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SOVIET ATOMIC ENERGY

АТОМНАЯ ЭНЕРГИЯ
(ATOMNAYA ÉNERGIYA)

TRANSLATED FROM RUSSIAN

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SOVIET ATOMIC ENERGY

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A translation of *Atomnaya Énergiya*

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CONTENTS

CONTENTS

(continued)

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CONTENTS

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ARTICLES

NUCLEAR PHYSICS RESEARCH CENTERS IN THE USSR

V. V. Goncharov and V. F. Kuleshov

UDC 621 039

Fifty years have elapsed since the historic First All-Union Congress of the Soviets. In those 50 years of intensified labor and struggle on the part of all of the peoples of our multinational Soviet State in the name of the bright ideals of communism, science has made enormous progress in our country.

The foundations of science laid down during the first years of Soviet power and the constant support and attention given to science by the State, have contributed to the country's scientific progress, and have made it possible to solve such major and complex scientific and technical problems of our times as the conquest of atomic energy, space flight, building of modern communications networks, aviation transportation, computer technology, and so forth.

Naturally, at each stage in the history of the Soviet State the development of science has been indissolubly bound up with concrete problems in the social and economic transformation of our country, and with the building of a socialist society.

The Leninist national policy pursued by the CPSU [Communist Party of the Soviet Union] has aimed at raising the level of the national economy and of the culture of all nations and nationalities in the USSR. This has led, within a short historical period, to the overcoming of age-old backwardness suffered by many of the peoples, and to the flourishing of science in all of the republics. In close collaboration with one another, and with the fraternal aid of the Russian people, the republics of the Union emerged to a new, and hitherto unexperienced, level of development of science and culture.

The organization of Soviet science has undergone radical change within the past five decades. The various republics of the Union now all boast of their own academies of sciences, which are major scientific-research centers successfully contributing to the development of science and to national culture.

It must be stressed that the growth of science in the country took place not only "horizontally," i.e., throughout the nation, in all parts of the nation, but also "vertically," i.e., in the direction of raising the overall level of research on a worldwide scale, encompassing a steadily broader scope of scientific and technical problems, and making available technology of increasingly greater sophistication.

This trend becomes particularly conspicuous in the establishment and development, in the republics of the Union, of research in the field of nuclear science and nuclear engineering, in the field of the peaceful uses of atomic energy.

As we know, during the very first postwar years nuclear physics research and applications of this research in other areas of science, engineering and industry, began to develop on a large scale in our country. Scientists from many of the republics in the Union took part in the very first research programs and efforts based primarily on utilization of radioactive isotopes in the solution of important problems in chemistry, biology, geology, engineering, medicine, agriculture, and other fields. Highly qualified teams of research workers known not only for their own work and accomplishments in the field of fundamental nuclear physics, but also for their contributions to the elucidation of the nature of the atom, had been formed by that time in some of the republics. That was the case, for example, with the Khar'kov Physics and Engineering Institute [KhFTI]. Much experience had been accumulated in Armenia, where research on cosmic radiation had been in progress as early as the period nearing the end of the war. A major nuclear physics center, the Erevan Physics Institute, equipped with a 6 GeV electron accelerator later developed in Armenia.

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Work in the field of peaceful uses of atomic energy gradually strengthened ties between the scientists of the sister republics of the Union and the central scientific-research institutions, and with the scientific collectives of other republics in the Union. The close ties between scientists and such major centers of nuclear science and nuclear engineering as the I. V. Kurchatov Institute of Atomic Energy, the Institute of Theoretical and Experimental Physics, the V. G. Khlopin Radium Institute, the A. F. Ioffe Physics and Engineering Institute, the Institute of Physics of the USSR Academy of Sciences [FIAN], and many others, and trips to the sister republics of the Union on the part of famous nuclear scientists, contributed to raising the overall level of local research work, and to bringing into being national scientific cadres in that field.

The building of experimental water-cooled, water-moderated reactors designed for a broad range of research programs on low-energy nuclear physics, radiation material science, radiation chemistry and radiation biology, as well as for the production of radioactive isotopes to meet the needs of science and of the national economy of republics, was begun in response to an idea put forth by I. V. Kurchatov and under the supervision of Academician A. P. Aleksandrov at the Institute of Atomic Energy, and provided a powerful stimulus to the development of research in the sister republics of the Union.

Research reactors of that type were built not only in Moscow, Leningrad, and other cities of the RSFSR, but were also built in Georgia, Uzbekistan, Ukraine, Latvia, Belorussia, and Kazakhstan. Those reactors provided a focus and base for the scientific institutions being rebuilt or developed still further in those areas, and that helped bring about fundamentally new conditions favoring the development of nuclear physics in the sister republics of the Union.

I. V. Kurchatov played a major role in setting up many of the atomic research centers.

Some of the research centers in the republics now have at their disposition not only reactors, but also sophisticated particle accelerators, high-level γ-irradiation facilities, hot laboratories, and other unique research equipment. The contributions being made to science (pure and applied) by these new research centers is continually on the increase, and many of the research projects carried out by scientists at these centers have long since passed beyond the range of work centered on techniques and work of narrowly applied character, and are now counted among the most advanced achievements of world science.

The history of nuclear physics research in the Ukraine is the richest in events; development of research in the Ukraine got its start essentially in 1932, when the lithium nucleus was successfully split at the Khar'kov Physics and Engineering Institute. During the postwar period, Ukrainian scientists, in addition to their active participation in determining the nature of the atom, also made significant contributions to many fundamental branches of atomic physics, and to the utilization of the techniques and equipment of nuclear physics in allied branches of science and in the national economy in the Ukraine. Neutron generators, a 1 GeV linear electron accelerator, the U-120 cyclotron, and various other custom-designed nuclear facilities, were developed and built. The 10 MW VVR-M reactor of the Nuclear-Research Institute (IYaI) of the Academy of Sciences of the Ukrainian SSR was started up in Kiev in 1960. With the aid of those facilities, and their reactor, Ukrainian scientists succeeded in completing major research projects in the study of the spectroscopy of slow neutrons and interactions of fast neutrons, as well as work on interactions of protons, deuterons, and α-particles with nuclei. The neutron constants of various structural materials were determined, the nonmonotonic behavior of neutron cross sections, and the isotope effect in elastic scattering of protons, were detected.

Theoretical physicists of the Ukraine have to their credit some research accomplishments of major importance on the statistical model of the nucleus, the theory of nonaxial nuclei, and on the compound and dual model of elementary particles.

At the present time, work on construction of the U-240 reactor designed for comprehensive investigations of nuclear processes over a broad range of energies is nearing completion at IYaI.

The startup of the VVR-M reactor enabled Ukrainian scientists to develop extensive work on applications of nuclear techniques in allied branches of science and engineering, and in the national economy. Research work centered around the reactor facility allows great leeway for studies on applications of inelastic scattering of slow neutrons in the study of the dynamics of matter in the condensed state. A special multidetector arrangement, built in collaboration with the I. V. Kurchatov Institute of Atomic Energy (IAE), has been used to carry out an interesting program of research on inelastic scattering of slow neutrons in polyethylene. That research has yielded data of great importance for reactor physics and solid

state physics, shedding light on neutron thermalization processes in polyethylene and the generalized frequency spectrum of polyethylene. The facility has also been utilized for work on studies of phonon spectra in such elements as iron, copper, and nickel.

A higher level of sophistication in measurement techniques and procedures and improvements in the performance of the facility made it possible to measure dispersion curves in crystals, which is of great interest in solid-state physics research. Research on the radiation physics of solids and radiation materials science are accorded no less an important place in the research program. In particular, possibilities for varying the parameters of various semiconducting materials in a prespecified direction have been greatly expanded as a result of research on radiation effects on the properties of semiconducting materials and on the n–p structure of semiconducting materials. One characteristic trait of work on radiation materials is the profound combination of the study of the structure of radiation defects and research on the most important technological properties of structural materials used in reactor design. Attention is centered on the development of adequate procedures and investigation of the properties of structural materials directly while the irradiation process is underway.

Biological research has experienced extensive development at the reactor facility of the IYaI AN UkrSSR, particularly in the field of radiation microbiology. This research is directed toward the development of effective methods for raising the productivity of industrially important microorganisms.

In their study of the general patterns of variability of bacteria due to fast neutrons, Ukrainian scientists have succeeded in sectioning a series of mutants with enhanced biosynthetic activity by combining neutron irradiation and chemical mutagens. In 1970, a radiation-induced mutant of a strain producing proteolytic enzymes was introduced into a process of production of triacetate films, where the culture broth of that microorganism is used in regenerating the films in order to wash off the gelatin substrate and the photographic emulsion. Starting with 1961, research in the field of biological effects of fast neutrons has been successfully pursued at that reactor facility. The purpose of the research was to clarify changes occurring in animals after they have been exposed to fast neutrons at different dosage levels, with simultaneous sampling of the most effective preparations protecting the organism from neutron injury. As a result of those investigations, it was found that fast neutrons bring about long-term functional and organic injuries in exposed animals, which show up in six or more months, and light was also shed on the protective mechanism of DNA in combatting radiation injury.

Research on the physical and experimental parameters of the reactor needed in order to work on modernizing, perfecting, and fabricating new systems and components, occupies a prominent place in the Institute's work program. In-pile tests showed that the reactor output level could be stepped up from 10 to 15 MW without any design modifications. Over 1200 containers with different specimens were irradiated in the reactor's experimental channels, for the benefit of many organizations in the Ukraine and in other republics of the Soviet Union.

The first in a series of IRT-2000 pool-type nuclear research reactors developed in the USSR was the reactor at the Institute of Physics of the Academy of Sciences of the Georgian SSR, which went into operation in October 1959, at a site near Tbilisi. The distinguishing feature of the research carried out with that reactor was the way low-temperature physics problems, which have been tackled with great success at the Institute of Physics of the Academy of Sciences of the Georgian SSR, have been organically combined with nuclear physics problems, solid-state physics problems, and biophysics problems. At the present time, the Tbilisi nuclear reactor has been converted into a modern cryogenic laboratory equipped with powerful refrigerator and liquefying machines, and special mainline devices for low-temperature radiation experiments. The Institute of Physics of the Academy of Sciences of the Georgian SSR is the leading organization in the development of low-temperature materials science.

The methods of radiation low-temperature solid-state research developed at the Institute, and the low-temperature irradiation channels built in the reactor for irradiating specimens and in-channel facilities for performing remote measurements, enabled the Georgian physicists to achieve impressive results in their work. For example, investigations of changes in the physicomechanical properties of solids exposed to reactor irradiation have been carried out over a broad range of temperatures, directly in the beam and with some time elapsed after the irradiation, and the investigations have shown that changes in the mechanical properties of a solid are more likely a secondary effect of the irradiation associated with a clustering of point defects into complexes. Great emphasis is laid by scientists on the radiation physics of solids in work related to directed changes in the physicomechanical properties of crystals exposed to

radiation bombardment in the pile, in combination with other effects. In particular, some interesting experiments have been staged (jointly with the Physics and Engineering Institute [FTI AN UkrSSR]) on studies of the motion of dislocations through the stress field in a beam. A diffusion-dislocation mechanism of plastic flow of crystals has been put forth on the basis of the experimental findings.

Another, no less important, trend in the work of the Institute of Physics of the Academy of Sciences of the Georgian SSR is the development of highly sensitive techniques for neutron activation analysis in determinations of trace quantities of heavy elements in human blood and in components of the blood (erythrocytes, leucocytes, serum, etc.), in the cerebrospinal fluid, and in tissues subjected to in vivo biopsies.

In experiments carried out at the macromolecular level (proteins, nucleic acids), investigations of the trace-element composition in biologically active substances helped in detecting an essential difference between the composition of trace amounts of metals in DNA molecules, RNA molecules, and collagen molecules extracted from normal and tumorous tissues of animals.

Important results were obtained by Georgian physicists in the field of the design of radiation indium–gallium loops, as new and high-level sources of γ-radiation for research and industrial use.

The generation of radioactive nuclei in the working materials of radiation loops, and the related deformation of the neutron fields and of the neutron spectrum in the region where activity generators are positioned, have been investigated. Methods have been proposed for more complete utilization of neutrons in attaining higher power output levels in the irradiators of radiation loops. In joint work with the L. Ya. Karpov Scientific-Research Physicochemical Institute [NIFKhI] and GIREDMET Rare Metals Institute, the most promising γ-carrier for radiation loops: a liquid-metal indium–gallium alloy, was selected, studied and subjected to long-term tests. The processes taking place at the interface between the molten indium–gallium alloy and the structural material were investigated in detail, and materials that might be useful in the design of high-level radiation loops for industrial application were singled out.

Over the course of the past five years, plans have been finalized for the building of the USSR's full-scale industrial facility for production of wood and lumber products modified with the aid of radiation technology, on the basis of developmental work carried out at the nuclear-research center of the Institute of Physics of the Academy of Sciences of the Georgian SSR.

Work on intensification of heat transfer, with the aid of an artificial surface roughness, applied by means of a method specially developed by Georgian physicists, to the surface of fuel elements was carried out in the core of the Tbilisi reactor. That work made it possible to increase the heat loading of the reactor core by a factor of 2.5.

At the present time, the possibility of using the artificial roughness to advantage, in raising the heat loading of high-output water-cooled reactors generating power for nuclear electric power stations, is under study.

A VVR-S reactor was started up in Uzbekistan in 1959 at the Institute of Nuclear Physics of the Academy of Sciences of the Uzbek SSR, which had been commissioned only three years earlier. The rapid constitution of an experimental facility fully up to modern standards in the republic, has since expanded so much that at the present time the experimental capabilities of the republic include not only the reactor, but a U-150 type cyclotron, a high-level γ-facility with irradiators, and other nuclear physics equipment and arrangements, has enabled Uzbek scientists to perform applied and fundamental investigations on the physics of elementary particles, low-energy physics and intermediate-energy physics, radiation physics of solids, chemical dosimetry, radiochemistry, and activation analysis.

In 1971, the VVR-S reactor was redesigned with the assistance of the I. V. Kurchatov Institute of Atomic Energy (IAE), so that its power output could be raised from 2 to 10 MW, and its experimental capabilities expanded appreciably. A two-channel loop facility was installed as part of the reactor system.

Investigations on inelastic coherent interactions of high-energy particles with complex nuclei, spectroscopy of heavy nuclides, and investigations of nuclear reactions with protons on light nuclides with completely filled shells, research on the radiation physics of crystals employed in quantum electronics and laser optics, occupy a prominent place in the research program of the Institute. Techniques of activation analysis of rare and scattered elements in specimens of rock and ores, and practical implementation of those methods at industrial enterprises of Uzbekistan and other republics, are being developed at an intensive pace. A highly efficient and productive automated complex for determining contents of precious metals in ores has been built alongside the reactor.

In recent years, the Institute has been conducting extensive research work in collaboration with scientific institutions and with industrial enterprises of Uzbekistan and other republics. In particular, some important work is underway on utilizing radiation exposure to improve the performance characteristics of field logging cables, power transmission cables, and other cableware intended for service under high-temperature conditions. A technology for irradiating large-size cableware of different design, as well as special dosimetric systems useful for monitoring the radiation process and radiation quality, and also the uniformity of the dose field in the irradiator facility, has been worked out.

Starting with 1964, the Institute of Nuclear Physics of the Academy of Sciences of the Uzbek SSR [IYaF AN UzSSR] has been engaged in fruitful collaboration with the Tashkent hemp mill. During that time, radioisotope weight gages for hemp strips on rough-combing machines have been developed and introduced into production work, thereby making it possible to save on expensive raw materials and cut down overall production costs. Development of a system for automatic control of the linear weight of hemp strips for the machines is now nearing completion.

Radioisotope arrangements for the building materials industry of the republic have been devised in joint programs with NIIstroiproekt Design and Planning Institute. One of these arrangements is a radioisotope facility for monitoring the state of the lining in rotary kilns, and is now in use at the Bekabad Cement Combine. Preparations are now being made for putting this instrument into mass production. Radioisotope facilities for monitoring the dust load of off-gases, for monitoring the degree of decarburization of material in the kiln, and the degree of readiness of the lime.

Radioisotope clinker scales were developed and fabricated for production use at the Kuvasai cement combine.

The close of the past year marked one full decade of service of the IRT-2000 research reactor of the Institute of Physics of the Academy of Sciences of the Latvian SSR. The startup of the reactor was associated not only with a substantial expansion of work in the fields of nuclear spectroscopy and activation analysis, which had by that time become something of a tradition at the young institute, but also with the appearance of a research trend, new at that institute, centered on the radiation physics of solids. Results of comprehensive investigations of the formation of radiation defects, radioluminescence and radiolysis in alkali halide crystals, as well as radiation processes in ferrites and oxides of the transition elements, have achieved widespread scientific acknowledgement in our country. These investigations have shed light on the radioluminescence mechanism in activated alkali halide crystals, and have aided in establishing the energy microstructure of luminescence centers and in obtaining more refined information on the nature of the chemical bond in impurities in irradiated alkali halide compounds. As a result of studies on the mechanism of energy transfer in ionic crystals, a rapid process of energy migration between color centers was detected. The discovered phenomenon provided an impetus to search for new ways to use the detected effect in the design and development of high-speed, high-capacity optical memory components, in which the memory matrix is constituted by an appropriate single crystal. Investigations of radiation effects in lithium fluoride crystals subjected to different modes of irradiation, led Latvian physicists to develop new thermoluminescent dosimeters with a broad range of doses accessible to measurement (10 Mrad to 10^9 rad). These dosimeters are suitable for recording x-rays, γ-rays, neutron radiation, and other modes of radiation. They are being employed with success in radiochemical research at the reactor, and have been adopted by the Ministry of Public Health of the USSR for regular use in radiological practice. Another type of dosimeter, based on thermoluminescent detectors developed at the Institute of Physics of the Academy of Sciences of the Latvian SSR and capable of recording doses from 1 to 10^4 Mrad while retaining the dosimetric information for a sufficiently long time, may find application in radiation work.

Investigations of the electrical and magnetic properties of ferrites and semiconductors subjected to radiation and an externally applied magnetic field have also paid off in the discovery of some intriguing phenomena. Some of them have already met with practical application. For example, a method of improving the performance of semiconductor diodes and transistors by irradiating them with fast neutrons has been developed.

Research and development work on high-level liquid-metal radiation loops as sources of γ-radiation are emphasized in the work of the Institute of Physics of the Academy of Sciences of the Latvian SSR. Such novel experimental facilities as an indium–gallium–tin radiation loop for experiments on radiation chemistry, physics, radiobiology, and other areas, have been developed at the reactor facility.

In the field of neutron activation analysis, the Institute's scientists have developed well over 40 methods for determining chemical elements in a variety of materials, and these methods have been put to successful use in chemistry, biology, medicine, and in various industrial enterprises of the republic.

Work on nuclear spectroscopy is proceeding in close liaison with the Joint Institute for Nuclear Research. Much of the work involves investigation of the structure of nuclei of the rare earths in terms of γ-emission and spectra of conversion electrons emitted when thermal neutrons are captured. A novel β-ray spectrograph with focusing of a broad electron beam has been designed and built for the study of the spectra of conversion electrons.

In Belorussia, the operation of the IRT-2000 research reactor of the Nuclear Power Institute of the Academy of Sciences of the Belorussian SSR [IYaE AN BSSR] went into service in 1963. Over the ensuing years, the material research capabilities of the republic increased considerably. At the present time, in addition to the reactor, whose output level has been brought up to 4000 kW, the Institute also has at its disposal several critical assemblies for the study of reactor physics, a few dozen test rigs for a broad research program on the heat-transfer properties of coolants, heat transfer in general, and related topics, and also modern computer hardware. Since 1968, the Institute has been operating one of the country's largest multichamber, γ-ray facilities with an irradiator activity exceeding 400,000 g-eq radium. The Institute's scientists have put forth and developed the concept of utilizing a gas hitherto never exploited in the power industry: dissociable nitrogen tetroxide, as a coolant in nuclear reactors and as the working fluid in turbines. The successful materialization of this concept in the construction of nuclear power stations may make it possible to reduce turbine size and the size of heat-transfer equipment severalfold while increasing efficiency and cutting electric power costs.

Important scientific information on the thermodynamic properties and transport properties of dissociable nitrogen tetroxide have been obtained, as a result of many long years of persistent investigation, in ranges of pressures and temperatures which are of practical interest for the nuclear power industry. Calculations of the thermodynamic cycles on chemically reactive coolants have been carried out, and a mathematical model has been worked out for calculation of the flow of such a coolant with dissociation kinetics taken into account. Convective heat transfer in the range of temperatures and pressures of practical interest was studied, as well as heat transfer in boiling and condensation of nitrogen tetroxide. The viability and thermal reversibility of the nitrogen tetroxide thermodynamic cycle and the radiation stability of nitrogen tetroxide were confirmed. Also the possibility of using corrosion-resistant structural materials manufactured by Soviet industry for all major components of power plants utilizing nitrogen tetroxide was demonstrated.

A large volume of theoretical, experimental, and design work associated with the development and design of fast reactors and thermal reactors using dissociating coolants has been done at the Institute. Close attention is also being given to comprehensive projects in the development of other promising nuclear reactor types, radiation chemistry, utilization of radiation in industry and agriculture, and in medicine. In particular, the scientific fundamentals of an industrial enterprise with a nuclear facility for simultaneous production of electrical- and thermal-power and various chemical products are under development.

In a joint program with Moscow State University (MGU), a new radiation method for production of ethylene glycol as an intermediate product in the production of lavsan [dacron], dyes, antifreeze agents, propylene glycol, butylene glycol, and other α-glycols, as well as glycerin, from methyl alcohol which is in ample supply, a new radiation processing method was developed.

Kilogram quantities of organochlorine compounds were obtained on a specially constructed facility by means of radiation synthesis, and additives for lubricants were obtained by this method. "Cold" radiation sterilization of disposable blood service systems on an industrial scale was achieved at the γ-irradiation facility of the institute.

The Nuclear Power Institute is taking an active part in the work of the Republic's agricultural institutions to work out techniques of presowing irradiation of seeds for use under conditions prevailing in Belorussia.

BSSR scientists have conducted research on radiation-induced modification of wood and concrete, have developed readily colorable and colored polymers and polyacrylonitrile fibers, and a fundamentally new class of compounds known as polymer pigments, which can be used in bulk in the coloration of fibers and films.

Nuclear physics research in the Kazakh SSR began its development about 15 years ago. In 1967, Kazakhstan physicists acquired a 10 MW VVR-K research reactor. In 1972, the country's first isochronous cyclotron, 150 cm pole diameter, was commissioned at the Institute of Nuclear Physics of the Academy of Sciences of the Kazakh SSR. The modern experimental capabilities and resources of the Institute enable the Republic's scientists to perform fundamental and applied research in the field of nuclear science and nuclear engineering. For several years, a systematic study of nuclear resonance has been in progress at the VVR-K reactor facility, using predominantly short-lived isotopes, and a pneumatic shuttle system has been installed to facilitate that work. The lifetimes of the excited states of a large group of nuclides, including Al^{27}, Mg^{24}, Ca^{42}, Zn^{66}, and Sn^{88}, etc., have been studied. Quasimonoenergetic γ-photons emitted in radiative capture of thermal neutrons at the center of the reactor core are being used as investigative tools in the study of some characteristics of fission of heavy nuclei. Since 1969, work has been underway on building a facility for the generation of ultracold neutrons. These neutrons were obtained midway through the past year on a facility built on the continuous channel of the VVR-K reactor. Yields of ultracold neutrons with different converters were measured on the facility, and the mean time of propagation of ultracold neutrons along the neutron duct from the converter to the detector was measured. A prominent place is given, in the research centered around the VVR-K reactor, to work on activation analysis to determine the content of impurities in ultrapure elements, and in determinations of platinum, silver, and rhenium in geological specimens. Studies of the dynamics of changes in the trace element composition in plants grown from irradiated seeds are now underway.

The installation of the reactor at the Institute of Nuclear Physics of the Academy of Sciences of the Kazakh SSR [IYaF AN KazSSR] paved the way for broad investigations in the field of radiation materials science, and more specifically research on the effect of simultaneous temperature, stress and neutron flux on the long-term strength of refractory materials. Studies are being made of the influence of reactor irradiation on the strength and ductility, in short-term tensile strength tests, and on the physical properties of refractory structural materials and on structural changes occurring in those materials, as well as changes in the adsorptivities and catalytic activities of oxides of silicon, aluminum, and beryllium exposed to bombardment by neutral particles.

A large volume of reactor work is being done in connection with the production of various radioactive isotopes needed by industry and needed by other scientific institutions and industrial enterprises of the republic.

The successes that scientists of the sister republics of the Union have achieved in the field of peaceful uses of atomic energy are due in considerable measure to the new organization principles underlying the use of such unique and novel nuclear physics facilities as the research reactors represent. The essence of these organizational principles lies in the fact that each research reactor is treated as a facility belonging to the republic as a whole, or even a facility shared by different republics, even though it is administratively under the jurisdiction and management of a specific institute.

Effective utilization of republic-owned reactors would be impossible if not for the development and materialization of new forms of organization of research projects, and the appropriate assistance in terms of scientific-research techniques on the part of central scientific-research institutions, most specifically the I. V. Kurchatov Institute of Atomic Energy. Periodic conferences attended by representatives of interested institutions for the purpose of discussing summaries and agreements on plans of scientific-research projects at reactor facilities became such a new form of research organization. The idea of conferences of that type is credited to I. V. Kurchatov.

The first such conference of representatives of all of the reactor centers of the USSR was held in March 1960, Moscow, under the supervision of Academician A. P. Aleksandrov; it attracted 80 persons from 26 organizations. Plans for scientific-research work at reactors built at various institutions under the jurisdiction of the Academy of Sciences of the USSR and of the academies of sciences of the various republics of the Union were discussed, and certain trends of research were recommended as guidelines to focus the research programs of each specific reactor facility, while measures to expedite materials logistics support of the projects were prepared.

Later on, as the level of skills and training of the cadres rose and as the materials logistics base for research at reactor facilities in the sister republics of the Union was strengthened, the agenda on the conferences organized under the joint auspices of the Academy of Sciences of the USSR and the GKAE [State Commission on Atomic Energy] and under the supervision of Academician A. P. Aleksandrov expanded to

take up not only scientific organizational problems, but also presentations of original scientific reports on the basic research trends at reactor facilities. The conferences began to draw representation not only from institutions of the Academy of Sciences of the USSR and the scientific-research institutions of the republics, but also from scientific institutions of the Ministry of Higher and Medium Special Education and other ministries and departments of the nation.

We get some idea of the scientific and scientific-organizational authority of those conferences, for example, from the fact that 344 representatives from 88 organizations of the Academy of Sciences of the USSR, and the academies of sciences of the sister republics, GKAE, and other ministries and departments, participated in the deliberations of the seventh coordination conference held at Minsk. Of course, regular joint discussion of research findings and plans for future research, as well as the steadily expanding scope of work by republic-level institutes working in unison with leading nuclear physics institutions of the country, have exerted a most fruitful effect on the situation regarding creative research work in the nuclear research centers of the republic, and have contributed to raising the effectiveness with which those centers make use of their available nuclear physics facilities.

And now, when our country celebrates the 50th anniversary of the USSR, Soviet science can be justifiably proud of both the general level of work in the field of peaceful uses of atomic energy and of the progress achieved in that direction in the various sister republics of the Union.

BOOK REVIEWS

A. M. Petros'yants

FROM SCIENTIFIC PROSPECTING TO ATOMIC INDUSTRY*

Reviewed by Yu. Klimov

The second edition of the book under review differs substantially from the first edition, mostly due to the rapid development of nuclear science and engineering in both qualitative and quantitative terms. During the two years that have elapsed since the publication of the first edition, some important events have occurred in the application of nuclear power in the Soviet Union, in other socialist countries, and throughout the world. Of course, this could not fail to be reflected in the book, which the author dedicates to the 50th anniversary of the USSR, and treats as a report scheduled for that major historical date.

In the second edition, as in the first, the reader can explicitly follow the author's intention to emphasize the achievements and significance for the national economy of the work done in the field of peaceful uses of nuclear power in the Soviet Union. In that sense, the contents fully correspond to the title selected for the book, and to a concentrated presentation of the idea expressed in the title.

Actually, the nuclear industry of the USSR, with its vast scientific, engineering, production, and labor potentials, has become an important and irreplaceable element of the productive force of the Soviet Union. This is demonstrated convincingly and from many points of view, particularly in the presentation of the material on the most productive aspects of the utilization of nuclear power: the nuclear power industry and the production of radioactive isotopes.

The volume of the text has been expanded in the second edition. The section on the nuclear power industry has been expanded appreciably. One major event of international interest was the IVth Geneva Conference on the Peaceful Uses of Atomic Energy, held in 1971, which was reflected in the form of communications and presentations of Soviet reports delivered at the conference. New data are brought forth, and information on channel-type uranium–graphite power reactors, water-cooled, water-moderated power reactors, and the developmental outlook for the nuclear power industry of the USSR, are provided in large amounts.

The section dealing with high-energy physics and charged-particle accelerators is supplemented with new data on the largest accelerators, and on the production of new superheavy transuranium elements. The section on thermonuclear fusion has been revised.

The section on the utilization of isotopes in the national economy of the USSR has been enriched with new and more complete information. The section on nuclear research centers in the Soviet Union has been expanded and its aims and achievements are expounded in greater detail.

On the whole, the book has been made more rigorous, and exhibits more of the character of a monograph. A certain trend toward refraining from popularization is noticeable and it sometimes seems that the text is now more difficult to read. The explanation probably lies in the author's intention to approach the various topics in a more rigorous and profound manner. The author himself writes in the preface that the book is of a higher quality in this second edition. In his work on the second edition, the author oriented himself not so much toward the untutored reader as toward the reader possessing a broad but at least minimum familiarity with "atomic" subject matter. This justifies the inclusion of ample factual, numerical, and tabular material, which makes the book useful for reference.

*Atomizdat, Moscow, 1972.

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In other words, it seems that the author was confronted with the decision of whether to continue the presentation in the genre of the typical popular science literature, or whether to lean in the direction of the monograph and fundamental treatment. It seems that the author has chosen the latter course.

The first edition was well received by the intended readership: the entire edition sold out in short time. It is gratifying to note that the book was awarded an honorary diploma by the Znanie ["Knowledge"] Society. It is to be hoped that the second edition will meet with no less well-deserved interest.

V. A. Dobromyslov and S. V. Rumyantsev

RADIATION INTROSCOPY*

Reviewed by P. K. Oshchepkov

This book is devoted to engineering diagnostics of the quality and reliability of industrial products and materials in the modern machinery industry, and specifically in the manufacture of power machinery. It is written authoritatively from an extensive familiarity with the essentials of the problems in question, and is provided with adequate factual material drawn from practical applications of radiation introscopy.

The problem of quality and reliability has been gaining in importance under the conditions of present-day technical advances. The development of techniques and equipment for attaining the most complete information, i.e., information covering many different relevant elements, on the spatial distribution of the principal physical and physicomechanical properties of the product or material to be inspected, meets the most stringent modern requirements. Basic theoretical and applied information on radiation introscopy are provided, rendering the book highly useful to engineers concerned with applications of those techniques in industry.

Special attention is given to active monitoring of the quality of metals and other materials during the production process, in the construction of modern high-cost objects and installations. Radiation methods of inspection and testing perform an irreplaceable service in many instances. In that sense, the book being reviewed will be useful for application and development of those methods, since radiation methods of inspection and quality control are by and large noncontacting methods.

Basic operating principles of existing electron optics converters of radiation and other types of converters are presented in the text. The operating principles of x-ray television circuits for industrial use are described, and their basic technical characteristics are cited.

The application and wide acceptance of x-ray and γ-ray closed-circuit television systems are dictated at present not only by the large volume of industrial inspection work on crucial parts and subassemblies, but also by the fact that further developments in this line of inspection equipment will be associated with the development of objective techniques for assessing inspection results. Closed-circuit television systems for inspection, supplemented by appropriate data-processing and computing equipment, will make it possible to raise the level of inspection work on crucial parts and products to a qualitatively new position. The book introduces the reader to the topic in a logical presentation. An extensive bibliography is presented, and should be highly useful not only to practising engineers, but also to workers at scientific-research institutes and laboratories engaged in the development and applications of radiation introscopy.

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ARTICLES

A MATHEMATICAL MODEL FOR LONG-TERM PREDICTION OF NUCLEAR POWER DEVELOPMENT BASED ON ECONOMIC CRITERIA

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UDC 621.039.003

Economic-mathematical modeling is suited to the problem of determining effective courses for the development of nuclear power (NP) to the greatest extent. Together with the study of NP as a part of the fuel-power economy of the country one should examine the interconnection of NP, the strength and nature of the relationships between separate elements of the fuel cycle, the interaction of various types of nuclear power plants (NPP) in the general dynamic development of NP – i.e., its internal structure.

For the study of these questions one should ascertain: the role of NPP's with fast reactors in satisfying demands for electric energy; the range of technical and economic features of NPP's with fast reactors and improved heat converters allowed by the restrictions of economic competition; the advisability of the introduction into NP of fast reactors and improved heat converters; the influence of the pace of NP development on the proportion of different types of reactors; the optimal times for the introduction of fast reactors into NP; the influence of the cost of nuclear fuel on the NP structure.

The most important parameter in making predictions is the prediction period. Its length is defined as the greatest interval of time, beyond which ideas about NP development do not have an effect on solutions adopted at the present time. Analysis of the time lag and the persistence of the investment cycle of a NPP shows that the required prediction period in NP is greater than 25-30 years [1].

The prediction method and the accuracy of the information permit determination of the length of a feasible prediction period. The use of the prediction method where there is increasing inaccuracy of information can lead to a situation where a feasible prediction period will be less than required. In this case one should use the feasible prediction period as a design prediction period, but for the analysis of the remaining period one should use another method. In choosing a long-range prediction method one should consider that on the basis of the present state of NP it is impossible to accurately predict its future state. Therefore the method of "exploratory prediction" is widely applied for long-range predictions of NP. This method permits the description of the future of NP in terms of likely conditions of its development. The concept of the "future" is a complex description which comprises the dynamics of investment in NP during the design period, the extent of the demand for primary nuclear fuel, the dynamics of accumulation and consumption of secondary nuclear fuel, the importance of secondary nuclear fuel in the system, the composition of the introducible power of NPP's according to the type of available nuclear reactors, the optimal time to begin the mass construction of NPP's with fast reactors, and the ultimate technical and economic limits of various types of NPP's, etc.

A preliminary analysis of the object of the study – the NP system – is an initial prerequisite of an exploratory prediction. Such an analysis permits a formulation of the principal characteristics of NP for the next 20-30 years.

1. The availability of NPP's with various types of nuclear reactors.
2. All NPP's operate from the common reserves of plutonium; therefore all NPP's are integrated by the relationships of the fuel cycle.

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3. The state of the NPP system at a future moment of time is determined to a significant degree by the course of the development prior to that moment, particularly in that stage where there exists a deficit in secondary nuclear fuel. This is explained by the fact that the reserve plutonium supply is determined by the integral nature of the consumption and production of plutonium by all NPP's introduced at a given moment. Thus the NPP system is a dynamic one.

4. The existence of a common reserve of plutonium together with the known differences in the breeding ratio (BR) in nuclear reactors makes an isolated comparison of NPP's impossible (without an analysis of the system), since, for example, such situations may arise: it is found that an NPP with a converter, using a high BR, is more expensive by reduced costs, while an NPP with a low BR is economically cheaper. Thus it becomes necessary to examine NPP systems as an aggregate of dynamic objects which are linked among themselves with direct and reciprocal relationships.

5. The transport component in the cost structure of a NPP is completely insignificant, since within the scope of a long-term prediction the NPP system can be represented in the form of a unijunctional model.

The above-mentioned characteristics of NP introduce the following requirements into the mathematical model for the prediction of NP development: 1) the model must exhibit the properties of a system of interrelated groups of NPP's; 2) it must be dynamic in structure; 3) the model must be realized as unijunctional; 4) the length of the design period must extend more than 25-30 years and must be modified in the process of investigation.

The mathematical model for the prediction of the NPP system development includes a description of the external and internal relationships of the modeled system and expressions for integral functions and for unknown functions describing the future of NP.

The equivalent description in the model of the actual relationships of NP is determined to a significant degree by the method of introducing the unknown variables. We denote by x the magnitude of the introducible power of a NPP. The subscripts for x – f, j, and τ – designate, respectively, the type of NPP, its mode of operation and the time interval during which power is introduced. The superscripts designate the interval during which the NPP is operated. We denote by y the amount of electrical energy generated by the NPP. We use the following unknown variables: $x_{f,j,\tau}$ is the established power introducible during the time interval τ of NPP's intended for operation in mode j (with the annual number of hours of operation $h^{r-\tau+1}_{f,j,\tau}$); $x^{r}_{f,j,\tau}$ is the power available in the interval r ($r = \tau, \tau+1, \ldots$) of NPP's put into operation in the interval τ; $x_{f,\tau}$ is the established power introduced in the interval τ; $y^{r}_{f,\tau}$ is the electrical energy generated in the interval r from NPP's put into operation in the interval τ.

Let the total number of reactor types examined be n, the number of different modes m, and the number of time intervals R'. Then using the notation $x_{f,j,\tau}$ for the unknown variables, the total number of variables will be nmR'. It is evident that the transformation to another notation increases the number of variables. At the same time a series of supplementary equations must be introduced into the model, since the available power and the generation of electrical energy must be related to the established power of a NPP within a given interval of the operating period. These propositions are illustrated by the data presented in Table 1.

The first method, which led to a more compact mathematical model, is evidently preferable in the present case. The disadvantage of this method – the inability to reflect changes in power of introduced NPP's in their operation period – can be eliminated by the introduction of appropriate coefficients for the unknowns $x_{f,j,\tau}$.

The structure of the time scale plays a significant role in the mathematical model. A uniform time scale with one-year intervals permits the fullest consideration of the influence of inertia in the external fuel cycle. However, its use results in a large dimensionality for the model, which in a number of cases is undesirable. Therefore the application of a nonuniform time scale seems promising, with the initial intervals having a duration of one year, and increasing to three, five, or 10 years. The nonuniform scaling is more preferable first from a practical point of view since it permits a sharp reduction in the dimensionality of the mathematical model, and, second, as a theoretical consequence of the fact that the applied initial information becomes more and more ambiguous according to the amount of movement along the time scale.

TABLE 1. The Influence of the Method of Introducing Unknown Variables on the Dimensionality of the Model

Characteristics	Method number		
	1	2	3
Form of the unknowns	$x_{f,j,\tau}$	$x^r_{f,j,\tau}$	$x_{f,\tau}, y^r_{f,\tau}$
Number of supplementary constraints	0	$n\frac{R'(R'-1)}{2}$	$nR'(R'+1)$
Number of supplementary unknowns	0	$n(m-1)\frac{R'(R'-1)}{2}$	$nR'(1{,}5-m+0{,}5R')$

It is natural then that the mathematical model must also become less detailed. The nonuniform time scale allows us at the same time to approximate the realization of equal accuracy for the separate parts of the model among themselves in connection with the nature of the initial information.

External Relationships of the NPP System

The problem entails the study of the NPP over T' years. This period is composed of R' time intervals, each of which consists of S_τ years. The NPP system interacts with the joint power system (JPS) of the country through direct and reciprocal external relationships. The direct relationship is expressed in the influence of demand for the production of electrical energy on the structure of introducible power and other characteristics of future NP development. The reciprocal relationship is the influence of NP on the closing costs for electrical energy in the general JPS structure, and through changes in their costs on the economically preferable structure of the introducible power from NP. The direct relationship may be described by two balanced equations in which the electrical energy generated in each interval of time and the electrical power from NP used by the joint power system is expressed through structural parameters characterizing the composition of NPP types.

To include the reciprocal link in the mathematical model one must know the functional dependence of changes in closing costs of electrical energy in the JPS on changes in generated electrical energy from NP. Since this problem is to a large degree an independent one, it follows that in the present model the description may be limited only by the direct external relationships of NP. In describing the elements of the NPP system one must take into account not only the possibility of their improvement during the years of their introduction through an increase of unit power of the blocks and the influence of other factors, but also the possibility of changes in their technical and economic characteristics after the NPP is put into operation, i.e., not only the technical progress of NPP's under construction, but also their improvement during operation.

The balanced equations for the available power and the generation of electrical energy in the NPP system for each time interval have the form

$$\sum_{f=1}^{n}\sum_{j=1}^{2}(a_{f,r}x_{f,j,r}+\Delta a_{f,r-1}x_{f,j,r-1})=N_r,$$

$$\sum_{f=1}^{n}\sum_{j=1}^{2}[a_{f,r}h^1_{f,j,r}x_{f,j,r}+(h^2_{f,j,r-1}-a_{f,r-1}h^1_{f,j,r-1})x_{f,j,r-1}+\sum_{\tau=1}^{r-2}(h^{r-\tau+1}_{f,j,\tau}-h^{r-\tau}_{f,j,\tau})x_{f,j,\tau}]=\dot{E}_r, \qquad (2)$$

where $a_{f,\tau}$ is the coefficient for assimilation of the established power in the interval τ for an f-type NPP introduced in this period; $\Delta a_{f,\tau}=1-a_{f,\tau}$ is the increase in the assimilation coefficient in the second year of operation of an f-type NPP introduced in the interval τ; $h_{f,j,\tau}$ is the number of hours of operation for the respective NPP's in the time interval θ of operation after startup, $\theta=1, 2, \ldots, R'-\tau+1$; E_r, N_r are, respectively, the necessary increase in the production of electrical energy in the interval r and the available power of all NPP's in the NP system, $r=1, 2, \ldots, R'$.

In deriving (1) and (2) we have assumed that the mode of operation of the NPP existing at the beginning of the design period does not change.

With the introduction into the system of a new type of nuclear reactor (for example, a fast reactor) the time which is needed for the changeover of the machine construction industry, for the assimilation of equipment, etc., must be considered. Therefore it is expedient to provide in the model for conditions limiting the introduction of a new type of NPP during this period of time. Let F be the multiplicity of all types of nuclear reactors under consideration and F_H the multiplicity of new types of reactors; r^0_{fH} is the initial time interval for the extensive construction of the f_H-type, where $f_H \in F_H$, and r_f is the duration of the assimilation period and of the changeover for the industry manufacturing the essential equipment. One

can then note the following limits on the total introducible power of f_H-type reactors:

$$\sum_{j=1}^{2} x_{f,\,j,\,r} \leqslant N_{f_{H},\,r}, \tag{3}$$

where $r = r^0_{f_H}, \ldots, r_{f_H} + r_f$; $N_{f_{H,}\,r}$ is the allowable increase in established power of the f_H-type NPP in the interval r.

Internal Relationships in the NP System

The basic structural relationship between the separate types of NPP's is expressed by the equation of balance of secondary nuclear fuel (in the present work the uranium–plutonium cycle is considered) between its consumption by breeder reactors and its production by all NPP's. This restriction on the development can be formulated in the following way: in each moment of time the reserve in the system of plutonium ready for charging in a breeder reactor must not be negative.

Let F_a be the multiplicity of NPP types using uranium as nuclear fuel; F_b is the multiplicity of NPP types on plutonium. We introduce the following notation: $g^0_{f_b,\tau}$ is the specific consumption of plutonium in the initial charge of a NPP with an f_b-type reactor ($f_b \in F_b$), introduced in the time interval τ, per unit of established power; $g^{r-\tau+1}_{f_b,r}$ is the specific consumption of plutonium for recharging per unit of electrical energy generated in the interval r; $g^{r-\tau+1}_{f,\tau}$ is the specific discharge of plutonium per unit of electrical energy generated in the interval r; r_0 is the first interval, beginning with which NPP's with f_b-type reactors can be introduced; r_{BH} is the inertia of the external fuel cycle, i.e., the time elapsing from the discharge of the fuel elements to their charge after processing in the reactors; α_r is the reserve of prepared plutonium available in the system at the end of the interval r.

We assume that the transfer time of secondary nuclear fuel in the external fuel cycle is equal to one year. This means that the plutonium manufactured in the year $(r-2)$ may be used to charge NPP's introduced in the year r. We will assume also that in the first year of operation of a NPP with breeder reactors, consumption of fuel for recharging is equal to zero. Taking into account these assumptions for time intervals where $S_\tau = 1$, the quantity of plutonium suitable for charging a NPP in the interval r is determined by the relation

$$\mathcal{P}_r = G_{r-2} + \sum_{f=1}^{n}\sum_{j=1}^{2}\sum_{\tau=1}^{r-2} a^{r-\tau-1}_{f,\tau} g^{r-\tau-1}_{f,\tau} h^{r-\tau-1}_{f,\,j,\,\tau} x_{f,j,\tau} + \alpha_{r-1}, \tag{4}$$

where G_{r-2} is the plutonium manufactured in the $(r-2)$ interval by NPP's operating at the beginning of the design period.

The quantity of plutonium being charged in f_b-type NPP's introduced in the interval r is equal to

$$\mathcal{P}^0_r = \sum_{f_b \in F_b}\sum_{j=1}^{2} g^0_{f_b,\,r} x_{f_b,\,j,\,r}, \tag{5}$$

where $r = r_0, \ldots, R'$.

The consumption of plutonium for recharging in the interval r is found from the relation

$$\overline{\mathcal{P}}_r = \sum_{f_b \in F_b}\sum_{j=1}^{2}\sum_{\tau=1}^{r-1} a^{r-\tau+1}_{f_b,\tau} h^{r-\tau+1}_{f_b,\,j,\,\tau} \overline{g}^{r-\tau+1}_{f_b,\,\tau} x_{f_b,\,j,\,\tau}. \tag{6}$$

Combining (4)-(6), we obtain conditions of nonnegativity for the system's plutonium reserve:

$$\mathcal{P}^0_r + \overline{\mathcal{P}}_r - \mathcal{P}_r + \alpha_r = 0, \tag{7}$$

$$\alpha_{r-1} \geqslant 0, \tag{7'}$$

$$\alpha_r \geqslant 0, \tag{7''}$$

where $r = r_0, \ldots, R'$.

In perspective, the most important condition for the development of the NPP system is the possible limitation on the primary nuclear fuel used, which may be related either to an increase of its cost as a function of the size of the involved resources of natural uranium or to limitations on the size of the involved resources due to a possible lag in construction of enterprises in the external fuel cycle.

Let us divide the reserves of natural uranium into three categories, where the cheap reserves are withdrawn first, and then the more expensive ones. We introduce the notation: C_i is the specific cost of natural uranium of the i-th category; Q_i is the corresponding size of the reserves; X_i^r is the size of the natural uranium resources used in the time interval r from the reserves of the i-th category by all NPP's; $Q^r(x_{f,j,\tau})$ is the natural uranium consumption by the corresponding NPP's in the time interval r, expressed linearly in $x_{f,j,\tau}$; Q_0^r is the natural uranium consumption in the interval r by existing NPP's. The constraints on natural uranium are then written in the form

$$\sum_{f=1}^{n}\sum_{j=1}^{2}\sum_{\tau=1}^{r} Q^r(x_{f,j,\tau}) + Q_0^r = \sum_{i=1}^{3} X_i^r; \tag{8}$$

$$\sum_{r=1}^{R'} X_i^r \leqslant Q_i. \tag{9}$$

Accumulated experience in mathematical modeling of economic systems in problems of long-range planning and prediction shows that in describing objects it is possible to neglect nonlinear effects associated with the fact that construction customarily proceeds in several stages. An analogous linearization for NPP's is used in the present study.

The choice of a target function for the model is a complex question in problems of long-range prediction in general and of NP in particular. In this study the economic criteria most fully corresponding to the posed problem were used in its financial expression. Variants of the different NP structures under comparison have identical values for the generation of electrical energy in each interval of the design period, and therefore it is possible to use financial criteria in the form of reduced costs. Since conditions for the functioning of NPP's put into operation are such that it is possible to vary the cost of nuclear fuel, where this variation depends on the entire operation of the NPP, it seems unsuitable in the present study to use expressions for reduced costs, which are generally applied through statistical estimates for isolated objects:

$$З = \sigma_н К + И. \tag{10}$$

At present the dynamic criterion for the system is written in the form

$$З = \sum_k \sum_{t=1-t_c}^{T} (К_{k,t} + И_{k,t})(1+\sigma_{pr})^{1-t} + \sum_k \sum_{t=T+1}^{\infty} И_{k,t}(1+\sigma_{pr})^{1-t}, \tag{11}$$

where T is the length of the design period; σ_{pr} is the norm for introducing nonsimultaneous costs; $K_{k,t}$ is the investment in the object k in the year t; $I_{k,t}$ is the operating cost of object k in the year t; t_c is the construction period of the NPP.

It is usually assumed [2] that the operating costs do not change with time for the duration of the prediction period. Then (11) takes the form

$$З = \sum_k \sum_{t=1-t_c}^{T} (К_{k,t} + И_{k,t})(1+\sigma_{pr})^{1-t} + \sum_k И_{k,н} \frac{(1+\sigma_{pr})^{1-T}}{\sigma_{pr}}, \tag{12}$$

where $I_{k,H}$ are the fixed operating costs of the k-th object for the duration of the prediction period.

The practical use of criterion (12) is associated with a number of difficulties because of the nonconstructional determination of costs over the duration of the prediction period, especially since the accuracy of the information is extremely low for the range of possible durations for the prediction period. In [1] it is shown that a modified criterion in the form

$$З = \sum_k \sum_{t=1-t_c}^{T} (К_{k,t} + И_{k,t})(1+\sigma_{pr})^{1-t}, \tag{13}$$

may be used for the target function.

The length of the design period consists of two time segments: optimal development for the duration T' and quasioptimal for T – T'.

The following factors are taken into account in the target function: the change in investment during the years of construction; differences in the specific investment for NPP's; technical progress, appearing in the decrease of specific investments during the prediction period due to the increase in the unit power of a NPP energy-block, improvements in technology, etc.; the costs of the initial charging of a NPP and the possibility of their change during the design period; depreciation deductions and operating costs in a NPP; differences in the fuel components for various NPP types as a function of the year of the NPP's introduction and the year of their operation; the influence of the closure of the fuel cycle of primary and secondary nuclear fuel on the amount of fuel components; the necessity for reserve power for the NPP system. Condensing electric power stations were used for reserves, and the effect of unequal introduction of power within the intervals on the costs in these intervals was taken into account.

The question of the importance of plutonium in the NPP system is a principal one. One of the possible restrictions consists in the fact that the system does not expend plutonium elsewhere and does not obtain it externally. It then follows to take the cost of plutonium discharged in a NPP to be equal to zero, and the cost of the plutonium charged in the reactor to be determined by expenditures for the extraction of plutonium and the preparation of fuel elements from it.

It was indicated above that a concept of the future of NP includes not only a description of the NP structure, i.e., the relationships of various types of NPP's, but also a series of other features: the dynamics of investment, the consumption of primary and secondary fuel, and the dynamics of expenditures in NP.

The unknown features are written in the form of equations and are expressed through the fundamental variables of the model $x_{f,j,\tau}$. For example, the consumption of plutonium in the period from the initial charge until $\tau = R$ is determined by the expression

$$G_1 = \sum_{f_b \in F_b} \sum_{j=1} \sum_{\tau=\tau_0}^{R} g^0_{f_b,\tau} x_{f_b,j,\tau}. \tag{14}$$

Equation (14) may be written in another form:

$$\sum_{f_b \in F_b} \sum_{j=1}^{2} \sum_{\tau=\tau_0}^{R} g^0_{f_b,\tau} x_{f_b,j,\tau} - G_1 = 0 \tag{15}$$

and may be considered to be one of the constraints of the model in which a new unknown – G_1 – is present. It is evident that equations of the type (15) do not influence the choice of optimal compositions of NPP's, and therefore we will call them pseudoconstraints of the model, and the unknown features, analogous to G_1, are pseudounknowns.

From the preceding it follows that the model described for long-range prediction of the development of the NP system – which includes a complex of constraints and pseudoconstraints and also an expression for the target function – contains only linear forms of the fundamental unknowns of the posed problem, $x_{f,j,\tau}$. Therefore, the method of linear programming must be applied in its investigation, which allows a determination of the optimal structures of NPP's and of the values of the pseudovariables corresponding to them.

The mathematical model developed was widely tested in computational experiments, the results of which confirmed the appropriateness of its application to the analysis of the most suitable courses for the development of NP. At the same time the adequacy of the mathematical model was confirmed by the problem posed by the investigation. In particular, the model allows us to determine the influence of the following factors on the NP structure: the pace of growth of NP; the changes of the average annual number of hours of NPP use along the intervals of the prediction period; the initial time interval in which fast neutron reactors may be introduced; possible constraints on the part of the machine-construction industry; economic features of the reserves of natural uranium; the relative rise in price of NPP's with fast reactors; the technical and economic features of different types of NPP's, and others. Together with these such questions were resolved as the role of different types of nuclear reactors (including NPP's with thermal and fast reactors using primary and secondary nuclear fuel), the optimal times for the introduction of NPP's with fast reacors, the importance of secondary nuclear fuel in a developing NP system, the dynamics of the

consumption of nuclear fuel during the years of the design period, and the integral and differential economic expenditures on NP development, etc.

An analysis of computational data is not presented in the present work since this would have excessively broadened its scope.

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A MATHEMATICAL MODEL FOR THE OPTIMIZATION OF THE STRUCTURE OF THE NUCLEAR POWER INDUSTRY ON THE BASIS OF MINIMUM NUCLEAR FUEL REQUIREMENTS

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UDC 338.4:620.9

The most correct criterion for estimating the optimal structure of a developing nuclear power industry should be an economic criterion expressed in monetary terms (for example, the value of the adjusted costs of the development of the nuclear power industry for the forecast period). However, in the absence of a single unit of currency, and of a basis for a comparative study of the economic effects produced in various countries by the development of the nuclear power industry in the member states of the Council for Mutual Economic Assistance (CMEA), an estimate of the cooperative development of the nuclear power industry expressed in monetary terms is not yet possible. Optimization on the basis of natural indicators, the most important of which is the consumption of natural fuel, can also be of interest for investigations of the development of nuclear power industries on a national scale when there is a scarcity of natural uranium.

In the proposed model it is assumed that the nuclear power industry is a subsystem of the nation's fuel–energy balance (the overall power system), which imposes certain conditions that act as an external restriction on the development of the nuclear power industry. During the entire forecast period the following restrictions, expressed in the form of balance equations, will apply during each interval of time.

1. The balance of the amount of nuclear power put into service each year. It is assumed that the total nuclear power during the calculation period is defined on the basis of conditions governing the combined study of the total fuel–energy balance of the CMEA member countries.
2. The balance of electrical energy generation, which is also defined by the conditions of the fuel –energy balance of the CMEA member countries.
3. The plutonium balance of the nuclear power industry. This restriction is due to the internal characteristics of the nuclear power industry. It is assumed that the total amount of plutonium produced in the system is used up in the system (in breeders) and no plutonium is introduced from outside.

Thus, in the general case the problem is formulated as follows: it is required to minimize a target function, the amount of natural fuel consumed during the forecast period, subject to the above three restrictions. We shall consider the development of a nuclear power system based on the use of three or more types of reactors. The independent variables x_{it}, each of which represents the amount of power contributed by reactors of type i in year t, appear in the target function and in the equations. The coefficients of the x_{it} are independent of the variables.

Thus, in the general case the problem of optimizing the structure of the nuclear power industry on the basis of natural-fuel consumption is an optimization problem in linear programming and can be solved by the appropriate methods.

Derivation of the Target Function, the Equations, and Coefficients. The derivation of the equations and coefficients for the independent variables can be illustrated by using the example of the development of a nuclear power industry using three types of reactors: fast breeders, fast converters, and thermal converters.

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The target function in general form is written as follows:

$$\sum_{i=1}^{I}\sum_{t=1}^{T} G_{it}, \tag{1}$$

where G_{it} is the natural uranium requirement for a reactor of type i in year t; T is the forecast period for the calculations, expressed in years; I is the number of types of reactor used. In the general case the number of independent variables is equal to the product $I \cdot T$.

For three types of reactors the target function will have the form

$$a_{11}x_{11}+a_{21}x_{21}+a_{31}x_{31}+\sum_{t=2}^{T}\left(a_{1t}x_{1t}+a_{2t}x_{2t}+a_{3t}x_{3t}+a_{4t}\cdot\sum_{\tau=1}^{t-1}x_{1\tau}+a_{5t}\cdot\sum_{\tau=1}^{t-1}x_{2\tau}+a_{6t}\cdot\sum_{\tau=1}^{t-1}x_{3\tau}\right)\rightarrow \min. \tag{2}$$

Here x_{1t}, x_{2t}, and x_{3t} are independent variables which determine the amount of power (in millions of kilowatts) contributed to the nuclear power industry by reactors of the first type (fast breeders), the second type (fast converters), and the third type (thermal converters) in year t; a_{1t}, a_{2t}, and a_{3t} are indicators which determine the proportion of natural uranium [in $t/10^6$ kW (electrical)] charged into fast breeders, fast converters, and thermal converters, respectively, in year t ($a_{1t} = 0$, since the amount of depleted or spent uranium charged into the fast breeders is not taken into consideration here); a_{4t}, a_{5t}, and a_{6t} are indicators which determine the proportion of natural uranium (in $t/10^6$ kW · yr) used for recharging fast breeders, fast converters, and thermal converters, respectively, in year t ($a_{4t} = 0$ for reasons similar to those mentioned above for a_{1t}); the coefficients $a_1, \ldots, a_6$ can be expressed in terms of the parameters of the nuclear power stations and of the fuel cycle in the following way:

$$\left.\begin{aligned}
a_1&=0;\\
a_2&=g_2^{x_{init\,2}}(1+\delta g_2^{x_{init}})\cdot\frac{x_{init\,2}-y}{c_0-y};\\
a_3&=g_3^{x_{init\,3}}(1+\delta g_3^{x_{init}})\cdot\frac{x_{init\,3}-y}{c_0-y};\\
a_4&=0;\\
a_5&=\frac{3.65\cdot10^5\cdot\varphi_2}{\bar{B}_2\cdot\eta_2}\cdot\left[\frac{x_{init\,2}-y}{c_0-y}-A_2\left(\frac{x_{fin\,2}-y}{c_0-y}\right)\right];\\
a_6&=\frac{3.65\cdot10^5\cdot\varphi_3}{\bar{B}_3\cdot\eta_3}\cdot\left[\frac{x_{init\,3}-y}{c_0-y}-A_3\left(\frac{x_{fin\,3}-y}{c_0-y}\right)\right];
\end{aligned}\right\} \tag{3}$$

where $A_i = 1 - \varepsilon_U$; $g_2^{x_{init2}}$, $g_3^{x_{init3}}$ represent the amount of uranium of initial enrichment x_{init} charged into fast converters and thermal converters, respectively, expressed in $t/10^6$ kW; $\delta g_2^{x_{init}}$, $\delta g_3^{x_{init}}$ represent the reserves of fresh fuel at atomic power stations (in the fractions $g_2^{x_{init}}$ and $g_3^{x_{init}}$, respectively); c_0 is the U^{235} content of the natural uranium, equal to 0.00714; x_{init2} and x_{init3} represent the initial enrichment of uranium in the fast converters and thermal converters, respectively, expressed in fractions of unity; y is the U^{235} content of the tailings of the separation plant, expressed in fractions of unity; φ_2 and φ_3 are the average values of the plant power utilization factor, expressed in fractions of unity, for fast converters and thermal converters, respectively, in year t; $\bar{B}_2$ and $\bar{B}_3$ represent the average depth of burnup in fast converters and thermal converters, respectively, in year t, expressed in MW · days/t; η_2 and η_3 are the efficiencies of power stations using fast converters and thermal converters, respectively, in year t, expressed in fractions of unity; x_{fin2} and x_{fin3} represent the final enrichment of the uranium in fast converters and thermal converters, respectively, expressed in fractions of unity; ε_U represents the uranium losses in the external cycle, expressed in fractions of unity.

Where the forecast period is longer than the nominal service life of the power station (t_{serv}), we must include in the target function the condition that spent nuclear power stations will be replaced by new ones. Then the target function will be

$$a_{11}x_{11}+a_{21}x_{21}+a_{31}x_{31}+\sum_{t=2}^{T}\left\{\left[a_{1t}x_{1t}+a_{2t}x_{2t}+a_{3t}x_{3t}\right]+a_{4t}\sum_{\tau=1}^{t-1}x_{1\tau}+a_{5t}\sum_{\tau=1}^{t-1}x_{2\tau}+a_{6t}\sum_{\tau=1}^{t-1}x_{3\tau}\right\}$$

$$+\sum_{t=31}^{T}\{[a_{1t}x_{1(t-t_{serv})}+a_{2t}x_{2(t-t_{serv})}+a_{3t}x_{3(t-t_{serv})}]+a_{4t}\sum_{\tau=t_{serv}}^{t-1}x_{1(\tau-t_{serv})}+a_{5t}\sum_{\tau=t_{serv}}^{t-1}x_{2(\tau-t_{serv})}+$$

$$+a_{6t}\sum_{t=\tau_{serv}}^{t-1}x_{3(\tau-t_{serv})}\}\to\min. \tag{4}$$

It should be noted that the coefficients a_{jt} are independent of the variables x_{it}; consequently, the target function is a linear function of the independent variables x_{it}.

We shall consider below the restrictive conditions under which the nuclear power industry is to develop. The power balance equation has the form

$$x_{1t}+x_{2t}+x_{3t}=a_{7t}, \tag{5}$$

where a_{7t} (in 10^6 kW/yr) = N_t is the annual power generation of the entire nuclear power industry. In this case, if $T > t_{serv}$, we have

$$a_{7t}=N_t-N_{t-t_{serv}}, \tag{6}$$

starting from year $t > t_{serv}$, expressed in years.

The second equation defines the plutonium balance in the nuclear power system. It is assumed that all the plutonium obtained in the system is used up in it; there is no plutonium introduced from outside. It is also assumed that the plutonium is used as a fuel only in fast breeders.

Plutonium Balance Equation for Year t. 1) Plutonium consumption

$$a_{8t}x_{1t}+a_{9t}\sum_{\tau=\theta+1}^{t-\theta}x_{1\tau}. \tag{7}$$

Here a_8 is the proportion of fissionable plutonium charged into the breeders (in $t/10^6$ kW):

$$a_8=g_{a.z.}Z_{init} \tag{8}$$

($g_{a.z.}$ is the total amount of fuel charged into the active zone, in $t/10^6$ kW; Z_{init} is the initial fissionable plutonium content of the fuel, expressed in fractions of unity);

$$a_9=\frac{3{,}65\cdot10^5\varphi_1}{\overline{B}_1\eta_1}, \tag{9}$$

where a_9 is the proportion of fissionable plutonium used for recharging the breeders, expressed in $t/10^6$ kW · yr; φ_1 is the average plant power utilization factor for breeders, expressed in fractions of unity; $\overline{B}_1$ is the average depth of fuel burnup in the active zones of the breeders, expressed in MW (fuel) · days/t; η_1 is the efficiency of a nuclear power station using a breeder, expressed in fractions of unity.

2) Plutonium production

$$a_{10(t-\theta_1)}\sum_{\tau=\theta+1}^{t-\theta_1}x_{1\tau}+a_{11(t-\theta_2)}\sum_{\tau=1}^{t-\theta_2}x_{2\tau}+a_{12(t-\theta_3)}\sum_{\tau=1}^{t-\theta_3}x_{3\tau}. \tag{10}$$

In formulas (7) and (10), θ is the residence time of the fuel in the cycle; $\theta_0 = |T_{fl_1}/\varphi_1|$, in years ($T_{fl_1}$ is the fuel lifetime in breeders, expressed in years); $\theta_1 = |T_{fl_1}/\varphi_1 + T_p|$, in years ($T_p$ is the length of the external fuel cycle, expressed in years); $\theta_2 = |T_{fl_2}/\varphi_2 + T_p|$, in years ($T_{fl_2}$ is the fuel lifetime in fast converters, expressed in years); $\theta_3 = |T_{fl_3}/\varphi_3 + T_p|$ (T_{fl_3} is the fuel lifetime in thermal converters, expressed in years); a_{10}, a_{11}, and a_{12} represent the buildup of plutonium (in $t/10^6$ kW · yr) in breeders, fast converters, and thermal converters, respectively:

$$a_{10}=(1-\varepsilon_{Pu})\frac{0.39\cdot\varphi_1}{\eta_1}(KH_{a.z.}+KH_{e.s.}+KH_{s.s.}) \tag{11}$$

(ε_{Pu} represents the relative plutonium loss in the external cycle, expressed in fractions of unity; $KH_{a.z.}$, $KH_{e.s.}$, and $KH_{s.s.}$ are the coefficients of accumulation of fissionable plutonium in the active zone, in the end shields, and in the side shields, respectively, expressed in grams of plutonium per gram of all separated nuclei);

$$\left.\begin{aligned} a_{11} &= (1-\varepsilon_{Pu})\,\frac{0.39KH_2\varphi_2}{\eta_2}\,; \\ a_{12} &= (1-\varepsilon_{Pu})\,\frac{0.39KH_3\varphi_3}{\eta_3}\,, \end{aligned}\right\} \tag{12}$$

where the coefficients KH are expressed in grams of plutonium fission per gram of separated nuclei and the indices 2 and 3 refer to fast converters and thermal converters, respectively.

Under the conditions we have adopted, when only the plutonium accumulated in the system is used until plutonium is obtained from the regenerate, i.e., when $t \leq \theta = \min\{\theta_2, \theta_3\}$, we have $x_1 = 0$. The plutonium balance equation becomes meaningless in this case and can be used only for $t > \theta$.

Thus, the plutonium balance equation has the following form:

$$a_{7t}x_{1t} + a_{8t}\sum_{\tau=\theta+1}^{t-\theta_0} x_{1\tau} - a_{9(t-\theta)}\sum_{\tau=\theta+1}^{t-\theta} x_{1\tau} - a_{10(t-\theta_2)}\sum_{\tau=1}^{t-\theta_2} x_{1\tau} - a_{11(t-\theta_3)}\sum_{\tau=1}^{t-\theta_3} x_{3\tau} = 0. \tag{13}$$

The electrical power generation balance equation has the following form:

$$\sum_{i=1}^{3}\sum_{t=1}^{T}\varphi_{it}\sum_{\tau=1}^{t} x_{it} = E_{ct}, \tag{14}$$

where φ_i is expressed in h/yr; E_{ct} is the total amount of electrical energy generated by all the nuclear power stations, beginning at time t = 0, expressed in 10^6 kWh/yr.

In all equations, we have $a_{it} \geq 0$; $x_{it} \geq 0$; $a_{i0} = 0$; $x_{i0} = 0$. The coefficients a_{it} are independent of the variables x_{it}, and therefore the equations can be solved as a system of three linear equations in three unknowns.

Substituting the solution of the system into the target function yields the amount of natural uranium consumed. The nuclear power industry structure obtained as a result of the solution is the optimal structure with respect to natural uranium consumption.

The choice of the dynamics of the variation in the plant power utilization factor (φ_{it}) for different types of reactor should be studied in somewhat more detail. The model provides for different methods of determining this factor. In the first method, φ_{it} is defined on the basis of processing and extrapolation of experience acquired in operating thermal electric power stations; in this case the φ_{it} are independent of the x_{it}, and the model is linear. In the second method, the φ_{it} for breeders is assumed to take on its maximum possible value (having due regard, of course, for the requirements of the power system), while the φ_{it} for converters is selected in the model itself, as a function of the structure of the system in previous years, and the difference between the φ_{it} values is specified. This method is somewhat approximate, since, strictly speaking, φ_{it} will depend on x_{it}. Lastly, there is a variant in which several possible operating regimes are specified for converters of different types, and in the model we take the optimal combination of the specified regimes from the point of view of minimizing the target function. In this case the model is again linear.

In the above model, fast breeders are put into operation as plutonium is accumulated in the system. It is a matter of great practical interest, however, to investigate a model in which the breeders are put into operation in large numbers simultaneously, starting in some specified year, using plutonium accumulated earlier in the nuclear power system. Such a situation can develop in a nuclear power industry, since its development is based almost exclusively on thermal reactors, whereas considerable time is required for developing single experimental or pilot-plant power stations using fast breeders. During this period of time it is impossible to introduce a large number of high-power fast reactors into the inudstry. This is an additional restriction imposed on the development of a nuclear power industry where the internal characteristics are taken into consideration. We assume that the fast breeders will be brought into operation starting from some particular time (year) ϑ ($\theta + 1 < \vartheta < T$). Up to the year ϑ, naturally, $x_1 = 0$. In this case the fast-breeder power that can be introduced in the year ϑ, on the basis of the available reserves of plutonium, will be

$$x_{1\vartheta} = \sum_{t=\theta+1}^{\vartheta}\frac{a_{10(t-\theta_2)}\sum_{\tau=1}^{t-\theta_2} x_{2\tau} + a_{11(t-\theta_3)}\sum_{\tau=1}^{t-\theta_3} x_{3\tau}}{a_{8t}}. \tag{15}$$

If ϑ is greater than θ, then $x_{1\vartheta}$ may exceed $a_{7\vartheta} = N_\vartheta$, the total amount of power introduced in the entire nuclear power industry during this year. Then $x_{1\vartheta} = a_{1\vartheta}$, while $x_{2\vartheta} = x_{3\vartheta} = 0$, i.e., no converters are put into operation in the year ϑ.

The excess plutonium produced in the system during this year should be used during the following year, $\vartheta + 1$. For the year $\vartheta + 1$ we solve an ordinary system of equations, but to the value $x_{1(\vartheta+1)}$ found from the second equation, we add the quantity $x_{1\vartheta} - a_{7\vartheta}$. Then we find $x_{2(\vartheta+1)}$ and $x_{3(\vartheta+1)}$ in the usual manner. If the value of $x_{1(\vartheta+1)} + x_{1\vartheta} - a_{7\vartheta}$ is greater than $a_{7(\vartheta+1)}$, then the excess plutonium will be used in the year $\vartheta + 2$, and so on.

If in the solution of the ordinary system of equations (the introduction of breeders as plutonium is accumulated in the system) in the year t' the value of $x_{1t'}$ exceeds $N_{t'}$, then we follow a procedure analogous to the one described above. If the number of types of breeder reactors is greater than three, then additional independent variables are introduced, in a manner analogous to x_1, into the target function and the equations; if the new reactors are converters, we add variables which are introduced in a manner analogous to x_2 and x_3.

Consideration has also been given to the case in which a nuclear power industry is developed on the basis of two types of reactors: a fast breeder and a thermal or fast converter. Here x_{1t} and x_{2t} are determined from the plutonium and power balance equations, and the electrical energy generation balance equation is used to select φ_{2t} for given values of φ_{1t} and E_{ct}.

Taking Account of the Diversity Factor in the Demand for Natural Uranium. The mathematical model described above enables us to consider a large number of variants of possible development of a nuclear power industry within the framework of the CMEA on the basis of a specified demand for original nuclear fuel. Obviously the different variants may differ not only in their integral demand but also in their differential demand for natural uranium.

At the same time, it is known that it makes a difference to a state (or a group of states) whether a specified amount of natural uranium is required at present or will be required at some future time. In other words, the economic significance of the demand for each unit of natural uranium for the years of the forecast period should be estimated with a coefficient less than 1, and this demand should be reduced to the value for the beginning or the end of the forecast period. In this case, when we reduce the integral requirement for natural uranium to the beginning of the forecast period, the target function (1) can be written as

$$\sum_{i=1}^{I} \sum_{t=1}^{T} G_{it} (1 + \sigma_t^{fp})^{1-t}, \tag{16}$$

where σ_t^{fp} is the norm used for taking into consideration the uranium requirement for the years of the forecast period. From the above considerations, we see that the value of σ_t^{fp} will be determined by the efficiency of utilization of natural uranium in the nuclear power industry for the condition of optimal development (from the viewpoint of minimum consumption of natural fuel).

The model described above enables us to verify how the structure of a nuclear power industry is affected by factors such as the coefficient of plant power utilization, the physical and heat-engineering characteristics of the reactors and of the fuel cycle (depth of burnup, reproduction of secondary fuel, efficiency of the fuel residence time in the cycle, U^{235} content of the tailings of the separation plant, etc.). The need to replace spent atomic power stations must be taken into account.

In addition, the model provides for the possibility of stockpiling the plutonium accumulated in the system and introducing breeders using this plutonium when it is practical to do so. The choice of nuclear power station operating regimes has been shown to be possible. At the output of the calculation program there is also a provision for obtaining a number of dynamic indicators from the values of enriched-fuel consumption for different types of reactors, the amount of spent fuel being sent to regenerating plants, and other indicators of a developing nuclear power system. Calculations have shown that the model corresponds to the problems of normative and investigative forecasting.

OUTFLOW OF RADIOACTIVE NOBLE GASES FROM THE COOLANT INTO THE GAS SPACES OF THE FIRST CIRCUIT OF THE BOR-60 REACTOR

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During the operation of reactors with a sodium coolant under conditions in which the hermetic state of the fuel elements may be disrupted, especially when using fuel elements involving emission of gaseous fission products, it is very important to know the proportion of gaseous radioactive products passing from the coolant into the gas spaces of the reactor. The level of radioactivity in the gas spaces determines the protective measures which have to be applied to the gas supplies under normal operation and the demands made upon the gas purification system, as well as the system for monitoring the vacuum tightness of the cans. A description of the BOR-60 reactor, its parameters, purpose, and prospects are set out in [1, 2].

In order to study the distribution of the radioactive gases in the first circuit of the BOR-60 reactor we measured the activity of these in the gas spaces and in the sodium coolant. The activity of the noble gases in the gas spaces of the reactor and the circulating pump of the first circuit were measured in gas-spectrometric loops by a nonsampling method, using a scintillation γ-spectrometer with a collimator [3].

For a more detailed study of the isotopic composition and a determination of low activities of noble gases in gas samples we used a Ge(Li) detector (sensitive volume 25 cm^3, resolution with respect to Cs^{137} approximately 1.2%).

The activity of the Ne^{23} in the sodium coolant was determined by a nonsampling method in the tubular conduit of the first circuit for a thermal power of the reactor up to 100 kW.

Table 1 shows the specific activities of the noble gases in the gas spaces of the reactor and the pump.

The levels of activity of these gases in the gas spaces, and also the ratio of the activities, indicate that the fuel elements are hermetically sealed. The maximum burnup of the heavy nuclei in the latter extends to 3%.

The relationship between the activities of noble gases with different half lives in the gas spaces of the reactor and the pump enables us to calculate the probability of gas leakage from the coolant. The equation for the transfer of radioactive noble gases in the first circuit, allowing for the two pumps involved in this, may be written in the form

$$\begin{aligned}\frac{dN_i^c}{dt} &= c_i - \lambda_i N_i^c - a_i N_i^c - 2b_i N_i^c + d_i N_i^r + 2f_i N_i^p;\\ \frac{dN_i^r}{dt} &= a_i N_i^c - \lambda_i N_i^r - d_i N_i^r;\\ \frac{dN_i^p}{dt} &= b_i N_i^c - \lambda_i N_i^p - f_i N_i^p;\end{aligned} \qquad (1)$$

where N_i^c, N_i^r, N_i^p are the number of atoms of the i-th isotope in the coolant, the gas space of the reactor, and the gas space of the pump, respectively; c_i is the rate of access of the i-th isotope, sec^{-1}; λ_i is the decay constant of the i-th isotope, sec^{-1}; $a_i b_i$ are coefficients characterizing the probability of a leakage of the i-th isotope from the coolant into the gas spaces of the reactor and pump, respectively, sec^{-1}; $d_i f_i$ are coefficients characterizing the probability of the coolant capturing the i-th isotope from the gas space of the reactor and the gas space of the pump, respectively, sec^{-1}.

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TABLE 1. Isotopic Composition of Radioactive Noble Gases in the Gas Spaces of the Reactor and Pump for a Reactor Power of 40 MW

Isotope	Half life $T_{1/2}$	Specific activity, Ci/liter	
		gas space of reactor	gas space of pump
Ne^{23}	37,5 sec	$(3\pm0,5)\cdot10^{-2}$	$(5\pm0,8)\cdot10^{-5}$
Ar^{41}	1,83 h	$(1,3\pm0,2)\cdot10^{-3}$	$(2,5\pm0,5)\cdot10^{-6}$
Kr^{85m}	4,5 h	$(3,8\pm0,5)\cdot10^{-7}$	$(1\pm0,2)\cdot10^{-7}$
Kr^{87}	1,27 h	$(4\pm0,5)\cdot10^{-7}$	$(1\pm0,2)\cdot10^{-7}$
Kr^{88}	2,79 h	$(1\pm0,2)\cdot10^{-6}$	$(2,1\pm0,3)\cdot10^{-7}$
Xe^{133}	5,3 d	$(4\pm0,5)\cdot10^{-6}$	$(8,5\pm1)\cdot10^{-7}$
Xe^{133m}	2,26 d	$(1,2\pm0,2)\cdot10^{-8}$	—
Xe^{135}	9,15 h	$(2,7\pm0,4)\cdot10^{-6}$	$(6,2\pm1)\cdot10^{-7}$
Xe^{135m}	15,7 min	$(8\pm1)\cdot10^{-8}$	—
Xe^{138}	17 min	$(2,5\pm0,5)\cdot10^{-7}$	—

Note: 1) Flow of coolant in the first circuit 1000 m^3/h. 2) Temperature of coolant at reactor outlet 450°C. 3) The system includes a cold trap for sodium oxides.

TABLE 2. Coefficients Characterizing the Leakage Probabilities of Noble Gases from the Coolant into the Gas Spaces of the Reactor and Pump, and Mean Times Spent by These Gases in the Coolant

Element	a, h^{-1}	b, h^{-1}	τ, h
Xe	$0,98\cdot10^{-1}$	$1,1\cdot10^{-2}$	8,3
Kr	$1,08\cdot10^{-1}$	$1,2\cdot10^{-2}$	7,6
Ne	$1,96\cdot10^{-1}$	$1,4\cdot10^{-4}$	5.1

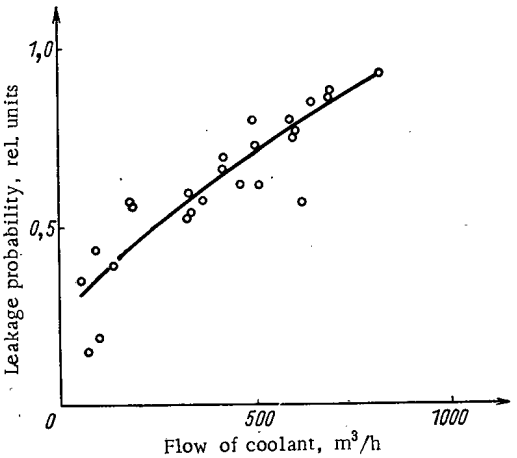

Fig. 1. Dependence of the coefficient characterizing the probability of a leakage of Ne^{23} into the gas space of the reactor on the flow of coolant in the first circuit (temperature of coolant 220°C).

Substitution of the experimental values (Table 1) into the system of equations (1) for three isotopes of one element (i, k, l), on the assumption that $a_i = a_k = a_e$, $b_i = b_k = b_e$, $d_i = d_k = d_e$, $f_i = f_k = f_e$, for the steady-state case ($dN_i^c/dt = 0$; $dN_i^r/dt = 0$; $dN_i^p/dt = 0$), gives $d_i \ll a_i$ and $f_i \ll b_i$. This is also supported by the manner in which the coolant moves at the interface with the gas spaces. The solution of Eqs. (1) may be written in the form

$$Q_i^r = \frac{a_i}{\lambda_i + a_i + 2b_i} c_i;\quad Q_i^p = \frac{b_i}{\lambda_i + a_i + 2b_i} c_i, \tag{2}$$

where Q_i^r and Q_i^p are the activities of the i-th isotope in the gas space of the reactor and in the gas space of the pump, respectively.

For the fission products of U^{235}, when this contaminates the cans of the fuel elements and the coolant

$$\frac{c_i}{c_k} = \frac{\eta_i}{\eta_k}, \tag{3}$$

where η_i, η_k are the yields of the i-th and k-th isotopes for one fission of U^{235}. Then, on the assumption that the coefficients characterizing the leakage probabilities are equal, we derive the following from Eq. (2) with due allowance for (3) for two isotopes i and k of one element:

$$a = \frac{\lambda_i A_i^r \eta_k - \lambda_k A_k^r \eta_i}{\eta_i \left(A_k^r + 2A_k^p \frac{V^p}{V^r}\right) - \eta_k \left(A_i^r + 2A_i^p \frac{V^p}{V^r}\right)};\quad b = \frac{A_i^p}{A_i^r} = \frac{V^p}{V^r} a, \tag{4}$$

where A_i^r, A_i^p are the specific activities of the i-th isotope in the gas space of the reactor and pump, respectively, V^r, V^p are the volumes of the gas in the reactor and the pump, liters. For the BOR-60 reactor $V^p = 0.5$ m^3, $V^r = 1.2$ m^3.

The results of our calculations for the isotopes Xe^{133}, Xe^{135m}, Kr^{85m}, and Kr^{88} are given in Table 2, in which the mean period spent in the coolant was calculated from the equation

$$\tau = \frac{1}{a + 2b}. \tag{5}$$

If we use Eq. (2) together with the experimental values of the specific activities of Xe^{133}, Xe^{135}, Kr^{85m}, and Kr^{88} (Table 1) and the calculated coefficients a and b (Table 2), we may find the number of U^{235} fissions

taking place on the surface of the fuel elements and in the coolant, $3.6 \cdot 10^9$ fissions/sec, which corresponds to a maximum surface contamination of the fuel elements with $1.2 \cdot 10^{-9}$ g of U^{235}/cm^2.

In order to verify the computing method under consideration and also the constants so obtained, we used Eq. (2) and calculated the specific activities of the isotopes Kr^{87}, Xe^{133m}, Xe^{135m}, Xe^{138} for the gas spaces. All the calculated specific activities agreed with the experimental values, within the limits of measuring error (Table 1).

The coefficients characterizing the leakage probability of neon into the gas spaces from the coolant were determined by reference to the Ne^{23} isotope, it being remembered that in the steady state

$$c_i = Q_i^c + Q_i^r + 2Q_i^p.$$

From Table 1, allowing for the volumes involved, we have $Q^r(Ne^{23}) = 36$ Ci, $Q^p(Ne^{23}) = 2.5$ Ci.

The nonsampling measurements in the tubular conduit of the first circuit showed that, for a reactor power of 40 MW, $Q^c(Ne^{23}) = 1.2 \cdot 10^4$ Ci. Hence for Ne^{23} $Q^r \ll Q^c$, $Q^p \ll Q^c$. It follows from Eq. (2) and the results of Table 1 that $b \ll a$, so that Eq. (2) simplifies

$$a = \frac{Q^r}{Q^c}\lambda; \quad b = \frac{Q^p}{Q^c}\lambda. \tag{6}$$

The coefficients a and b calculated from Eq. (6) for Ne^{23} are shown in Table 2. In order to compare the leakage of Ne^{23} and the gaseous U^{235} fission products, a correction must be introduced for the decay of Ne^{23} during the motion through the first circuit.

For a flow rate of 1000 m^3/h the circulation period of the coolant in the first circuit is 85 sec; then the specific activity of the Ne^{23} at the entrance into the active zone is five times smaller than at the outlet.

For all the gaseous U^{235} fission products considered, the specific activities in the reactor tank and pump are equal to their mean specific activities in the coolant of the first circuit. For Ne^{23} the specific activity in the reactor tank is twice as great as the mean specific activity, while in the pump tank it equals the mean specific activity. Thus the coefficient characterizing the leakage probability of neon into the gas space of the reactor would be $9.8 \cdot 10^{-2}$ h^{-1} for a well-agitated coolant, in agreement with the foregoing values for xenon and krypton.

At the boundary with the pump gas space, the coolant is 85 times poorer in Ne^{23} than the circulating coolant in the pump. This is due to the fact that in the reactor and pump of the BOR-60 installation the coolant passes to the interface with the gas spaces along annular gaps of fairly small dimensions. The height of the coolant in the gaps is considerable, and in the pump it is two or three times greater than in the reactor. This reduces the proportion of short-lived noble gases passing into the gas space of the pump.

The experimentally-obtained values of the specific activities of Ne^{23} in the gas space of the reactor were inserted into Eq. (6), and as a result of this the coefficients characterizing the leakage probability of Ne^{23} were calculated in relation to the flow of coolant (see Fig. 1). With increasing rate of flow the leakage probability also increases as a result of the better exchange of coolant in the gaps. An increase in the temperature leads to a reduction in the leakage probability into the gas space of the reactor.

If the coolant does not capture any gas bubbles from the gas spaces, as is indicated by the solution of Eqs. (1), with due allowance for the experimental results and a consideration of the structural features of the reactor and pump, the amount of argon in the sodium may also be estimated. After closing the gas line connecting the gas spaces of the reactor and the pump, the activity of Ar^{41} in the gas space of the pump remained undiminished. This indicates that the argon passes into the gas space of the pump from the coolant (the activation of the argon in the gas space of the pump may be neglected). If we then take the coefficient characterizing the probability of a leakage of argon from the coolant into the gas space of the pump as equal to the corresponding coefficient for krypton and xenon, Eq. (4) gives $C_i(Ar^{41}) = 1.25 \cdot 10^9$ sec^{-1}.

The passage of Ar^{41} into the coolant is due to the activation of the Ar^{40} dissolved in the coolant in the reactor (the passage of Ar^{41} into the coolant during the dissolution of the gas may be neglected). In this case we may use the equation

$$N(Ar^{40}) = \frac{c_i(Ar^{41})}{\Phi\sigma},$$

(where Φ is the neutron flux in the reactor, $cm^{-2} \cdot sec^{-1}$; σ is the cross section of the reaction $Ar^{40}(n, \gamma)Ar^{41}$ in cm^2) to determine the amount of argon in the coolant. In the BOR-60 reactor we determined the product $\Phi\sigma$ for the reaction $Na^{23}(n, \gamma)Na^{24}$ experimentally and converted the result to the $Ar^{40}(n, \gamma)Ar^{41}$ reaction by making use of the ratio of the cross sections of these reactions for the corresponding neutron spectrum. The estimate showed that the amount of argon in the sodium coolant for a mean temperature of 400°C was $4 \cdot 10^{-6}$ wt. %.

In some experiments aimed at discovering the effect of the oxide cold trap on the isotopic composition of the noble gases, we found no increases in the activity of the long-lived gas when the trap was disconnected, as was observed in [4]. This indicates that the precursors of the xenons, the iodines, occur mainly on the surface of the construction materials in the fuel-element cans.

The foregoing analysis leads to the following conclusions.

1. For the 3% burnup of the heavy nuclei which is reached in the active zone, no fuel elements deficient in vacuum-tightness occur; the levels of activity of the noble gases are determined by the initial contamination of the first circuit by the fuel ($1.2 \cdot 10^{-9}$ g of U^{235}/cm^2).
2. The capture of gas bubbles by the coolant from the gas spaces of the BOR-60 installation is a very unlikely event.
3. For all the radioactive noble gases with $T_{1/2} > 10$ min the mean period spent in the coolant is constant and equal to ~8 h.
4. The proportion of short-lived noble gases passing from the coolant into the gas spaces is low (for $T_{1/2} > 10$ min no greater than 3%).
5. The amount of argon dissolved in the sodium at 400°C is about $4 \cdot 10^6$ wt. %.
6. The methods of computation and spectrometric measurement here employed are in practice very accurate, sensitive, and reproducible, and may be used for testing the vacuum-tightness of fuel-element cans in reactors with molten metal coolants.

The foregoing picture of the outflow of gases from the coolant in no way contradicts the generally-accepted view [5]; however, the probability of a leakage of radioactive noble gases from the coolant may differ considerably for different reactors [4, 6], depending on their structural peculiarities.

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POSSIBLE RECUPERATION OF THE ENERGY OF A BEAM OF CHARGED PARTICLES IN A SYSTEM OF TAPERED DIAPHRAGMS

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A system for the direct conversion of the energy of charged particles leaving a mass of plasma in an open trap through the magnetic stoppers was described in [1-3]; after the conversion of the transverse energy into longitudinal energy and the separation of the electrons, the rarefied ion flux passes into a recuperator in which the ions are retarded and energy-separated by means of steady-retarding and alternating-deflecting electrostatic fields. Both calculations and physical experiments show [3] that the efficiency of this recuperator for a rarefied beam (one free from the influence of space charge) is about 85%.

As we shall later show, the influence of space charge may conveniently be estimated from the ratio d/r_{d_i} of the beam width d at the entrance to the Debye radius $r_{d_i} = \sqrt{W/2\pi e^2 n_{0i} Z^2}$, where W is the maximum initial energy Z of the charged ions; n_{0i} is the initial beam density.

For the reactor conditions of [3] (in which the energy W of the deuterons equals 800 keV, d = 1 m, j $= 4 \cdot 10^{-4}$ A/cm^2), the ratio d/r_{d_i} equals 0.18. In the absence of external fields the beam is initially parallel, but after passing through a distance of about 5.5d it increases its width by a factor of 1.5 under the influence of its own space charge. According to the results of a quantitative experiment, the efficiency of the Post system [2] with an alternating-deflecting field is in this case 70-75%.

If the deflecting field is made steady, in the same way as the retarding field, the particles retain the energy $W_\perp$ of their motion perpendicular to the total electrostatic field vector E along their whole trajectory, which makes the flow of particles more "rigid" and easy to control than using an alternating-deflecting field, when the particles in the flow almost completely lose their "memory" regarding the initial parts of their trajectory.

In the next section we shall present the results of a computer experiment involving the calculation of ion trajectories in a system of tapered diaphragms (proposed in [4]) under conditions in which the space charge exerts a substantial influence. The nature and results of the corresponding physical experiment are presented in the second section.

Calculation of the System Allowing for the Effects of Space Charge. The efficiency of the direct conversion of charged-particle energy in a system of tapered diaphragms was calculated and measured in [4] for a rarefied beam.

Let us now present some additional theoretical estimates. We note that if the angle of entry into the electric field $\pi - \alpha$ is optimized for particles with a slowing down length corresponding to an energy W_0 and $\varepsilon_{min} = \varepsilon_0$ it will no longer be optimum for particles with other energies, and the relative losses ε suffered by particles with a longitudinal energy W will be determined as follows in terms of W, W_0, and ε_0 as in Eqs. (3)-(5) in [4]:

$$\varepsilon(\varepsilon_0, W_0; W) = \frac{\varepsilon_0}{2}(1 + W_0^2/W^2). \quad (1)$$

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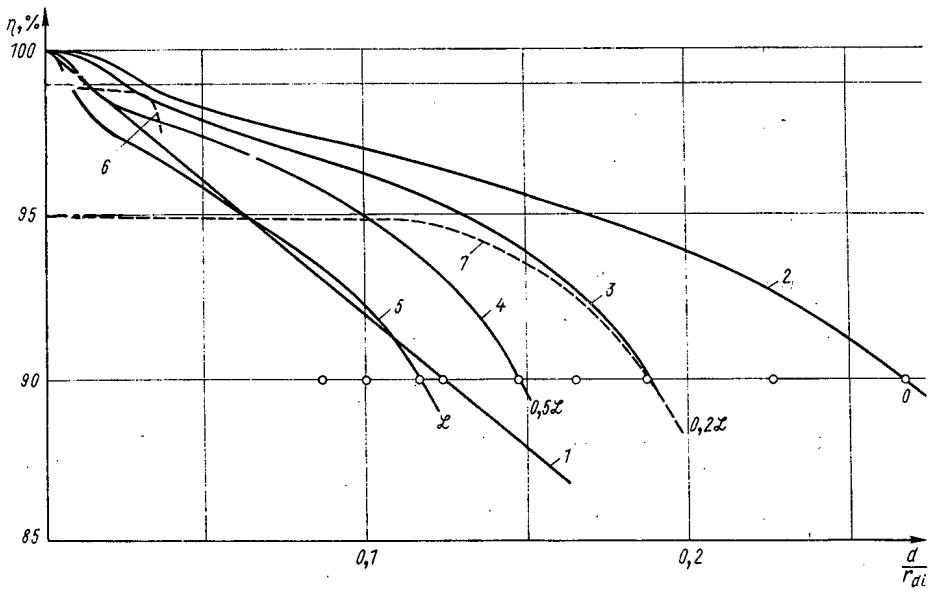

Fig. 1. Recuperation efficiency as a function of d/r_{d_i}: 1) uniform parallel beam; 2) $\delta L = 0$; 3) $\delta L = 0.2\mathscr{L}$; 4) $\delta L = 0.5\mathscr{L}$; 5) $\delta L = \mathscr{L}$; 6) system optimized for operation with small d/r_{d_i}, $\delta L = 0.2\mathscr{L}$; 7) system optimized for operation with large d/r_{d_i}, $\delta L = 0.2\mathscr{L}$.

If we take $W_0 = (3/4)W_{max}$ (W_{max} is the maximum energy of the particles in the beam), for energies $(1/2)W_{max}$ and W_{max} the relative losses will be $1.63\,\varepsilon_0$ and $0.78\,\varepsilon_0$. The system is not optimum for the particles with $W = W_{max}$, because by appropriately selecting the incident angle α while retaining the length of the system $\mathscr{L}$ a value of $\varepsilon = 0.71\,\varepsilon_0$ might be obtained for these.

Let us estimate the effect of space charge on the efficiency $\eta = 1 - \varepsilon$. A single-energy parallel plane beam with an initial velocity v_0, width d, and density n is completely retarded in a uniform retarding field with a slowing down length λ in a time $t = 2\lambda/v$, the final thickness being related to the original by

$$d(\lambda) = d(1 + 2\lambda^2/r_{d_i}^2),\quad r_{d_i}^2 \equiv \frac{v^2 M}{4\pi e^2 n}. \tag{2}$$

Hence the spread Δx in the vertices of the extreme ion trajectories relative to the central trajectory in a system of tapered diaphragms will be approximately equal to $\frac{d}{2}\left(1 + \frac{2\lambda^2}{r_{d_i}^2}\right) + \lambda\theta$, where θ is the initial angle of divergence of the beam. By analogy with [4] we may estimate $\tan^2\alpha_{opt}$ and ε_{min} as follows (allowing for the recuperation of the transverse energy $W_\perp$)

$$\operatorname{tg}^2\alpha_{opt} = \frac{1}{2\lambda}\sqrt{\frac{\varkappa}{\varepsilon_\perp}}\left[\frac{d}{2}\left(1 + \frac{2\lambda^2}{r_{d_i}^2}\right) + \lambda\theta;\right.$$
$$\left.\varepsilon_{min} = \sqrt{\varkappa\varepsilon_\perp}\left[\frac{d}{2\lambda}\left(1 + \frac{2\lambda^2}{r_{d_i}^2}\right) + \theta\right],\right. \tag{3}$$

where $\varepsilon_\perp$ is the efficiency of recuperation of the transverse energy; $\varkappa$ is a coefficient depending on the initial distribution of the particles over the beam cross section $n_0(x)$ ($x = 0$ is the center of the beam):

$$\varkappa \approx \frac{4}{d^2}\frac{\int_{-d/2}^{d/2} x^2 n_0(x)\,dx}{\int_{-d/2}^{d/2} n_0(x)\,dx}. \tag{4}$$

In the case of a uniform distribution of particles over the cross section of a plane beam $\varkappa \approx 1/3$, for a circular beam $\varkappa \approx 0.27$, for a plane beam with $n_0(x) \sim |x|$, $\varkappa \approx 1/2$, for $n_0(x) \sim 1 - \frac{2|x|}{d}$ $\varkappa \approx \frac{1}{6}$.

TABLE 1. Positions of the Centers of the Focusing Intervals

Parameters	First system				Second system				Third system			
	$\varepsilon_p = 0{,}02;\ \frac{d}{\lambda\varepsilon} = 0{,}0346$				$\varepsilon_p = 0{,}02;\ \frac{d}{\lambda\varepsilon} = 0{,}06$				$\varepsilon_p = 0{,}1;\ \frac{d}{\lambda\varepsilon} = 0{,}3$			
$\delta L/\mathscr{L}$	0	0,2	0,5	1	0	0,2	0,5	1	0	0,2	0,5	1
W_{max}	0,50	0,80	1,15	1,40	0,55	0,75	1,05	1,45	0,45	0,70	0,90	1,25
$^3/_4 W_{max}$	0,45	0,65	1,05	1,35	0,50	0,70	0,95	1,30	0,35	0,60	0,85	1,15
$^1/_2 W_{max}$	0,35	1,30	—	—	0,40	0,55	—	—	0,30	0,60	—	—

The quantity ε_{min} is determined from Eq. (3) as a function of λ/r_{d_i} (fixed ratio d/r_{d_i}) and has a minimum

$$\varepsilon_{min} = \sqrt{\varkappa\varepsilon_\perp}\left(\frac{\sqrt{2}\,d}{r_{d_i}} + \theta\right) \tag{5}$$

on satisfying the conditions

$$\frac{\lambda}{r_{d_i}} = \frac{1}{\sqrt{2}} = 0.706;\quad \mathrm{tg}^2\alpha_{opt} = \frac{1}{2}\sqrt{\frac{\varkappa}{\varepsilon_\perp}}\left(\frac{\sqrt{2}\,d}{r_{d_i}} + \theta\right). \tag{6}$$

Thus the geometry of the doubly-optimized system (slowing down length λ and field slope α) is determined by the parameters of the entrance beam r_{d_i}, $\varkappa$, d, θ, and the efficiency depends on the dimensionless parameters d/r_{d_i}, $\varkappa$, θ.

It follows from conditions (5) and (6) that the focusing of the beam at the end of the slowing down length ($\theta = -d/\lambda$) should lead to a rise in η and that for a uniform plane parallel beam ($\varkappa = 1/2$, $\theta = 0$), even without recuperation of the transverse energy ($\varepsilon_\perp = 1$), the ε of the optimum system is determined by the relation $\varepsilon = 0.815\ d/r_{d_i}$ (curve 1, Fig. 1).

In order to elucidate the possibility of increasing η by using preliminary focusing of the beam, we carried out a computer calculation of the ion trajectories in the system.

The space-charge field was simulated as being equivalent to the field of a charge uniformly distributed within the beam over each cross section of the latter. On passing from section to section, the charge varied in accordance with the beam cross section and particle velocity. The beam boundaries were found by successive approximations (three of these were sufficient).

For particles of three different energies (W_{max}, $(3/4)W_{max}$, $(1/2)W_{max}$) we calculated three pencils of trajectories in each case, starting from the edges and from the middle of the initial section. The vectors of the initial velocities of the extreme trajectories of the lateral beams were directed toward the ends L_1 and L_2 of the range of focusing δL on the axis of the beam, and those of the middle trajectories toward the center of δL.

The initial velocities of the extreme trajectories of the central beam were directed toward the points of intersection (most distant from the axis) of the aiming directions for the extreme lateral beams. The central trajectory was directed along the beam axis.

The boundaries L_1 and L_2 of the focusing range δL were chosen by successive approximations in such a way as to give the greatest efficiency η for a specified value of $\delta L = L_2 - L_1$. Curve 2 in Fig. 1 illustrates the highest possible level of η, for which focusing is required to occur at a single point ($\delta L = 0$). Curves 3-5 correspond to a δL of $0.2\mathscr{L}$, $0.5\mathscr{L}$, $1\mathscr{L}$. It is assumed that the transverse energy $W_\perp$ is not recuperated ($\varepsilon_\perp = 1$).

We calculated three specific systems: 1) for $\varepsilon = 0.02$ and $d = 0.0346\lambda$, we used Eq. (6) with $\varkappa = 1/3$; 2) the system for $\varepsilon = 0.02$ and $d = 0.06\lambda$ was better in the case of a strong space-charge effect (for a weak space-charge effect the focusing interval was smaller in this case than in the first system); 3) the third system was calculated for a smaller efficiency from the very beginning, in order to increase the reserve of stability with respect to the space-charge effect; it was similar to the second system in relation to $d/\lambda\varepsilon$.

The centers of the focusing intervals are indicated for each pair of values (W, δL) of systems 1, 2, and 3 in Table 1. It follows from this table that, on contracting the interval δL to zero, correspondingly increasing the space-charge effect to the greatest acceptable level, the focal length contracts to roughtly half the slowing down length.

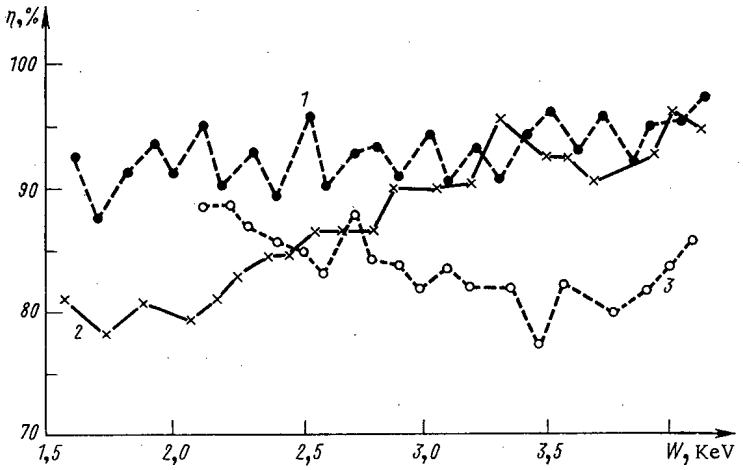

Fig. 2. Experimental dependence of the recuperation efficiency of the energy of a beam of N^+ ions on the beam energy for various α: 1) $\eta_{av} = 93\%$, $\alpha = 6°20'$; 2) $\eta_{av} = 88\%$, $\alpha = 4°50'$; 3) $\eta_{av} = 83\%$, $\alpha = 8°47'$.

Each particular arrangement using the focusing principle can only have an efficiency η approaching the curves corresponding to the highest values of η over a fairly narrow range of values of d/r_{d_i}, those to which it is "tuned." Figure 1 shows the efficiency η for two arrangements respectively calculated for working with small (curve 6) and large (curve 7) values of d/r_{d_i}. We see by comparing curves 6 and 7 that the system with the lower calculated efficiency is much more stable toward fluctuations in space-charge density, because each particle has a considerable energy $W_\perp$ right up to the point at which it approaches the focusing diaphragms; this enables it to overcome the repulsive action of the space-charge forces more successfully. For the examples calculated we first chose the relation between r_{d_i} and λ in the form of (6) but found that for a stronger space-charge effect it was better to take $\lambda < r_{d_i}/\sqrt{2}$.

We see from Fig. 1 that, even without using recuperation of the transverse energy, the system of tapered diaphragms may have $\eta \approx 0.91$-0.94 with $\delta L \approx 0.1$-0.24 for the "reactor" value of $d/r_{d_i} = 0.18$.

Description and Results of the Physical Experiment. In order to verify the theoretical considerations just discussed, we calculated a specific model of a direct-conversion system (the geometry of this system is indicated in [4]) for perfectly specific beam input parameters.

We estimated the effect of the angular deviation of the beam axis from the axis of the system on the efficiency of the latter. A theoretical estimate gives the following expression for the case of a uniform particle distribution over the cross section of a circular beam:

$$\varkappa(\delta\alpha) \approx 0.27 + 4\lambda^2 \frac{(\delta\alpha)^2}{d^2}. \qquad (7)$$

In order to verify this, we made some experiments with a beam of N^+ ions having parameters $d = 0.1$ cm, $\theta = 1°$, current density $j = 5 \cdot 10^{-4}$ A/cm^2, and $W_{max} = 4$ keV.

The geometry of the system and the potential distribution were left as before [4], except for the diaphragm D, where the entrance aperture had a diameter of $d = 0.1$ cm. The construction of the model enabled us to rotate the system in a plane perpendicular to the direction of the taper of the diaphragms. The particle energy varied over the range 1.5–4 keV. The spectrum of the N^+ ions was monoenergetic; the beam was modulated at a frequency of 50 cps.

For a weak space-charge effect (determined by the nonoptimum value of the ratio r_{d_i}/λ), the optimum angle α_{opt} is determined from (3). For a beam with the parameters just indicated this equals $\alpha_{opt} = 6°20'$. The experimentally-measured curve of retardation efficiency for this angle is shown in Fig. 2 (curve 1). The discrete disposition of these points is attributable to the finite number of diaphragms employed.

The optimum angle $\alpha_{opt} = 6°20'$ was calculated for a system $\lambda = 15$ cm long. For different values of the slowing down length, i.e., for particles with different initial energies, the optimum angle has different

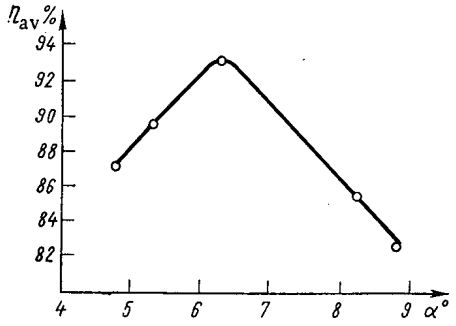

Fig. 3

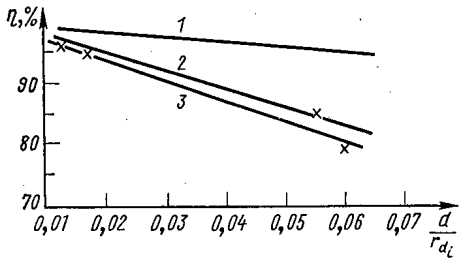

Fig. 4

Fig. 3. Dependence of the mean efficiency of recuperation on the angle α.

Fig. 4 Dependence of the recuperation efficiency on d/r_{d_i}: 1) theoretical relationship for $\theta = 0$ and $\lambda / r_{d_i} = 0.7$, calculated from Eq. (5); 2) theoretical relationship for $\theta = 1°$ and $\lambda/r_{d_i} = 3.2$, calculated from Eq. (3); 3) experimental relationship for $\theta = 1°$ and $\lambda/r_{d_i} \approx 3.2$.

values. Whereas the optimum condition with $\varepsilon_{min} = \varepsilon_0$ is achieved by suitably choosing the angle α_{opt} for particles with energy W_0, in the case of a rarefied beam, for particles with energy W, the value of ε is determined from Eq. (1). On the basis of this condition, the average theoretical efficiency in the energy range 1.5–4 keV equals 94%. According to the experimental measurements the value was 93%. In order to study the critical state of the system with respect to α, we measured the efficiency for various angles (Fig. 2). The relatively low efficiency for angles differing from the optimum value arose from the following circumstances. For $\alpha > \alpha_{opt}$ the particle trajectories lie at a small angle to the line of taper of the diaphragms, so that particles traveling quite close together fall on diaphragms spaced a long way apart from each other. Hence the currents of particles with one particular energy are distributed over four or five diaphragms at once, which reduces the direct conversion efficiency. For $\alpha > \alpha_{opt}$ the system has a high efficiency at high energies and correspondingly a low efficiency at low energies, as may readily be understood from the geometry of the particle trajectories. The low efficiency for $\alpha < \alpha_{opt}$ may be explained in a similar manner.

It follows from Eq. (7) that the angular deviation of the beam axis from the axis of the system in the x, y plane (the x coordinate is perpendicular to the beam axis, the y coordinate lies along it) by an amount $\delta\alpha$ leads to an increase in the value of $\varkappa$. The theoretically-expected efficiency, allowing for the change in $\varkappa$, was 89% for $\alpha = 8°47'$ and 91% for $\alpha = 4°50'$.

Figure 3 shows the manner in which the mean recuperation efficiency depends on the angle of entry of the beam into the system. We see that a slight deviation of α from the optimum value ($\delta \approx 3°$) reduces the efficiency to 80%. However, if we consider a beam with a divergence of $\theta \approx 6°$ and a weak space-charge effect, this is retarded with an efficiency greater than 80%, since most of the ions lie in the central part of the flow. Thus the divergence of the beam is less dangerous than the deviation of the entrance angle from its optimum value.

As already indicated, in the presence of space charge Eq. (3) has a minimum at $\lambda/r_{d_i} = 0.7$. If $\lambda/r_{d_i} > 0.7$, the slowing down time of the particle in the system increases, and hence so does the broadening of the beam under the influence of the space charge.

Our experimental investigation into the manner in which the efficiency of the installation varied with space charge involved the effects of both d/r_{d_i} and λ/r_{d_i} (Fig. 4). We see that, for reasonably small ratios $d/r_{d_i} \leq 0.06$, the dominant role in determining the conversion efficiency is played by the deviation of the slowing-down length from optimum, whereas a change of 0.014–0.06 in d/r_{d_i} produces only a slight change in the efficiency (4–5%).

We may draw the following conclusions from the foregoing discussion:

1. The state of a single-component beam before entering the recuperation system may reasonably be described by three-dimensionless parameters. The ratio d/r_{d_i} of the width of the beam d to the Debye radius r_{d_i}, calculated from the initial velocity and density of the ions, describes the influence of space charge. The angular spread of the initial velocities of the ions in radians θ at

each point of the initial cross section of the beam characterizes the possibility of increasing the recuperation efficiency by virtue of preliminary focusing. The ratio of the minimum energy of the particles to the maximum W_{min}/W_{max} (energy spread of the beam particles) determines the construction of the recuperator (whether it is to be in the form of a Faraday cylinder or a system of diaphragms).

2. The reactor conditions estimated earlier by R. Post correspond to $d/r_{d_i} \approx 0.18$; $W_{min}/W_{max} \leq 1/2$. The efficiency of a system of tapered diaphragms, allowing for the possible recuperation of the transverse energy and preliminary focusing, is then 90-95%, according to theoretical estimates (computer experiment).

3. The experimentally-measured efficiency of direct conversion for ion beams of small aperture ($d = 0.3$ cm, $\theta = 2°$) subject to a weak space-charge effect ($d/r_{d_i} = 0.014$) was 96% [4].

4. We have examined the effect of space charge in a system of tapered diaphragms over the range $d/r_{d_i} = 0.014$-0.06. Even for a nonoptimal choice of system parameters, the efficiency η equalled 85% in the region of greatest space-charge influence.

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RADIOACTIVE DETERMINATION OF IRON, COBALT, ZINC, SCANDIUM, CESIUM, AND ANTIMONY IN SOILS AND PLANTS

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UDC 621.039.8:543.7

The possibilities of instrumental activation analysis in biological soil studies were demonstrated, for example, in [1-3].

It is known that through thermal neutron activation of soil and plant samples, activities of more than 200 isotopes are induced in them which differ in the nature and energy of the radiation, in half-life and activity. In principle, simultaneous determination of a large number of isotopes immediately after activation is possible. Such a method is used comparatively rarely in practice and the analysis is carried out over a definite time span (activation time and lapse time) which makes it possible to increase the sensitivity and reliability of the analysis at the cost of an insignificant reduction in the number of simultaneously determined elements. From a practical viewpoint, it is advantageous to carry out the determination with respect to medium- and long-lived isotopes in developing techniques suitable for use in production-line analyses.

The purpose of this paper is to discuss the possibilities of soil and plant analysis for long-lived isotopes after the decay of the short- and medium-lived radioactive isotopes (sodium, potassium, manganese, etc.), which create considerable difficulty in the determination of other elements because of their high activity. A Ge(Li) detector with a volume of 7.3 cm^3 was used in conjunction with an amplifier assembly and a 1024-channel pulse height analyzer. Soil and plant samples weighing 200 and 500 mg, respectively, were irradiated along with standards in a flux of 10^{13} $n/cm^2 \cdot sec$ for 20 h. After lapse times of 15 and 30 days, measurements of the γ-ray spectra were started after first calibrating the equipment by means of emitters with radiations of known energy.

Figure 1 shows a typical γ-ray spectrum of a soil sample in the range 603-1408 keV, where lines from the isotopes Sb^{124}, Cs^{134}, Mn^{54}, Sc^{46}, Zn^{65}, Fe^{59}, Co^{60}, and Eu^{152} predominate. Spectra from plant samples were similar. For a shorter sample "cooling" time, a sharp line from the isotope Rb^{86} is observed in the spectra.

To separate unresolved peaks, ratios of the photopeaks in the standards were calculated and partial intensities computed by successive subtraction of normalized spectra.

Because the calculation of the height of the photopeak from radioactive zinc at 1115 keV is performed by subtraction from the total photopeak at 1115 + 1120 keV and because of the high ratio of scandium activity to zinc activity (10 – 1), we carried out a radiochemical analysis. The following technique was developed for this purpose: after activation, the samples were dissolved in a mixture of fluoric and sulfuric acids in the presence of a zinc carrier (5 mg). After decomposition of the samples, the solution was heated until the evolution of sulfur anhydride vapor stopped. The residue was dissolved in concentrated hydrochloric acid. Sodium citrate was added as a masking agent to prevent the extraction of iron. The zinc was extracted by dithiozon (diphenylthiocarbazone) in carbon tetrachloride from a weakly alkaline medium at a pH 8.3-9.3 [4]. An aliquot of the organic phase was activated again after measurement and then measured

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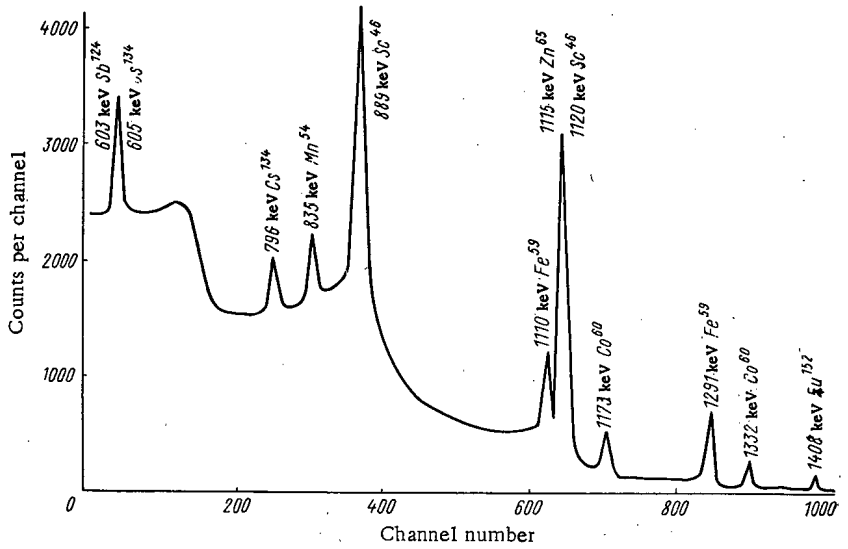

Fig. 1. γ-Spectrum of plant samples in the range 600-1410 keV after one month of "cooling."

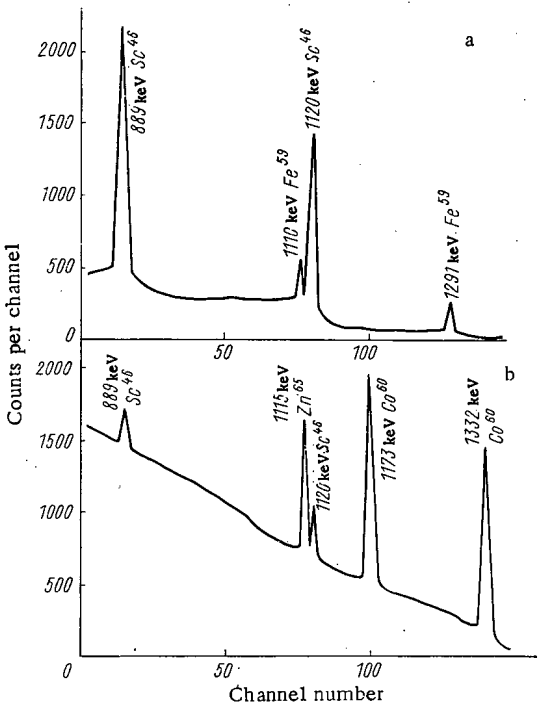

Fig. 2. γ-Spectra in (a) water and (b) organic phase (typical sierozem).

TABLE 1. Activation Determination of Zinc (in mg/kg)

Sample		Instrumental determination	Determination by radiochemical separation
Soil	№ 1	60,0±6,0	56,5±3,0
	№ 2	100,0±10,0	94,5±5,0
	№ 3	122,0±12,2	118,0±6,0
Cotton plant	№ 1	7,1±0,7	6,5±0,3
	№ 2	8,6±0,9	9,0±0,4
	№ 3	12,8±1,3	13,0±0,7

to determine the chemical yield, which was 60-80%. The factor for removal of scandium from zinc was greater than 100. Figure 2 shows γ-ray spectra for the separated phases. As is clear from the figure, the efficiency of this method was completely satisfactory.

The content of the elements specified was calculated by a comparison of photopeak areas in standards and samples. Analysis using an NaI(Tl) detector in accordance with a previously developed technique [5] was carried out for the comparison. Table 1 gives the results of the comparison of the determination of zinc by the instrumental method and by using radiochemical separation from scandium. As is clear from the table, the method of determination was completely satisfactory from the viewpoint of accuracy and simplicity of execution. Because of the use of the most intense lines [6], the method does not require extended times of measurement even when using comparatively small-volume semiconductor detectors.

The calculated sensitivity was $4 \cdot 10^{-7}$ g/g for the determination of cobalt, $8 \cdot 10^{-7}$ g/g for scandium, $2 \cdot 10^{-6}$ g/g for antimony, $1.4 \cdot 10^{-6}$ g/g for cesium, $4 \cdot 10^{-4}$ g/g for iron, $5 \cdot 10^{-5}$ g/g for zinc with the instrumental method and $5 \cdot 10^{-7}$ g/g for radiochemical separation, and $6 \cdot 10^{-10}$ g/g for europium.

The method was used for the determination of elemental composition of soil and plants from Uzbekistan. Samples were collected during the month of July in the Samarkand region. The purpose of the study was a more precise determination of the degree of depletion of soil microelements under extended use in

TABLE 2. Elemental Concentrations in Sierozem (in mg/kg)

Micro-element	Sierozem, irrigated for more than 20 years		Newly developed sierozem	
	soil	cotton plant	soil	cotton plant
Iron	2870±140	197±9,7	3006±150	270±13,5
Cobalt	5,92±0,60	0,53±0,06	11,4±1,14	0,97±0,09
Zinc	107±21	8,6±1,7	122±24,4	12,8±2,5
Scandium	7,09±0,36	0,61±0,03	14,3±0,71	1,07±0,06
Cesium	2,73±0,27	0,26±0,03	11,8±0,12	0,76±0,08
Antimony	1,82±0,27	0,25±0,04	3,54±0,52	0,52±0,08
Europium	0,01±0,0005	0,008±0,0004	0,03±0,0015	0,007±0,0003

connection with the increased frequency of appearance of verticillium wilt on cotton plants in long-irrigated fields in comparison with newly developed fields.

The results are shown in Table 2. As is clear, the old, irrigated soils are distinguished by a considerably reduced microelement content in plants and soils, which may be the background favoring the increased frequency of appearance of verticillium wilt. If one compares the content of each element in soils and plants, one can note a linear relationship between them. In our opinion, this is evidence that the variation of soil composition for all elements lies in a range proportional to storage of these elements by plants.

The material presented is evidence of the reliability of a simultaneous determination of a large number of elements in a single small sample. The method described above can be supplemented by methods for medium- and short-lived isotopes, which make it possible, in principle, to expand considerably the number of simultaneously determined elements.

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ORIGIN OF TRACKS OF FISSION PRODUCTS IN LEAD GLASSES

G. N. Flerov, V. P. Perelygin, and O. Otgonsurén

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The possible existence of a new group of relatively stable, superheavy transuranium elements has received attention from many researchers, both theoreticians and experimentalists. But experiments on the synthesis of, and the search for, heavy transuranium elements in samples found in nature and in primary cosmic radiation have not yet resulted in unambiguous conclusions concerning the existence of a new stable group with atomic numbers close to Z = 114 (or 126). The theoretical and experimental investigations of the problem have been reviewed in detail in [1-3].

The aim of the present work is to establish the origin of tracks of heavy, charged particles, which were found in 1968 when we etched lead glass for shielding purposes and 19th century crystal glass [4]. Similar results were obtained in [5] where samples containing bismuth, tungsten, and mercury were examined, along with a large number of lead glasses. The observed effect in several samples of lead glasses corresponded to 2-10 fissions/g lead/yr.

Control experiments in which activation analysis was used [4-6] have shown that spontaneous fission of uranium traces contained in the glasses could account for at most 5% of the observed number of tracks. The background originating from the fission of thorium contained in these glasses ($\leq 10^{-5}$ g/g [4]) was negligibly small. Estimates of the contribution of the nucleon component of the cosmic radiation to the fission of lead nuclei [4] gave values of less than 2 fissions/g lead/yr.

These estimates were based on the extrapolation of the results of tests concerning the determination of the fission probability of bismuth and thorium nuclei at altitudes of 13-20 km and of about 4 km (Pamyr) [7]. The estimates were also based on fission probability data of thorium at sea level, which were obtained for the latitude of Moscow [8]. In quantitative estimates, the fact that the cross section of lead-nuclei fission by fast neutrons is at least two times smaller than the cross section of bismuth fission in the energy range up to 200 MeV [9] was taken into account. Furthermore, it was borne in mind that the nucleon component is absorbed by floors and walls of buildings. Apart from this, the assumption was that the spectrum of the nucleon component, which penetrated into the atmosphere and reached sea level, is shifted toward lower energies [7] and that in the transition from the 38° latitude (Pamyr) to higher latitudes, the intensity of the neutron component at sea level remains practically constant [10].

Based on the estimates, the conclusion of [4] was that the background produced by the cosmic radiation forms only a fraction of the effect detected in lead glasses, though "due to lack of direct experimental evidence of the fission of lead by cosmic-ray particles at sea level, the effect cannot be completely ruled out at the present time" [4]. Apparently, this result was independently confirmed in tests with large proportional counters in which two to four fissions/g lead/yr were obtained with lead glasses [11]. On the other hand, in experiments with counters loaded with the isolated Pb^{208} lead isotope, a total of four pulses were recorded during a 160-day exposure, which corresponds to about 0.5 fissions/g/yr.

The results of [4, 5, 11] were not reliably confirmed in experiments in which spontaneous fission events were detected with the aid of He^3 neutron detectors [2]. The experiments were made under ground at a depth of about 400 m, which eliminated the contribution of the fission-inducing component of cosmic radiation.

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It was established in these experiments that the number of double and triple coincidences of pulses produced by neutrons is several 10 times smaller for the lead fraction of complex ore and lead glass than could be expected from the assumption that the average number ν of instantaneous neutron-induced fissions of an unknown, spontaneously decaying atom amounts to 5-10. According to the estimates of [2], the effect which has been previously observed in these samples can be explained by spontaneous fission of superheavy nuclei, provided that the average number of instantaneous neutrons is assumed to be $\nu = 1.5$, which is very unlikely.

It was therefore necessary to make checks in which other methods of recording products of nuclear fission were employed.

Control Experiments with Direct Determination of Products Resulting from Spontaneous Fission in Lead Glasses

Two samples of x-ray shielding glasses, for which the previously observed effect corresponded to 7 ± 2 and 4 ± 1.5 fissions/g lead/yr [4, 5], were selected for control experiments.

Since the known techniques of recording rare fission events of nuclei are either very laborious [12] or do not allow unambiguous identification of the fission products of nuclei [13], we developed a special low-background technique which is based on recording coincidences of fission tracks in two layers of polymer foil.

The samples were ground to a fine powder in a ball mill. The powder was then applied to a Lavsan foil (layer thickness about 2 mg/cm^2) which also served as the detector of the fission products. Two thin plastic foils were directly applied to the sample. The plastic foil in contact with the sample had a thickness of about 6 μ, which is much less than the average range of fission products in the plastic material (range amounting to 16-18 μ); the second layer had a thickness of about 70 μ. The foil with the sample under inspection covered an area between 1 and 4 m^2, exposure time ~100 days.

Experiments on the recording of spontaneous nuclear fission in lead glasses were made under ground at a depth of 30 m, which, according to the data of [14], reduced the background produced by cosmic radiation at least 100 times.

After the exposure, the plastic detectors were placed in a 25% NaOH solution; the etching lasted for 40-50 h at a temperature of 20°C. Spark-induced breakdown of the thin plastic foil was used as an indicator in the search for tracks of fission products. The plastic foil for this purpose was placed on a polished metal surface and covered with an aluminum foil. A potential difference of 500-900 V was maintained between the metal surfaces.

Breakdown occurred along transverse openings which had been produced by etching the tracks of fission products. Since etching of surface defects on Lavsan leads also to the breakdown of the thin plastic film, the second layer was examined in the region close to the breakdown point, in order to obtain an unambiguous identification of the tracks of fission products. The number of background events was less than two to four/dm^2, and the above method made it possible to accelerate the search for rare events of nuclear fission several hundred times over the usual microscopic scanning of the area of the foils.

On 6 m^2 foil exposed with lead glass, the coincidence method revealed three fission products in place of 25 nuclear fission products which could be anticipated from the measured density of heavy-product tracks in the glasses. According to calculations, the background produced by spontaneous fission of uranium traces contained in the protective lead glasses must account for one or two tracks. This detail implies that the effect observed in lead glasses may possibly not be explained by spontaneous fission of remote transuranium elements.

Additional careful analyses of the possible reasons for the background in the glasses were required.

Origin of Tracks of Heavy Particles Detected in Lead Glasses

Apart from the fission of uranium and thorium nuclei, spontaneous fission of technogenic isotopes of transuranium elements, viz., plutonium, curium, and californium, may be one of the sources of the

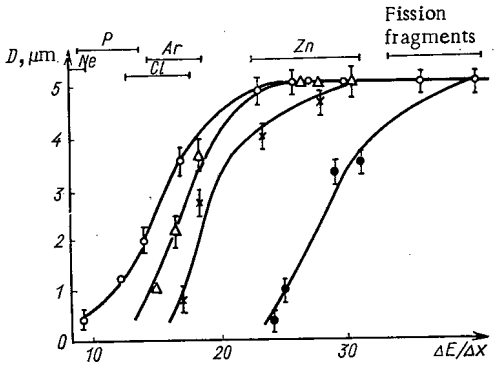

Fig. 1

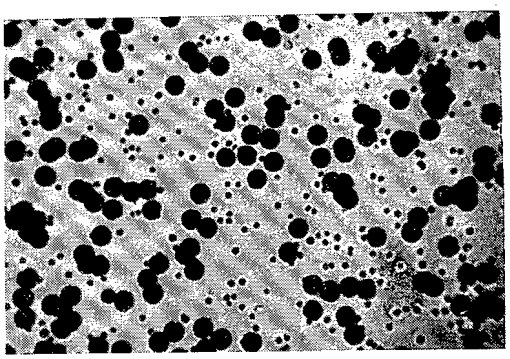

Fig. 2

Fig. 1. Dependence of the diameters of tracks upon the ionization produced by Ne^{22}, P^{32}, Cl^{35}, Ar^{40}, and Zn^{66} ions and by-products of Cm^{244} fission; various glasses: ○) silicate glass; Δ) 19th century crystal glass; ×) shielding lead glass; •) synthetic glass.

Fig. 2. Tracks of zinc and sulfur ions on silicate glass.

background in glasses. The concentration of admixtures undergoing spontaneous fission in the samples under consideration determines the upper sensitivity limit of the detection method used for remote trans-uranium elements, when the excess track density of fission products is considered [15]. The contribution of spontaneous fission of uranium can be determined by measuring the uranium concentration (e.g., by activation analysis [6]), yet elimination of the background produced by spontaneous fission of californium and curium isotopes requires additional analysis [16]. In fact, when very small effects are to be detected, Cm^{244} concentrations of up to 10^{-14} g/g and Cf^{252} concentrations of up to 10^{-20} g/g may become important.

For our work we used samples in the form of monolithic beams of lead glass which had been manufactured before the spontaneously decaying technogenic nuclei appeared. The etching procedure used made it possible to detect tracks of fission products in the bulk of the glass. The experiments impose certain conditions on the purity of the etching solutions. However, simple estimates show that dangerous concentrations of curium and californium in hydrofluoric acid or in NaOH amount to 10^{-6} and 10^{-12} g/g, respectively, in the case of 20 min etching. This contamination of the chemical agents can be ruled out completely.

Neutrons and protons generated in the atmosphere by primary cosmic radiation may be another source of background tracks in lead glasses.

The interaction of the nucleon component with the nuclei of elements forming part of the glass composition leads, firstly, to the formation of fast, heavy products which can lead to the appearance of background tracks imitating the tracks of fission products; secondly, these particles may induce fission of the nuclei of heavy elements (lead, bismuth, and thallium), when these elements are components of the glass.

In order to take these effects into consideration, the storage conditions of the lead glass samples must be accurately known.

For the purpose of determining the sensitivity to heavy particles, lead and silicate glass samples were irradiated perpendicularly to the surface with Ne^{22}, P^{31}, S^{32}, Ar^{40}, and Zn^{66} ions and with products of the fission of Cm^{244}. The energy of the bombarding particles was varied within wide limits. The samples were developed under identical conditions and thereafter, the diameters of the tracks produced by the bombarding particles were measured. The results of the measurements are depicted in Fig. 1. From Fig. 1 it can be seen that the various glasses are characterized by different recording thresholds for charged particles. The curves of Fig. 1 end in a plateau in the region of the specific ionization produced by zinc ions. When the ionization increases further, the diameters of the developed tracks cease expanding. This fact indicates saturation in the zone of defects produced by strongly ionizing particles in glasses.

It follows from Fig. 1 that silicate glass can record ions beginning from the Ne^{22} ion. Even the tracks produced by sulfur ions differ considerably in their dimensions from the tracks of zinc ions (Fig. 2). A similar pattern was observed in crystal glass made during the beginning of the 19th century and in the x-ray shielding glass, but when the lead concentration increased, the detection threshold was shifted toward increased specific energy losses of the bombarding particles (see Fig. 1). Samples of synthetic glass, in which the track diameters produced by zinc ions were much smaller than the track diameters produced by

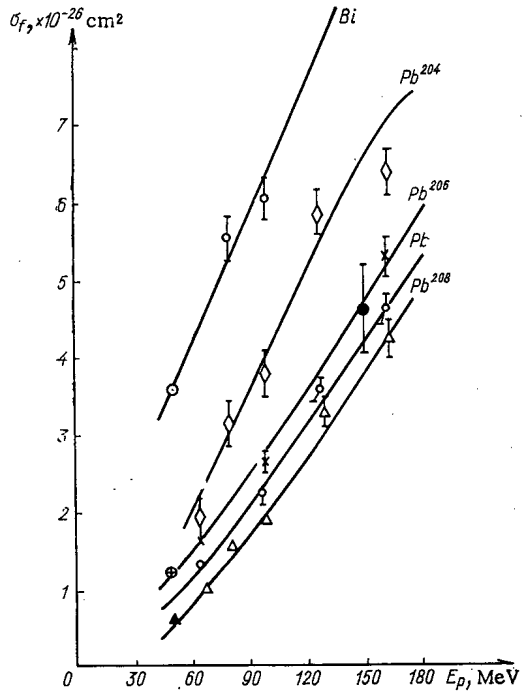

Fig. 3

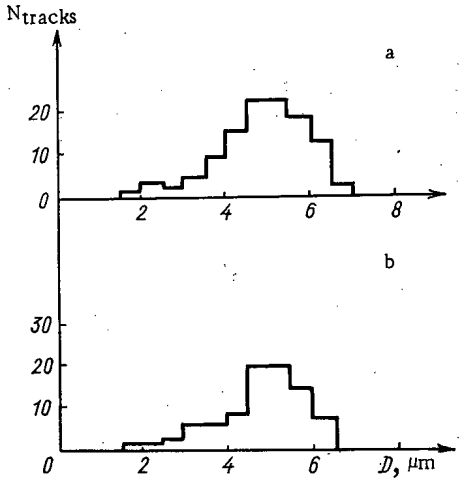

Fig. 4

Fig. 3. Energy dependencies of the corss section of proton-induced fission of Pb^{204}, Pb^{206}, and Pb^{208} nuclei of natural lead and of bismuth. , , ▲) denote the data of [16] for proton energies of 49.6 MeV; •) data of [9].

Fig. 4. a) Distribution of the diameters of tracks of the fission of U^{235} by thermal neutrons; b) distribution of the diameters of tracks of lead fission by protons with an energy of 120 MeV; x-ray shielding glass.

fission products, had the lowest sensitivity. It follows from Fig. 1 that the tracks of heavy products up to argon ions are characterized by diameters much smaller than that of fission products in lead glasses.

According to [17], the cross section of the formation of products with $Z \geq 18$ is less than 10^{-27} cm² in the interaction between 5.5 GeV protons and uranium.

Since the main fraction of the secondary nucleons produced by cosmic radiation at sea level has an energy of at most several hundred MeV, the probability of forming nuclei with $Z \geq 18$ is much smaller than the corresponding probability at the energy 5.5 GeV and, evidently, the background produced by heavy products in glasses can be ignored.

Let us consider the available information on the fission of heavy nuclei by the nucleon component of cosmic radiation. According to the results of [7, 14, 18], this component consists of 85-90% of neutrons. The neutron spectrum is characterized by an exponential decrease in the energy region $10-10^4$ MeV [18].

Since information on the fission of lead by cosmic particles at sea level is not available, data on the fission of bismuth by these particles were used. According to [18], the probability of bismuth fission on the geomagnetic latitude $\lambda = 44°$ N (USA) amounts to 22 fissions/g/yr. Extrapolation of the results obtained at the latitude of the Pamyr [7] provides the following value at the latitude of Moscow: about nine fissions/g of bismuth/yr.

The results of a calculation of the fission rate of lead nuclei split by nucleons of the cosmic radiation at sea level have been compiled in [14]. The geomagnetic latitude for which the calculations were made was not stated. According to [14], the rate of lead fission by the nucleon component amounts to 10-12 fissions/g/yr.

Based on the energy spectrum of the neutrons of the cosmic radiation, the conclusion is that nucleons in the energy range beginning at the fission threshold of lead (40-50 MeV) up to several hundred MeV give the main contribution to the fission of lead nuclei. According to the literature data, the cross section of

lead fission by fast neutrons in this energy range is two to three times smaller than that of bismuth [19]. Accurate figures for the cross section of the fission of lead and its isotopes are not available in this energy interval. We can only assume for the double magic Pb^{208} nucleus exposed in a proportional counter that the cross section of fission by fast particles is much smaller than in the other lead isotopes.

We have made experiments in order to determine the cross section of proton-induced fission of lead and the lead isotopes Pb^{204}, Pb^{206}, and Pb^{208}, and of bismuth in the proton energy range 60-160 MeV. $PbSO_4$ targets with a thickness of 10 mg/cm^2 were used in the experiments; the thickness of the targets was much greater than the range of fission products. Thin mica sheets were used as detectors. The proton beam was monitored with the aid of calibrated uranium samples. The experiment was made with the outside proton beam of the synchrocyclotron of the Laboratory of Nuclear Problems. Figure 3 shows the dependence of the fission cross sections of lead and its isotopes on the energy of the bombarding particles. The errors (±15%) which were made in the measurements of the absolute values result primarily from the inaccuracy with which the effective layer R_{eff} of the material ($PbSO_4$) was determined [6]. The statistical error amounted to ±4% for each point. The data displayed in Fig. 3 for the fission cross sections of Pb^{206} and Pb^{208} are in good agreement with the fission cross sections obtained for these isotopes at the proton energy 49.6 MeV (see [20]). Thus, taking into account the form of the spectrum of the nucleon component of the cosmic radiation [18], it is safe to say that the fission of lead by nucleons with energies of 50-200 MeV provides the major contribution to the observed effect.

Moreover, we can draw the conclusion that the probability of fission of the Pb^{208} isotope by protons is approximately 1.5 to 2 times smaller than the fission probability of natural lead nuclei.

Since it is impossible to make sufficiently reliable quantitative estimates, direct experiments were made to determine the probability of lead fission by cosmic particles ($\lambda = 56°$ N).

To this end, several square meters of lead foil with the natural isotope composition were placed in contact with double layers of detector foils. The foils were thereafter exposed for 92 days, without an absorber, under the open sky. Thirty-six tracks of products resulting from the fission of lead by fast neutrons were recorded per m^2 of the foil surface. Taking into account the efficiently working layer [6], the probability of lead-nuclei fission amounted to 15 ± 4 fissions/g/yr, which corresponds to the apparent spontaneous fission period of natural lead (about $1.4 \cdot 10^{20}$ yr). On the same foil area, which was in contact with lead in the region of proportional counters (first floor of a two-story building, ceiling thickness about 160 g/cm^2) five fission products were recorded during the same time, which corresponds to approximately two fissions/g lead/yr.

Fission of lead nuclei by nucleons of cosmic radiation can also be used to explain the effect which was observed in the first experiments with Lavsan exposed, while in contact with a lead layer, for 100 days under ground [4]. As a matter of fact, underground exposures were made before the arrangement was mounted and brought to the storage area; the sample was returned and disassembled after exposure. The sample was then in a building, the thickness of the concrete ceiling of which was less than 30 cm.

Since the entire operation took less than 3 days, the fission of lead nuclei during manipulation caused an observable effect of 1.0 + 0.5 fissions/g/yr.

We attempted to discriminate the tracks of products resulting from the fission of lead nuclei by fast protons (120-160 MeV) from the tracks resulting from the fission of uranium by neutrons. The distributions of the track diameters of fission products were measured in these experiments performed with 19th century crystal glass and with samples of shielding glass.

As can be inferred from Fig. 4, the diameter distributions of the tracks of products obtained by fission of lead by fast protons and of uranium by neutrons are similar.

According to a private communication of A. L. Fleischer (USA), the recoil effect obtained in the interaction of fast protons with lead nuclei must result in typical V-shaped tracks of fission products in glasses. However, our determination of the form of tracks in lead glass irradiated with 120 MeV protons parallel to the glass surface has shown that recoils in the form of V-shaped tracks are observed in less than 4% of the cases. Thus, we can conclude that tracks of products produced by the fission of lead by fast nucleons are not distinguished in diameter or form from the tracks of fission products of heavier nuclei.

CONCLUSIONS

It was previously shown [4, 5] that tracks of fission products which could not originate from the fission of uranium or thorium nuclei are present in certain samples of old lead glasses. The hypothesis

of spontaneous fission of nuclei of unknown superheavy elements is one of the possible explanations of this effect.

The additional experiments concerning the analysis of possible reasons for the appearance of background tracks have shown that a few possible sources of background tracks can be disregarded. However, it was established that fission of lead nuclei by fast nucleons of the cosmic radiation may fully account for the observed effect, provided that the glasses were located far from walls and in buildings with a concerte ceiling of less than 10 cm thickness, i.e., a thickness which does not render adequate shielding from cosmic radiation.

Since the storage conditions of the lead glasses which we examined were unknown, the previously obtained result is only an indirect proof of the existence of a long-lived spontaneously decaying nuclide in nature.

The glasses which contain lead or other heavy elements can be used for detection purposes only when the glasses are known to have been stored under a thick cover layer ($d \geq 200$-300 g/cm^2) since their manufacture.

It was established that in glasses which are free of heavy elements, the cosmic radiation does not create a background imitating tracks of fission products. This fact makes it possible to use a large number of synthetic and natural glasses for the future search for remote transuranium elements.

Our technique of searching for rare nuclear fission events by way of observing coincidences of tracks of products in plastic foil layers allows recordings of the effect at levels as low as 0.1-0.2 fissions/yr.

In conclusion, the authors express their sincere gratitude to G. M. Ter-Akop'yan, N. K. Skobelev, and A. G. Popenko for their continued help in the present work and for useful discussions, and to V. I. Kudryavtsev, G. S. Khodakov, and A. G. Pil'kov for their assistance in adopting comminution techniques for solids and application of thin homogeneous layers to Lavsan substrates. The authors are also indebted to L. V. Dzholos, A. V. Sergeev, K. I. Merkin, and T. I. Rybakov who participated in the development of the coincidence method and took care of the evaluation and scanning of the numerous glass and foil samples.

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NONASYMPTOTIC NEUTRON SPECTRUM IN A TWO-COMPONENT MEDIUM WITH ENERGY-DEPENDENT CROSS SECTIONS

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UDC 621.039.51.12

In treating the resonance absorption of neutrons in infinite homogeneous media the energy structure of the neutron flux is ordinarily determined by the asymptotic solution of the neutron slowing-down equation [1]:

$$\Psi(u)\sum_{i=1}^{n}\int_{u-q_i}^{u}\Psi(u')\,h_i(u')\,f_i(u-u')\,du'+\delta(u),$$

where $\Psi(u)$ is the neutron collision density; $h_i(u)$ is the relative probability of neutron scattering from the i-th kind of nuclei of the medium; $f_i(u) = \alpha_i e^{-u}$ is the neutron energy distribution function for elastic scattering; $\alpha_i = (A_i + 1)^2/4A_i$; A_i is the atomic number of the i-th nucleus; and $q_i = 2\ln(A_i + 1)/(A_i - 1)$.

In a number of problems of neutron slowing-down theory, however, it is necessary to take into account nonasymptotic (Placzek) oscillations of the neutron collision density which can occur in a mixture

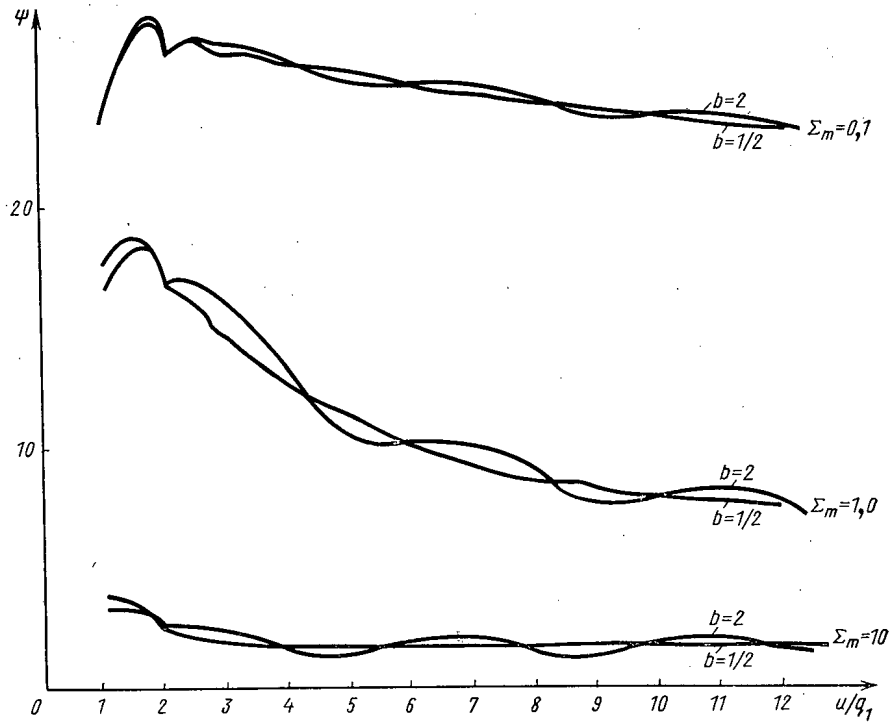

Fig. 1. Placzek function Ψ for a mixture of hydrogen and heavy nuclei for $\Sigma_m = 0.1$, 1.0, and 10; b = 2 and 1/2.

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of light and heavy nuclei. An exact calculation of the Placzek function is made by using an algorithm for the solution of the slowing-down equation in a multicomponent medium without any assumptions about the character of the energy dependence of the cross sections of the elements. This algorithm is based on the reduction of the integral slowing-down equation with a monoenergetic source, whose solution has discontinuities of the first kind [2], to a differential equation with a retarded argument whose solution has discontinuities at these same points on the energy axis, and the use of familiar numerical methods, such as the Adams method, for solving the differential equation.

This algorithm allowed us to determine correctly the behavior of the neutron collision density close to the source energy and to calculate the asymptotic behavior of the Placzek function. As an example we have examined the main features of the behavior of the Placzek function in a two-component hydrogeneous medium when the scattering cross section for the heavy component (iron) has the form

$$\sigma_s = \sigma_p + \sigma_r \sin^2\left(\frac{\pi u}{2q,\ b}\right),$$

where b is a parameter to vary the distance in lethargy units between the maxima of the function σ_s.

Figure 1 shows the form of the Placzek function for three values of the hydrogen scattering cross section Σ_m and two values of the parameter b. Here $\sigma_p = 4.5$ b and $\sigma_r = 10$ b.

The calculations permit an estimate of the validity of the narrow resonance approximation; for periods less than the magnitude of the maximum energy loss in elastic scattering from nuclei of the heavy component the asymptotic collision density does not depend on energy (e.g., for $b = 1/2$). Obviously the same result will hold for broad resonances ($b \gg 1$), and only for $b \approx 1$ (intermediate resonances) does the asymptotic collision density oscillate periodically with energy, correlated with the energy structure of the cross section. This effect can be important in calculating the effective resonance integral in individual energy groups, in averaging the group cross sections, and in analyzing the nuclear temperature effect in a fast reactor.

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FOURIER SERIES SOLUTIONS OF CERTAIN HETEROGENEOUS PROBLEMS

S. S. Gorodkov

UDC 621.039.51

Some problems associated with heterogeneous reactor calculations in the diffusion approximation can be reduced to the solution of a two-dimensional diffusion equation with an infinite set of δ-sources of the form

$$\sum_{m,\,n=-\infty}^{+\infty} \delta(x - ma)\,\delta(y - na). \qquad (1)$$

In addition to the problem of the Green's function for an infinite square lattice such problems include the Green's function for an infinite lattice with triangular and hexagonal cells, or a single rod in an assembly of rectangular cross section without a reflector.

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The solution of the diffusion equation with a source of the form (1) in an infinite moderator can be written as a double Fourier series. This series diverges logarithmically, however, at the points $x = ma$, $y = na$, and this complicates the use of the solution in computer calculations.

This difficulty can be removed by using the function

$$l(x, y) = -\frac{1}{4\pi} \ln \left(1 - \frac{1}{2}\cos\frac{2\pi x}{a} - \frac{1}{2}\cos\frac{2\pi y}{a}\right), \quad (2)$$

which has the same singularities at the same points as the solution of the differential equation with source (1). If this function is added to the solution in the form (2) and subtracted in the form of the double Fourier expansion, a series difference is obtained which does not have singularities and converges well at every point. The function $l(x, y)$ does not depend on the parameters of the moderator and in this sense is universal. Simple formulas are found for the Fourier coefficients of this function.

As an example of the possibilities of the proposed method a calculation is made for an exponential experiment performed on a few-channel assembly of square cross section without a reflector.

DETERMINATION OF PROTON RANGES

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UDC 539.125.4

Ranges of protons of energy E_p in matter are computed from the relation

$$R(E_p, X) = \frac{A_X}{2Z_X} R(E_p, C)\, k(y_X). \quad (1)$$

Here A_X and Z_X are the mass number and atomic number of the stopping medium; $R(E_p, C)$ is the range of protons of energy E_p in carbon; $k(y_X) = 1 + a_1 y_X + a_2 y_X^2 + a_3 y_X^3$; $y_X = \log(I_X / I_C)$ where I_X and I_C are the ionization potentials of the stopping material and carbon.

The coefficients a_1, a_2, and a_3 have been calculated as functions of the energy E_p from 10 to 500 MeV. The values obtained are approximated by the expressions

$$\left.\begin{aligned} a_1 &= 0.635 E_p^{-0.15} \\ a_2 &= 0.47 E_p^{-0.31} \end{aligned}\right| 10\ \text{MeV} \leqslant E_p \leqslant 500\ \text{MeV};$$
$$\left.\begin{aligned} a_3 &= 0.8 E_p^{-0.57} \end{aligned}\right| 10\ \text{MeV} \leqslant E_p \leqslant 100\ \text{MeV};$$
$$\left.\begin{aligned} a_3 &= 0.138 E_p^{-0.18} \end{aligned}\right| 100\ \text{MeV} \leqslant E_p \leqslant 500\ \text{MeV}. \quad (2)$$

Using this procedure the ranges of 10-500 MeV protons were calculated for such materials as Plexiglas, quartz glass, alloy IMI-680, titanium, lithium hydride, asbestos textolite, boron carbide, ablative material (30% phenol + 70% SiO_2), biological tissue, etc. Values of proton ranges in aluminum, copper, and lead obtained by the present method and by integration [1] are presented for comparison. They differ by less than 1% over the whole energy range.

The following effective parameters were used in calculating the ranges: atomic number Z_{eff}, atomic weight A_{eff}, ionization potential I_{eff}. The values obtained for the ranges can be expressed by

$$R = bE_p^c. \quad (3)$$

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The values of the coefficients b and c depend on the kind of material and are given for the materials listed. The values calculated by Eq. (3) are accurate to 1-5%, which is considerably better than the 7-15% estimated by the authors of the existing approximations [2, 3] for this same energy range.

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CALCULATION OF PHOTOELECTRIC CROSS SECTIONS FOR THE STATISTICAL SIMULATION OF TRANSPORT PROCESSES

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UDC 539.122.173

In solving γ-ray transport problems numerically by the Monte Carlo method it is expedient to express the dependence of the integrated cross sections of elementary interaction processes on the γ-energy ε by a formula. There are no exact theoretical or empirical formulas for the photoelectric cross section over a broad range of γ-energies. We consider a possible variation of the analytic approximation of the energy dependence of the photoelectric cross section $\tau(\varepsilon)$ based on tabulated reference data.

In [1], characterizing the state of theoretical γ-interaction cross section studies up to 1970, summary tables are presented of total and partial cross sections for 100 elements in the 0.001-100 MeV energy range. The error in the numerical values of the photoelectric cross sections is no more than 3% in the 0.006-0.2 MeV energy range, and no more than 10% outside this range.

The tables in [1] show that the energy dependence of the photoelectric cross section $\tau(\varepsilon)$ can be approximated by

$$\tau(\varepsilon) = \sum_{i=0}^{4} \tau_i \cdot \varepsilon^{-i} \qquad (1)$$

with an error of no more than 3% over a wide energy range $\varepsilon \geq \varepsilon_K$. The values of τ_i calculated on a computer by the method of least squares using power type weighting functions [2] are tabulated for 48 elements most frequently encountered in applied and technical nuclear physics problems.

The approximation of $\tau(\varepsilon)$ in the range $\varepsilon_{L_I} \leq \varepsilon \leq \varepsilon_K$ is also of practical interest. In this case $\tau_0 = \tau_1 = 0$ and Eq. (1) takes the form

$$\tau(\varepsilon) = \tau_2\varepsilon^{-2} + \tau_3\varepsilon^{-3} + \tau_4\varepsilon^{-4}. \qquad (2)$$

The coefficients $\tau_{i \geq 2}$ are listed for 21 elements. The maximum error of the approximation is 2.6%.

It should be noted that Eq. (1) is valid for 100 elements when $\varepsilon \geq \varepsilon_K$, and Eq. (2) for 73 elements beginning with nickel ($\varepsilon_{L_I} = 0.001008$ MeV) when $\varepsilon_{L_I} \leq \varepsilon \leq \varepsilon_K$.

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Translated from Atomnaya Énergiya, Vol. 33, No. 6, pp. 987-988, December, 1972. Original abstract submitted March 3, 1972.

THE BEHAVIOR OF FLUOROELASTOMERS UNDER γ-IRRADIATION

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UDC 661.48

We conducted a comparative study of the processes of radiative cross-linking and degradation of fluoroelastomers and their vulcanizates in the free and stressed states in air and in a vacuum, as well as of the changes in properties as the rubber was aged. We used copolymers of vinylidene fluoride with hexafluoropropylene (SKF-26), trifluorochloroethylene (SKF-32), and perfluoroalkylvinyl ether (SKF-1). Table 1 shows the values of G_c, the radiochemical yield of the network chains (the cross-linking), and G_d, the degradation coefficient, as determined by measuring the solubility and equilibrium swelling in acetone.

Since the number of chains actually formed in the irradiation process is characterized by the quantity $G_c + G_d$, it can be seen that the inhibiting influence of oxygen on the true rate of radiative cross-linking of the SKF-26 and SKF-1 rubbers is slight; however, in an air medium the rate of radiative degradation of these polymers increases considerably. In the case of SKF-32 the oxygen reduces the rate of cross-linking but has much less effect on the process of radiative degradation. The sensitivity of fluoroelastomers to radiation increases in the following order: SKF-32, SKF-26, SKF-1.

When the fluoroelastomers were irradiated in the deformed state, the rates of radiative degradation measured with a continuous relaxation of the stress were found to increase sharply; this is apparently due to the mechanical activation of the processes of chain breakup owing to the migration of a charge or to the electronic excitation of a weakened bond. The polymer molecules undergo radiative degradation more than the transverse bonds; the broken ends of the chains become attached to polymer molecules, forming branched networks.

When packed cured rubbers made with SKF-32 raw-rubber base are subjected in the undeformed state to radiative aging, their resistance to rupture decreases rapidly, and their relative elongation decreases much more slowly than in cured rubbers using SKF-26 raw rubber as a base. The decrease in the resistance to rupture in a cured rubber with an SKF-32 base is more marked than in the case of SKF-26. The radiation resistance of unpacked cured rubber using the above-mentioned raw-rubber bases does not exceed ~40 Mrad when tested in the free state and ~25 Mrad when tested in the compressed state.

TABLE 1. The Effect of Irradiation Conditions on the Values of G_c and G_d in Fluoroelastomers

Type of elastomer	Irradiation in air			Irradiation without access to air		
	G_c	G_d	$G_c + G_d$	G_c	G_d	$G_c + G_d$
SKF-32	0.20	0.18	0.38	0.32	0.16	0.48
SKF-26	0.78	0.62	1.40	1.08	0.24	1.32
SKF-1	0.90	0.72	1.62	1.46	0.37	1.83

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LETTERS TO THE EDITOR

THE APPLICATION OF RADIOMETRIC METHODS IN THE MAPPING OF DISCONTINUOUS STRUCTURES (FAULTS) AND IN SEARCHING FOR RARE-METAL DEPOSITS

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UDC 550.835

According to data published by Soviet investigators [1-3], tantalum deposits of apogranitic type (analogous to the Nigerian deposits of the Jos plateau and the Kaffo valley) are localized in the apical portions of differentiated granitoid massifs and are closely connected in position with large tectonic dislocations (fault lines) which act as incurrent canals or ore-containing structures. The tantalum mineralization in such deposits is paragenetically related to uranium and thorium mineralization.

Accordingly, reconnaissance trips undertaken in order to select promising areas for prospecting for tantalum deposits of the endogenous type were designed primarily to detect discontinuous dislocations in an area corresponding to the development of intrusions of granitoidal composition and involved the use of SRP-2 scintillation-type field radiometers.

In a massif of differentiated granitoids of the Devonian age which had burst through Proterozoic and Cambrian metamorphic rocks, we observed the following petrographic varieties: porphyritic biotite granites, granite porphyries, granophyres, fine-grained granites, granosyenite–porphyries, adamellites, granites of medium and large grain size, granosyenites, and alkaline (riebeckite) granites. The dike facies are represented by vein aplites, microgranites, microgranite–porphyries, diabases, dibase porphyrites, quartz diabases, and hornblendites.

Even before the start of our exploration, we noted that the state geological map, on a scale of 1 : 200,000, showed little tectonic detail; an area of more than 1000 km^2 showed only five fault lines with discontinuities (including some which were merely hypothetical), with sublatitudinal and northeastward strikes.

Laboratory study of the topographic sheets and geological–geomorphological analysis strongly suggested that there was an incomparably larger number of fault lines with discontinuities which were hardly reflected at all on the map but showed up in a sketch of the hydrographic network of the area, which was formed by river and stream valleys of sublatitudinal and submeridional courses. Field observations confirmed the presence of faults with discontinuities and of weakened zones along which lie the valleys of today's rivers and streams.

The rapid detection of previously unmapped tectonic dislocations with discontinuities was facilitated by the fact that in most cases the fractures had been "healed" by the injection of dikes from the closing stages of the Middle Devonian magmatic cycle along these fractures; these were noticeably different in radiological properties from the rocks of the parent massif.

The most interesting findings were the meridional dislocations (strike azimuths from 20 to 340°C), marked by dike belts of complex composition. The latter usually consist of a veined body of fine-grained leucocratic albitized microgranite, sometimes bright red in color (hematized), with a thickness of 20-50 m, and two or three parallel diabase dikes 2-5 m in thickness, with abundantly ingrained sulfides. Within the dike belts, on the surface, we frequently find pits 10-15 m in width, extending linearly (parallel to the dikes), with slip traces along vertical boundary walls. It is possible that these are water-worn zones of mylonization, or the results of the leaching of carbonate or sulfide bodies. The length of the dike belts is more than 1 km, with a total thickness of 70-100 m (see Fig. 1).

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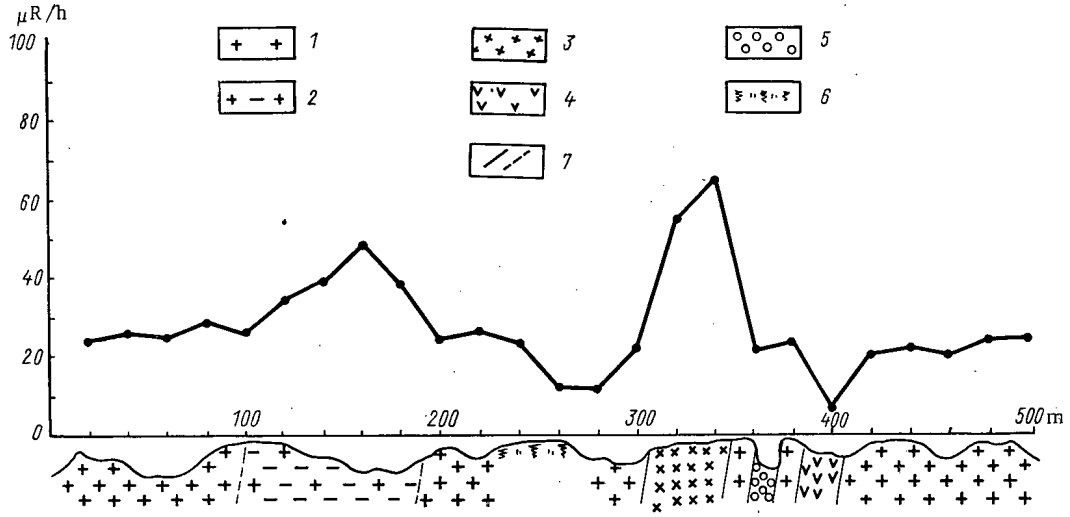

Fig. 1. Radiological characteristics of the dike belt. Rocks of the parent massif: 1) biotite granites; 2) riebeckite granites. Dike belt: 3) acid and alkaline derivatives of veined type; 4) diabase dikes; 5) linear collapse funnels (water-worn zones of mylonization); 6) sod-covered segments; 7) boundaries, confirmed or hypothetical.

Since the dike belts are relatively thin, we cannot be sure that they will be detected by ordinary geological mapping unless photographs on a scale of 1 : 5000-1 : 10,000 are used, since the veining differences in rocks lying in a mass of stony taluses which are similarly colored and covered with lichens cannot be distinguished visually from blocks of parent granites.

This explains why the 1 : 200,000 scale maps failed to show the dike belts and faults.

By using SRP-2 scintillation radiometers in our reconnaissance trips, even with a very widely spaced grid (scale 1 : 200,000), we were able to detect the dike belts with certainty from the sharp changes in the γ-activity of the rock. Against a parent-granite background of 19-22 μR/h, the albitized and hematized microgranites are distinguishable by peaks of up to 70-80 μR/h, while dips to 5-8 μR/h are characteristic of the diabase dikes. If individual discrete increases in the background along the reconnaissance route are caused by the presence of parent granites enriched by schlieren accessories or segments, an alternation, within a 50-100 m interval, of a sharp increase and an immediately-following equally sharp decrease in γ-radiation may in most cases be ascribed to a dike belt (see Fig. 1).

By detailed geological-radiometric photography on a scale of 1 : 2000 to 1 : 10,000 in the zone of the dike belts and meridional faults detected on the reconnaissance trips, we were able to discover a number of segments with complex mineralization, including U, Th, Li, Zr, TR, Nb, and Ta.

In these regions, in the actual zones of the meridional faults, between the albitized and hematized microgranites of the veined facies, we found an increase in radioactivity ranging up to several thousand μR/h. As we moved farther away from the central zone, the activity decreased, and over a 15-20 m interval it smoothed itself out to values characteristic of two-mica granite country rocks of the parent massif (19-22 μR/h). The mineralization of such segments is essentially uranium–thorium (U/Th = 1 : 5-1 : 8). The tantalum content does not amount to commercially useful values but is many times greater than the norm for this type of rock. The tantalum–niobium ratio is 1 : 2.

Other segments gravitate toward meridional faults, along which boss-shaped bodies of riebeckitic granites with low-grained complex mineralization have intruded. A higher percentage of U, Th, Zr, Li, TR, Nb, and Ta was established in the apical parts of the intrusion at the intersection of a meridional fault and a sublatitudinal dislocation acting as an ore-containing structure. Here the radioactivity of the albatized granitoids varies from several tens of microroentgens to hundreds of microroentgens in 1 h, and only at points with a high concentration of dark-violet fluorite do we find values of 1000-1100 μR/h. The uranium–thorium ratio in these ore manifestations ranges from 1 : 2 to 1 : 4, and the Ta/Nb ratio is 1 : 16. The zirconium content is more than 1%.

In one exocontact of the massif of granitoids, among the old metamorphic strata, by radiometric methods we were able to detect a number of sublatitudinal dislocations which showed a radioactivity value 3-5 times as high as usual. As was shown by aureole (halo) photographs, the Ta, Nb, Sn, Zr, Pb, and TR content in the zone of these faults is between 50 times and 10 times the norm, i.e., the concentrations are the same as in the granitoids of the parent massif outside the limits of the ore-bearing segments. The surrounding rocks near the sublatitudinal faults show unmistakable traces of metasomatic alterations. In the detailed investigation and tracing of the sublatitudinal faults and metasomatic alteration zones we found quartz–albite–hematite bodies with background radioactivity values in which we measured high tantalum concentrations at a Ta/Nb ratio of 1:1.

In addition to mapping and prospecting, radiometric methods may be successfully used for determining the contours of rare-metal mineralization (tantalum, niobium) and lithium mineralization in orebody outcrops. Such investigations may use either spectrometric pictures or ordinary γ-photographs on scales of 1:2000 to 1:10,000. From the data found in spectrometric photographs, we determined the following ratios between tantalum and the radioactive elements: U/Ta from 1:1 to 2:1, Th/Ta from 4:1 to 6:1. The most practical method (sufficiently exact one for the initial stage) is ordinary γ-photography of the surface, taking into consideration the ratios of tantalum to the total radioactivity.

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RADIOACTIVITY OF THE BOR-60 REACTOR COOLANT

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UDC 621.039.534.63:539.16

The widespread introduction of atomic power plants with sodium-cooled fast reactors necessitates a careful study of their safety and maintenance; in particular, the sources of radioactivity and the transport and accumulation of radioactive isotopes in deposits along the coolant circuit and in the reactor equipment must be examined. In order to perform such studies on the BOR-60 reactor an experimental complex was constructed including sodium and gas spectrometer loops fitted out with special equipment. By using interchangeable collimators and scintillation and semiconductor Ge(Li) detectors the radioactivity of various isotopes in the coolant and on the walls of the piping and equipment could be determined without taking samples [1]. In addition the radioactivity of the coolant was measured by radiochemical and γ-spectrometric analyses of samples using Ge(Li) detectors and NaI(Tl) crystals.

During reactor operation most of the ionizing radiation from the equipment and pipes comes from Na^{24}. The γ-dose rate reaches 7 R/sec in a box of technological equipment at a reactor power of 40 MW. The Na^{24} activity measured at various reactor powers and normalized to a maximum power of 60 MW remains practically constant. The specific activity of the other short-lived isotopes Ne^{23} and F^{20} is approximately inversely proportional to the coolant flow rate. Table 1 lists the values of the activities of the main short-lived isotopes in the sodium coolant at the reactor outlet.

After a reactor shutdown of 8-12 days the activity of Na^{24} becomes less than that of Na^{22} ($T_{1/2}$ = 2.6 yr) and Ag^{110m} ($T_{1/2}$ = 253 days) which subsequently determine the radiation environment in boxes of technological equipment of the primary circuit. The average activities of the long-lived isotopes in the sodium coolant and their changes with the time of reactor operation are listed in Table 2. It is characteristic that after 216 effective days of reactor operation at 40 MW the activity of the principal corrosion elements (Cr^{51}, Mn^{54}, Fe^{59}, Co^{58}, Co^{60}) in the sodium coolant does not exceed $0.5\text{-}3 \cdot 10^{-7}$ Ci/kg of sodium. In spite of the presence in the core of a bundle of fuel elements with assemblies which were not gas-tight (2.7% burnup), the activity of the fission products in the sodium coolant while a burnup of 5.9% was being achieved in the core was lower than the sensitivity of the method used in the analysis ($\leq 5 \cdot 10^{-8}$ Ci/kg of sodium).

Extrapolation of the measured values of the Na^{22} activity to saturation for a reactor power of 60 MW gives $2 \cdot 10^{-3}$ Ci/kg of sodium. The reference value assumed in the reactor design was $4 \cdot 10^{-3}$ Ci/kg of sodium [2].

Radioactive isotopes appearing in the coolant as a result of the activation of impurities in the sodium are shown in Table 2. An additional source of silver, zinc, and antimony probably arises from selective corrosion of steel. This is confirmed by determining these elements in sodium and steel by chemical analysis and by spectroscopy.

Analysis of the data in Table 2 shows that the Ag^{110m} activity changes more rapidly with time, and that of Zn^{65} less rapidly than follows from the laws of activation and decay. Calculations predict an average increase in silver in the circuit of 4 mg/day and a decrease in zinc of 3 mg/day. The calculated results are in good agreement with the spectral analysis of the silver in sodium, confirming the increase of silver from $7 \cdot 10^{-7}$ to $5 \cdot 10^{-5}$ wt. %. The removal of zinc from the coolant was shown also by γ-spectrometric measurements of the sodium oxides cold trap showing a concentration of Zn^{65} 15-20 times larger than in the primary circuit piping.

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TABLE 1. Radioactivity of Sodium Coolant Determining Most of the Ionizing Radiation from Pipes and Reactor Equipment for a Reactor Power of 60 MW and a Coolant Flow Rate of 800 m^3/h

Isotopes	Half-life	Reaction forming isotope from Na^{23}	Experimental value of specific activity, Ci/kg of sodium [2]	Calculated value of specific activity, Ci /kg [2]
Na^{24}	14.8 h	(n, γ)	43±5	57
Ne^{23}	38 sec	(n, p)	1.8±0.8	1.5
F^{20}	11.5 sec	(n, α)	1.6±0.3	-

TABLE 2. Radioactivity of Sodium Coolant Determining Most of the Ionizing Radiation from Pipes and Equipment after Reactor Shutdown

Time reactor was shut down	Radioactivity, Ci/kg of sodium				
	Na^{22}	Ag^{110m}	Zn^{65}	Rb^{86}	Sb^{125}
August 1970	$4 \cdot 10^{-5}$	$1{,}2 \cdot 10^{-5}$	$0{,}9 \cdot 10^{-5}$	—	—
November 1970	$4{,}9 \cdot 10^{-5}$	$1{,}6 \cdot 10^{-5}$	$0{,}95 \cdot 10^{-5}$	$3{,}8 \cdot 10^{-5}$	$2 \cdot 10^{-7}$
February 1971	$7{,}3 \cdot 10^{-5}$	$2{,}5 \cdot 10^{-5}$	$1{,}2 \cdot 10^{-5}$	$1 \cdot 10^{-4}$	$1{,}4 \cdot 10^{-6}$
August 1971	$1{,}9 \cdot 10^{-4}$	$0{,}9 \cdot 10^{-5}$	$2{,}0 \cdot 10^{-5}$	$1 \cdot 10^{-4}$	$1{,}6 \cdot 10^{-6}$

Measurements on piping without taking samples showed that the Zn^{65} activity $\leq 2 \cdot 10^{-4}$ Ci/m^2 and that of $Ag^{110m} \leq 10^{-3}$ Ci/m^2.

The isotopic composition obtained for the sodium coolant differs from the value assumed in the calculation for operating with hermetically sealed fuel elements (Na^{22}, Mn^{54}, Co^{58}) [2], and from the value found experimentally for other reactors (Na^{22}, fission products) [3].

Thus the investigations show that after long reactor operation with hermetically sealed fuel elements and with fuel element bundles having assemblies that are not gas-tight there are no radioactive fission products or corrosion products from the main components of the structural materials in the sodium coolant. The radioactivity of the coolant is due to the activation of sodium and its impurities and to the selective escape of silver, zinc, and antimony from steel into the sodium.

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REACTOR WITH AN ECCENTRICALLY PLACED ACTIVE ZONE

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UDC 621.039.51

In reconstructing the experimental heavy-water reactor ITEF the active zone was placed eccentrically with respect to the heavy-water tank (Fig. 1). This greatly reduced the background of fast neutrons and γ-quanta in the existing experimental channels. It was desired to estimate the bending of the neutron field in view of the fact that the thickness of the heavy-water reflector surrounding the active zone of the reactor was variable in azimuth. Since the reflector nevertheless remained fairly thick, the bending of the field in the active zone was only slight. Of greatest interest was the bending of the "surge" of thermal neutrons in the reflector.

In order to describe the neutron field in the reflector we used the two-group diffusion approximation. Allowing for the uniformity of the properties of the medium with respect to height in both zones of the reactor, the height dependence of the neutron fluxes may be regarded as proportional to $\sin(\alpha_z z)$.

The amplitude–radial dependence of the thermal and delayed neutron fluxes may, respectively, be expressed in the form:

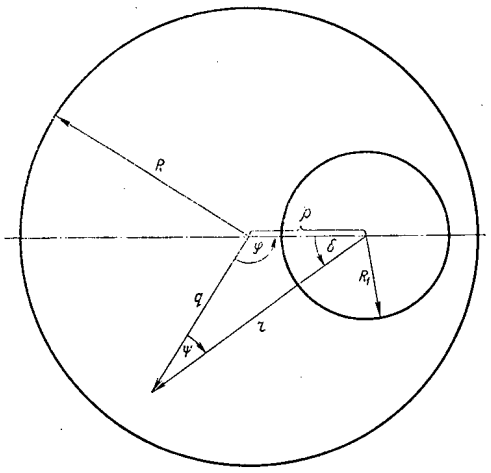

Fig. 1

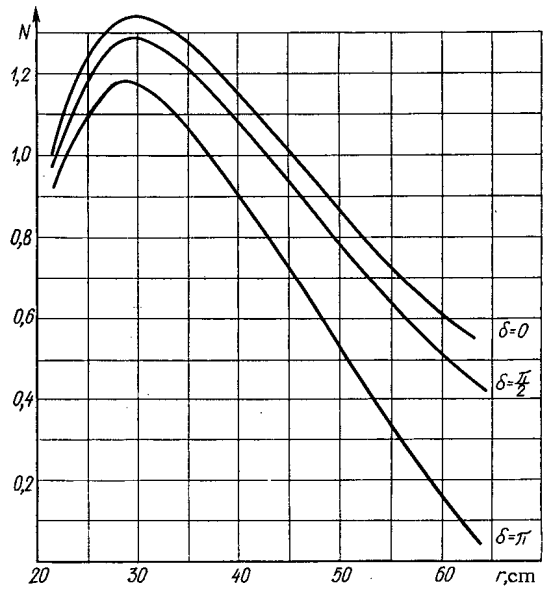

Fig. 2

Fig. 1. Arrangement of the active zone in the ITEF reactor heavy-water tank (reconstruction design): (q, φ) and (r, δ) are, respectively, the cylindrical coordinates of an arbitrary point in the reflector relative to the center of the tank and the center of the active zone.

Fig. 2. Relative distribution of the neutron flux density in the reflector as a function of the distance from the center of the active zone along different directions (b).

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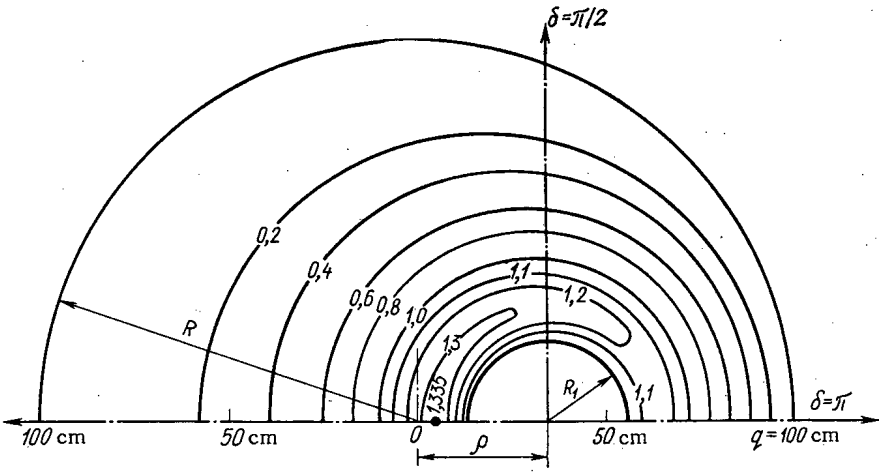

Fig. 3. Lines of constant neutron flux density in the reflector of the reactor with an eccentrically-placed active zone.

TABLE 1. Physical Characteristics of the Reactor

Parameter*	Active zone	Reflector
k_∞	1,8156	0
L^2	37,0	3055
τ	190	141
w	0,389	0
δ	0,890	1
L_s	3,21	3,22
$\bar{L}_s$	3,66	4,26

*Here k_∞ is the breeding factor, L^2 is the square of the diffusion length, τ is the square of the slowing down length, L_s, $\bar{L}_s$ are the scattering lengths for thermal and hyperthermal neutrons, w and δ are respectively the absorption and multiplication of the neutrons while slowing down.

TABLE 2. Some Computed Results

Angular harmonics of the flux at the boundary of the active zone	Perturbation theory	EKTSENT program
N_1	0,0205	0,0228
N_2	—0,0025	—0,0028
N_3	—	0,00025

$$N(r, \delta) = \sum_{-\infty}^{\infty} N_k(r)\, e^{ik\delta};$$

$$n(r, \delta) = \sum_{-\infty}^{\infty} n_k(r)\, e^{ik\delta}$$

(this nomenclature for the geometrical quantities is illustrated in Fig. 1, while the remainder of the notation is taken from the book by Galanin [1]).

Let us introduce the vectors $\mathbf{N}(r)$ and $\mathbf{n}(r)$, the components of these being angular harmonics of the densities of the thermal and delayed neutrons $N_k(r)$ and $n_k(r)$, respectively. Then if we consider the active zone as a unit multiplying and absorbing the neutrons we may replace it by boundary conditions of the form

$$\mathbf{N}'(R_1) = \hat{\lambda}\mathbf{N}(R_1) + \hat{\nu}\mathbf{n}(R_1);$$

$$\mathbf{n}'(R_1) = \hat{\bar{\lambda}}\mathbf{n}(R_1) + \hat{\bar{\nu}}\mathbf{N}(R_1),$$

where $\hat{\lambda}$, $\hat{\nu}$, $\hat{\bar{\lambda}}$, and $\hat{\bar{\nu}}$ are diagonal matrices, the elements of which may be expressed extremely simply in terms of the two-group parameters of the active zone. At the outer surface of the reactor

$$\mathbf{N}(R) = \mathbf{n}(R) = 0.$$

In this case we may make use of the heterogeneous matrix equation of Galanin [2]

$$\hat{M}\mathbf{N}(R_1) = \mathbf{N}(R_1), \tag{1}$$

where $\hat{M}$ is a matrix with elements capable of being expressed in terms of the parameters of the reflector and the eccentricity of the active zone, using the boundary conditions indicated. The procedure for calculating the matrix elements is described in [2].

Equation (1) enables us to find the critical condition of the system and the bending of the neutron flux at the boundary of the active zone and the reflector. This equation was solved by perturbation theory. The solution was compared with the exact solution obtained by a numerical method. As unperturbed system we took a system with an infinite reflector. In this case all the nondiagonal terms of the matrix $\hat{M}$ become zero, and the solution of the equation

$$\hat{M}^{(0)}\mathbf{N}^{(0)} = \mathbf{N}^{(0)}$$

is the vector $N^{(0)}$, which has only one nonzero "zero" component N_0. In the first approximation of perturbation theory we obtain the following expression for the components of the vector $N^{(1)}$:

$$N_i^{(1)}/N_0 = M_{i0}/(1 - M_{ii}^{(0)}).$$

For the numerical solution of Eq. (1) the computer program EKSTSENT was written in Algol and applied to the M-220 computer. After the solution of Eq. (1), i.e., the determination of the critical radius R_1 of the active zone and the vector $N(R_1)$ of the amplitudes of the angular harmonics of the thermal neutron flux at the boundary of the active zone, the program EKSTSENT enabled us to calculate the distribution of the thermal flux in the reflector.

The calculations were carried out for a system with the physical parameters given in Table 1 and the following geometrical parameters:

Radius of tank R.	100 cm
Radius of active zone R_1 (determined from the critical condition)	22 cm
Distance between the center of the tank and the center of the active zone	35 cm
Axial Laplacian	$6.85 \cdot 10^{-4}$ cm^{-2}

From Table 2 we see the correspondence between the amplitudes of the angular harmonics at the boundary of the active zone respectively calculated by perturbation theory and numerically (approximating the distribution by four angular harmonics).

The distribution of the thermal neutron flux density in the reflector is illustrated (in arbitrary units) in Figs. 2 and 3. It follows from Fig. 2 that the maximum "surge" of neutron flux in the reflector everywhere occurs at approximately 8 cm from the edge of the active zone, and its magnitude depends on the angle. The maximum value of this "surge" is approximately 13% above its minimum value.

The authors are grateful to A. D. Galanin for useful comments and constant interest in this work.

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FUSIBILITY DIAGRAMS OF TERNARY SYSTEMS CONTAINING MAGNESIUM AND CALCIUM CHLORIDES AND URANIUM TRICHLORIDE OR TETRACHLORIDE

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UDC 546.791.6

Fused salts containing uranium trichloride or tetrachloride are used for electrochemical refining and manufacture of uranium metal [1-3]. However, insufficient work has been done on the behavior of uranium chlorides in ternary melts of alkaline earth metals. Below we give the results of an investigation of ternary systems containing magnesium and calcium chlorides and uranium trichloride or tetrachloride.

The investigation was performed by the differential thermal method [4,5]. Alkaline earth chlorides of ar grade were obtained by the procedure in [6]. Uranium tetrachloride was obtained from freshly prepared UO_2 [7]. Uranium trichloride was obtained from the tetrachloride and reduced in a current of hydrogen as follows: $UCl_4 + 0.5H_2 \rightarrow UCl_3 + HCl$. After reduction, uranium trichloride was subjected to vacuum treatment at 850°C for 3 h. The initial substances, prepared in this way, had the following melting points: $MgCl_2$ 716 ± 2°C; $CaCl_2$ 780 ± 2°C; UCl_4 590 ± 2°C; UCl_3 835 ± 2°C. These values agree closely with the literature data [8]. The atomic ratio of chlorine to uranium (IV, III) for uranium chlorides was 4.01 and 3.02, respectively. Being markedly hygroscopic, the chlorides were kept in a dry chamber.

For differential thermal analysis, a weighed sample of the substance was placed in a boron nitride crucible, located in a quartz tube and tightly sealed by a rubber stopper. The cell was evacuated, filled with purified argon, and the sample was fused. The melt was retained for 1 h to obtain complete homogenization, and then cooled at a rate of 3-5 deg/min. The thermograms were interpreted by known procedures [9, 10].

On the basis of these results, Fig. 1 shows the primary crystallization surfaces of the systems $MgCl_2-UCl_4-CaCl_2$ and $MgCl_2-UCl_3-CaCl_2$. The liquidus surface was constructed from data on binary systems. The system $MgCl_2-CaCl_2$ was investigated by Mende [11] and Ivanov [12]. The investigated systems $MgCl_4-UCl_4$, $MgCl_2-UCl_3$, and $CaCl_2-UCl_4$ are simple eutectics. The melting point of the eutectic UCl_4–26 mole % $MgCl_2$ is 540 ± 2°C; the melting points of the eutectics $MgCl_2$–36 mole % UCl_3 and $CaCl_2$ –60 mole % UCl_4 are 670 ± 2°C and 490 ± 2°C, respectively. The system $CaCl_2-UCl_3$ exhibits a eutectic at 41 mole % UCl_3 with a melting point at 630 ± 2°C and two regions of solid solutions based on calcium chloride and uranium trichloride.

To construct the liquidus surface in the system $MgCl_2-UCl_4-CaCl_2$, we investigated 10 polythermic cross sections, passing through the apex of the concentration triangle corresponding to the composition $CaCl_2$ across to the opposite side. It will be seen from Fig. 1a that the system $MgCl_2-UCl_4-CaCl_2$ is a simple eutectic. The crystallization fields of $MgCl_2$, UCl_4, and $CaCl_2$ converge to form a eutectic E of the following composition: 9 mole % $MgCl_2$–53 mole % UCl_4–38 mole % $CaCl_2$, with a melting point of 487 ± 2°C. The x-ray diffraction patterns of specimens of different compositions confirmed the absence of new phases.

In the system $MgCl_2-UCl_3-CaCl_2$ we investigated nine polythermic cross sections. The primary crystallization surface of the system $MgCl_2-UCl_3-CaCl_2$ (Fig. 1b) exhibits three crystallization fields of

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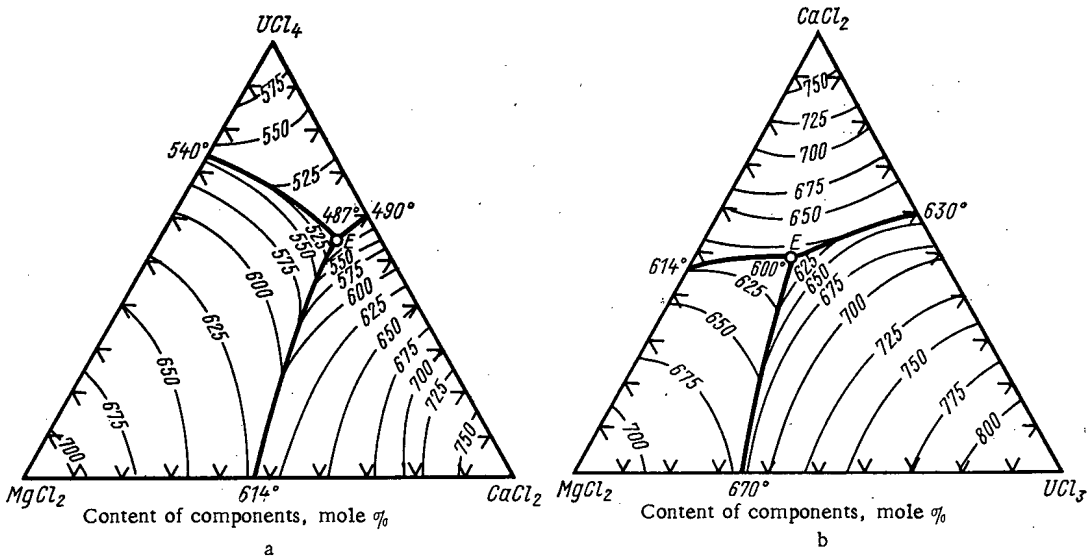

Fig. 1. Fusibility diagrams of the systems $MgCl_2-UCl_4-CaCl_2$ (a) and $MgCl_2-UCl_3-CaCl_2$ (b).

$MgCl_2$, UCl_3, and $CaCl_2$, which converge to form a eutectic E of the following composition: 31 mole % $MgCl_2$–20 mole % UCl_3–49 mole % $CaCl_2$ with a melting point of 600 ± 2°C. The x-ray diffraction patterns of specimens of various different compositions showed that melts rich in UCl_3 and $CaCl_2$ are characterized by the presence of a solid solution, which disappears as the concentration of the third component increases. We failed to establish the exact point at which the solid solution disappears. The x-ray diffraction patterns of $MgCl_2$-rich specimens revealed the presence of the initial components.

The primary crystallization surfaces of the ternary systems $MgCl_2-UCl_4-CaCl_2$ and $MgCl_2-UCl_3-CaCl_2$ have thus been investigated. It has been established that these systems are of the simple monoeutectic type.

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FISSION BARRIERS OF HEAVY NUCLEI

P. E. Vorotnikov

UDC 539.173.4

The liquid-drop model of the fission of heavy nuclei, which appeared immediately after the discovery of this phenomenon, predicted a rapid fall in the fission barrier height B_f with increasing fissility parameter Z^2/A [1]. The experimental data for nuclei with $Z \geq 90$ did not confirm this tendency, and the model had to be made more complicated. It was modified in two stages: account was taken of the one-particle structure of the nucleus only in equilibrium deformation (see, for example, Bell [2]); and a more detailed calculation was made to include shape asymmetry over a wide range of deformations [3, 4]. In the former case, the barrier remained one-hump; in the second, it was shown that it consists of two humps of different shapes and heights.

Experimentally, B_f is manifested in a sharp break in the energy dependence of the fission probability or cross section. Recently-published data enable us to compare the experimental and theoretical fission barriers for nuclei with $227 \leq A \leq 253$. Figure 1 plots the values of B_f obtained by analysis, with allowance for competing processes, of the neutron fission cross sections $\sigma_f(E_n)$ of some target nuclei which are even–even or odd–odd in Z and N [5-11]. The experimental values of the neutron binding energy B_n, required for the conversion from E_n to the excitation energy E^* of the compound nuclei, were taken from the tables in [12]. In cases where the value of B_n in [12] has a large uncertainty, we used the results of systematic classifications which show that for even–odd isotopes with $A = 4n \pm 1$ the values of B_n are close together and change by about 0.5 MeV on transition from one to the other in such a pair. The values given by this procedure for B_n were as follows: for Pu^{245}, 5.0 ± 0.2 MeV; for Cm^{247}, Cm^{249}, and Cf^{253}, 5.3 ± 0.2 MeV (as against the values 4.6 ± 1.4, 5.2 ± 1.0, 4.7 ± 1.0, and 4.8 ± 1.0 MeV, respectively, given in [12]).

It is remarkable that B_f is constant for nuclei with $Z \geq 90$: this differs sharply from the relation predicted by the liquid-drop model, which is shown in Fig. 1 by the dashed curve. Since the deviation of B_f

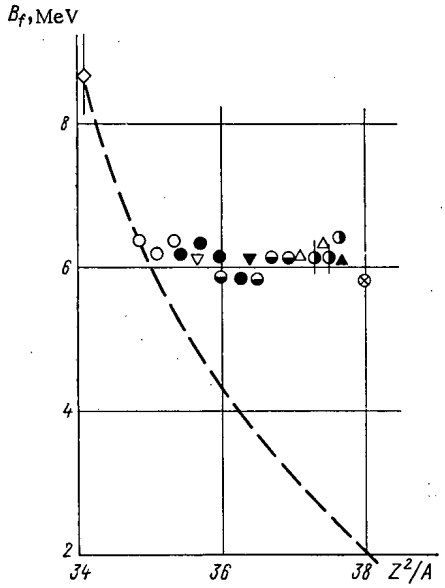

Fig. 1. Heights of fission barriers of heavy nuclei, plotted vs Z^2/A: ◇) Ra; ○) Th; ▽) Pa; ●) U; ▼) Np; ◒) Pu; △) Am; ◑) Cm; ▲) Bk; ⊗) Cf.

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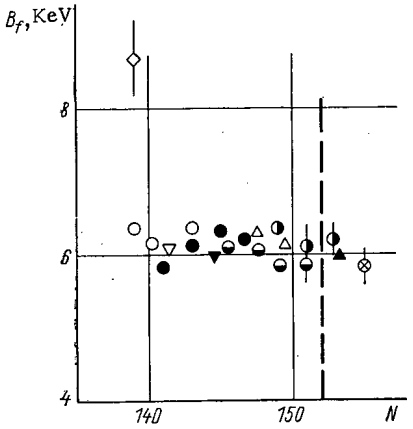

Fig. 1

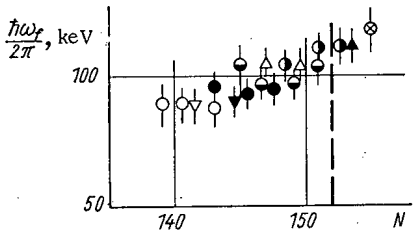

Fig. 2

Fig. 2. Fission barriers of heavy nuclei versus numbers of neutrons in nucleus. (Notation is the same as in Fig. 1.)

Fig. 3. Graph of $\hbar\omega_f$ vs number of neutrons in nucleus. (Notation is the same as in Fig. 1.)

from this relation is usually attributed to shell effects, in Fig. 2 we have plotted the same B_f values vs N, the number of neutrons in the nucleus. According to the calculations of Pauli and Ledergerber [4], in the two-hump barrier model one would expect a local increase of B_f by about 2 MeV in the region of the subshell with N = 152. This clearly fails to occur: the barriers of the 20 nuclei from Th^{229} to Cf^{253} remain equal to 6.1 MeV with a root-mean-square deviation of ±0.2 MeV. Although the one-hump barrier model [2] gives a different curve for B_f from that given by the two-hump model [4], in both cases the deviation of the calculated values of B_f from the experimental values is 1.5 MeV, while the root-mean-square value is 0.75 MeV. Only for Ra^{227} is the value of B_f higher, and this is perhaps due to the transition to the liquid-drop law for lighter nuclei.

The subbarrier curve of $\sigma_f(E_n)$ enables us to determine the value of $\hbar\omega_f = dE^*/d\ln[2\pi p\Gamma_f(E^*)]$; it is usually considered that this characterizes the shape of the barrier near its peak. In the two-hump model of the barrier for nuclei with $A \leqslant 242$, the second hump is higher, i.e., it determines $\sigma_f(E_n)$; for heavier nuclei the first hump is higher [4]; the values of $\hbar\omega_f$ should differ by a factor of about two [13]. As we see from Fig. 3, the experimental values of $\hbar\omega_f$ do not display this sudden change; they increase slowly, deviating from their mean curve only within the error of about 5-10%, and do not even react to the transition through the subshell at N = 152. Note that the indicated increase in $\hbar\omega_f$ is sufficient to explain the change in the periods of spontaneous fission with B_f approximately constant.

These graphs of B_f and $\hbar\omega_f$ vs N clearly do not allow us to regard the single-particle or shell effects as responsible for the marked deviations of the height and shape of the fission barrier from the liquid-drop laws. Thus it is difficult to explain the fission of the actinides in terms of a deformed liquid drop, and we must seek other possible mechanisms for this phenomenon [14].

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A METHOD FOR DETERMINING THE DIFFUSION MOBILITY OF INERT GASES IN SOLIDS

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UDC 533.15

In recent years, increasing attention has been paid to the study of the diffusion mobility of inert gases in metals, which is determined primarily from the kinetics of their emission from specimens being investigated during the process of annealing [1, 2]. However, since the solubility of inert gases in metals is very low, it is very difficult to introduce their atoms into a crystal lattice. Consequently, this is done chiefly by the injection of ions into the metal. In each individual case, the solution of the appropriate diffusion equation is used for determining the coefficient of diffusion of inert gases in materials from the kinetics of their emission during annealing [3-5]. The important point in solving the diffusion equation is the geometry of the specimen, and the geometry of the grain is disregarded. Because of this, the calculation formulas involve the dimension of the specimen being investigated as one of the experimentally determined parameters. Thus, in determining the diffusion coefficient of xenon in β- and γ-uranium [5], Perraillon, Levy, and Adda used the solution of the diffusion equation for a solid of finite dimensions with connecting boundaries (a uniformly saturated plate of thickness l):

$$\frac{Q_t}{Q_0} = 1 - \sum_{n=0}^{\infty} \frac{8}{\pi^2} \cdot \frac{1}{(2n+1)^2} \exp\left[-\frac{(2n+1)^2 \pi^2}{l^2} Dt\right],$$

where Q_t is the amount of gas emitted from the specimen during an annealing time equal to t; Q_0 is the original quantity of gas in the specimen; D is the diffusion coefficient; l is the thickness of the specimen.

However, since inert gases are practically insoluble in metal, the geometry that must be taken into consideration in calculating the diffusion coefficient from the kinetics of gas emission is grain geometry, not specimen geometry.

In the isothermal annealing of specimens uniformly saturated with gas (in order to prevent adsorption of the gas being analyzed, the surface layer of the specimen must be removed by electrical polishing), the kinetics of the gas emission must be determined at the moment when volumetric diffusion of inert-gas atoms toward the surface of the specimen begins in the grains situated directly beneath the surface (at this stage the contribution of boundary diffusion to the gas emission is small). In this case the volumetric diffusion of inert gases in the grains situated in deeper layers of the specimen should not have any great effect on the emission of gases from the specimen, since the inert gases coming out to the boundaries of the grains will either be partially accumulated there or partially migrate along these boundaries.

Therefore, in calculating the coefficients of volumetric diffusion of inert gases from the kinetic curve of their emission from a uniformly saturated specimen, we can use only that portion of the curve which corresponds to the emission of gas from the grains that lie near the surface of the specimen. As a rule, the fraction of the emitted gas is proportional to the square root of the annealing time ($t^{1/2}$), whereas in the case of gas emission through diffusion along the boundaries of the grains, there is a nearly linear relationship between gas emission and time. The calculated formulas must take account of the grain geometry, not the specimen geometry, provided that the size of the grains is much smaller than the geometric dimensions of the specimens. If the grain size is comparable to the specimen dimensions (the limiting case being a single-crystal specimen), we must use the solution of a differential equation that corresponds to

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the grain geometry. If the geometric dimensions of the specimen are much greater than the grain dimensions, all the cases can be reduced to the case of a uniformly saturated plate whose thickness is equal to the size of the grain.

When specimens are saturated with inert gases in an accelerator, the coefficient of volumetric diffusion of gases in the specimen material can be determined from the kinetics of gas emission only if the distance of the gas-saturated layer from the surface of the specimen is less than the size of the grain.

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CAPTURE EFFICIENCY IN INJECTION OF BEAM BUNCHED AT FREQUENCY OF SYNCHROTRON ACCELERATING RESONATORS

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UDC 621.384.634

Operating experience with an electron synchrotron at high energies has shown that the effectiveness of capture of injected particles is enhanced appreciably when a buncher excited at the operating frequency of the synchrotron accelerating resonator cavities is employed [1-3]. The bunching makes it possible to remove, beforehand, those particles entering the accelerator in phases differing substantially from equilibrium phase, and to increase the particle density near the equilibrium phase. That alleviates the beam load on the high-Q accelerating system and lowers the amplitude of coherent radial-phase oscillations, so that particle losses are diminished.

A buncher for the injector beam, with an operating frequency of 132.79 MHz, was developed and investigated on a test rig at the Erevan electron synchrotron facility [4, 5]. This article cites some results of adjusting the buncher in the injector-plus-synchrotron system. The two-resonator single-frequency buncher was installed at the input to the injector section of the linear accelerator.

The first resonator on the beam path (with a Q of 1500) was positioned 60 cm from the input to the injector section, while the second resonator (with a Q of 2000) was positioned 30 cm away. The buncher was supplied from a three-stage amplifier with plate modulation 20 kW per pulse, and using a GI-7B tube in the output stage. Continuous-wave excitation at 50 W average power was obtained from the oscillator driving the accelerating resonators of the synchrotron.

The effectiveness of particle capture by the synchrotron was determined by comparing the beam pulse current on the first few circuits through the machine with all of the synchrotron systems optimally adjusted, and at different bunching conditions (with buncher switched off, and with the first and second resonators alternatively energized, and finally with both resonators activated simultaneously). Oscillograms taken of the first 19 circuits of the beam, for those sets of operating conditions, are cited in Fig. 1. The difference in the performance of the first and second resonators stems from the fact that the voltage across the first resonator gap is much lower. Table 1 lists results of quantitative measurements of the pulse current taken on different circuits of the beam, relative to the pulse for the first circuit. As the beam circulates further in the machine, for the case where two resonators are functioning simultaneously, the pulse current undergoes virtually no variation till the termination of the acceleration cycle. In the absence of bunching, the pulse amplitude subsequently varied by about 60%.

Families of phase characteristics, displayed in Fig. 2, were taken in order to investigate the buncher operating conditions. They represent the dependence of the current I_f of particles accelerated in the synchrotron to the final energy on the phase shift of the buncher voltage relative to the voltage of the synchrotron radio-frequency oscillator. The accelerated current was measured with the aid of a calibrated resonator cavity. Before each curve was taken, all of the accelerator systems were carefully tuned so that the contribution made by the slow variation of the operating conditions of the distinct systems would be attenuated to the maximum. Curve 1 presents an arbitrary picture of the magnitude of the accelerated current when the voltage across the two-electrode electron gun $U_g = 41.8$ kV (corresponding to gun emission current $I_g = 30$ μA) when the buncher is deenergized. Curve 2 was obtained with only the first resonator operating, at a voltage $U_r = 2$kV across the resonator gap. Curves 3, 4, and 5 were taken with only the

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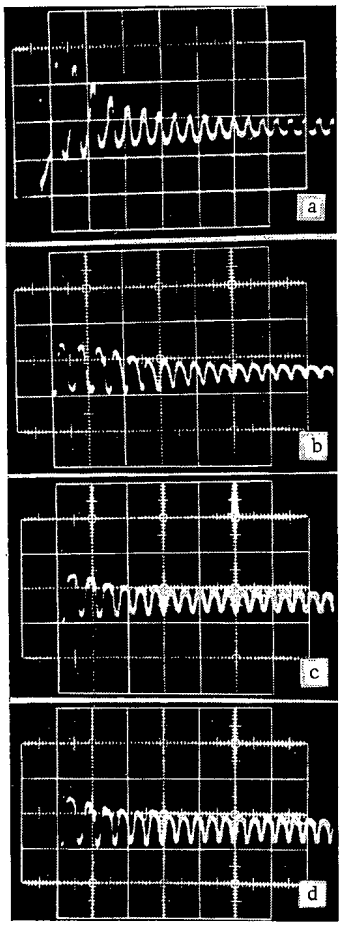

Fig. 1

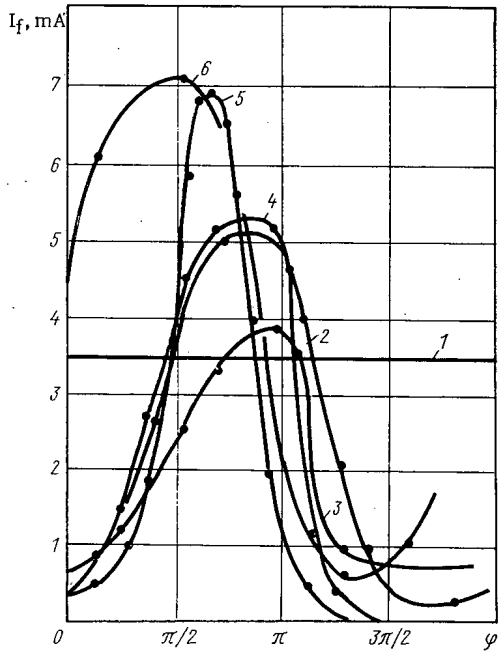

Fig. 2

Fig. 1. Beam current oscillograms taken in first few circuits through the synchrotron (vertical scale uniform for all the oscillograms, and equal to 15 mA/cm): a) no buncher; b) only first resonator activated; c) only second resonator activated; d) both resonators activated.

Fig. 2. Dependence of accelerated current in synchrotron on phase of buncher voltage with respect to voltage of synchrotron radio-frequency oscillator: 1) $U_r = 0$, $U_g = 41.8$ kV, $I_g = 30$ μA, $P_{inj} = 0.8$ MW; 2) $U_{r_1} = 2$ kV, $U_g = 20$ kV, $I_g = 18$ μA, $P_{inj} = 0.85$ MW; 3) $U_{r_2} = 5.1$ kV, $U_g = 32$ kV, $I_g = 21$ μA, $P_{inj} = 0.83$ MW; 4) $U_{r_2} = 5.1$ kV, $U_g = 24$ kV, $I_g = 18$ μA, $P_{inj} = 0.84$ MW; 5) $U_{r_2} = 9$ kV, $U_g = 22$ kV, $I_g = 15$ μA, $P_{inj} = 0.84$ MW; 6) $U_{r_2} = 13$ kV, $U_g = 22$ kV, $I_g = 15$ μA, $P_{inj} = 1.03$ MW.

second resonator operating, for three voltages U_r across the resonator gap. The voltage across the gun U_g was decreased with increasing U_r, in order to keep the energy of the electrons to be injected into the first section of the linear accelerator constant, and that in turn brought about a drop in the gun emission current I_g. Comparison of those curves shows that the current accelerated in the synchrotron increases with increasing voltage across the resonator gap, while the phase width of the train of bunches injected from the linear accelerator into the synchrotron narrows down. At voltages U_r upwards of 3.8 kV, the accelerated current varies by roughly 100% depending on the phase of the second resonator of the buncher system. Moreover, with increasing U_r there is a pronounced displacement of the phase about which bunching takes place to the zero point, and that corresponds to an increment in the average energy of the particles injected [4].

Clearly, the efficiency of the buncher also depends on the operating conditions of the linear accelerator injector section. We therefore also took curve 6 for the case of enhanced power P_{inj} in the accelerator section. The increase in P_{inj}, in conjunction with the increase in the accelerated current I_f, is responsible for the increased phase width of the train of bunches. Figure 3 shows the relationship between the current

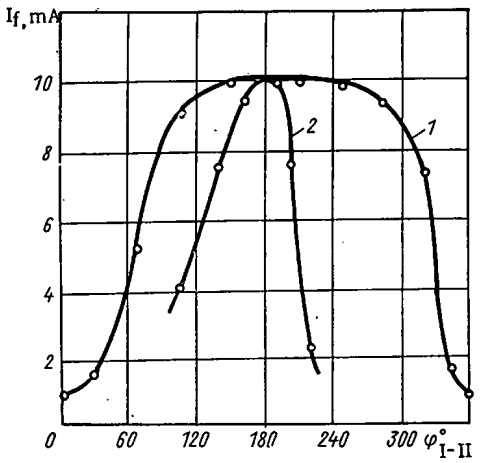

Fig. 3. Relationship between accelerated current and phase of voltage between the first and second resonators (1) and between the radio-frequency accelerating system and buncher when the phase shift between the resonators is constant (2).

TABLE 1. Sychrotron Accelerator Capacity

Sequential number of beam circuit, k	Relative magnitude of current pulse in K-th circuit			
	No buncher	Only first resonator activated	Only second resonator activated	Both resonators activated
1	1	1	1	1
2	0,64	0,75	0,76	0,83
3	0,58	0,75	0,75	0,83
4	0,46	0,75	0,75	0,83
5	0,41	0,68	0,72	0,83
6	0,39	0,62	0,65	0,8
7	0,32	0,62	0,65	0,77
10	0,3	0,53	0,62	0,73
19	0,23	0,36	0,5	0,7

accelerated in the synchrotron when two resonators (I, II) are energized and the change in phase between the two resonators when there is a constant phase shift between the accelerating voltage of the synchrotron and the voltage of the second resonator, and also the dependence on the change in the phase between the accelerating voltage and the voltage across the buncher when the phase shift between the buncher resonators is constant. The use of the buncher array made it possible to increase the peak accelerated current by a factor of 1.8.

Note that, since the injector of the Erevan synchrotron facility has a two-electrode gun in which the voltage, current, and beam focusing are interdependent, it would be difficult to optimize conditions for the injection facility as a whole. The buncher efficiency can be improved by using a three-electrode gun with constant anode voltage in order to inject the electrons into the linear accelerator, since a gun of that design would make it possible to exercise independent control over the energy, current, and focusing of the beam of particles, and would make it possible to define a narrower energy spectrum.

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SPATIAL DISTRIBUTION OF INTENSITY OF IONIZATION PROCESS NEAR PLANAR RADIOACTIVE SURFACE OF FINITE DIMENSIONS

A. S. Rozenkrants and I. E. Shishkova

UDC 621.317.7

The spatial distribution of the intensity of the ionization process exerts a substantial influence on voltage–current characteristics and on the electric field near a radioisotope neutralizer of static electricity. However, the literature[1] lists results of calculations of the spatial distribution of ionization intensity only near a planar unbounded radiating surface. Since the active layer of the neutralizer is of finite dimensions, it is of interest to take this fact into account when calculating the spatial distribution of the ionization intensity.

As pointed out in an earlier article [1], treatment of the effect of the fringe of the active layer is handled by introducing a coefficient K(r) dependent upon the shape of the active surface into the integrand expression of function (15) in [1]. This function (15) then acquires the form

$$N_i = \frac{\tau}{2l_m d_a^*} \int_0^{r_m} \frac{\Delta N_\alpha r \cos \alpha K(r)}{r^2 + x^2} dr. \quad (1)$$

Here N_i is the intensity of the ionization process; τ is the number of α-particles appearing in the active layer under a unit surface area in unit time; l_m is the total path length in air of α-particles emitted by the isotope in question; d_a^* is the relative thickness of the active layer; x is the shortest distance from the observation point to the neutralizer surface; α is the angle between the path traversed by the particle and the normal to the surface; r is the radius of the element of area irradiating the observation point; $\Delta N_\alpha = N_\alpha(x'_{max}) - N_\alpha(x'_{min})$, where x' is the distance remaining to the end of the particle range; N_α is the total number of ionizations brought about by the particle on its path x'.

Consider the definition of the function K(r) for a bounded planar surface. In that case the radiating layer is broken up into annular elemental areas, each of which brings about ionization dN_i at the observation point, when N_i is calculated using Eq. (1). One such area is shown in Fig. 1. Its center is displaced a distance y from the centerline of the layer, while the radius r is much greater than half (b) the longitudinal dimension of the layer (we assign to the longitudinal dimension of the coating the value 2b, along the direction in which the process material is moved).

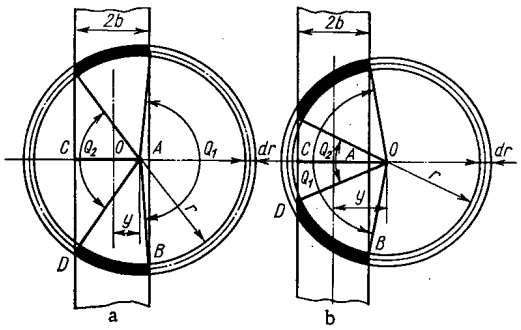

Fig. 1. Illustration of definition of coefficient K(r).

The coefficient K(r) is numerically equal to that part of the entire elemental annular area which lies within the confines of the active surface and irradiates the observation point. It can be defined, consequently, as the ratio of the sum of the arcs of a circle of radius r lying within the confines of the active surface to the entire length of that circle. If the ratio of the arcs of the circle is replaced by the ratio of the corresponding central angles, we obtain the formula

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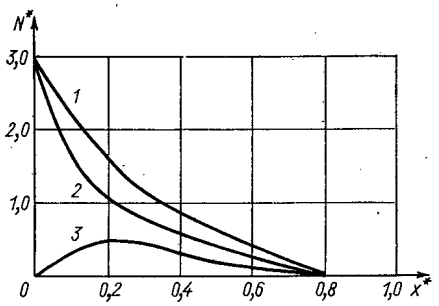

Fig. 2

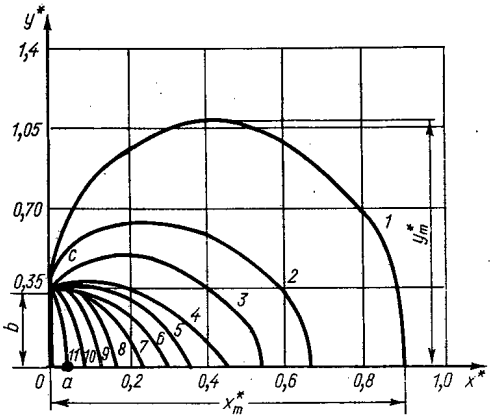

Fig. 3

Fig. 2. Distribution of relative ionization intensity near radioisotope neutralizer of static electricity, at different distances from neutralizer centerline: 1) $y^* = 0$ ($y^* = y/l_m$); 2) $y^* = 0.34$; 3) $y^* = 0.475$.

Fig. 3. Family of curves of equal ionization intensity near planar bounded active surface ($d_a^* = 0.3$; $d_{co}^* = 0.1$; $b = 0.34\,l_m$) coated with Pu^{239} layer. Curve 1 corresponds to $N^* = 0$, for each subsequent curve the value of N^* is increased by $\Delta N^* = 0.25$.

$$K(r) = \frac{2\pi - Q_1 - Q_2}{2\pi}, \tag{2}$$

where

$$Q_1 = 2\,\mathrm{arctg}\,\frac{\sqrt{r^2-(b-y)^2}}{b-y} \text{ (see Fig. 1a } \triangle OAB), \tag{3}$$

$$Q_2 = 2\,\mathrm{arctg}\,\frac{\sqrt{r^2-(b+y)^2}}{b+y} \text{ (see Fig. 1a } \triangle OCD). \tag{4}$$

In those cases where $r < b + y$, the angle Q_2 is set equal to zero.

Note that Eq. (2) is valid only when the center of the elemental annular area lies within the confines of the active layer, i.e., $y < b$. But if the observation point is projected onto the surface not occupied by the radiating layer, or on the edge of the layer ($y \geq b$), a part of the annular area either equal to or less than half the layer area will be found within the confines of the active layer:

$$K(r) = \frac{Q_1 - Q_2}{2\pi} \tag{5}$$

(when $y = b$, $Q_1 = \pi$). In addition, we have to bear in mind the fact that, when $r < (b - y)$ (the entire elemental area is within the confines of the active layer), $K(r) = 1$, and when $r < (y - b)$ (meaning that the entire elemental area lies outside the ionizing coating), $K(r) = 0$. Those conditions are taken into account by the algorithm compiled for digital computer calculations.

The distribution $N' = f(x^*)$ was calculated for different distances from the centerline of the radioactive coating, and beyond the confines of the coating (transition to relative units performed in a manner similar to that described earlier [1]), in order to clarify the effect of the boundedness of the active layer on the spatial distribution of the ionization intensity near the planar surface.

Figure 2 shows the $N^* = f(x^*)$ curves near the planar surface coated with a layer of Pu^{239} ($d_a^* = 0.3$, $d_{co}^* = 0.1$, $b = 0.34\,l_m$). It is clear from those curves that the intensity of the ionization process near the planar radioactive surface of finite dimensions is a function of two variables (x^*, y^*), so that the results of the calculations can be conveniently represented as a family of curves of equal ionization intensity in the x^*, y^* system of coordinates (see Fig. 3).

A family of those characteristics plotted for a layer of specified parameters (material of radioactive coating d_a^*, d_{co}^*, b) makes it possible to:

1. Determine the shape and maximum extent of the region of active ionization in two directions (according to Fig. 3, $x_m^* = 0.9$ at $y^* = 0$; $y_m = 1.06$ at $x^* = 0.395$).
2. Estimate the rate of change of the ionization intensity in the direction perpendicular to the radiating surface (from the extent to which the plotted iso-curves draw together). In Fig. 3, the highest rate of change in the ionization intensity occurs at points with coordinates $0 < x^* < 0.15$; $0.2 < y^* < 0.4$.
3. To clarify the effect of the distance from the observation point to the edge of the active layer on the value of N^*. For example, when the observation point is displaced from the middle of the active layer beyond the edge of the layer (transition from point d to point c), the ionization intensity undergoes a fivefold change ($N_a^* = 2.5$; $N_c^* = 0.5$).

From a comparison of families of curves of equal ionization intensity with different values of one of the layers of the parameter (e.g., $d_a^* + d_{co}^*$), we can determine the degree of the dependence of N^* at any observation point on that parameter.

All of this information is needed for more accurate calculations of the electric field and of voltage–current characteristics of radioisotope neutralizers of static electricity.

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III INTERNATIONAL CONFERENCE ON THERMIONIC DIRECT CONVERSION OF ELECTRICAL ENERGY

D. V. Karetnikov

The III International Conference on Thermionic Direct Conversion was held during June 5-9, 1972, at the Nuclear Research Center at Julich (West Germany). The conference attracted over 200 scientists; 140 papers covering all aspects of direct conversion were discussed. Soviet delegates were afforded the opportunity of becoming familiarized with the principal specialized scientific-research laboratories and firms of West Germany.

A general discussion was held, within the framework of this conference, on the future outlook for thermionic conversion, with state representatives and scientists responsible for the development of thermionic direct conversion in the major industrial countries taking part. During the time elapsed since the II International Conference on Thermionic Direct Conversion held in 1968, the principal landmark in the development of the thermionic method of converting heat energy into electrical energy has been, without question, the successful tests of the two thermionic reactors Topaz-1 and Topaz-2 in the Soviet Union. While the field of thermionic reactor design was characterized as of 1968 by a shift from scientific research and development work to applied engineering activity, the state of the art in 1972 is characterized by practical realization of methods and devices. Operating costs, reliability, and other topics marking the beginning of a transition from the first experimental prototypes to full-scale industrial systems are now on the agenda. Of course, the role of scientific problems is not at all diminished in this area, as demonstrated by, if nothing else, the subject matter and practical significance of the papers presented at the conference.

In the USA research and development program on thermionic energy systems, the principal prospective consumers of thermionic electric power are space systems. Thermionic direct conversion is preferred over all rival systems for the kilowatt power range. The greatest difficulties currently standing in the way of practical realization of thermionic direct conversion systems lie in the area of technology, and primarily in the technology of electric generating channels (EGC). Efforts of USA scientists and engineers have been concentrated on the problem of how to devise reliable electric generating channels with service lives of at least 20,000 h. It is assumed that, by using unitized modular electric generating channels, it will be possible to build several thermionic direct conversion systems based on fast reactors in the range of 10 to 120 kW. The need for service lives of 20,000 h is dictated by the fact that the heaviest prospective consumers of thermionic direct converted energy will be electric jet propulsion engines, which must be capable of powering a large number of planned long-duration space flights. The main hopes for engineering realization of EGC with service lives in that range are pinned on the use of sophisticated technological methods capable of bringing about limiting ultrahard vacuum conditions, and very moderate operating conditions remote from the limiting operations of the EGC per se. Table 1 lists the basic parameters of the first-generation American thermionic direct conversion systems in the design stage.

It is characteristic that, the design of a six-component built-in EGC with a fuel element ventilation channel, which is the optimum design for a 120 kW reactor, has been envisaged for all of the thermionic direct conversion reactors being designed. Replacement of some of the EGC by $U-ZrH_x$ type fuel elements is being proposed for the thermionic direct conversion systems in the intermediate power range. The most important reactor parameters for thermionic direct conversion systems are listed in Table 2. Prior to building a land-based prototype, the Gulf General Atomics Corporation, responsible for designing thermionic reactors, proposes conducting reactor tests on a chain of EGC with total electric power output of

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TABLE 1. Basic Characteristics of Thermionic Converters Designed in the USA

Application	Space vehicle format	Basic parameters							
		useful electric power, kW	reactor total electric power, kW	number of EGC	total weight of thermionic converters kg	weight of reactor, kg	weight of shielding, kg	weight of heat-transfer system, kg	weight of auxiliary systems, kg
Interplanetary system with electric jet propulsion engine	32 modular mercury transverse-thrust engines	120	164	162	3 171	1250	630	843	448
Unmanned communications satellite (two variants)	Reactor and instrument compartment coupled by rigid pipe	5; 10	8; 13,5	19; 19	1 035; 1 190	545; 545	173; 173	228; 274	89; 193
Manned space laboratory	Reactor and compartment coupled	40	55	60	15 950 *	—	—	—	—
	Reactor and manned compartment joined by three kilometer cable	40	65	60	6 270 †	—	—	—	—

*Including 1800 kg of connecting body.
†Including 1090 kg connecting cable.

TABLE 2. Some Parameters of USA Thermionic Direct Conversion Reactors

Thermal power rating, kW	Electrical power rating, kW	Number EGC	Voltage across EGC output, V	Nominal specific EGC power, W/cm²	Peak emitter temperature, °C
350	10	19*	10	1,2	1570
1235	40	60*	9	3,0	1730
1480	120	162	23	3,4	1630

*With moderator.

about 10 kW, and testing a thermionic reactor core on a general-purpose test stand with a thermal power output of 3 MW. Those tests are scheduled for 1977. By that time, research and development work on EGC with service lives of 20,000 h should have been completed. At the present time, the service lives of single electric generating elements (EGE) for in-pile installation have attained 15,000 h, while the service life of the EGC prototype is 5000 h.

Work is now continuing on other concepts of thermionic direct conversion reactors: a fast reactor with externally placed fuel, a reactor with heating of emitters by thermal tubes, a fast reactor operating at high output heat loads with "dry" heat insulation.

In France, the outlook for the development of offshore and ocean-bottom oil fields in the basins of the North Sea and Mediterranean Sea have given a powerful impetus to interest in sources of electric power for ocean-bottom drilling operations. Cost estimates have shown that thermionic direct conversion systems are economically competitive with rival power sources if the cost of the thermionically converted electric power is not greater than 1 dollar per kWh. French direct conversion specialists view those figures as realistic. The volume of research and development work on thermionic direct conversion has increased sharply as a result. While the earlier goal of the French developmental program was limited to development work on EGC, current work in this area extends to reactor design and materials science, etc., with new teams of specialists being drawn into the work in all of France's leading nuclear-research centers. Building a prototype water-moderated, water-cooled reactor with an electric power rating of 20 kW is proposed as a first step. The use of three-element EGC with separate adsorption type cesium sources based on the Sirene type EGE has been proposed; the service life of this EGE is in excess of 5000 h. Tests on the prototype are scheduled for 1975-1976. Below, we cite the basic characteristics of such an ocean-bottom thermionic reactor:

Electric power output, kW	20
Thermal power output, kW	2000
Number of diodes	128
Minimum immersion depth, m	50
Maximum diameter, m	4
Total length, m	10
Total weight, tons	25

Research and development work on a thermionic reactor for spaceflight application, specifically for a satellite-aided television broadcasting system, is being spearheaded in West Germany by the consortium of firms Interatom, Siemens, and Brown-Boveri. Several state-sponsored scientific centers have been drawn into this work. At the present time, work on plans for the first such reactor are nearing completion. The principal characteristics of that reactor are:

Electric power rating, kW	20
Thermal power rating, kW	1400
Number EGC	19
Number of booster fuel elements	400
Output voltage, V	6.5-8.4
Sodium coolant rate, kg/sec	11.5
U^{235} loading, kg	16

The use of zirconium hydride high-temperature moderator in the reactor necessitates a search for ways of keeping hydrogen from gaining access to the EGC, as hydrogen is capable of seriously increasing mass transfer from the emitters to the EGC collectors. Service lives of 3700 h were attained in reactor tests of shortened (three-element) EGC. The consortium must obtain a government order for the fabrication of the reactor prototype in the immediate future; tests are to be staged around 1976.

The generality of the basic solutions was stressed in all of the papers and reports on the building of thermionic direct conversion reactors, so that widespread international collaboration on the basic aspects of thermionic direct conversion is highly favored at the present time.

One of the characteristic traits of recent work on thermionic direct conversion physics is the reanalysis and rethinking being done on the basis of the current level attained in knowledge and technology in this area. For example, several sets of conditions or side reactions and subsidiary phenomena in thermionic direct conversion which were previously dismissed as uninteresting are now being used to good advantage in thermionic direct conversion diagnostics. The long service times required necessitated limiting sterile conditions both in EGC technology and in laboratory experiments, and those special conditions have led to singling out such criteria of vacuum preparation as make it possible not only to monitor, but also to regulate and control, the composition of residual gases in the diode. Under those conditions, long-term and reproducible improvements have been successfully attained in the voltage–current characteristics by adding a very small amount of pure oxygen to the effective gap in thermionic direct converter devices.

The presentation by the leader of the Soviet delegation, the first assistant to the chairman of the USSR State Committee on the Peaceful Uses of Atomic Energy [GKIAE] I. D. Morokhov, in the general discussion on the trends of research on thermionic direct energy conversion in the USSR, was met with great interest, as was the paper presented by V. A. Kuznetsov at the conference on development and testing of the Soviet thermionic direct conversion reactors Topaz-1 and Topaz-2. Thermionic direct conversion reactors of low output ratings (units to tens of kilowatts) and high output ratings (in the megawatt range) are being investigated in the Soviet Union. From the standpoint of practical engineering realization, the investigations are being concentrated on low-output reactors with no booster region with hydride moderator, i.e., on simpler and low-cost prototypes that will be a useful tool at the present state of the art in dealing with general problems specific to thermionic direct conversion systems. The principal problem in this testing program is to confirm the serviceability of EGC clusters as a component part of a reactor system and the reproducibility of performance parameters from one specimen to the next. The tests have yielded a wealth of information on EGC in-core behavior (total service life within reactor about one million component-hours). Each of the reactors has been functioning under power generation conditions for about 15,000 h. Several land-based tests of the Topaz reactor are now scheduled.

Papers by Soviet specialists on fundamental thermionic direct conversion problems occupied a prominent place at the conference. Analytical models of a low-voltage arc whose role has become enhanced recently in connection with the development of thermionic direct conversion diagnostic techniques based on voltage–current characteristics, investigation of stationary and dynamic properties of thermionic direct conversion plasma, stability problems, investigation of voltage–current characteristics, adsorption phenomena, materials testing and materials science problems – constitute the main research trends represented by USSR papers at the conference.

The conference marked an important stage in the development of the thermionic direct conversion method. The conference drew a balance sheet of the basic achievements in realization of national development programs, and enabled leading specialists to discuss the results of scientific-research work, experimental design work, production technology development, and cost investigations.

CONFERENCE OF IAEA EXPERTS ON GAS-COOLED FAST REACTORS

N. M. Sinev and V. M. Shmelev

A conference of IAEA experts on problems connected with gas-cooled fast reactors was held in Minsk, July 24-28, 1972, responding to an invitation by the USSR State Committee on Peaceful Uses of Atomic Energy [GKIAE].

The present state of research and development work on gas-cooled fast reactors was discussed; their characteristics were compared to the characteristics of fast reactors with liquid-metal coolants; estimates of losses and of efforts needed to develop gas-cooled fast breeders; the outlook for international collaboration in that field.

Twenty-one reports on fast breeders with helium, CO_2, or dissociable N_2O_4 gas as coolants were presented and discussed. Successes achieved in recent years in various countries in the development of high-temperature gas-cooled reactors were noted. Close attention was given to discussions of the requirements for a promising fast breeder type, particularly in relation to doubling time and safety conditions.

The participants at the conference noted that a gas-cooled fast breeder can, in principle, attain the necessary doubling time (6 years) and can be regarded as a trend parallel to or backing up liquid-metal fast breeders.

Two concepts were put forth on fuel utilization and on the doubling time. Some of the participants felt that the basic problem facing fast breeders during the next 20 to 25 years is the production of the cheapest electric power competitive with the electric power obtainable from nuclear power stations with thermal reactors, rather than cutting down on the needs for natural fuel. These specialists feel that fast breeders must be optimized not for the minimum doubling time but for the minimum cost of electric power when the doubling time is greater than seven or eight years.

The concept entertained by the second group is based on the fact that fast breeders must solve a raw materials problem within the next 20 to 25 years, since the doubling times for breeder reactors must be much shorter than the doubling times in conventional and nuclear power generation.

Reports by S. M. Feinberg and N. N. Ponomarev-Stepnoi (USSR) suggested two variants for building a helium-cooled fast breeder: 1) a small fast breeder with a plutonium core, with an expected breeder time of the order of three years, helium pressure 300-400 atm, and reactor exit temperature 700°C (hard neutron spectrum and high breeding ratio ≈2); 2) a large fast breeder with a diluted core and longitudinal –transverse (cross) flow of gas through clusters with fuel microelements of breeding ratio ≈1.6, and helium pressure 150 to 200 atm. Those papers discussed several variants of designs of nuclear power stations with small-core reactors: a reactor with a multiregion plutonium core (several plutonium cores accommodated within a single metallic uranium reflector unit in a common metallic or concrete pressure vessel); a modular nuclear power station layout (combining several, i.e., 5 to 12, reactors with plutonium cores, each within a separate metallic vessel ~2 m in diameter). The reactor pressure vessel together with the steam generator units, gas blower, etc., are housed in a reinforced concrete containment structure, i.e., the layout of the facility is structurally integrated.

P. Fortesquieu (USA) made a survey of the work done by the Gulf General Atomic Corporation on gas-cooled fast breeders. He presented layouts and designs of the basic components and subsystems of a

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demonstration fast helium-cooled 300 MW(e) reactor, and cited the basic characteristics of the prototype of a commercial helium-cooled fast breeder designed for 1000 MW(e) output. These characteristics were compared to the corresponding characteristics of liquid-metal fast breeders. In Fortesquieu's opinion, when the temperature of the gaseous coolant does not exceed the temperature of the sodium, a helium-cooled breeder provides a solution of the problem of safety and servicing. The corporation is making use of extensive experience accumulated in its work in the design of gas-cooled thermal reactors. The corporation considers the performance of the 1000 MW(e) gas-cooled fast breeder in a system of four high-temperature 1000 MW(e) gas-cooled thermal reactors to be a feasible arrangement.

An assessment of the present status and developmental outlook of gas-cooled fast breeders was given in papers submitted by a representative of the European Association of Gas-Cooled Breeder Reactors, Sherman. The association unites 15 organizations rooted in eight European countries (Austria, Belgium, Switzerland, West Germany, France, Great Britain, Italy, and Holland). The problems under consideration include assessments of the potential capabilities of gas-cooled fast breeders, developments of preliminary projects, coordination of research, and exchanges of information on the topic.

At the present time, three tentative projects of 1000 MW(e) reactors are under consideration: one with rod-type fuel elements and reactor exit helium temperature 587°C, another with fuel in microelements and CO_2 temperature 700°C at the reactor exit, a third with fuel in microelements and reactor exit CO_2 temperature 650°C. An integrated layout in a prestressed reinforced concrete containment structure is used in all three variants.

Very close attention is given, in the Association's program, to development and testing of fuel elements, fuel compositions (based on oxides and carbides of plutonium and uranium) and structural materials. It is shown that a gas-cooled fast reactor with rod-type fuel elements is close, in its capital outlay, to a high-temperature gas-cooled thermal reactor.

The status of work on gas-cooled fast breeders in West Germany was described by M. Dallidonne. In 1971, the West German nuclear-research centers at Karlsruhe and Julich entered upon a joint program (1971-1974) involving a total sum of 6.5 million dollars, in the development of gas-cooled fast breeders. About 90% of that sum is earmarked for the development of a 1000 MW(e) reactor project featuring gas cooling and ventilated oxide fuel elements in stainless steel cladding. A nuclear power station with a gas-cooled fast breeder is being built in a two-loop steam-turbine arrangement. The remaining 10% is earmarked for development research (vanadium cladding for fuel elements, clad microelements, a gas-turbine cycle).

The report takes into account the fact that a fast breeder project featuring gas coolant, stainless steel-clad oxide fuel, and a steam-turbine cycle, can be realized in the immediate future. But the following steps will be required for that engineering realization:

1) radiation tests on fuel compositions and models of the prospective fuel elements in existing research reactors;

2) test-stand trials of the thermodynamic and gas-dynamic characteristics of the fuel elements;

3) rapid completion of the first drafts and engineering designs and detailed engineering cost analyses based on them, and safety measures based on quite rigorous requirements.

A staffmember of the Swiss Federal Reactor Research Institute, G. Sarlos, gave an account of research done at the Institute.

An analysis of several variants of nuclear power stations with fast gas-cooled breeders is being carried on in a joint project with Swiss industry. Specific emphasis is being laid on a single-loop variant with a gas turbine and a double-loop variant with a steam turbine, with integrated and nonintegrated equipment arrangements. The single-loop gas-turbine variant with nonintegrated equipment arrangement is, so far, in front as a solution of engineering cost problems.

But difficulties in the development and fabrication of equipment have led to some preference for the double-loop variant and a steam-turbine cycle, even though the latter is inferior to the single-loop gas-turbine variant in engineering cost performance, in that it does have distinct advantages to offer in technology and in the solution of safety problems.

Tanaka (Japan) reported that the development of a gas-cooled fast breeder has been underway in Japan since 1967. Starting with 1970, options have become narrowed to the development of a 1000 MW(e) reactor with helium coolant, rod-type fuel elements with stainless steel clad uranium dioxide and plutonium dioxide fuel, and a double-loop arrangement with a steam-turbine cycle.

Reports by Comellini (Italy) and Videra (Sweden) presented a picture of the status of work on gas-cooled fast reactors in their respective countries.

International collaboration in west European countries within the framework of the European Atomic Energy Agency was dealt with in a report by a representative of EAEA, L. Boxer.

There was substantial interest shown in reports by A. K. Krasin, V. B. Nesterenko, A. M. Sukhotin, and G. A. Sharovarov on various aspects of the development of gas-cooled fast reactors using dissociable gases as coolants. The reports demonstrated that results of theoretical and experimental research on a 1000 MW(e) fast reactor with N_2O_4 coolant and working fluid, carried out at the Nuclear Power Institute (IYaE) of the Belorussian SSR Academy of Sciences, supports hopes that nuclear power stations based on such reactors will meet all of the requirements imposed on fast breeders. Efficiencies of 30-33% and doubling times of six to seven years are possible for such a reactor at loop pressures of 160 atm and exit gas temperatures of 350-450°C.

During a visit to the Nuclear Power Institute of the Belorussian SSR Academy of Sciences, the conference participants were shown test stands devised for the study of various topics related to uses of N_2O_4 as coolant in the reactors and as turbine working fluid.

The conference reached the conclusion that research and development work on gas-cooled fast reactors now in progress in many laboratories of the countries represented at the conference indicates a promising trend in nuclear power development. The feasibility of continuing and intensifying research and development work in that direction was acknowledged.

With that object in mind, the conference recommended that IAEA review ways of bringing about further exchanges of information between the countries interested in the development of gas-cooled fast breeders, either within the framework of the existing international work group on fast reactors, or in whatever alternative convenient way.

On the whole, the conference showed that interest in gas-cooled fast reactors is growing, and that research, development, designing, and planning work aimed at the building of prototype facilities is well under way in a number of countries.

The conference proceedings will be published by IAEA.

VI INTERNATIONAL SYMPOSIUM ON RADIATION EFFECTS ON STRUCTURAL MATERIALS

P. A. Platonov

The VI International Symposium on the effects of radiation on building materials, organized under the auspices of the American Society for Testing Materials, was held June 26-28, 1972, in Los Angeles (USA).

The symposium was attended by leading specialists from American national centers and from firms conducting research and development work in the field of nuclear power, as well as by representatives of research centers and universities of other countries. Forty-four papers were presented on the following topics: 1) mechanics of failure of reactor pressure vessel steels; 2) effect of structure and impurities on the properties of pressure vessel steels; 3) microstructural changes in response to neutron bombardment (pore formation, separation of phases, pore formation through the action of charged particles); 4) mathematical simulation; 5) mechanical properties (ductility and plasticity, creep, fatigue, strength).

The reports under the first heading dealt primarily with development work on reliability criteria for pressure vessels based on materials failure mechanics, and finding correlations between those criteria and standard mechanical properties. The possibility of ascertaining the failure strength coefficients from conventional Charpy type specimens with notch fatigue cracks introduced prior to the tests was discussed, as well as several new methods for estimating resistance to brittle failure (resistance to crack propagation) in static and dynamic bending tests.

As we know, the behavior of several low-carbon and low-alloy steels employed in the fabrication of reactor pressure vessels to radiation-induced embrittlement is quite different. In some reports on the effect of the composition of pressure vessel steels on their stability to radiation, the mechanism by which individual impurities (copper, nitrogen, carbon) act has been studied from the standpoint of the interaction between those impurities and defects present. Stress was placed on the role played by nitrogen impurities in the state in which the steel exists (in solid solution or in a bound state in the form of nitrides and carbonitrides). Those papers do introduce a certain amount of clarity into our understanding of the mechanism underlying the harmful effects of copper on the radiation stability of steel, and on the temperature stability of radiation-induced changes.

The problem of radiation damage to stainless steels, and above all pore formation and swelling in steels, was discussed in greatest detail. Data were cited on swelling of stainless steels bombarded by integrated neutron fluxes greater than 10^{23} neutrons/cm^2, or doses of irradiation by heavy ions (C^+, Ni^+, Ta^+) equivalent to those fluxes, at temperatures from 450 to 700°C. The most important conclusions forthcoming from those studies are: 1) saturation of the swelling effect in stainless steels is absent at integrated fluxes 10^{23} to $8 \cdot 10^{23}$ neutrons/cm^2; 2) swelling increases approximately as a linear function of the irradiation dose. According to data on exposures of steel specimens in a fast reactor (EBR-II), swelling of grade steel 304 in an integrated flux $1.1 \cdot 10^{23}$ neutrons/cm^2 amounted to 11 vol. % (R. Fish et al., USA).

When steels of grades 304 and 316 were irradiated by heavy ions in doses of $3 \cdot 10^{23}$ neutrons/cm^2, swelling amounted to 30%. Grade 304 steel swells to 30 vol. % at integrated fluxes to $2 \cdot 10^{23}$ neutrons/cm^2.

Note that preliminary deformation of the steels reduces swelling (twice or three times as much as in the case of austenitized steel) only in the temperature range below 600°C (V. Johnston et al., USA). At integrated fluxes from $3 \cdot 10^{23}$ to $4 \cdot 10^{23}$ neutrons/cm^2 at 625°C, swelling is virtually the same for either

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austenitized steel or previously cold-worked steel. Although all of those data were obtained from exposures of steel specimens to beams in heavy-ion accelerator systems, there could hardly be any doubt as to their representativeness. Detailed investigations have demonstrated the reasonably good correlation between the pile-irradiation data and the accelerator-exposure data over a broad range of exposure doses.

Methods devised for measuring swelling of thin specimens irradiated in accelerator facilities directly are of interest in that respect. Those methods can be used to demonstrate the adequate representativity and the quantitative nature of the results obtained when transmission electron microscopy is relied on.

Hence, the results presented at the symposium indicate that some difficulties can be encountered in the selection of material and in the solution of reactor core engineering problems, in attempting engineering realizations of fast reactor projects designed for high burnup levels. Note also that preliminary cold working of steel, which seemed earlier to be a solution of the swelling problem, is not a sufficiently efficacious measure at higher integrated fluxes and at elevated temperatures, and may result in real gains only when the temperature is brought below 600°C.

In referring to ways of diminishing swelling of materials through alloying, we must point out that there is as yet no unified point of view on the nature and underlying mechanism of the effect of alloying components on the behavior of the materials.

Some of the papers presented at the symposium (K. Garr et al., USA; V. Appleby, W. Wolf, USA; K. Ehrlich, N. Pakan, West Germany) demonstrated the essential role played by separation of excess phases within the grain, which can serve as nuclei for pore formation in some instances. Hence the negative estimate of stabilized 347 steel as compared to unstabilized grades 304 and 316. We could also point out the lesser swelling of the stabilized steel 15Gr–13Ni–Nb (4988) fabricated in West Germany, as compared to steel grades 304 and 316. That grade 321 steel swells less than grades 304 and 316 was also pointed out.

Some results of an investigation of high-nickel alloys (H. Breger, J. Straalsund, USA) reporting detection of a slight increase in swelling over that typifying steels is of interest from the vantage point of estimating the effect of composition on swelling. It is assumed that swelling is hindered by disperse particles (Ni_3Nb and Ni_3Ti) present in dispersion-hardening steels (such as PE-16), in that those particles impede the formation of dislocation loops, and thereby contribute to mutual annihilation of Frenkel pairs. The absence of appreciable swelling in alloys that have no disperse precipitates in their structure (e.g., Incaloy-800) does not yet have a sufficiently warranted interpretation. While exhibiting advantages over steels as far as swelling is concerned, these alloys are however subject to rather intense high-temperature embrittlement.

Among the papers not directly concerned with the practical aspects of the study of radiation effects on materials, we might mention a few on investigations of the formation of an ordered structure of pores and dislocation loops in pure metals. A survey paper presented at the symposium (G. Kulginski and J. Brimell, USA) gives grounds for the assumption that valuable information on the mechanism of pore formation and methods of controlling swelling can be obtained in the future in this area.

Presentation of reports on investigation of such properties of fast-reactor stainless steels which can be deemed applicable to thermal reactors as well, though to a lesser extent (fatigue, creep, long-term strength, ductility), was a typical feature of the symposium. Data presented in the papers attest to the substantial effect of irradiation by fast neutrons on the long-term strength characteristics (decrease in long-term strength and ductility, rise in creep rate). The effect of those changes (e.g., 10 to 100 times increase in creep rate) emphasizes the need for detailed treatment of those effects when designing structural components for the cores of fast reactors.

We also draw attention to papers on investigations of radiation effects on the mechanical properties of Zircaloy-2. Creep in specimens and tubes made of that alloy was investigated under irradiation conditions (D. Wood, Britain). Equations relating creep rate to testing conditions at testing times to 40,000 h. The properties of Zircaloy-2 tubes irradiated in the form of fuel-element cladding (D. Hardy, Canada). One major conclusion in that paper is that the initial ductility of the alloy must be 25 to 30% if fuel-element cladding is to be serviceable.

In addition to investigations of the use of materials in thermal and fast reactors, papers were presented on the study of radiation effects on refractory materials (niobium and niobium alloys) with applications to their use as structural materials in thermonuclear machines.

A broad range of materials science problems in reactor design of maximum urgency was covered in the symposium. The symposium materials are to be published as a collection of articles in early 1973.

JULY 1972 INDC SESSION

G. B. Yan'kov

The International Nuclear Data Committee (INDC) has been known by that title since 1968. An International Nuclear Data Work Group existed earlier. The regular (fifth) scheduled session of the INDC was held in Vienna, July 17-21, 1972. Representatives of many countries, as well as representatives of IAEA, Euratom, and the Saclay nuclear data center, took part in the deliberations. Chairmen of the committee are reelected every two years; a representative of the USSR (L. N. Usachev) has been chairman of the group since January 1, 1972.

After the minutes of the preceding session had been approved, discussions of follow-ups on concrete recommendations and the report of the 1970-1971 INDC chairman G. Kolsted (USA) were heard, and the committee proceeded to a discussion of experimental research performed during the preceding year.

Attention continues to be centered on measurements of constants in fields where the greatest discrepancies of data exist for isotopes where reactor performance is seriously concerned; α ratios for U^{235}, Pu^{239}, fission cross sections of U^{235}, Pu^{239}, and other nuclides, inelastic scattering cross sections and neutron capture cross section of U^{238}, average number and spectra of fission neutrons, and capture cross sections for fissile and structural materials. The experiments have been characterized not only by a decrease in the error of data in measurements taken at different points, but also by efforts to secure the most precise results over a broad range of neutron energies under the same experimental conditions. For example, values for 0.02 eV to 400 keV have been obtained for the α of Pu^{239} (USA data), from 0.01 to 1 MeV for the α of U^{235} and Pu^{239} (USSR data), from 0.8 to 30 MeV for (n, f) of U^{235} (West German data). No success was achieved over the past year in clarifying the reasons for this discrepancy in the data obtained under different conditions, and for such important quantities as the value of $\bar{\nu}$ for Cf^{252}, or characteristics of the spectra of prompt fission neutrons of U^{235} and Pu^{239}; the last topic was the subject of a special international gathering of consultants held in August, 1971; the recommendations forthcoming from that gathering were approved by INDC.

Reference standard cross sections used played a significant role in comparing the results of experiments carried out with high precision. The following reactions recommended as suitable standards and presented as such by an INDC subcommittee on standard cross sections were approved at the July session: H^1(n, p); Li^6(n, α); B^{10}(n, α); Au^{197}(n, γ); U^{235}(n, f); and He^3(n, p). It was acknowledged that further precision will be needed in cross sections of those processes, e.g., U^{235}(n, f), and particularly so in the energy range from 0.1 to 1 MeV, or Li^6(n, α) at energies upwards of 100 keV. Standard reactions will be discussed in detail at a special IAEA conference to be held in November, 1972.

At the present time the volume of work on estimates of nuclear data, with generous use of computers and data processing technology, has expanded appreciably. Several countries, the USSR included, are transmitting their own estimated data to IAEA to expedite data exchanges. However, the committee has an understanding on exchanges of estimated data solely for the standards enumerated above. In view of that, the problem of how to go about publishing an international bulletin on estimation work in progress or being planned has not met with a positive solution. An INDC subcommittee on discrepancies in the most important nuclear data and on estimates of nuclear data has decided to restructure its work by distributing responsibilities for those variables and constants among the members of the subcommittee: each subcommittee member must submit to the subcommittee chairman, three months prior to a regularly scheduled INDC session, a review of the data accumulated on the aspect to which he has been assigned, so that the entire assemblage of relevant data can be discussed at the scheduled INDC session.

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A topical seminar on inelastic scattering of fast neutrons that was held concurrently with the INDC session heard eight reports devoted to developments in techniques, measurements, estimates, and theoretical calculations of data; the greatest attention was placed on new U^{238} investigations.

Four research centers (Brookhaven, Vienna, Obninsk, Saclay) are engaged in collecting, reviewing systematizing, processing, and storing nuclear data, organising estimates and exchanges of nuclear data. Exchanges of experimental data are expedited on the basis of the EXFOR system; the committee accepted a proposal to include data on fission products in that system. Results of a conference held by those four research centers in October, 1971, were also discussed, as well as various aspects of data exchanges, and specifically times required for obtaining data requested and reasons for delays in answering inquiries and requests. A step-up in the flow of information in recent years can be followed against the number of reports in SINDA compendia: 55,000 in 1969, 75,000 in 1971, 87,000 in May, 1972.

The first edition of the world list of requests for measurements of nuclear data (WRENDA) scheduled for the end of this year was discussed in detail at the session, and the sequencing of subsequent editions was also discussed. It was resolved to turn to the participating nations and to the reactor division of IAEA with a request to begin a continuing work program on engineering cost justifications of allowable errors in reactor and shielding parameters traceable to inaccuracies and uncertainties in nuclear data. It was pointed out that one standard deviation constitutes an error in WRENDA. The committee approved, and recommended for rapid dissemination, a list of requests for nuclear data drawn up by the subcommittee on guarantees and prepared at present on the basis of data from the USSR, the USA, and West Germany, for the purpose of assisting in the development of engineering methods in a system of guarantees, and also discussed the work done to date on compiling a similar list for thermonuclear reactors.

Nonneutron nuclear data were high on the INDC agenda. Comprehensive calculations of fission reactors and radiation shielding, engineering work on thermonuclear reactors, engineering methods in a system of guarantees, activation analysis, applications of radioisotopes and accelerators in medicine, biology, and industry, and also astrophysics and nuclear physics, require the collection and storage, review and systematization, compilation and estimates of large volumes of nuclear spectroscopic information of a nonneutron nature. In addition, we are still lacking systems of international exchange of nonneutron information. IAEA has set up an international work team responsible for data on the structure of the atom and nuclear reactions. In March of 1972, the team held its first session in Vienna, to discuss the present state of the art in the field of compiling and estimating nonneutron nuclear data in various countries, and discussed fields of application of those data. A preparatory committee (Canada, USSR, USA, France) then drew up an agenda for an international symposium on applications of nuclear data (both neutron data and nonneutron data) in science and industry, to be held under IAEA auspices in March, 1973, Paris. The entire discussion at the INDC session revolved around a single topic, specifically, whether that group should be an INDC subcommittee or an independent unit in the solution of all related matters except certain recommendations of a financial nature. The decision was put off to the following INDC session, in order to allow an opportunity to take into account information becoming available while the symposium was being prepared and held.

In 1972, IAEA set aside 15,000 dollars to assist developing countries in acquiring specimens and targets for research in nuclear data. The committee familiarized itself with requests coming in, and approved a procedure for distribution of specimens and targets, emphasizing the point that support must be given to work projects that dovetail with the lists of demands and requests.

The third IAEA conference on nuclear data is scheduled for 1974. The first session of the IAEA preparatory committee will meet in November, 1973.

The next INDC session will be held in Vienna in October, 1973.

SECOND INTERNATIONAL CONFERENCE ON HIGH-β PULSED PLASMA

Yu. V. Skvortsov

The second conference on high-β pulsed plasma was held July 3-6, 1972, at the Max Planck Plasma Physics Institute in Garching (West Germany). The conference drew participation from 143 physicists hailing from 11 countries, including the Soviet Union.

A total of 64 reports was heard, and these were discussed at seven panels which did not overlap in time allotments.

The first panel dealt mainly with fast toroidal Z-pinches and θ-pinches. The audience heard with special interest a report on results of experiments conducted on the toroidal sector (1/3 the total length) of the Scyllac machine, which features a pinch whose major diameter is 5 m (Los Alamos, USA). The purpose of the experiment was to compensate the toroidal drift of the plasma with the aid of additional spiral ($\ell = 1$) windings and a cusped magnetic field formed by circular recesses in the chamber walls ($\ell = 0$). The experiments showed that the plasma lifetime τ in the system was 1 to 2 μsec when the correcting fields were shut off. When the parameters and configuration of the compensating fields were carefully assigned, τ could be increased successfully to 10 μsec. That value is commensurate with the time it takes a hot plasma to flow through the end faces of the Scyllac sector.

The remaining papers presented in this panel dealt with toroidal systems with longitudinal current. The configuration of the magnetic fields in some of those systems was similar to the configuration brought about in tokamak type quasistationary machines. The experiments generally amounted to a verification of the Kruskal–Shafranov criterion for a plasma such that $\beta = 0.1$ to 0.5.

The second panel was devoted to stability theory and equilibrium theory. That panel heard nine reports, dealing mainly with the reasons behind the higher stability (as an acknowledged experimental fact) of high-systems compared to machines where $\beta \ll 1$. The reports were presented by the two groups of scientists from Garching and Los Alamos. The difference in the techniques employed in the work of these two teams is quite intriguing. The former conducted their research in what could be termed classical style: they analyze the equations describing the process and attempt to find solutions for the rates of development of instabilities, spectra of oscillations, etc., in analytical form. The Los Alamos scientists, using the increasing capacity of digital computers to advantage, "break through" the solutions of the most complicated systems of equations by brute-force numerical methods, and seek out new effects in that manner. A report by Freidberg and Lewis using a numerical solution of the kinetic equation to show that the rate of development of instabilities such at $m \geq 2$, in a high-β plasma, can be well below the rates computed with the aid of a linearized system of equations in ideal magnetohydrodynamics.

The third panel discussed linear θ-pinches. Modifications of this system made their appearance: a θ-pinch with a multipolar magnetic field, and a "negative" θ-pinch [$B(t) = B_0(1 - \cos \omega t)$]. These devices fail to yield any distinct advantages in generating a dense and hot plasma.

It is worth pointing out that the role of numerical experimentation clearly comes to the forefront in the reports on direct θ-pinches. It becomes an important method for investigating plasmas, in fact. At the present time, experiments are being planned in such a way that direct measurements and numerical calculations will supplement one another. In that way, the energy balance on the Isar II machine (West Germany; θ-pinch 1 m in length, 10 cm in diameter, energy storage of capacitors 500 kJ) was arrived at, and it was shown that plasma energy losses are determined by the electronic thermal conductivity and by the outflow of plasma, similar to the gas flow pattern in Laval nozzles.

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Results of measurements performed on the 5 m linear θ-pinch powered by the Scyllac battery (Los Alamos) were somewhat surprising. It was found that energizing the magnetic mirrors brings about a rapid development of an instability in the plasma column. Those data are at variance with experiments conducted on the Scylla IV machine (also at Los Alamos), and the reasons for those discrepant effects are under scrutiny.

A separate session discussed collisionless shock waves. This method of generating a high-temperature plasma is proposed for use either in order to fill up magnetic traps of the stellarator type or, when the plasma lifetime is sufficiently long, in order to provide an independent source of neutrons. In the latter case, the skin time must be large, and the physics of thin current sheaths must be known. High ion energy is required in both cases. In a convergent cylindrical shock wave it is proportional to $E \cdot n_0^{-1/2}$, where E is the intensity of the electric field in the plasma, and n_0 is the initial density of the gas. In the reports submitted (mostly by German scientists), the voltage on the periphery on the coils varied from several tens of kV to 500 kV, the initial density varied from $3 \cdot 10^{10}$ to $5 \cdot 10^{12}$ cm^{-3}. The rate of implosion in these experiments was 10^8 cm/sec and the neutron pulse commenced well in advance of the cumulation of the plasma on the system axis, since the target was a cold gas. During the implosion stage, the plasma temperature reached 10 keV, and the density was 10^{14} cm^{-3}, $\beta = 0.5$. A fairly broad range of values is cited for the width of the current sheath, in both theoretical and experimental contributions to the literature. The experimentally derived values varied from 3 to 8 c/ω_{pe} (Garching) to c/ω_{pi} (Julich). Only one of the theoretical contributions showed close agreement with experiment. That article treated the buildup of a nonlinear ionosonic wave moving opposite to the discharge current. It was shown that the critical value of the current velocity is $2 \cdot \sqrt{(m_e/m_i)} V_{T_e}$ over a broad range of assigned parameters, while the width of the sheath is c/ω_{pi}.

Investigators of plasma-focus systems were extremely active at the conference. In addition to the planned session set aside for the presentation of the largest number of reports on the topic, an additional discussion was arranged to cover the dynamics of current sheaths and mechanisms underlying generation of high-temperature plasma and fast particles. There is virtually complete clarity on the first topic (dynamics): careful comparisons of experimental findings and calculations demonstrates complete agreement all the way up to the point of peak compression. Nothing definite can be said yet of heating. At least three plausible reasons for the appearance of a heated plasma by the time the peak intensity of neutron radiation is reached (second pulse) are under discussion: adiabatic compression, turbulent heating, and dissipation of the magnetic energy associated with current vortices (Bostick, USA). As for the mechanisms underlying generation of fast particles, reports by N. V. Filippov (USSR) and Bernard (France) demonstrated the existence of special sets of conditions under which fast protons are generated. In Bernard's view, these conditions are typical for discharges with a low initial pressure. All of these inaccuracies and discrepancies can apparently be removed with success in the coming years – a procedure featuring a resolving time of 1 nsec was demonstrated at the conference. The main point is that the plasma implosion process in plasma-focus systems is being calculated in broad outlines right up to level of quantitative estimates of neutron yields. That makes it possible to proceed ahead to the planning, design, and building of new and more powerful machines. This trend is supported not only by the desire to come closer to the Lawson criterion in thermonuclear research, but also by the practical interests of firms engaged in the development of pulsed x-ray and neutron sources. In any case, the design of a 1 MJ facility has been worked out at Frascati (Italy), and this project will be completed in the immediate future under Euratom auspices.

The basic topic covered by the sixth panel was θ-pinches with a discharge chamber of rectangular cross section ("ribbon" chambers, "thimble" chambers). Only projects of experimental facilities and their theoretical foundations were discussed at the 1971 Madison conference, while the first experimental results were reported out at this conference. It was found that the plasma in these devices is quite stable, and the plasma lifetime is determined by the diffusion time of the magnetic fields. A Soviet report (Yu. F. Nasedkin, G. B. Levadnyi, et al.) on a ribbon θ-pinch project was heard with great interest and discussed actively in the corridors. It differed from foreign machines in the high value of the magnetic fields (100 kG).

The last panel ("Miscellaneous") was devoted to a broad range of topics: from stabilization of instabilities of feedback to calculations in the design of a pulsed thermonuclear reactor. The continuing and unremitting searches for new ways of generating high-temperature plasma is one of those topics that deserves special attention here. In addition to familiar suggestions on heating the material by means of electron beams and laser radiation, some brand new ideas also came forth for discussion. For example, Cheng (USA) passed 1 MA current through a thin-walled deuterium-loaded metallic tube. The overvoltage occurring during the current rest as the tube exploded was found to be sufficient to cause breakdown of the

deuterium. The gas was heated by the discharge current to a high temperature, and the disintegration of the gas was slowed down by the heavy envelope of vapor from the tube metal. Neutron radiation was recorded. Cheng is of the opinion that $n \cdot \tau \approx 10^{14}$ can be attained at a current of 25 MA.

A report by Boris et al. (USA) on a machine project featuring an ultrastrong (2.5 MG) magnetic field and a positive energy balance was also of interest. The field is established by a collapsing metallic liner of initial diameter ≈1 m and length 10 m. The energy stored by the inductive storage units is 200 MJ. Preparations for practical realization of this project are now underway.

The work of the panel, and with it the conference, ended with a report by Ribi (USA) on the results of self-consistent calculations for a thermonuclear reactor based on a two-stage toroidal θ-pinch. The reactor design power is 1750 MW, the thermal load on the wall is 3.5 MW/m^2; the plasma parameters: $n \approx 10^{16}$ cm^{-3}, and T = 15 to 30 keV. The plasma in the reactor is generated by fast shock heating, and is then confined for 100 msec by a slow field. The duty cycle of the firing pulses is 10^2. Some advantages of pulsed reactors over stationary reactors are pointed out (small size, absence of flows of liquid metals in magnetic fields, etc.).

VI INTERNATIONAL CYCLOTRON CONFERENCE

N. I. Venikov

The VI International Conference on cyclotrons was held in Vancouver, Canada, July 18-21, 1972. Approximately 80 reports were presented.

Very few new projects were reported on at this conference, in contrast to previous conferences. This demonstrated the unflagging attention being given to acceleration of heavy ions and the steadily increasing interest in the application of cyclotrons in various branches of science and industry, and especially in medicine and materials science.

The most remarkable achievement in cyclotron engineering in the course of the past year has been acceleration of heavy ions in a two-cyclotron system at JINR (Dubna). By injecting ions accelerated in a 3 m cyclotron into a 2 m cyclotron, and effecting charge transfer between the ions in the latter machine, colleagues of G. N. Flerov have succeeded in obtaining xenon ions with energies of 7 MeV/nucleon, and beam intensities of $2 \cdot 10^{10}$ particles $\cdot$ sec^{-1}, and have begun experiments on the synthesis of superheavy transuranium elements. The Alice facility built for the same purpose at Orsay (France), and consisting of a linear injector-accelerator and a 2 m isochronous cyclotron, produces krypton ions of 505 MeV energy, with beam intensity of $2 \cdot 10^{8}$ sec^{-1}. Reports submitted to the conference gave accounts of intensive work on the production of accelerated heavy ions on the isochronous cyclotrons built at Berkeley, Oak Ridge, and the University of Texas (all USA).

Two projects of hybrid accelerator facilities were developed primarily for the production of accelerated heavy ions. The first is the Oak Ridge laboratory project, envisaging injection of ions into a ring cyclotron from a 20 MV tandem accelerator; this project has been modified slightly. Now, in addition to the tandem accelerator, an isochronous cyclotron in service at that laboratory is to be incorporated into the project. Uranium ions will be accelerated possibly to energies of 10 MeV/nucleon in the first case, and tantalum ions to 6 MeV/nucleon in the latter case.

The second project incorporates the annular cyclotron of the Hahn-Meitner Institute in Berlin, where a 7 MV electrostatic accelerator now in service has been proposed as the injector. Ions from carbon to argon are to be accelerated at intensities $\approx 10^{12}$ sec^{-1}, to energies of 20 MeV.

The Blosser group at the University of Michigan has been conducting preliminary research on a project involving a compact cyclotron with variable energies up to a peak proton energy of 200 MeV. A 50 MeV machine already in service has been used as a model in this project. The new cyclotron will be a steeply spiral action machine, but with the same pole gap, and four sectors instead of three. The characteristics of the extracted proton beam will be: spread $2 \cdot 10^{-4}$, horizontal emittance 0.3 mm$\cdot$mrad, vertical emittance 2 mm$\cdot$mrad. The cost of the cyclotron itself will be about 2,000,000 dollars.

Construction work is continuing on five large cyclotrons, and the status of the work was reported on at the conference.

1. The largest cyclotron involved is the Triumph machine (being built at Vancouver, Canada), pole diameter $\approx$17 m; for accelerating negative hydrogen ions to 500 MeV at 100 μA beam intensity. Assembly of the cyclotron components is now in progress, and the ion source and injection system are receiving their finishing touches. The first accelerated beam of negative ions is scheduled for the end of 1973. The cyclotron plus the building housing it will cost 29,000,000 dollars.
2. The SIN meson factory in Zürich (Switzerland), consisting of two cyclotrons: a 72 MeV injector and a 590 MeV annular cyclotron. The injector cyclotron was built by the Philips firm, and can

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also be used independently, while its design is geared not only to proton acceleration, but also to acceleration of deuterium ions, helium ions, and various heavy ions. The proton beam is to be produced at design power during 1974.

3. The annular cyclotron, a 200 MeV proton accelerator, in Indiana (USA), with a cyclotron injector. The first accelerated beam must be produced in late 1973. Acceleration of heavy ions all the way up to calcium is proposed with this machine.
4. The U-240 cyclotron of the Nuclear-Research Institute of the Academy of Sciences of the Ukrainian SSR [IYaI AN USSR] at Kiev, accelerating protons to 80 MeV in the spectrometric mode. A system of external monochromatization of ions by the rf field of the buncher, as proposed by Yu. G. Basargin, is planned as part of this cyclotron. The system must furnish 20 μA current to the target with an energy spread of $2 \cdot 10^{-4}$. Plans call for using the cyclotron as a high-level neutron source for time-of-flight studies. A 10 MV tandem injector is to be installed in the future, and that should make it possible to accelerate very heavy ions (up to xenon) at beam intensities $\approx 10^{12}$ sec^{-1}. The cyclotron is scheduled to go into service during 1973.
5. The cyclotron of the Catholic University (Belgium) designed for 80 MeV proton energy, and fabricated by the CSE-Tomson firm (France). Adjustment work is in its final stages.

Work is in progress on the conversion of three large synchrocyclotrons: one at CERN (Geneva), one at Columbia University (USA), and one at JINR (Dubna), and the changes should result in qualitative improvements in the experimental capabilities of those machines.

Interest in axial injection of ions has fallen somewhat from its peak at the preceding conference. Axial injection is apparently viewed only as a future tool for the production of polarized or negative ions. It was precisely for that purpose that the axial injection systems at the cyclotrons in Grenoble, where an extracted beam of polarized protons with intensity to 0.13 μA, and in Birmingham, were improved.

Several reports dealt with applications of digital computers in the control systems of both accelerators now being built (the Triumph machine in Canada and the Indiana cyclotron) and machines already in service (the Oak Ridge cyclotron and the University of Michigan cyclotron, the cyclotron at Eindhoven in the Netherlands, the Julich cyclotron in West Germany). Close interest was shown in the development of instruments for sensing the phase, intensity, and position of a beam.

Reports on applications of cyclotrons in various branches of science and industry, and particularly in medicine and in materials science and materials testing, were received with keen interest. Cancer research programs are using the Harvard University synchrocyclotron and the compact cyclotron at the Argonne hospital, while programs of cancer treatment, beam surgery, diagnostics utilizing short-lived isotopes, activation analysis of calcium content in bones, etc., are centered around the cyclotron at the University of Washington, and many other machines. Techniques and results of therapy in treating malignancies of various organs by irradiation with cyclotron-produced fast neutrons and protons have been discussed in reports submitted by medical physicians at conferences. The advantages of a beam of fast protons over γ-ray therapy have been demonstrated convincingly: there is substantially less skin burn, and there is a more favorable depth distribution of exposure dose. A special 200 MeV (proton) cyclotron with a large number of beams for irradiating patients (and capable of handling 220 patient exposure sessions a day) has been proposed in Sweden for cancer therapy. The cost of this accelerator, with the necessary accessory equipment and the building housing the machine, will be 7,000,000 dollars.

A report was made at the conference on one further important application of cyclotrons: in the study of the effect of radiation damage on the mechanical properties of structural materials used in nuclear reactors (stainless steel, aluminum, graphite). This problem has acquired special importance in connection with the development of fast reactors. The research is being conducted with the cyclotrons of the Oak Ridge National Laboratory and Argonne National Laboratory in the USA, at Karlsruhe in West Germany, and especially at Harwell in Great Britain, where beams of both protons and helium ions, as well as beams of heavier ions (carbon, iron, nickel), i.e., precisely those elements to be found in the composition of stainless steels, are being employed. Radiation damage in materials detected in reactors after several years of exposure to radiation is brought about in cyclotrons in a matter of hours, which accounts for the advantages in using cyclotrons to study the effect of radiation damage on the mechanical properties of materials.

VIII INTERNATIONAL CONFERENCE ON NUCLEAR PHOTOGRAPHIC EMULSIONS AND SOLID-STATE TRACK DETECTORS

N. P. Kocherov and L. I. Shur

The eighth international conference on nuclear photographic emulsions and solid-state track detectors was held in Bucharest on July 10-15, 1972. There were 163 delegates from 19 countries present. Approximately 100 papers were delivered, on the following general topics: 1) mechanism underlying formation of tracks in track detectors; 2) physicochemical processes occurring in track detectors; 3) properties of AgCl crystals as track detectors; 4) investigation of cosmic rays with the aid of track detectors; 5) applications of track detectors in biophysics; 6) study of nuclear fission; 7) applications of track detectors in high-energy physics; 8) applications of track detectors in dosimetry; 9) investigation of lunar rocks and geochronology; 10) automation of track detector measurements.

Reports under the first heading were devoted principally to improvements in methods for ascertaining the charge on heavy and superheavy nuclei of high energy. From that standpoint, we must single out those papers devoted to the applications of Katz' (USA) theory of track formation in track detectors for the practical photometry of track delineation. That approach enabled K. S. Bogomolov (USSR) to suggest a novel method for identification of superheavy nuclides. Matjesen (Sweden) developed a special photometer for determining the ionization characteristics of tracks. Reports by L. I. Shur (USSR) dealt with radioluminescence in nuclear photoemulsions, and the relationship between that phenomenon and the properties of photographic emulsions and of some sensitizing agents.

The next heading includes papers discussing those factors that affect the sensitivity of solid-state track detectors, techniques for fabricating solid-state track detectors, and optimum selection of detector material in the solution of specific physical problems. Chambaudier (France) discussed the effect of the degree of crystallization of solid-state track detector materials on the track characteristics.

Schopper (West Germany) reported on a technique for producing AgCl single crystals sized 25 × 25 × 5 mm, with sensitivities to ionizing radiations variable over a wide range by introducing different concentrations of cadmium ion (Cd^{2+}). The latent image in the AgCl crystals is stable only in the case where the crystal is illuminated by yellow light simultaneously with the radiation to be recorded. That makes it possible to switch the sensitivity of the crystal on and off by remote control at the necessary instants of time. Background fog in the AgCl crystal detectors appears at dosages upwards of 100 r.

Enge (West Germany) reported on results of applications of solid-state track detectors to obtain spectra of cosmic rays at great heights. The accuracy he attained in determining charge and mass is very high. For example, mass resolution $\Delta M = \pm 0.7$ mass units and charge resolution $\Delta Z = \pm 0.1$ charge units in the range of 5-8 charges were obtained for cosmic particles of energy ~100 MeV/nucleon. The resolution attainable in the range of 25-30 charges is ±0.3 units of charge. A report by Pinsky et al. (USA) proposing a new method for identifying primary nuclei in cosmic radiation such that $Z \geq 60$ should be singled out for attention. The authors make use of a Cherenkov detector surrounded by a thin film of high-sensitivity emulsion, in combination with conventional thick-layered nuclear photoemulsion. Relativistic superheavy nuclides traversing the scintillator produce a flare which is recorded by the high-sensitivity film. The darkening brought about by those flares is readily detectable by visual means, while the shape of the spot is determined by the velocity of the particle, i.e., it yields an additional parameter to aid in the identification of the particles. A team headed by Benton (USA) used solid-state track detectors and nuclear photoemulsions for in-flight monitoring in Apollo space flight dosimetry.

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Results of investigations of the nature of the light flashes observed by astronauts during space flights (McNalty, USA) are of interest. During their rest periods, astronauts observed one or two such flashes per minute. It was shown that these are due to the passage of fast charged particles through the retina of the eye. According to calculations by Jung and Fukui (USA), those flashes must be observed by passengers on supersonic aircraft.

Physical results were mainly tentative in nature.

Reports were presented on dosimetry in flights aboard supersonic passenger aircraft, at industrial plants, and also dosimetry of neutron flux levels. Zosefowicz (Poland) showed that track diameters in nitrocellulose solid-state track detectors depend on the neutron energy. She concluded that nitrocellulose detectors are consequently also useful in qualitative estimates of the energy spectrum of neutrons, such as personnel at plants are exposed to. Spurny (Czechoslovakia) developed a dosimeter for the benefit of personnel working with α-active preparations. Nitrocellulose film is the sensitive element in the dosimeter.

Paretzky (West Germany) reported on a facility for track delineation and track counting in solid-state track counters, by using the line-scanning beam of standard television equipment, and coherent optics. A report by Tommasino (Italy) described a spark counter for counting fission product tracks in solid-state track detectors. It should be pointed out that the Kodak and Deissel firms have already begun the manufacture of materials specially designed for use as solid-state track detectors.

III SESSION OF THE ALL-UNION SCHOOL ON THEORETICAL NUCLEAR PHYSICS

N. Ya. Smorodinskaya

The III All-Union Theoretical Nuclear Physics School was in session June 1 through June 15, 1972, not far from Kalinin. Sixty young scientists from different institutes took part in this session.

All of the lectures, interconnected under the common topic "Interaction of high-energy particles with nuclei," reflected the most interesting trends in contemporary nuclear physics.

The school session was opened with a series of lectures delivered by M. S. Marinov (Institute of Theoretical and Experimental Physics – ITEF) devoted to interactions of high-energy hadrons and atomic nuclei. A fast particle not only becomes deflected slightly upon traversing a nucleus, but does not have sufficient time to effect a change in the nucleons of the nucleus. Hence, the nuclear nucleons, each of which has an entirely negligible momentum imparted to it in the collision, remain as if observed at rest, and the collision they experience with the impinging fast particle falls under the heading of diffraction phenomena. As a consequence, the entire interaction can be described within the framework of a theory similar to the theory of optical diffraction. This approach was proposed about 15 years ago by the American physicist R. Glauber, and is now known as the "Glauber approximation." The diffraction theory was developed at more or less the same time, and independently of Glauber, in the contributions of A. I. Azkhiezer and A. G. Sitenko. The Glauber approximation offers an excellent description of the available experimental data, and provides reasonable and warranted physical predictions in cases where experimental confirmation is still lacking. In addition to the classical Glauber theory, the lectures also went into various corrections to the theory (treatment of nucleon recoil, multiple scattering, inelastic shielding), as well as interpretation of the theory in terms of relativistic diagrams.

The students showed great interest in the lectures by I. S. Shapiro (ITEF) discussing interaction of nonrelativistic antinucleons ($\bar{N}$) with nucleons (N) and nuclei. The basic idea in this problem, and the statement of the problem, are credited to I. S. Shapiro, and all of the subsequent research on the problem was conducted by him in joint work with ITEF colleagues. The restriction to low energies of the interacting particles makes it possible to apply the potential approach to the $N\bar{N}$-interaction, despite annihilation in the $N\bar{N}$ system. That also makes it possible to explain theoretically some properties of systems containing nucleons and antinucleons. The problem of the existence of quasinuclear $N\bar{N}$ states is especially interesting. Repulsion exists in the $N\bar{N}$ system at a distance of 0.5 fermi, and that is due to the ω-meson in the one-meson exchange potential. Accordingly, strong attraction must prevail in the $N\bar{N}$ channel, and it is found that that is the case indeed for most s-, p-, and d-states out to distances of 1-2 fermi, because of the effect of all of the meson exchanges. The effective potential well for $N\bar{N}$ interactions is consequently roughly five times deeper than for the $N\bar{N}$ system at the same radius. We must therefore anticipate that the spectrum of $N\bar{N}$ bound states must be much richer (about 20 levels) than the spectrum of two nucleons, which features only one bound state – the deuteron. The resonance levels in the $N\bar{N}$ system are obtained by calculating the Regge trajectories. The theory predicts the presence of near-threshold resonance with nonzero orbital moments. This result, as well as the existence of a large number of $l \neq 0$ bound states, is in agreement with experimental findings on $N\bar{N}$ annihilation arrived at in recent experiments.

Lectures by V. S. Popov (ITEF) based on the author's work on the spectrum of atomic levels for $Z > 137$ proved extremely useful. It is well known that the Dirac equation for an electron in the coulomb field of a point charge has an exact solution. The energy of the ground state $\varepsilon_1 = \sqrt{1-\xi^2}$, where $\xi = Z_\alpha = Z/137$.

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When $\xi = 1$ ($Z = 137$), the $\varepsilon_1(\xi)$ curve has a root singularity. Formal continuation onto the $\xi > 1$ region leads to imaginary values of ε_1, which is a physical impossibility. The lectures analyzed in detail how the difficulty can be gotten around, and discussed the phenomena occurring at the "critical point" (i.e., at $Z = Z_c$). Unfortunately, experiments must be staged with nuclei such that $Z > 170$ in order to verify the theory, while the heaviest of the presently known elements has a charge well below 170. Hence, the author put forth still another method of experimental verification revolving around the fact that two "bare" nuclei the sum of whose charges exceeds Z_c collide.

Quite a few new results were contained in the course of lectures on "Interaction of pions with nuclei at low and intermediate energies" given by V. M. Kolybasov (ITEF). His lectures were devoted principally to elastic scattering of pions on deuterons and on complex nuclei and pionic atoms. Capture of negative pions by nuclei was discussed briefly at the end. The amplitude of elastic πd-scattering was calculated within the framework of the Bruckner approach, but with the binding energy of the deuteron energy taken into account. The πd-scattering length was calculated to a high degree of accuracy by taking recoil of nucleons upon scattering into account (with the resulting substantial corrections), as well as some kinematic corrections whose need is felt in transitions between the reference systems and p-wave πN-interactions.

The next lectures constituted a review of the present state of theory and experiment on scattering of pions by complex nuclei and pionic atoms. In conclusion, V. M. Kolybasov gave a brief account of his work on capture of π-mesons at rest by light nuclei.

The lectures by L. A. Kondratyuk were devoted to electromagnetic (EM) effects in scattering of high-energy particles by nuclei. The discussion centered around two main problems: 1) interference of nuclear (or strong) interactions and EM-interactions in the scattering of high-energy hadrons on nucleons and nuclei into small angles; 2) two-photon exchange in scattering of high-energy electrons on light nuclei with large momentum transfer. In studies of the properties of the purely nuclear interaction, that mode of interaction must be separated out from EM-interactions. How this is done was demonstrated in the lectures, and examples driving home the need to do this were cited. The timeliness of the second problem relates to the recent appearance of very exact experimental data on form factors of some light nuclei at large transferred moments. It was shown that extrapolation of existing formulas to the range of large transferred momenta yields an incorrect estimate of the two-photon correction.

The course of lectures delivered by G. A. Leksin (ITEF) on experimental investigations of the mechanism underlying nuclear reactions at high energies stimulated great interest. The possibility of investigating the properties of nuclei with the aid of knock-out reactions was discussed in detailed fashion, and an experimental program for setting up a complete experiment, that had been worked out at ITEF, was cited. The program contains nine points, including, as an example, verification of various isotopic relationships, obtaining momentum distributions of a recoil nucleus, etc. The importance of utilizing spark chambers was demonstrated, and results were cited from the first experiments conducted at ITEF. These experiments are to date the only ones of their kind in the world, and will unquestionably make an enormous contribution to nuclear physics.

The lectures by M. I. Ryazanov (MIFI) discussed bremsstrahlung of ultrarelativistic particles in condensed media.

In addition to these lectures, seminars were conducted at which those auditing the lectures could present original papers.

The concluding session, at which each lecturer shared his thoughts on the perspectives of his work, and on the most interesting problems encountered in his work, was unusual and highly useful. The possibility of engaging in a constant interchange with the instructors, and discussing a variety of problems with them, was of great benefit to the students.

The school was competently organized. Outlines of the lectures were published for the opening of the school, and can be obtained from the Moscow Engineering and Physics Institute (MIFI).

SOVIET SPECIALISTS IN THE FIELD OF SUPERCONDUCTING ENGINEERING VISIT THE USA

E. Yu. Klimenko

A delegation of Soviet specialists on superconductivity (E. Yu. Klimenko, G. V. Trokhachev, N. A. Chernoplekov) visited the USA, from April 30 through May 14, 1972. The members of the delegation, who were in fact returning an earlier visit by American specialists, took part in the proceedings of the traditional national conference on applied superconductivity (held at Annapolis), and visited the USA national laboratories at Oak Ridge, Argonne, Ames, and Brookhaven, where they were brought up to date on research in physical and applied superconductivity.

Superconducting dc magnet systems are widely used in physics laboratories. Superconducting solenoids, including the largest ones, are not only more convenient than conventional electromagnets and solenoids, but are much more reliable in operation. The conference discussed several projects of large-scale magnet systems now operative in the USA; the largest of these is the superconducting solenoid for the bubble chamber at the National Accelerator Laboratory in Batavia (3 T in 30 m^3). A magnet array for the Bumpy Torus thermonuclear research machine has been tested at the NASA Lewis research center. A minimum-B dipole–quadrupole magnet system is now in successful operation at Oak Ridge National Laboratory. Tests on the superconducting sector of a hybrid solenoid designed to generate a 23 T field are being continued at the Massachusetts Institute of Technology. G. Bogner, a representative of Siemens AG, delivered a review of major superconducting systems now being planned or already in service in western Europe. In his account of the progress to date in bringing to fulfillment the widely familiar projects of the large European bubble chamber and the Omega spark chamber, he took note of some serious difficulties in the production of a vacuum in the simulating space of the bubble chamber cryostat, as well as the fact that the spark chamber will go into service at first with only one magnet sector, because of deformation suffered by one of the power plates of the Omega solenoid.

Most of the reports presented dealt with the building of superconducting magnets for accelerators. Materials suitable for the construction of dipoles and quadrupoles which will give an acceptable, though comparatively high, level of energy losses at alternating-current frequencies from 0.1 to 1 Hz, have been developed on the basis of niobium–tantalum alloys. Simulation of such systems is being pursued intensively in both the USA and in Europe. At Brookhaven National Laboratory, for example, analogs of superconducting dipoles for a storage ring with colliding proton beams are being developed (in the "Isabella" project). Information on tests of small (33 kJ) storage rings were provided in a report by representatives of the Los Alamos laboratory (within the framework of a θ-pinch type thermonuclear reactor project). A series of small storage rings of 500 J energy capable of operating at very high frequencies to 5 pulses/sec (charging time 4.5 msec, discharge time 0.21 msec) for 40 sec, before their coils overheat, has been tested in the course of developing 100 kJ storage rings.

Reports assessing the outlook for application of superconductors in the magnetic systems of thermonuclear reactors, synchrotrons, and electric power transmission lines were optimistic as a rule. At the present, USAEC is preparing to undertake the development of comprehensive nuclear electric power generating station projects incorporating superconducting electric power generators and superconducting electric power transmission lines.

To date, the basic material employed in superconducting magnet systems has been niobium–tantalum alloy. The technology of products made of this alloy has been raised to a high level of competency. Representatives of the IMI corporation (Great Britain), for instance, reported on a conductor containing over

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10,000 niobium – titanium filaments up to 5 μ in diameter embedded in a copper stabilizing matrix, and barriers of copper – nickel alloy hindering the appearance of shielding currents in the conductor are introduced into that same matrix. Significant advances have also been scored in the technology of intermetallic materials. Japanese and American concerns are mastering the production of multicore conductors using niobium – tin and vanadium – gallium intermetallic compounds. An opinion is current to the effect that production of such materials will be no more expensive than the production of conductors from niobium – titanium alloy.

Intensive searches are underway in USA laboratories in advancing the production technology of winding materials based on niobium – gallium compounds, which present the best critical parameters of all the binary compounds: 20.3°K, 34 T at 4.2°K. Certain hopes are also placed on ternary alloys with a Laves phase structure. The critical parameters for these materials are slightly inferior to those of the intermetallic compounds, but they are more than adequate for the technology.

Work is progressing on investigations of the effect of neutron irradiation and electron irradiation of superconducting materials in connection with the problem of building linear electron accelerators with superconducting resonators, and preparations for building new high-output thermonuclear facilities and accelerators with superconducting magnet systems.

Close attention is being given to applications of superconducting materials in the design of electrical machinery. It appears that electrical machinery manufacturing will be the first branch of industry to offer industrial-scale applications for superconductivity. Heavy hopes are being placed on superconducting electric machinery in shipbuilding. This level of interest is accounted for by the undisputed advantages offered by electrical propulsion of ships with gas turbine power plants, when superconducting electrical machinery is employed. Actually, the use of superconducting electrical machinery cuts capital investment almost in half, while improving efficiency and simultaneously reducing the weight and size of the propulsion engines. The possibility of decentralizing the shipboard propulsion equipment further simplifies the layout of the ship and increases its viability. The maneuverability of the ship is improved as an added benefit.

VISIT OF FRENCH SPECIALISTS ON RADIATION SAFETY TO THE SOVIET UNION

A. V. Fisun

In accordance with the program of cooperation between the USSR State Committee on the Use of Atomic Energy (SCUAE) and France's Commissariat on Atomic Energy (CAE), a delegation of French specialists on radiation safety visited the Soviet Union from May 23 to June 5, 1972.

The purpose of the visit was to become familiar with the problems of safeguarding the labor and health of workers, and also to visit scientific organizations which use sources of ionizing radiation in their work.

The delegation included a group of doctors who are specialists in radiation shielding, headed by Mr. Michali, the director of a CAE department, and also Mr. Vertu, an expert on the construction and use of handlers and other apparatus for working in radioactive surroundings.

The French guests visited several institutes of a medical character in various cities of the Soviet Union. In particular, in Moscow, they visited the Institute of Labor Hygiene and Occupational Diseases, the Institute of Biophysics, and the Central Institute for the Advancement of Doctors. The members of the delegation learned of the working conditions at various industrial enterprises of our country, individual cases of occupational diseases, and the methods of curing radiation disease and its prevention. The French specialists became acquainted with the organization of scientific research in the laboratories and at various experimental sites of the Institute of Radiation Hygiene in Leningrad, with the scientific research on radiation safety and with the working conditions at the Physics-Energy Institute in Obninsk, at the Scientific-Research Institute of Atomic Reactors in Dimitrovgrad, and at the I. V. Kurchatov Institute of Atomic Energy in Moscow.

It must be noted that the French specialists were interested not only in the problems of radiation shielding and work safety in the Soviet Union; they gladly shared their impressions and told about the organization of radiation safety in France. Dr. Michali spoke about the radiation-safety service in the CAE, emphasizing, in particular, many general aspects.

Mr. Vertu gave several talks concerning the application of various types of handlers used in hot cells. The Soviet engineers and technicians examined with great interest the samples of French equipment shown by Mr. Vertu using slides and movies.

At the All-Union Soviety "Izotop," the guests were told in detail about the use of various isotopes in agriculture, and they were familiarized with several working equipment models.

The members of the delegation remarked many times on the high technical level of Soviet equipment and technology, and they spoke with great respect of the Soviet press, emphasizing its highly qualified scientific level and the high competence of Soviet authors.

At the final meeting at the SCUAE, Mr. Michali said that "the contacts made will tie a new knot in the close, many-sided cooperation which is continuously growing between France and the USSR."

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INDEX

SOVIET ATOMIC ENERGY

Volumes 32-33, 1972

SOVIET ATOMIC ENERGY
Volumes 32-33, 1972

(A translation of Atomnaya Énergiya)

E

F

G

I

J

K

L

M

N

O

P

R

S

T

U

V

Y

Z

SOVIET ATOMIC ENERGY
Volumes 32-33, 1972

(A translation of Atomnaya Énergiya)

Volume 32, Number 1 January, 1972

Volume 32, Number 2 February, 1972

Volume 32, Number 3 March, 1972

Volume 32, Number 4 April, 1972

ABSTRACTS

LETTERS TO THE EDITOR

Volume 32, Number 5 May, 1972

Volume 32, Number 6 June, 1972

Volume 33, Number 1 July, 1972

Volume 33, Number 2 August, 1972

Volume 33, Number 3 September, 1972

Volume 33, Number 4 October, 1972

Volume 33, Number 5 November, 1972

Volume 33, Number 6 December, 1972

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