy will be different. In the opinion of the experts, however, the most applicable solution consists of a small number of primary or secondary standards being stored in the IAEA whereas the inspecting enterprises would have standards which would be verified with IAEA standards or calibrated in IAEA apparatus.

The fourth section considers methods of calibrating apparatus as well as reducing errors of measurement stemming from the difference between sample and standard. To reduce these errors, in the opinion of the experts, it is advisable to create apparatus with a low sensitivity to parameters (differences in chemical composition, dimensions, matrices, etc.) which are not a measured quantity or apparatus which automatically makes correction dependent upon the difference. To date considerable experience has been acquired in the development of such apparatus. Built-in or pocket microcomputers are used to process the results of measurements in inspection practice. The need for higher speed and accuracy in carrying out inspections necessitates the use of available miniature computing technique as well as the development of special-purpose microcomputers.

ALL-UNION SEMINAR ON THE PROCESSING OF PHYSICAL INFORMATION

O. P. Fedotov

The seminar, the second, was held in Erevan in September 1977, with the participation of representatives of 20 institutes of the Soviet Union, the Joint Institute for Nuclear Research, CERN, and some institutes of Hungary, the German Democratic Republic, Great Britain, and Italy. The subject matter of the seminar covered practically all of the principal aspects of automation of scientific research.

Considerable attention was paid to the introduction of computational networks in experimental physics, system software and specialized programming languages, and methods of calculation and optimization of measuring and computing complexes. The papers reflected the modernization of computers, the provision of computers with a large memory for accumulation of experimental data, and standardized interfaces for peripherals. Interesting data about these subjects were presented in papers by N. N. Govorun et al. (JINR), G. Davis (Gt. Britain), B. Talyer (CERN), B. M. Bobenko et al. (Institute of Theoretical and Experimental Physics ITEF), A. I. Bagin et al. (Radiotechnical Institute of the Academy of Sciences of the USSR), S. Z. Abelyan et al. (Erevan Physics Institute), and others. A separate session was devoted to the representation of information and active man-computer interaction. Papers by A. V. Ekimov (Institute of High-Energy Physics IFVE), N. A. Abramov et al. (Radiotechnical Institute), A. S. Grui et al. (JINR), and V. B. Anikeev et al. (IFVE) pointed out the broad capabilities of graphical and textual displays, interesting designs of displays based on a color TV, cathode-ray storage tubes, etc. All of this work is especially important at the present time in view of the need to increase the capacity of computing technique by optimizing computing systems and developing systems for collective use.

Other topics which were also taken up extensively in papers included the development of standardized program-controlled electronics and its use to create various automated systems for data acquisition and control with the aid of computers. Most experimental facilities today employ standardized electronics built according to CAMAC logic and electrical standards. Papers by workers of IFVE, JINR, Institute of Nuclear Physics of the Siberian Branch of the Academy of Sciences of the USSR (IYaF SO AN SSSR), ITEF, etc., presented concrete systems for counting information from wire chambers, complexes of apparatus for automation of experiments, including automation of accelerator monitoring and control. Programmed control with CAMAC systems was also considered. Some papers were devoted to a new line of development in CAMAC systems which has recently come into being and has gained widespread recognition. This concerns the development of "intellectual" CAMAC and electronic systems with autonomous "self-control" based on the application of microprocessors (papers by S. Szamori, Hungary; E. M. Gleibman et al., JINR; etc.).

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Considerable attention at the seminar was devoted to the development of measuring systems and new methods of measuring filmed information from a physics experiment. It was with great interest that the participants heard papers on the construction of new scanners based on cathode-ray tubes (JINR), on a system for the control of laser beam intensity (Moscow Engineering Physics Institute MIFI), on a system of automated stereoprojectors (ITEF), on new units and modules of measuring apparatus (JINR), on a method of calibrating automatic measuring machines (IFVE), etc. Notwithstanding the considerable advances made in the construction of measuring systems, these topics continue to be of interest to specialists owing to the continual emergence of new problems in the processing of film data.

Many papers at the seminar were devoted to the mathematical treatment of measured information and especially to the automatic data identification and filtration. There were reports on programs for on-line filtration in an HPD (JINR), filtration on a small computer in a system of PSP-2 automata (ITEF), off-line filtration in a Spiral Reader system (IFVE), on the method of analysis of the results of data processing in an interactive mode (JINR), etc.

Considerable interest was aroused by a communication from the ITEF and CNAF (Italy) on the development of means for the high-speed realization of processing algorithms by apparatus and the creation, on this basis, of a system for automatic trajectory-data recognition in a "minimum reading" mode from the human side. Instrumental methods of recognition and filtration makes it possible for processing algorithms to be realized tens and hundreds of times more quickly than in present-day computers.

A distinctive feature of the seminar was the fact that it paid particular attention to methods and means of automation developed in experimental physics, in other areas of scientific research, and for applied purposes. Two decades of development of automation of physics experiments have yielded a wealth of experience in creating processing methods and systems of unique accuracy and speed; many of them have proved to be extremely effective for application in other areas of science and technology. It was noted that these methods and means have been applied in medicobiological research (papers by Erevan Physics Institute and IFVE), in processing flight data (MIFI), processing of stereo photographs and many other forms of half-tone graphical information in geodesy, astronomy, etc. Measuring and computing technique has come into widespread use for the production and diagnostics of electronic apparatus and other applied purposes.

The seminar took great interest in the review papers presented (R. Pose, German Democratic Republic; D. Wiscott, CERN; V. M. Kotov et al., JINR; V. L. Mamaev et al., MIFI; etc.). The seminar demonstrated the substantially increased level of research and development in the realm of automation of scientific experiments, more technical equipment in Soviet institutes, and considerable successes in the application of methods and means used in the automation of physics experiments to other research and to the solution of applied problems.

The seminar was useful and furthered the exchange of information and coordination of efforts by various institutes in work on the automation of scientific experiment. The considerable organizational work done by workers of the Erevan Physics Institute contributed to a substantial degree to the success of the seminar.

The seminar proceedings will be published in 1978.

INTERNATIONAL CONGRESS ON CONCENTRATION OF USEFUL MINERALS

I. P. Kondakov

The Twelfth Congress, with the participation of 550 scientists from 41 countries, was held in Sao Paulo, Brazil, in August-September 1977. At 10 plenary sessions the Congress discussed 47 papers. The technology for processing raw materials difficult to concentrate was considered at three seminars.

The theory and practice of the process of preparing crude ore and concentrating it by magnetic, flotation, and other methods were reflected at the Congress and it was noted that there was a trend toward increasing the use of hydrometallurgical methods of ore dressing.

In the domain of ore preparation attention should be drawn to a communication on a unique method of automatic monitoring of wet grinding on the basis of measurements of the viscosity of the pulp (Australia), on the effect of chemical additives on grinding (U.S.A.), and on new approach by Italian specialists to screening which makes it possible to determine the principal kinetic parameters of the process for designing equipment or predicting the results of the process.

An interesting cylindrical apparatus may find application in the atomic industry for separating fine classes of complex phosphate ores in heavy suspensions. In the opinion of the U.S. manufacturer, this apparatus has an advantage in comparison with a similar process in hydrocyclones.

A laser photometric separator for ore sorting (U. S. A.), in operation in industry, was demonstrated during one paper and at the exhibition. Elements of its construction could be used to improve domestic radio-metric separators.

Papers which described the application of magnetohydrostatic separation are worthy of mention. The most interesting of the new designs of separators with an intense magnetic field is a polygradient separator of the carousel type with a throughput of up to 400 tons/h with a higher field strength than in Jones separators (a joint U.S.-Swedish development).

In the area of flotation the Congress devoted much attention to the study of the physicochemical processes on the surface of minerals and the structure of flotation reagents on the basis of thermodynamic, electrochemical, and other modern methods of research employing computers. Some papers dealt with improvements in selective flotation processes by the addition of reagents (Sweden) as well as by vibroacoustical treatment (Bulgaria) and discussed the results obtained in the U.S.A. and Canada by introducing this process for hematite slurries. Considerable interest was awakened by a paper on the effective introduction of flotation machines of the OK-16 type with a chamber volume of up to 16 m³ in ore-dressing plants in Finland.

A considerable proportion of the papers was devoted to combined ore dressing flowsheets including leaching, chemical and electrochemical deposition, sorption, and extraction, in addition to the traditional pyrometallurgical methods. Some papers considered heap and underground leaching of oxidized copper, uranium, and other types of ore which are abundant in the U.S.A., Canada, Mexico, and Latin American countries.

West German scientists reported the successful combination of grinding, leaching, and deposition of copper in a vibrating mill with subsequent flotation which increases the copper extraction from the ore in comparison with the usual multistage flowsheet.

Of the papers on process automation and control mention should be made of the introduction of the Proskon-103 system with a Courier-300 x-ray analyzer for controlling flotation in ore-dressing plants (Finland), the analysis, simulation, and optimization of large capacity mills, as well as grinding and flotation (Australia) and the control of iron-ore-dressing plants in Brazil. Some papers told of the achievements of Swedish, Finnish, and British companies in constructing and applying monitoring control equipment in iron-ore dressing and pelletizing: measurement of the moisture content from the reflectivity, determination of the iron content in ores

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and concentrates by radioactive methods, and a capacitive method of continuously determining the hematite content and an induction method of measuring the magnetite content on a conveyor, both methods being applicable for other forms of raw material.

In the area of design, some interest was aroused by a paper on the development of large conveyor systems in Brazil for the transportation of ore over distances of several kilometers at the rate of up to 16,000 tons/h per conveyor (belt width 1500-2200 mm, speed of travel 3-4 m/sec) and a paper on the "radial" (monoblock) principle of composing structures and construction of ore-dressing plants ensuring two to three times the throughput achieved with a "linear" arrangement while providing an enhanced coefficient of plant utilization.

Some papers at the Congress were devoted to environmental protection. A noteworthy method was presented for purification eliminating chromium and cyanide from effluents by a settling-flotation technique and for processing the waste from hydrometallurgical production of zinc; by this method valuable incidental components are extracted and contamination of the atmospheric and aqueous environment is prevented (Japan). Interesting studies were made on the development of flowsheets for a closed circulating water supply for ore-dressing plants (Italy, Rumania, etc.). Some of these measures are being studied with a view to their application in the atomic industry.

Certain attention was paid by the Congress organizers and delegates to a seminar on dressing of phosphate raw material which frequently constitutes uranium-phosphorus ores. Papers were presented on the study of the types of ore, their reserves, location, and geology, on the mineralogical nature of the samples tested, as well as on studies on ore dressing. Samples of varying composition of low-grade phosphate raw material were used to consider traditional methods of dressing (grinding, roughing, washing off, classification, deslurrying, and separation in heavy media) and new methods, including flotation, calculation with washing off with water and leaching reagents, and magnetic and electric separation in various combinations.

During the visit to Brazilian enterprises, the Soviet delegation became acquainted with the approach to the development of deposits and to design. All the plants are intended to process considerable raw material reserves of quite a high grade and have a high throughput: 5-46 million tons/yr (iron-ore plants), 3-12 million tons/yr (phosphate plants), and 300,000 tons/yr (niobium plants).

Noteworthy requirements are imposed on the design of reprocessing plants. Thus, the designing is preceded by a careful study of the material composition of the ore in the deposits, determination of a rotational technology for processing the raw material, and development of a technological flowsheet (as a rule, protected by patent) in two or three years for a pilot plant, and testing of commercial specimens of new equipment. The integrated processing of raw materials is ensured by the existence of developed technology for the extraction of incidental components; otherwise, the design, construction and commissioning of an enterprise are envisaged alternately with storage of intermediate products or reservation of ore containing incidental components.

An invariable condition is that the processing plants be as close as possible to the deposits, including the organization of preliminary crushing at the edge of the open-cut mine with subsequent transportation by conveyor to the main site of the dressing plant. An obligatory element in the ore-dressing flowsheet is that of laminar blending of crushed phosphate ore (after preliminary crushing or before grinding) in open or closed heaps with a 3-4 day capacity, formed by layer-by-layer pulling by means of a system of conveyor belts and a distributing bogie. Ore blending is all the more important for complex uranium raw material. The ore is taken away by machines made by Robins (U.S.A.) or Salzburger (Federal Republic of Germany). The system of iron-ore blending is even more complex. It is important when designing to concentrate the main technological operations as much as possible and to place them in a monoblock, to organize a system of total intraplant water cycle by dehydrating the final products in large thickeners with smaller quantities of thickened waste into a settling pond with an earth dike, etc.

The proceedings of the Congress are of considerable interest for specialists of the atomic industry who can obtain valuable information about the most recent achievements on other countries in the realm of the engineering and technology of ore dressing which has much in common with the initial stage of uranium ore treatment.

The Congress proceedings (in Russian) as well as a set of catalogs of the exhibition can be studied at the Institute of Information on Nonferrous Metals (Tsvetmetinformatsiya), at 101 Prospekt Mira, Moscow.

MEETING OF IAEA CONSULTANTS ON THE CHOICE
OF SITES FOR BURIAL OF RADIOACTIVE WASTES

M. K. Pimenov

As is well known, early in 1977 the International Atomic Energy Agency drew up a long-term (8-10 years) comprehensive program for the coordination of work by interested member-states of the Agency on the burial of radioactive waste in geological formations in the solid, liquid, and gaseous state. On the basis of the experience acquired in the world in storing and burying radioactive waste in geological formations as well as on the basis of national programs elaborated, the program is intended to provide a number of handbooks and methodological instructions which would regulate safe burial.

The task of the meeting, which was held in November-December 1977, in Vienna, Austria, was to map out further directions and time for work on drawing up handbooks for determining the geological, geophysical, hydrogeological, geomechanical, and physicochemical studies necessary to establish the suitability of a site for burial of radioactive waste in continental rock. This refers to burial of high-activity waste, containing long-lived isotopes, in loose rock and in salt, clay, and crystalline formations.

In the course of the discussion on the national programs on the burial of radioactive waste it was ascertained that practically nowhere (apart from the USSR) is there special legislation regulating the burial of radioactive waste in geological formations. At the same time, however, the U.S.A., France, and the Federal Republic of Germany, e.g., do have legislation which spells out the conditions for the injection (pumping) of industrial effluents into deep-lying permeable horizons of the earth's crust. Most of the consultants voiced the opinion that problems of the injection of radioactive gaseous waste into deep porous horizons of the earth's crust are not of top priority and may be considered at a later date.

At the meeting the consultants adopted a final document which takes note of the urgency in preparing handbooks specifying the composition and methods of conducting investigations to establish the suitability of a site for burial of radioactive waste in continental rock and agreed that priority should be given to the preparation of three handbooks. The first concerns establishment of the suitability of salt structures, crystalline rock, and clay deposits for burials of high-activity waste, the second deals with the geological structures and storage strata for burial of liquid radioactive waste in deep-lying permeable horizons of the earth's crust. The third handbook should establish the suitability of structures for the burial of radioactive waste by employing the hydraulic fault of the stratum. All three handbooks will be ready in 1979.

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MEETING OF IAEA EXPERTS ON PROTECTION
OF POPULATION IN MAJOR
RADIATION ACCIDENT

Yu. V. Sivintsev and V. A. Klimanov

The meeting, which took place in Lisbon, Portugal, in October 1977, discussed the structure, contents, and text of a handbook on the preparation of measures for the event of a major radiation accident. Major is taken to mean an accident in a nuclear facility, e.g., in an atomic power plant, which results in the ejection of large quantities of radioactive substances beyond the limits of the forbidden zone and there is a danger of part of the population being affected. The new document deals not only with protection of the population, but rather more with a set of measures which can be undertaken outside the site of the nuclear facility in the event of a radiation accident, primarily in an atomic power plant. It was agreed that the IAEA plan of work will include the preparation of recommendations for the elaboration of measures to be taken on the site of the nuclear facility.

The handbook consists of six parts. The first part discusses methods of analyzing the scale of the radiation accident and the possible consequences for the population of the adjacent territory. It points out that these problems must be studied carefully in advance as applicable to local conditions, type of facility, and scale of the accident.

The document is based on the initial assumption that operation of an atomic power plant with a rate of release which does not grow above the maximum permissible (MPR) is radiation-safe. To the extent that it was noticed that a considerable (but brief) increase in the power surge above the MPR does not entail any significant increase in the radiation risk for the population, the integrated concentrations will be low. Accordingly, no additional measures outside the atomic power plant are needed. In cases when the rate of release exceeds the MPR for a long time and reaches the "investigation level" (according to the terminology of the latest recommendations of the International Commission on Radiological Protection ICRP), additional information must be obtained about the activity and isotopic composition of the release. This makes it possible to estimate the concentration of radionuclides in objects of the external environment and the intake of these radiators into the human organism. If this intake for the critical group of the population is below the maximum annual intake (MAI), as fixed by the ICRP and domestic radiation safety standards, then the only action necessary outside the atomic power plant site is for reserve groups of the external dosimetry service to carry out functions according to an expanded program for monitoring objects of the environment. If the results of radiometric and spectrometric monitoring in an atomic power plant indicate that the rate of release significantly surpassed the investigation level and the intakes exceeded the MAI, then the "intervention level" must be employed. Above these values the population of the district adjacent to the power plant may be injured and property and objects of the environment may suffer damage; measures may thus be required to protect them from radiation overexposure.

The second part considers possible protective measures such as moving the population to shelters, prophylactic application of stable iodine, evacuation of the population, deactivation of the people, food, and water, and the attendant risk and cost. The protective measures should be taken with due account of their risk, time after the accident, and only provided that the radiation rise is reduced significantly.

The document deepens and supplements the classification of measures into immediate (upon passage of a cloud of gaseous discharge), intermediate (days, weeks), and long-range (months, years).

The third part is devoted to a discussion of the concept of intervention level and consideration of various factors which affect the choice of level. It was decided that intervention level should mean a predetermined radiation dose which, if received by the population in the territory adjacent to the nuclear installation, would not require any measures.

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In addition to radiobiological data, in establishing numerical values for intervention levels one must take account of the social and economic conditions, types of nuclear facilities, characteristics of the environment, and other factors. The experts therefore deemed it is advisable to recommend the same values for all countries and supplemented the handbook with an appendix of examples of concrete values adopted in IAEA member-countries. However, bearing in mind that the intervention level differs considerably in the various countries and that unjustifiably low levels may hinder the development of nuclear power, the experts decided to take only the levels adopted in the USA, Gt. Britain, and the USSR for illustrative material in the appendix.

The fourth part of the handbook discusses the characteristics for prior planning of protective measures. This planning includes analysis of the potential danger of a radiation accident in a concrete nuclear installation from the point of view of possible consequences for the population and environment, demarcation of the most "dangerous" areas in the adjacent district, and estimation of possible resources which could be used to diminish the consequences of an accident.

The fifth part analyzes various restoring measures, recommended for the latter phase of the radiation accident when the situation in the nuclear facility itself has been partially normalized and there is no danger of any further release of radioactive products.

The last, sixth, part discusses the responsibility of various organizations and their cooperation in making decisions and eliminating the consequences of a radiation accident.

The IAEA plans to publish the handbooks in the first half of 1978.

IN THE INSTITUTES

METHODS OF HIGH-ENERGY PLASMA TECHNOLOGY

V. G. Padalka and V. T. Tolok

Fundamental research carried out by the Kharkov Physicotechnical Institute of the Academy of Sciences of the Ukrainian SSR (FTI AN USSR) on the physics of high-temperature plasma in connection with the problem of controlled thermonuclear fusion has resulted in the creation of an applied field, viz., high-energy plasma technology, for the purpose of solving many problems of present-day technology.

Metal-plasma accelerators based on a cathodic vacuum arc are extremely promising for application [1, 2]. The high current density in the cathode spots ($1-10 \text{ MA/cm}^2$) and the specific power released ($10-100 \text{ MW/cm}^2$) cause intensive evaporation of the cathode and generation of a plasma stream consisting of products of cathode erosion. The degree of ionization is quite high (up to 80-100% for the plasma of high-melting-point metals) and doubly charged ions with an energy of about 100 eV constitute the great majority. The effect of vacuum-arc generation of high-speed plasma streams permits an anomalously high ion current (up to 10 A or more, $\sim 10\%$ of the discharge current) to be obtained from these streams. The energy efficiency of plasma generation may reach 20-30% while the plasma condensate is deposited at a rate of up to $0.5 \mu\text{/min}$.

The process of deposition of plasma condensates, which has been dubbed CIB (condensation of material in a vacuum from a plasma stream under ion bombardment of plasma condensate), occurs in the Bulat apparatuses [3] (Fig. 1). The vacuum chamber of the apparatus is pumped down to a pressure $\sim 6.6 \cdot 10^{-4} \text{ N/m}^2$ by a diffusion pump with a pumping speed of 1000 liters/sec. The electric-arc plasma accelerator is powered from a rectifier with a steep volt-ampere characteristic (no-load voltage 60 V, operating current up to 300 A). The cathode may be made of any conducting material whereas the vacuum-chamber vessel is the anode. The product being worked has a negative potential (of several hundred volts to several kilovolts) applied to it to ensure extraction of ions from the plasma stream and imparting the necessary additional energy to them. To obtain chemical compounds (nitrides, carbides) an alloying gas is introduced into the chamber. Any cathode material

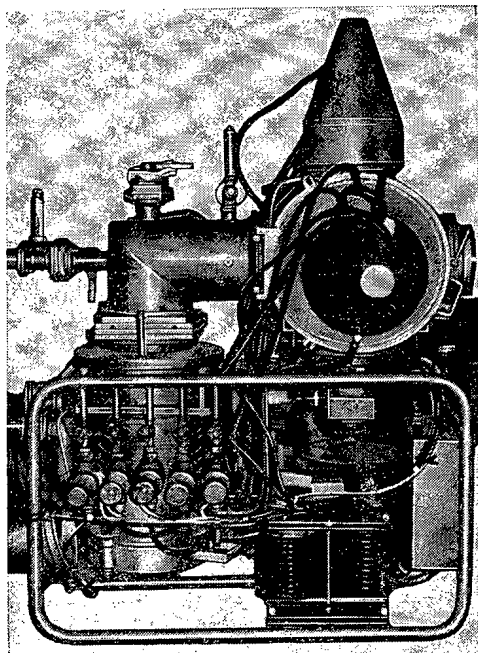


Fig. 1. The Bulat-3 apparatus.

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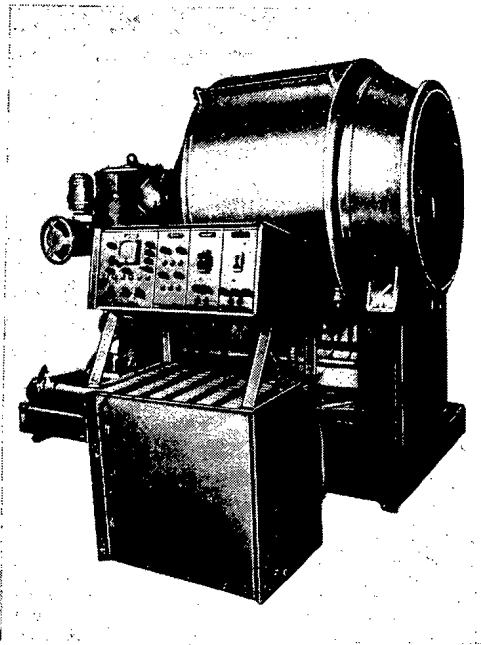


Fig. 2. AVED-40/800 electric-arc machine.

(tungsten, tantalum, molybdenum, niobium, graphite, and semiconductors) can be evaporated as a result of the high temperature of the cathode spot of the vacuum arc. The introduction of an alloying gas into the vacuum chamber and its ionization in the discharge permit plasma beams to be produced with a controlled content of various ionic components.

The ionic state of the substance near the surface of the sample is activated by the interaction reaction, thus making it possible to obtain coatings with excellent physicomechanical properties (extrahard, wear-resistant, etc.). Since they are formed directly from the ion beam, the coatings are subjected to ionic treatment over their entire thickness (in contrast to ionic alloying). The composition and energy of the ion beams formed from the plasma stream can be regulated to vary the phase composition, structure, and properties of the plasma coatings. The resultant coatings, consisting of the pure starting materials evaporated and their solid solutions, compounds, and heterogeneous alloys. Multilayered compositions have been produced with programmed variation of the content of various components.

Accelerated high-density ion beams obtained from a plasma stream allow the surface of the sample to be cleaned very efficiently by sputtering. This is a distinctive feature of the plasma stream method. Similar adhesion cannot be ensured by any other existing method of applying coatings.

One of the most important areas in which coatings obtained by vacuum condensation of plasma streams can be applied is that of producing wear-resistant layers of nitrides of high-melting-point metals (especially molybdenum and titanium). The phase composition of the coatings and the physicomechanical properties can be altered by varying the conditions of the experiment (pressure and composition of the alloying gases, ion current density and energy, substrate temperature). The materials obtained have high values of the determining parameters. In the case of coatings based on molybdenum nitride it was found that the crystal lattice had a high nitrogen content, that the lattice was highly distorted, and that the microhardness was up to 3500 kg/mm². The process makes it possible to achieve various modes of strengthening: solid-solution and dispersion strengthening, and strengthening by forming heterogeneous laminar structure [4].

Tools made of hard alloys and high-speed and tool steels were strengthened by the application of coatings based on titanium and molybdenum nitrides. Laboratory and industrial tests of the strengthened tools were carried out in the treatment of structural steels and alloys, cast irons, and nonferrous metals. It was established that strengthened by the CIB method increases the wear resistance of the tools two- to eight-fold, depending on the type of material treated, the conditions of the treatment, and the material of the coating.

The CIB method is being introduced into industry. Specialized production sections for strengthening cutting tools have been in operation in the Malyshev and GPZ-8 plants (Kharkov) since 1975 with Bulat-2 machines. Other enterprises in the country have been provided with Bulat-3 machines. The annual economic effectiveness of the introduction of Bulat-3 in a large engineering plant runs to about 100,000 rubles.

The conditions necessary for the formation of a diamond structure can be created by the condensation of a stream of carbon plasma on a substrate. Upon striking the substrate, the energetic ions produce a local pressure which reaches hundreds of kilobar and the thermal energy released locally may correspond to a temperature of several thousand degrees. If the heat is removed from these "hot" points by cooling the substrate, metastable crystal forms, including diamond forms, can be quenched to room temperature. For crystallization of a diamond structure to occur it is sufficient for the ions to have an energy of several tens of electron volts. Under optimal conditions of carbon plasma deposition the density is 2.9 g/cm^3 , the resistivity is $10^8 \Omega \cdot \text{cm}$, and the microhardness of the coating is $9.8 \cdot 10^{10} \text{ N/m}^2$. The coating is a two-phase material consisting of amorphous carbon and finely dispersed graphite, which have a low hardness, and a crystalline intermediate form of carbon with a high hardness [5].

Stationary erosion accelerators of metal plasma developed by the Kharkov Physicotechnical Institute are being used successfully in sorption titanium pumps as a means of regenerating the getter. In this case active gases are sorbed by titanium deposited on the inner surface of the pump chamber, inert gases are pumped out by an oil vapor pump with a pumping speed of about 1-2% of that of the nitrogen sorption pump.

The AVED-40/800 electric-arc machines (Fig. 2) built on this principle are used extensively to pump down technological vacuum equipment (vacuum metallurgy, electron-beam welding, and furnaces for annealing titanium alloys) [6].

The AVED-40/800 specifications:

Pumping speed, 10^4 liters/sec:	
nitrogen	4
hydrogen.....	8
air	3
Specific capacity, liter \cdot N/m ² \cdot sec:	
nitrogen	267
hydrogen.....	4
Service life in continuous operation before replacement of	
titanium (depending on gas load), h	300-3000
Maximum titanium consumption, mg/sec.....	10
Start-up pressure, N/m ²	1.3-13
Maximal residual pressure (with water cooling of condensation surface), N/m ²	$1.3 \cdot 10^{-5}$

The results of both the development and improvement of plasma technology and its use in industry demonstrate that the application of plasma streams opens up broad possibilities for the solution of scientific and industrial problems. It is quite justifiable to assume that this technology will become widespread in industry.

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A. P. Shirenko

RADIOISOTOPIC METHODS OF MEASURING ALTITUDES*

Reviewed by E. R. Kartashev

In aircraft (airplanes, helicopters, etc.) the altitude can be determined by radio altimeters which, however, do not provide sufficient accuracy in low-altitude measurements (below 40-50 m). At the same time, exact knowledge of low altitudes is particularly important in such important stages of the flight as landing (especially under difficult meteorological conditions and with poor visibility). Determination of high altitudes (tens of kilometers) is also an urgent problem.

A promising method of determining both low and high altitudes, according to many researchers, is a radiation method based on measurement of the degree of back scattering or absorption of the radiation of various radioactive nuclides. The problems of measurement of altitudes by the radiation method have not been sufficiently illuminated in the scientific and engineering literature. Hence the great interest in the book under review which considers the various aspects of this method on the basis of data from the literature and research done by the author.

The book goes very deeply into the theoretical foundations of the radiation method of measuring aircraft altitudes from a fraction of a meter to 100 km. Particular attention is paid to substantiation of the possibility of determining flight altitude by using γ radiation backscattered by the air and the underlying surface and absorption of β radiation by the air.

The author considers structural schemes, altitude characteristics of radiation altimeters, and the effect of various factors on the accuracy of their readings. He assesses the advantages and disadvantages of such altimeters, presents some recommendations on reduction of the effect of possible disturbances and on the choice of radiation sources and detectors most suitable for use in radiation altimeters. The results of researches carried out by the author are of considerable interest to specialists. The theoretical and experimental altitude characteristics, in particular, are in good agreement with allowance for the assumptions made for the calculations.

The book under review, however, is not free of shortcomings. Evidently, it was not the author's undertaking to develop an on-board radiation altimeter. The models used in the experimental investigations were made up of standard instruments manufactured in preceding years and the separate units were based on radio tubes. Accordingly, the detailed description of this apparatus as well as other instruments used in the experiments was unnecessary since it is of no assistance to the designer with modern microelectronic techniques at his disposal nor is it needed by the experimenter who most often has access to a set of other instruments. Such descriptions only clutter up the text and draw attention away from a book that is interesting on the whole.

*Atomizdat, Moscow (1977).

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