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SOVIET ATOMIC ENERGY

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August, 1978

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SEVENTY-FIFTH BIRTHDAY OF A. P. ALEKSANDROV

February 13, 1978, was the 75th birthday of that eminent Soviet physicist, Academician Anatolii Petrovich Aleksandrov, President of the Academy of Sciences of the USSR, and Director of the I. V. Kurchatov Institute of Atomic Energy.

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A. P. Aleksandrov was born in the town of Tarasche in the Ukraine in the family of a teacher. Upon finishing technical high school in Kiev he worked as an electrician. In 1923 he taught physics and chemistry in school and at the same time studied in the Kiev University Department of Physics and Mathematics, from which he graduated in 1929.

His first scientific paper "High-voltage polarization in ceresin," published in 1929, attracted the attention of Academician A. F. Ioffe who invited Aleksandrov to the Leningrad Physicotechnical Institute (LFTI). It was here that Aleksandrov became a scientist.

In his first years at the LFTI, Aleksandrov worked on dielectrics. He did research on breakdown in dielectrics and on the properties of polystyrene, a promising new material for electrical and radio engineering. In the mid-1930s the foundations were being laid for a new science, the physics of polymers. In view of this, it became of considerable practical, as well as scientific, interest to ascertain the electromechanical properties of polymers. It was precisely this area of research that attracted Aleksandrov most of all. Foreseeing an enormous future for high-molecular compounds, together with his co-workers (and in the case of some studies, in collaboration with P. N. Kobeko) he pursued physical research on polymers.

All of the investigations carried out by Aleksandrov during this period are characterized by an endeavor to extract the maximum practical results from fundamental research. This has been especially clear in his subsequent work.

During the Second World War Aleksandrov was in charge of naval work to provide protection for ships against magnetic mines by methods developed before the war in his laboratory. In addition to his immediate co-workers, he was actively assisted in this work by many co-workers from other LFTI laboratories, including I. V. Kurchatov. Protection for ships by this method made a great contribution to the successful operations of the Soviet navy.

It was in this period that the talent of Aleksandrov was forcibly revealed, not only as a scientifist but also as an organizer of scientific–engineering development and design and as a skillful leader in the practical implementation of such developments.

A profound knowledge of physics, the ability to see the engineering aspects of a problem and possible ways of solving them, and authority as an attentive, benevolent, but at the same time strict and insistent person are qualities which help Anatolii Petrovich solve major and responsible problems.

The year 1943 was noteworthy in the history of science and technology of our country. That was the year that Soviet physicists began work on a major scientific–engineering problem of the 20th century, that of harnessing nuclear energy. As is known, Igor' Vasil'evich Kurchatov was in charge of the scientific side of the work. Aleksandrov was involved in the work with his laboratory and soon came to head a large body of scientists and engineers.

The greatest development of the activities of Aleksandrov has been associated with the application of atomic energy in many areas of the national economy. In 1948, when he was appointed deputy to Kurchatov, Aleksandrov devoted his talent as a scientist and his great experience and energy to the development of reactor construction. His amazing versatility and erudition have been displayed in reactor development. An outstanding physicist, he has directed and organized the work of designers, technologists, materials scientists, and electrical engineers, and with his brilliant comprehension of all the details he has proposed solutions and evaluated the results. Aleksandrov sees not only the general outline and the principal features of any design, he also sees the fine details. Such an approach gives confidence that the solutions adopted are correct and this is the approach he teaches to others.

Choice of clear-cut and feasible problems, sensible organization of research and experimental work, his attraction for designers and industrial organizations in the early stages, and, finally, his enthusiasm enable Aleksandrov to avoid the "submerged rocks" associated with the promotion of scientific advances and to maintain close, fruitful ties with industry.

Under his scientific leadership, major scientific–engineering work has been done on the construction of the atomic industry in the USSR. The construction of the first atomic power plants, the development of a series of research reactors (VVR, SM, IGR, etc.) were the first successes on this road. Special mention should be made of the fact that the construction of research reactors in various scientific centers of the country has led to intense development of a number of areas of physics, biology, and chemistry.

After the death of Kurchatov in 1960 Aleksandrov succeeded him as head of the Institute of Atomic Energy. Under Aleksandrov the reliable and economic reactor plants VVÉR-440 and RBMK-1000 were developed for atomic power plants and are now built in the Soviet Union and abroad.

While paying much attention to the development of concrete plants for the first atomic power installations, Aleksandrov clearly saw the prospects of further development of nuclear power and took care that the results of atomic research be introduced on a broad scale in other branches of the national economy. In 1968 at the Seventh World Power Engineering Congress (Moscow) he said that. . . . "in the long term nuclear power stands out as a power industry of multipurpose complex plants engaged in electricity generation and other forms of production. . .. Clearly, the development and all-round extension of the forms of technology which can be converted to nuclear energy resources is one of the cardinal practical tasks confronting our generation along with the development of fast breeder reactors with a high breeding ratio. . .." These ideas are being actively developed at the I. V. Kurchatov Institute of Atomic Energy and in other organizations in the form of new energy and technological reactor plants.

Aleksandrov was the initiator of the application of atomic energy in shipping. Under his direct guidance and participation, high-quality marine power plants have been developed and built. Atomic icebreakers operating on the most difficult segments of the northern sea route have transformed the strategy and tactics for convoying ships. The atomic icebreaker Lenin, the world's first atomic-powered surface vessel, went into service in 1959, and has been used to appreciably extend the shipping season. The atomic icebreaker Arktika, fitted with an improved power plant, has reinforced the successes of the icebreaker Lenin; navigation in the western sector has become almost year-long. In 1977 the Arktika completed its unprecedented voyage to the North Pole in a record short time, thus showing that for our icebreakers there are no unattainable places in the icy seas.

The expanse of the scientific interests of Aleksandrov is exceptional and hence the development of many areas of basic and applied research, ranging from thermonuclear fusion to biology, within a single institute is not surprising.

Aleksandrov has taken an unflagging interest in the physics of the condensed state, an area of science in which he worked in his youth. This interest is heightened by the fact that the development of atomic science and engineering has confronted solid-state physics with new questions and has at the same time placed in the hands of researchers new equipment and methods for studying the properties of solids. Aleksandrov attentively follows and supports work on solid-state physics both at the I. V. Kurchatov Institute of Atomic Energy and at other research organizations of the country.

Along with this research, Aleksandrov supports and develops work on the practical application of superconductivity for the needs of atomic engineering and the national economy as one of the major directions of the present scientific–technological revolution. And here once again one sees the ability of Aleksandrov to combine scientific research with development for industry and by his knowledge and persuasion to unite sizeable staffs of scientific, design, and industrial organizations for solving major scientific–engineering problems.

While he heads an institute with a huge staff and diversity of scientific–technological subject matter, Aleksandrov looks after not only the construction of plant and the financing of work but, perhaps above all, is concerned about maintaining an atmosphere of goodwill and of enthusiasm for the work. He has succeeded in doing this by virtue of his enormous personal charm and extremely respectful attitude to each employee of the Institute and his work, but, obviously, mainly by arousing enthusiasm for any unknown phenomenon, new problem, or new instrument. To comprehend a new theory, to become aware of new experimental facts, and to examine a different, nontraditional approach to any known problem are all important and interesting to Aleksandrov.

Aleksandrov is an eminent specialist who has participated directly in the solution of a multitude of applied problems. He has widely advocated and collaborated in every way in the development of basic research. An inexhaustible curiosity in basic research and his encouragement of such research enabled him to use a new understanding of a physical effect or the ability to measure something to extract a more accurate method of solving an important engineering problem. Aleksandrov rarely observes the official hierarchy when solving scientific–engineering problems. In the evening in his office venerable academicians, as well as junior scitific workers and senior and ordinary engineers, have their heads bent over drawings spread out on the floor or over reports and they tell him about the results of an experiment that has just been completed or outline ideas for a new experiment.

The attention Aleksandrov pays to people is exceptional. No important matter, personal illness, or fatigue could prevent him from immediately coming to the assistance of someone who has fallen ill and regularly phoning in the evening to the home of a hospitalized colleague.

In 1943 Aleksandrov was elected Corresponding Member and in 1953, Academician to the Academy of Sciences of the USSR. For 15 years Aleksandrov was a member of the Presidium of the Academy of Sciences of the USSR and in 1975 he was elected President of the Academy. Aleksandrov has headed the Academy at a time when the importance of scientific research in the life of society, especially a developed socialist society, has been growing steadily, when there has been an extraordinary expansion of the areas of research and an increase in the scale of activities of the Academy of Sciences, and a growth of the complexity of the tasks of the Academy as the principal center of basic science and coordinator of scientific work in the country.

With a clear perception of the responsibilities and enormous tasks put before Soviet science and the Academy of Sciences of the USSR, Aleksandrov gives paramount attention to the choice of the most promising directions of scientific research, to the concentration of scientific forces and material resources upon the most important problems of present-day science and current goals of technical progress.

Bearing in mind the character of scientific work under modern conditions, Aleksandrov is constantly concerned with the development of the material and technical base of science, improving the level of equipment, and automating research.

In this work as President of the Academy of Sciences of the USSR, Aleksandrov has displayed scientific erudition, on the one hand, and a wealth of experience of work in collaboration with industry, on the other hand. Under the conditions today, when science has become a direct productive force, these qualitities of the head of the Academy are extremely important in solving problems of the practical realization of scientific achievements.

An important part of his activities as President concerns the development of science in the republics and in the branches and scientific centers of the Academy of Sciences of the USSR, refinement of planning of research and development, and improvement of the administration of all academic scientific and institutions.

For meritorious service to the country's science and technology Aleksandrov has been made a Hero of Socialist Labor on three occasions. He has been awarded the Order of Lenin eight times, the Order of the October Revolution, and other orders and metals. Aleksandrov is a Laureate of the Lenin Prize and of State Prizes of the USSR. At the Twenty-Third, Twenty-Fourth, and Twenty-Fifth Congresses of the Communist Party of the Soviet Union (CPSU), Aleksandrov was elected member of the Central Committee of the CPSU. Aleksandrov is a deputy to the Supreme Soviet of the USSR.

TWENTIETH ANNIVERSARY OF THE INTERNATIONAL ATOMIC ENERGY AGENCY (IAEA)

I. D. Morokhov*

The International Atomic Energy Agency (IAEA) achieved its 20th anniversary in 1977. The IAEA is an organization, which was founded by a group consisting of 60 countries, under the aegis of the United Nations.

The purpose of the IAEA according to Statute is the achievement of "the more rapid and more widespread utilization of atomic energy for the maintenance of peace, health, and prosperity throughout the whole world, ... The Agency guarantees that assistance given by it or through its requirement or under its supervision or control would not be used in such a way as to contribute to any military objective" [1].

Authorities and Structure of IAEA

The highest authority is the General Conference, at which each member-nation of this organization is represented by one delegate. The General Conference regularly, once per year, assembles in session. The Statute provides for the convening of special sessions of the General Conferences according to the requirements of the majority of member-nations or the Controlling Council.

Between sessions the Agency is guided by the Council, consisting of 34 managers. It assembles at times set by them (as a rule, 5 sessions/yr) and is guided in its work by the Statute of the IAEA and the resolutions of the General Conference.

As the highest authority, the General Conference discusses any problems specified by the Statute, and also selects the members of the Council of Managers, ratifies the acceptance of countries into membership of the IAEA, considers the annual report of the Council of Mangers, approves the submitted budget, reports of the Council for the United Nations, and also changes of the Statute, etc.

All information presented to the General Conference is considered and accepted by the Council of Managers. In addition, the Council appoints a General Director, who is then approved by the General Conference. He is the principal administative person and directs the Agency Secretariat.

The IAEA budget is comprised from the obligatory payments of the member-nations which, in 1977, amounted to 37 million dollars, and voluntary payments (amounting in 1977 to 6 million dollars), intended for rendering technical assistance to developing countries.

During 20 years, the IAEA has been transformed into an impressive international forum. Since 1957 the number of member-nations has grown from 60 to 110. In the work of the Executive – the Council of Managers – 34 countries now participate, as against 23 in 1957 and 25 countries in 1963. During this same period, the budget has increased, and also the strength of its personnel. At present, it amounts to about 1300 persons, of whom approximately one-third are specialists, and the remainder are technical and auxiliary personnel.

At the end of September and the beginning of October, 1977, the Twenty-First Jubilee Session of the General Conference of the IAEA took place in Vienna in the headquarters, which had conducted a total of 20 years of activity. The delegates listened with great satisfaction to the welcoming message of the General Secretary of the Central Committee of the Communist Party of the Soviet Union, Chairman of the Presidium of the Supreme Council of the SSSR, L. I. Brezhnev, in which, in particular, he said: "The problem standing before the International Agency of promoting the widespread utilization of Atomic Energy for maintaining peace, the health of the people and the prosperity of the nations, is close and understandable to us.

The Soviet Union actively cooperates and is ready to develop even further cooperation with other countries in the matter of the peaceful utilization of nuclear energy, included within the scope of the IAEA. Our country, widely utilizing nuclear energy for constructive purposes, is ready to share its rich experience and scientific–technical knowledge in this field, in the name of the future progress of mankind" [2].

*First Vice-Chairman of the State Committee for the Utilization of Atomic Energy in the Soviet Union.

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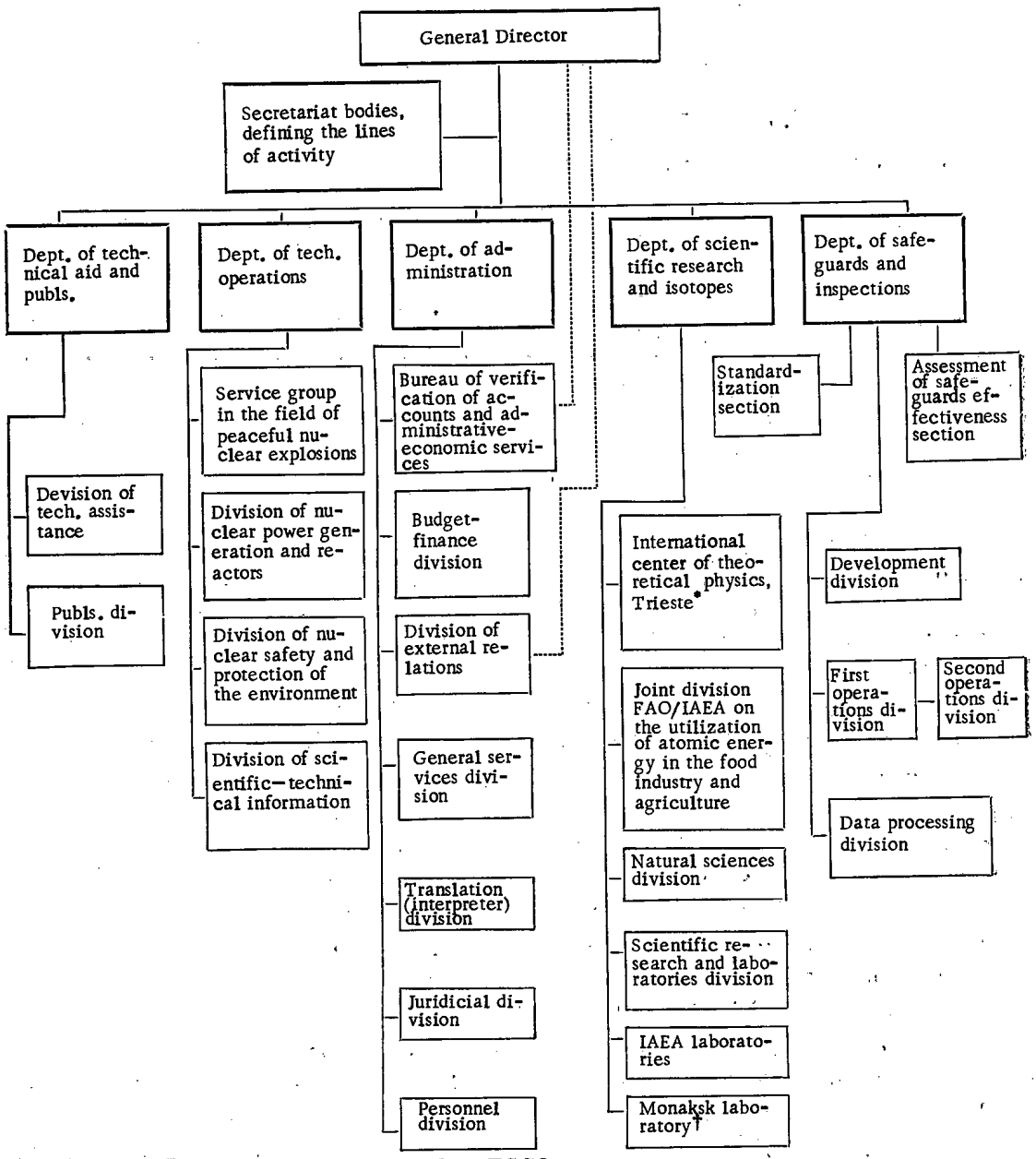

*Under joint supervision of IAEA and UNESCO;
†with increased participation of UNESCO and UNEP.

Fig. 1. Organizational Structure of the IAEA Secretariat.

Scientific – Technical Activity of IAEA

Over 20 years, the IAEA has carried out major work in the field of the peaceful utilization of atomic energy. For the assistance of member-nations, broad programs have been developed for research, for promoting the development of nuclear power generation, exchange of scientific–technical information in the field of nuclear science and technology, the application of nuclear explosions for peaceful purposes, ensuring the safety of the environment, new sources of power are being mastered, such as controlled thermonuclear fusion, etc. The Soviet Union has actively participated in the accomplishment of these programs.

The scientific–technical activity of the IAEA includes various programs on the introduction of nuclear energy in the various fields of economics of the countries of the world [3].

The aim of the IAEA program, conducted jointly with the Food and Agricultural Organizations of the United Nations (FAO), is the use of isotopes and radiations in the food industry and in agriculture. The program is oriented on the application of nuclear methods for increasing agricultural production, and also for raising the quality of food products and the protection of crops, domestic animals and foodstuffs from harmful

TABLE 1. Growth of Power, GW (electrical), and Power Generating Reactors

Region	1975		1980		1985	
	capacity	reactors	capacity	reactors	capacity	reactors
Europe	31,6	103	116,1	207	382,4	509
North America	52,5	77	124,8	149	299	278
Latin America	0,3	1	2,9	5	15,4	24
Africa					3,2	4
Asia and Australia	9,1	21	36,6	56	82,3	109
Total	93,5	202	280,4	417	782,3	924
Countries not possessing nuclear weapons	26,9	69	111,7	179	278,7	376

insects, sickness and injury. Important results have been obtained already for increasing the fertility of soil, due to the rational introduction of fertilizers and a water cycle, nuclear methods have been established and continue to be developed for determining the protein content in seed cultures, which is extremely important for increasing the quantity and improving the quality of protein by means of mutation induction, mutant selection and the development of methods of selection; genetic, nutrient and agronomic assessment of the mutants has been carried out.

Program in the Field of Natural Sciences

This has been accomplished jointly with the World Health Organization (WHO), for promoting the development of procedures and methods of using radioisotopes in medicine, biology and also for the preservation of the environment.

The Physics Program consists of the following divisions: nuclear physics, the use of research reactors, plasma physics and controlled thermonuclear fusion, industrial application and the chemistry, testing and analysis of materials, the production and industrial application of radioactive sources, nuclear data, atomic and molecular data.

One of the most important programs is that of nuclear power generation and reactors. This program in conjunction with the program on nuclear safety and protection of the environment occupies the greatest volume in the scientific–technical activites of the IAEA.

The nuclear power generation program covers all aspects of this problem – from the forecasting of economic questions to the study of improved methods of energy conversion. The program has such divisions as nuclear material resources, surveying assessment, supply and demand; fuel cycle technology, including fuel element technology, reprocessing of spent nuclear fuel and the handling of wastes; study of the regional centers of the nuclear fuel cycle, etc.

The program on the Nuclear Safety and Protection of the Environment has its aim in ensuring the safe utilization of nuclear power and the protection of people and the medium from the injurious effects of nuclear radiation from radioactive and nonradioactive effluents from nuclear facilities. Altogether, the work in the establishment of standards of safety, recommendations and guidance, asistance, and service given to the member-nations of the IAEA on standards of radiation safety are well known to specialists. They are considered mainly as the national standards of safety in many countries of the world, including the Soviet Union.

The modes of achievement of the IAEA programs are very varied: symposia and conferences, active working groups and groups of experts, meetings of specialists, etc. In this connection, the special importance for the future development of world nuclear power generation of the Salzburg Scientific–Technical Conference on Nuclear Power Generation and Its Fuel Cycle, held in May 1977, should be mentioned. The conference showed that the solution of the immediate and future points of the problem are being approached in different ways in the world, which is explained by the special features and requirements of the economics of individual countries. This discussion on the routes and tendencies of the development of nuclear power generation should be continued.

TABLE 2. Number of Plants for Reprocessing Fissile Materials

Plant	1975	1980	1985
For prod. of fuel from uranium	24	36	55
For prod. of mixed uranium–plutonium fuel	21	26	30
For enriched uranium	8	10	13
For reprocessing spent fuel	6	12	17

The Information and Technical Services to member-nations and the Secretariat occupy a special place in the activities of the IAEA.

The development of an automated system of collection and distribution of scientific–technical information (ISIS system) is a great achievement. The system, created on the initiative of the Soviet Union, started to operate in 1970 and has developed rapidly in recent years. The number of items processed annually has increased from 4000 in the first year of operation to 65,000 at the present time. Now the ISIS system is caused by 46 member-nations and 13 international organizations. The ISIS Atomindex is a unique international reference journal on nuclear science and technology.

The IAEA has available a library with a large stock of specialist literature. It has also connections with national libraries and there is a high rate of exchange of literature according to enquiries from member-nations and the Secretariat.

The IAEA carried out a widespread publishing activity and issues the journal "Thermonuclear Fusion," the series "Reviews on Atomic Energy," a monthly Bulletin, and also the proceedings of conferences, symposia, etc.

The Soviet Union participates actively and directly in the scientific–technical activities of the IAEA, sending its own specialists on scientific–technical and organizational means, directing highly qualified scientists, specialists and administrators to work in this organization. The Permanent Representation of the Soviet Union at international organizations in Vienna renders great assistance in liaison and cooperation with the IAEA. The participation of the Soviet Union in the work of the IAEA wins high praise from the Secretariat and member-nations. The role, importance and authority of the Soviet Union in the IAEA, undoubtedly has grown, especially over recent years.

Technical Assistance to Developing Countries

One of the first places in the activities of the IAEA is occupied by the rendering of technical assistance to developing countries, which includes the transmission of technical knowledge and skills in the fields of utilization of nuclear energy for peaceful purposes, support for efforts toward a more efficient achievement of work in the field of nuclear power generation and ensuring that the transmitted technical skills and knowledge could be applied after rendering this assistance.

The modes of the rendering technical assistance are diverse: services of experts, provision of plant, granting of scholarships, and training of national personnel.

Since 1958, 82 countries have utilized the services of 3000 experts and detached specialists. During this period, 20 million dollars worth of plants and materials have been supplied, ≈3000 scientists, engineers, and administrators have carried out training in more than 180 regional and interregional training establishments.

In attaching great importance to the rendering of technical assistance to developing member-nations of the IAEA, the Soviet Union has supplied to these countries at the requests of the Secretariat, plant and materials to the account of its voluntary payments, and has also trained national personnel.

From 1969 to 1976, of the total sum of voluntary payments of the Soviet Union of 2.8 million rubles in the national currency of the IAEA, more than 2 million rubles already has been realized. On the account of this payment, 15 scientific-familiarization trips of specialists from developing countries have taken place.

TABLE 3. Installed Capacity of Nuclear Power Stations, Number of Facilities, and Quantity of Nuclear Materials under Safeguards of the IAEA (on Jan. 1, 1977)

Items	1973	1974	1975	1976
Installed nuclear power station capacity, GW (el.)	5	8	10	20
No. of nuclear power stations	27	36	43	60
Other reactors	107	110	103	120
Facilities for the manufacture of fuel elements and for the chemical reprocessing of fuel	20	26	29	35
Other facilities or zones of material balance	100	116	140	150
Total facilities	254	288	315	365
Plutonium, kg	4730	6300	9035	12 000
Enriched uranium, tons				
element	1865	2305	3096	5 000
isotope	43	53	66,7	150
Raw material, tons	3370	3910	4440	6000

From 1977 annual trimonthly courses on the application of nuclear methods to agriculture will be held in the K. A. Timiryazev All-Union Agricultural Academy in Moscow. In 1978-1979, it is planned to organize a course at the Novovoronezh Nuclear Power station on the operation of water-cooled/water-moderated reactors. The possibility is being considered of founding annual courses in Moscow on the application of nuclear methods in medicine. For the first time, a scientific-technical tour has been organized and successfully conducted on safeguards, with a visit to nuclear facilities of the Soviet Union.

On the recommendation of the government, the Soviet delegation declared at the Twenty-First Jubilee Session of the General Conference of the IAEA an increase in the voluntary payment of the Soviet Union to the technical assistance fund, in the first place to developing country-participants of the Treaty for the Nonproliferation of Nuclear Weapons. This payment may be used for the purchase of Soviet plant, instruments and materials, and also for conducting IAEA educational-familiarization arrangements in the Soviet Union.

The effective combination of technical assistance with the necessary control measures will serve for the further consolidation of the policy of nonproliferation of nuclear weapons and, consequently, a more complete realization of the problems arising from the IAEA Statute and the conditions of the Nuclear Weapons Nonproliferation Treaty.

The Problem of Nonproliferation of Nuclear Weapons

It should be pointed out, however, that even if the activities of the IAEA in cooperation with the widespread introduction of atomic energy into the peace economics of member-nations of this organization do not prove fruitful, in the modern setting there is no more urgent problem than the cessation of the arms race and disarmament. The IAEA acknowledges cooperation in the achievement of these aims. At the moment, it is impossible to forget that the energy of the atomic nucleus can be used also as the most destructive weapon which mankind has ever known. Therefore, the efforts undertaken by the IAEA for the prevention of nuclear weapon proliferation acquire special importance.

At present, it can be seen with all authenticity that the development of nuclear power generation is proceeding with increasing rates, and an even greater number of countries are included in its orbit. Undoubtedly, its intensive development will allow the greater part of all forms of energy requirement to be ensured and will allow economy in the use of the large quantity of organic raw material for those purposes where its total replacement is more complicated, mainly for the chemical industry.

At the same time, in considering the positive aspects of development, it must not be forgotten that the significant increase of the quantity of fissile materials and the number of countries possessing them increases the potential hazard of using the accumulated nuclear materials for the creation of nuclear weapons.

Estimates show that the average doubling time of the world's nuclear power generation capacity in the next 2-3 decades may amount to 5 years, and the installed capacity of nuclear power stations expected by 2000 A.D. may amount to $4 \cdot 10^6$ MW (electrical). Even if these development times prove to be low but commensurate with the increase of all power generation as a whole, the capacity of nuclear power stations by 2000

A. D. will amount to $2 \cdot 10^6$ MW (electrical). However, even this minimum estimate shows the considerable scale of its growth [4].

The distribution of capacities and the numbers of nuclear power stations throughout the regions of the world in the forthcoming decade are shown in Table 1.

Thus, in 1985 the capacity of nuclear power stations in countries which do not possess nuclear weapons will have increased by a factor of 10, and the number of countries possessing nuclear power will have doubled.

The considerable increase of nuclear power stations leads to an increase of requirements for uranium, which will have increased from 25,000 tons in 1975 to 35,000 and 160,000 tons by 1980 and 1985, respectively Significantly, the requirements for enrichment will increase from 13,000 tons of sep. work units/yr in 1985 to 100,000 tons of sep. work units/yr in 1985; fuel manufacture will increase from 6000 tons in 1975 to 15,000 and 30,000 tons in 1980 and 1985. By 1980, more than 150 tons of plutonium converted to fissile fuel will have accumulated, and by 1985 this figure will amount to 500 tons.

It should be mentioned that the increase in the number of nuclear facilities is not identical in all stages of utilization of nuclear material and its reprocessing. Thus, if the number of nuclear power stations increases by more than 200 units by 1980, and by 600 units by 1985, in comparison with 1975, then over this same period only a few new uranium enrichment plants and plants for reprocessing spent fuel will appear (Table 2).

L. I. Brezhnev, in the salutory address at the Twenty-First Jubilee Session of the IAEA General Conference, wrote: "In supporting the development of the peaceful utilization of atomic energy, the Soviet Union is firmly resolved, together with other governments, to consolidate in every way the international policy of nonproliferation of nuclear weapons. It is essential to do everything possible in order that the international exchange of nuclear technology, involving in many countries a scientific-technical and industrial nuclear potential, does not become a channel for the proliferation of nuclear weapons.

"We cannot shut our eyes to the fact that in the world there will always be powers who would wish to receive in their hands nuclear weapons, in order to threaten nations with this weapon. Therefore, the problem of setting a reliable safeguard on the paths of nuclear weapon proliferation, and for preventing the hazard of a nuclear war, remains now just as acute as ever.

"In solving this problem of immediate importance, the International Atomic Energy Agency has played on important role, and we express the hope that the IAEA will apply all efforts to ensure that the atom will serve only the interests of peace."

Future consolidation of an international policy of nonproliferation, today as never before, is important and is connected directly with the maintenance of peace, safety and reduction of the threat of nuclear war.

The accelerated development of nuclear power generation, which is becoming one of the principal sources for satisfying the power generation requirements of countries, is related inevitably with the accumulation of large quantities of nuclear materials and, as a consequence, with an increase of the danger of nuclear weapons proliferation. The Soviet Union proceeds from the fact that the development of nuclear power generation in the world must be combined to the fullest extent with consolidation of the nonproliferation policy.

All governments who value peace highly, must actively strive for the Treaty on the Nonproliferation of Nuclear Weapons to become a genuinely universal instrument of international nonproliferation politics, encompassing all governments without exception. Unfortunately, not all countries who possess nuclear weapons, nor all countries with significant nuclear potential, have subscribed to the Treaty, and some of them, as for example the UAR, in fact are opposed to this Treaty and are actively preparing to carry out nuclear tests.

The campaign for a new stage of the nuclear arms race being conducted by certain western circles under the catchword of expansion of production of the so-called neutron bomb and other dangerous types of weapons, does not assist consolidation of the Treaty for the Nonproliferation of Nuclear Weapons.

System of Safeguards

During the 20 years of existence of the IAEA, considerable experience of monitoring activities has been built up. A system of legal standards has been worked out, monitoring equipment has been set up, procedures and methods of monitoring have been developed and introduced at many types of nuclear facilities. At the present time, the IAEA monitors the activity of many nonnuclear countries of the world. This is done completely regularly, because one of the functions of the IAEA, fixed by its Statue, is the implementation of

safeguards which have their aim in ensuring "that special fissile or any other materials, services, plant, technical facilities and data, presented by the Agency either according to its requirement or under its supervision or control, should not be used in such manner as to further study any military purpose, and to extend, according to the requirement of the parties, the use of these safeguards to any two-party or multiparty agreement or, according to the requirement of one or other government, to any forms of activity of this government in the field of nuclear energy."

The system of safeguards was formulated for the first time in 1961 in the form of INFCIRC-66 and contained monitoring procedures for small experimental reactors. Since then, it has been extended and modernized repeatedly, which has been reflected in other documents. The key stage was the decision of the participating countries of the Treaty on the Nonproliferation of Nuclear Weapons to guarantee to the IAEA the implementation of monitoring functions according to Article III–I of the Treaty, in accordance with the proposals of INFCIRC-153. Thus, at present, the IAEA monitors nuclear activity resulting from the agreements concluded on the basis of INFCIRC-153 and INFCIRC-66, Rev. 2. Based on INFCIRC-153, 44 agreements are concluded, of which 21 are with countries which do not possess a significant nuclear activity. Based on INFCIRC-66, Rev. 2, agreements on projects (11) and transfer of safeguards (21) are operative, and also agreements in connection with single-party organization of nuclear activity under safeguard (8).

Under the control of the IAEA, there are about 12 tons of plutonium, 5000 tons of enriched uranium and about 6000 tons of raw material (Table 3) [5].

Under the conditions when nuclear power generation in the world is developing and international trade exchange of nuclear materials and plant is expanding, the improvement of IAEA activities in the field of safeguards is being promoted in the first plan of a number of measures directed at consolidation of the policy of nonproliferation of nuclear weapons. The Soviet Union considers the efficient monitoring of the IAEA as one of the principal premises for widespread international cooperation in the field of the peaceful utilization of atomic energy.

The IAEA is entering at present a new stage of its monitoring activity, the characteristic feature of which is a sharp increase of the volume and complexity of monitoring. In connection with this, the problem of the maximum use of all possibilities set out in the system of safeguards arises in all its acuteness. At the basis of the system, as is well known, lies the principle of independent verification. The IAEA must use this entirely in its own right, independently of the extent of the development of registration and monitoring in individual governments of groups of governments. Moreover, it will be necessary in all countries using IAEA monitoring, that efficient systems of accounting and control of nuclear materials should be created and operated. The subsequent achievement by the IAEA and by countries of the regulations laid down in the IAEA system of safeguards is a pledge of effective international control in the field of nonproliferation of nuclear weapons.

As before, the question of the necessity for radical improvement of operation of the IAEA monitoring machine is acute. Recently, the Department of Safeguards and Inspection was reorganized. A second inspection division was set up and a section for assessing the effectiveness of the safeguards, intended to play the leading role in stepping up controls. It is important to strengthen the Department of Qualified Specialists and to raise to a new level the cooperation between its divisions and sections.

The necessity for a comprehensive analysis of the activities of the IAEA control machine has become imminent, and the implementation of long-term and short-term plans for its improvement. This would give the capability of more reasonably approaching a definition of the necessary manpower and financial resources, and would stimulate on a planned basis the development of procedures and methods of control, instruments and plant used in monitoring activities, and their operative introduction into practice, especially at the present time, when the IAEA is approaching achievement of safeguards in a number of large-scale facilities, which are "sensitive" from the point of view of nonproliferation of nuclear weapons. The question of the development of a model of effective safeguards also has been put on the agenda.

Due to the increase of volume of monitoring activities of the IAEA, the question of the volume of data received by the IAEA is important. Until recently, processing and analysis of this information received insufficient attention. The creation in the Department of a special division for the processing of information on safeguards, the development and operative introduction of an automated system of data processing, in principle, is of great value for the entire system of control.

The formulation of the problem of implementing within the framework of the IAEA a project for an international convention concerning the physical protection of nuclear materials, plant, and transportation is urgent.

In attaching great importance to the activities of the IAEA in the field of safeguards, the Soviet delegation made a statement at the Twenty-First Jubilee Session of the IAEA General Conference about the purposeful contribution introduced by the Soviet Union in the implementation of the technical aspects of safeguards in 1978 to the amount of 300,000 rubles in national currency. This contribution may be used, in particular, for conducting training for inspectors at the Novovoronezh nuclear power station, development of technical methods of monitoring at this nuclear power station, and for the organization in the Soviet Union of IAEA conferences and courses on safeguards. The Soviet Union, for their part, is prepared to render further assistance to the IAEA in work on the strengthening of the system of safeguards, which is important for peace [6].

It would be desirable to mention that governments who supply nuclear materials, plant, and technology should assume a special responsibility. Rigorous safeguards will be necessary, so that international cooperation in the field of peaceful utilization does not become a channel for the proliferation of nuclear weapons. This is not a commercial problem, but one of politics and safety. It is well known that a group of 15 supplier-countries of nuclear materials, plant, and technology have implemented guiding principles for nuclear export. At the conference of suppliers held in Sept. 1977 in London, understanding was reached to inform IAEA through its General Director concerning the policy followed by them for nuclear export control.

The guiding principles are intended as an obligatory condition for the granting to nonnuclear countries of export services, the official assurance of the government of the recipient-country that the imported nuclear materials, plant, and technology enumerated in the reference list supplied by the exporter-countries will not be used for the creation or production of nuclear explosive devices. The guiding principles require from the recipient assurances for the physical protection of the articles of the references list received, if it accepts the safeguards (monitoring) of the IAEA not only on the transferred items, but also on the materials and plant produced by means of the items received. The guiding principles provide for special IAEA control in the case of export of facilities, plant, and technology for the enrichment of uranium, and reexport regulations, in which cases reexport may be effected only with the agreement of the original exporter and in the same conditions of initial supply, and other regulations including sanctions in the event of violation by the recipient of the conditions of the guiding principles for nuclear export. In addition to this, the exporters have been obliged to render active assistance for improving and increasing the effectiveness of the monitoring (control) activities of the IAEA.

The task of intensifying control measures during report will be continued. With regard to the Soviet Union, it will subsequently strive for the acceptance of a principle of total control as a condition of supply of any materials, plant, and technology included in the agreed reference list.

Being a specialized international organization, the IAEA reacts tactfully to the political changes in the world. The scientific-technical direction of this organization is subjected to the influence of these political problems which stand before mankind. An example of this is the activities of the IAEA in consolidating the conditions for the nonproliferation of nuclear weapons, etc. It is important that concern about the assurance of peace on earth and the safety of mankind from a nuclear catastrophe are the initiating elements in the activities of the IAEA, and here the words of the salutory address of L. I. Brezhnev at the Twenty-First Jubilee Session of the IAEA General Conference are pertinent: "The Soviet Union, for its part, will even further render total cooperation to the IAEA in the achievement of the noble aims which stand before this authoritative international organization.

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ARTICLES

PROSPECTS FOR THE DEVELOPMENT OF CHEMICAL TECHNOLOGY OF FACTORIES OF THE NUCLEAR-POWER GENERATION FUEL CYCLE

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The necessity for the development of the nuclear industry in the Soviet Union [1] is conditioned by the increasing demand for power, the continually expanding use of radioactive isotopes for the intensification of technological processes in chemistry, control and automation of the various branches of industry, the use of the achievements of nuclear science and technology in agriculture, medicine, geology, and for controllable contamination of the atmosphere caused by concentrated sources of energy. All this is accompanied by an increase of the role of chemical and radiochemical processes in the treatment of natural uranium raw material and the regeneration of spent fuel, in the production of new types of fissile material and in other factories of the nuclear-power generation fuel cycle. Let us consider the achievements and future prospects for the development of these processes.

System Analysis and Mathematical Modeling of Production Development

In a nuclear-power generating complex, the decisive circuit is that of the fuel cycle, representing an assembly of interrelated different plants. The fuel cycle consists of four stages of the total technological process, each of which includes one or several plants.

The first stage is the manufacture of the nuclear fuel: extraction of uranium or thorium, concentration, production of uranium concentrate and uranium hexafluoride, isotope separation, fuel component manufacture, and fuel elements.

The second stage is the combustion of the nuclear fuel in reactors.

The third stage is the cooling of the spent fuel and its transportation to the reprocessing site.

The fourth stage is the reprocessing of the spent nuclear fuel (in closed cycles); extraction of valuable components, manufacture of uranium–plutonium fuel, reprocessing and storage of waste. The following plants occur in the structural layout of fuel cycles: structural materials for nuclear power station reactor cores, specialized plant, instruments for monitoring radioactive materials, and also spent fuel-element storage and production tailings during isotope separation.

In order to determine the prospects in detail of the second alternatives of production development of a nuclear-power generating complex, taking account of new technology and types of reactors, system analysis of the fuel cycle structure is of great importance. System analysis permits one:

to establish the mutual effect of the plants entering into the fuel cycle;

to show the technicoeconomic significance of each plant from the point of view of the long-term development of nuclear power generation;

to reveal the varied development factors of each plant and to establish their interrelation;

to determine the system of limitations when considering different alternatives and to select optimization criteria for production development.

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The analysis shows that the multivariability of production development is determined by the type of reactor, the form of the nuclear fuel (uranium, thorium, uranium–thorium, uranium–plutonium, etc.), the regeneration technology of the spent fuel elements, and also the treatment of the natural raw material, the structure of the capacities of the separation plant, which is characterized by the feasibility of using different physicochemical methods for the separation of uranium isotopes, and other factors.

These factors of the alternative development of individual production plants govern the fuel cycles. The choice of alternatives for their implementation is determined by the econimic competitiveness of each alternative, the balanceability of operation of the fuel cycle plants, the supply of raw materials and materials in short supply, and the readiness of industry for ensuring the production of the fuel cycle with the necessary facilities.

In order to investigate and optimize the alternatives for the development of the nuclear-power generating fuel cycle and to choose from them the best in different countries, including also the Soviet Union, mathematical models have been developed [2-6]. According to their nature, they are subdivided into optimization and simulation models. The first of these permits an all-round analysis of the effect of various factors in their interrelation in the development of the nuclear-power generation system (taking into account both the inherent special development features of nuclear power generation, and also interrelation with the fuel-power generating economy of the country). By means of simulation models, the effect of individual factors on the development of nuclear power generation can be investigated.

At present, it is advantageous to construct an interrelated set of mathematical models. Such a combination of models makes it possible:

to consider the large number of alternatives for the production development of the fuel cycle;

to compare acceptable alternatives and, taking into account their limitations, to recommend the best of them according to the chosen critiera;

to allow for the large number of influencing factors;

to carry out complex and laborious calculations for forecasting and estimating long-term development alternatives;

to operatively correct previously made calculations in proportion to the accuracy of the starting data (technological parameters of a different kind, technicoeconomic indices and restrictions, etc.), and to change the production structure;

to plan effective paths of scientific–technical progress and improvement of the fuel nuclear-power generating cycle.

Taking all this into account, it should be mentioned that for processing in detail of complicated valid decisions for determining the prospects of development of the nuclear-power generation fuel cycle like a large production-economic system, it will be necessary to use methods of program-objective planning and system-mathematical analysis. This approach allows a dynamic model of planning and control to be established in a development process and the introduction of new industrial technology and nuclear reactors, with an assessment of the long-term direct and indirect consequences of the solutions used.

It will be interesting to consider the prospects of development of certain production plants of the nuclear-power generation fuel cycle, taking account of the achievements of nuclear technology both in the Soviet Union and also abroad.

Processing of Uranium Raw Material

Forecasts of the development of nuclear power generation indicate a significant increase of capacity during the next decades; because of this, the requirement on uranium increases with every year. Therefore, the importance of bringing into line the scale of possible extraction and processing with the known natural resources increases.

In the Soviet Union, the most diverse problems have been solved successfully in the processing of uranium raw material and the prevention during processing of contamination of the environment. Theoretical calculations, laboratory investigations, semi-industrial tests and industrial practice substantiate the effective application of radiometric concentration to the majority of low-grade uranium ores [7]. Further reduction of the cost of sorting and improvement of the technological indices are possible by the use of new, higher-output separators and the utilization of methods based on the use of the artificial radioactivity of the ores.

Low-grade, resistant, and complex ores are processed by using autoclave processes. The use of elevated temperatures and pressures, together with the cheapest of oxidants (atmospheric oxygen), permits a profitable processing of the ore raw material to be organized, and permits a high uranium extraction to be obtained with a reduction of the consumption of reagents (e.g., acids) and a reduction of power costs (steam). Commercially manufactured autoclave equipment provides for carrying out oxidized leaching processes of uranium over a wide range of temperatures, pressures, and reagent concentrations.

In recent years, uranium from low-grade ores is being extracted on even greater scales by leaching out the useful component at the site of the ore deposit. The uranium in this case is extracted from the depths to the surface in the form of a solution. Soviet scientists reported for the first time on these investigations at the International Conference on the Processing of Low-Grade Uranium Ores, held in 1966 in Vienna. Underground leaching at present has been fashioned into a self-sustaining chemicotechnological process [8]. A technology has been developed which is intended for the recovery of uranium from hard (massif) rock and from sedimentary ores, deposited in stratified conditions (horizontal strata). Underground leaching has permitted the capital costs on production organization to be reduced, the cost of uranium production to be reduced, and the working conditions to be improved considerably. Moreover, the possibility has been given of processing local small-scale ore deposits, to include in the processing compensated ores treated by the usual method of mechanical extraction, and also deposits lying in complex mining-geological conditions. Experience in industrial operation shows that different low-grade uranium ores can be processed by this method.

New possibilities in the processing of low-grade and complex uranium ores are opened up by sorption processes [9].

The irrefutable advantages of these processes are due both to the aggregation state of granular ion-exchangers, which permit the separation process to be conducted easily, and also to the high exchange capacity of the majority of resin types. This, even at the beginning of the 1950s, permitted sorption processes from pulp to be carried out, which are predominant in the uranium industry of the Soviet Union.

The development of a filtrationless sorption method has led to the development of leaching and extraction desorption processes, which intensifies the uranium ore recovery processes and considerably improves the technicoeconomic indices, due to the elimination of laborious operations of repeated filtration and repulping of the filter cakes. The method has made it possible to include lower-grade uranium raw material in the processing, and simultaneously to separate valuable components: molybdenum, vanadium, rare-earth elements, scandium, and phosphorus [10].

Industrial experience has been built up of sorption from dense pulps up to solid:liquid = 1:1, which has led to an increase of productivity of the operative plants by a factor of 1.5-3, an increase of uranium extraction by 5-10%, an increase of work productivity of the basic workers, and a reduction of the consumption of chemicals, auxiliary materials, electric power and steam by a factor of several. In essence, an efficient technology, continuous in all its links, has been created with total and complex automation of the process, high-productivity equipment of large unit capacity with mechanical and pneumatic mixing for high-density pulp, and also equipment for the continuous regeneration of saturated sorbent.

The ionites manufactured in the Soviet Union with weakly acid and strongly basic exchange groups can be used for almost any (including even complex salt) systems. The production of granular ionites with high kinetics properties, sorption capability and a high mechanical strength, has expanded the use of ion-exchange processes. The production of new types of ionites—ampholites has permitted simultaneously the extraction of attendant elements.

In the Soviet Union ion-exchange resins have been produced for sorption from pulp and solutions and, especially the production of strongly basic anionites of helium structure AM, AMP, VP-1A, VP-3A, macroporous AMp, AMPp and VP-1Ap, bi- and polyfunctional anionites of the type AM-2B, medium-basic AM-3 and VP-1p, and also extremely promising carboxyl ampholites AMK, AMK-2, VPK and various phosphorus- and phosphorus—nitrogen-containing ionites (ampholites AFI-5, AFI-7, VPF-1, and VPF-2). These ionites have a high mechanical strength, which ensures minimum losses of resin under the most rigorous operating conditions [11].

When processing low-grade ores by underground and mound leaching, solutions are obtained with a low uranium content. An even lower uranium content is characteristic for natural and mine water. In order to extract uranium from the large volumes of solutions with a low concentration, an equipment has been designed which makes it possible to carry out the process at a high linear flow-rate of the solutions.

It is well known that extraction with organic solvents, from the point of view of physical chemistry, is similar to sorption with solid ionites. The most efficient and optimum regions of application of each of these processes has been determined from a comparative assessment. Sorption extraction from pulp usually is combined with extraction processing of the desorption solutions. Depending on the salt content of the solutions and the problem of supply, a suitable extractant can be used. For reextraction, it is most advantageous to obtain a uranium salt directly from the organic phase.

With the development of high-capacity equipment, it has become possible to carry out extraction directly from ore solutions. Two types of extractors, as usual, occupy the predominant position in the equipment layout of the processes – mixer-settlers and columns. The main trend of the future improvement of extractors consists in the search for optimum mixing conditions.

A considerable reduction of capital and operating costs can be achieved with extraction directly from dense pulps and nonaqueous leaching. However, these operations have still not emerged from the semi-industrial stage and test-rig experiments.

New possibilities in hydrometallurgy are being opened up by the creation of processes which combine the advantages of sorption and extraction methods. Such methods are the impregnation with organic solvents of porous granules, and desorption of uranium from solid ionites with acids or with neutral extractants.

Considerable research has been undertaken by Soviet scientists on the extraction of uranium from natural water with granular sorbents [12]. In the process of investigation of selective sorbents, more than 400 different ionite samples have been tested. The most efficient were found to be certain strongly basic anionites, with a capacity amounting to 2.5-5.3 mg/g. The regenerates, obtained by desorption of the anionites saturated with uranium, are reprocessed by extraction or sorption concentration.

The scientific–technical level achieved at the present time will permit the most diverse problems in the field of uranium raw material processing to be solved and will prevent contamination of the environment.

Wide possibilities in the inclusion of low-grade uranium-containing raw material in processing are being opened up by the extraction of uranium as a by-product or as a joint product in combination with other useful components.

The Soviet Union has available great production experience in the extraction of uranium and other valuable components from phosphate raw material, and also in the complex utilization of uranium–molybdenum ores.

Isotope Separation

At the present time, requirements for enriched uranium are met by gaseous diffusion plants, which are linked with a large demand on electric power [13]. With the development of nuclear power generation, interest is increasing with the realization of the possibilities of centrifuging, which is characterized by a significantly lower power requirement. The Federal Republic of Germany, Holland, and Great Britain have concluded an agreement on joint cooperation of separation plants with centrifuges. Investigations on the technology of centrifuging are being carried out in Japan.

The first work on the chemical and ion-exchange separation of uranium isotopes is related to the end of the 1940s. In 1953 a report appeared on the enrichment of uranium in the light isotope up to 2.8% by the ion mobility method. The separation of isotopes by the precipitation of oxalates with countercurrent migration is described. Ion-exchange chromatography, carried out by the use of anionites and cationites, occupies a special place (including solutions of phosphorus- and nitrogen-containing complex-forming agents), and also water and organic solutions, including hydrochloric, sulfuric, nitric, or chloric acid solutions of uranium (VI), uranium (IV) or their mixture. The isotope separation factor varied from 1.00006 to 1.0004. In the majority of cases, the results of the work on the separation of uranium isotopes are only satisfactory, the latter associated with an insignificant coefficient not exceeding 1.001. It is true that Japanese scientists have achieved an increase of the ^{235}U content over a single cycle by a factor of 1.017, by filtration of a solution through a sulfocationite JRA-120B in a column of height 1 m and with a cross section of 1 cm^2. Fractions enriched in ^{235}U emerged primarily from the column [14].

On the whole, sorption processes for uranium isotope separation have been widely investigated in the U.S.A., Yugoslavia, France, and the UAR.

The kinetics of the electron exchange of ^{235}U and ^{238}U, in the four- and six-valent states is being studied in aqueous or organic solutions (TOA and TBP*) in the presence of cationites and anionites. The purpose of these investigations is the achievement of a maximum rate of exchange for the subsequent use of suitable systems in ion-exchange fractionation of isotopes.

Extraction processes for the separation of uranium isotopes have been less studied. The achievement of a single separation factor of 1.002-1.00006 has been reported. In recent years, scientists in France and other countries have published the results of investigations into the separation of isotopes by extraction, which allow the advisability of further exploration in this field to be judged.

Ion-exchange and extraction methods of ^{235}U and ^{238}U separation could play an important role in the creation of a single water cycle for the regeneration of reactor fuel elements of low-enrichment uranium fuel. The solution of the problem of increasing the rate of electron exchange between isotopes in the ionite phase and the development of a high-capacity continuous chromographic process is imminent. The well-known extraction systems do not yet provide acceptable uranium isotope separation factors, although they are characterized by a high speed of attaining equilibrium. Efficient organic complex-forming agents and new principles for the organization of phase flows in the stripping and enrichment lines will be necessary.

Since 1970 reports have been appearing about the separation of isotopes by laser. Great attention was paid to this at the Eighth International Conference on Quantum Electronics (San Francisco, 1974) and at the International Conference on Uranium Enrichment Methods (London, 1975).

Laser separation of isotopes includes the stages: introduction of the starting material into the system, selective excitation, and extraction. As the starting material, at present the vapor of a mixture of isotopes in atomic and molecular forms is being used; there are indications, however, of the possibility of using starting materials in both the liquid and solid states [15].

Isotope separation is effected by laser by means of the selective excitation of the isotopes. In the interaction of radiation with a mixture of two isotopes, one of them is resonantly excited, while the other remains in the ground state. The excited isotope can be extracted by various physicochemical methods (photon ionization, photodissociation of the molecules, spatial separation of an atomic beam, chemical reactions). The most widely developed method is photon ionization. The extraction of laser-excited isotopes by means of chemical reactions is considered to be the most promising for industrial application.

The laser method is characterized by a high separation factor, which permits the same degree of enrichment to be achieved with a considerably smaller number of process stages; the degree of enrichment is sharply increased and the content of ^{235}U in the tailings is reduced to 0.03% [16]. In addition, with laser technology, the required power is proportional to the quantity of separated isotope, and not the starting material, as in the case of the methods being used at present, and therefore it is the least energy-consuming. The power consumption in the separation of a single atom of ^{235}U by different methods confirms this: gaseous diffusion, 3000 keV; centrifuge, 300 keV, and laser 100 keV.

The economic efficiency of the isotopes of certain metals (zirconium, iron, etc.) obtained by laser technology, which have a low neutron absorption cross section, for the manufacture of fuel-element claddings should be mentioned; this leads to a significant improvement in the use of neutrons and to a reduction of the requirements on the degree of uranium enrichment. Moreover, ^{232}U and ^{233}U can be separated by laser, which increases the efficiency of the uranium–thorium cycle, as it permits the use of ^{232}U as a radioisotopic source of heat. The industrial achievement of uranium isotope separation by laser technology is possible by 1985 [17].

Information has appeared on the possibility of commercial laser isotopic separation of plutonium earlier than for uranium isotopes. According to estimates of different researchers, laser technology is the most economical for the separation of highly radioactive ^{238}Pu from its other isotopes in spent nuclear fuel (the cost of 1 g of high-quality ^{238}Pu is reduced from 1300 to 125-250 dollars) [18]. ^{238}Pu is used as a compact energy source, e.g., for satellites and cardiological devices.

The introduction into production of laser technology will permit not only the utilization of uranium in thermal reactors to be improved, but will also permit optimization of the isotopic content of the fuel of breeder-reactors. Thus, ^{240}Pu undoubtedly can be more usefully used in breeder-reactors than in thermal reactors.

*Trioctylamine and tributylphosphate.

Manufacture of Fuel Compositions

In investigations in this direction, great attention has been paid to consideration and comparison of the various technological factors which determine the quality of the pelleted cores of uranium oxide, and discussion of methods of producing and using granulated oxide fuel in vibropacked fuel elements [19].

Thermal reactors of nuclear power stations, operating on enriched (2-4%) uranium, usefully use only ~1% of the required natural uranium. Therefore, in order to increase the utilization of natural uranium in the period preceding the bringing into operation of fast reactors, there is considerable interest in the conversion of thermal reactors partially to a plutonium fuel cycle, which provides for the repeated utilization of plutonium based on a mixed uranium–plutonium fuel. Numerous investigations of the technological and economic aspects of this process [20-26] have shown promise for the development of a cycle with this fuel in thermal reactors.

In the Soviet Union, as in other countries, investigations are being carried out on the design of fuel elements based on uranium–plutonium fuel for fast reactors [3, 27], including experiments on the irradiation of these fuel elements up to high burnups in the reactor [28]. Different technological schemes for obtaining a mixed oxide fuel for fast reactors are being analyzed.

In connection with the development of high-temperature, gas-cooled reactors and fast reactors, carbide, nitride, phosphide and other fuel compositions have been investigated [29, 30]. Many papers are devoted to the study of the physicomechanical, radiation, thermodynamic and other properties of these refractory uranium and plutonium compounds. In the investigations, an important place is being assigned to the manufacture of fuel elements based on microparticles (uranium dioxide or dicarbide) with multilayered protective coatings of graphite (of different density) and silicon carbide. The microfuel elements, when inserted in a graphite matrix, are grouped into elements of different geometry (rods, plates, and spheres) [30], and are characterized by a high degree of fission product retention (up to 1300-1400°C) [31].

The buildup at present of experience in the technology of manufacture of microfuel elements with coatings makes it possible even now to obtain coolant gas temperatures in nuclear reactors at 1000°C and somewhat higher. The production technology of microfuel elements is being advanced continuously. Although at the present-day stage of their production they are coating more than rod-type fuel elements, there is a basis for hoping that future improvement in the technology of manufacture of coated particles will bring their cost near to the cost of fuel-element rods. This permits microfuel elements with coatings to be considered as extremely promising fuel for future nuclear power stations.

Carbonitride fuel is considered to be the most promising for fast reactors. Possibilities are being developed for improving the technology for the production of carbides from oxides, and the design of continuous technological processes for obtaining carbonitride fuel, including also in granulated form, is promising.

Regeneration of Spent Fuel

In the nuclear fuel cycle, its regeneration is one of the most complex and most important technical problems. Regeneration remains one of the tightest points in the fuel cycle, from the point of view of guaranteeing production capacities essential for satisfying the requirements in the bulk production of fuel for nuclear power stations.

The industrial method of reprocessing the fuel from thermal reactors, which is unique in world practice, independently of its composition and degree of irradiation, is the continuous counterflow extraction of uranium and plutonium with solutions of tributylphosphate into diluents. The differences in the individual extraction schemes consist in the number of cycles of extraction purification, in the separation of uranium and plutonium in the first or second extraction cycle, in the method of separation, operations for the intercycle treatment of the uranium solutions, the presence of a nodal point in the final purification of the uranium (on silica gel, titanium phosphate, etc.), methods of concentration and refining of the plutonium.

The number of extraction cycles depends on the activity of the starting solution, which is determined by the type of fuel, depth of burnup, and cooling time. With approximately equal conditions, the decisive factor is the level of development of technology in a given factory, consisting in the correct choice of the optimum influence of factors which affect purification from fission products, such as the degree of saturation of the extractant with uranium, the acidity of the eluted solutions, temperature, the use of complexing agents, time of contact between phases in the extraction plants, chemical and radiation stability of the extractant and diluent, and the removal of certain fission products in preparatory operations.

Technicoeconomic requirements are met most completely by extraction schemes which ensure the following basic indices [32]:

Purification factor:	
uranium from plutonium	10^7
plutonium from uranium	10^6
uranium from fission products	10^7
in the first cycle	$2 \cdot 10^4$
in the second cycle	$5 \cdot 10^2$
plutonium from fission products	10^8
in the first cycle	$2 \cdot 10^4$
in the second cycle	$2 \cdot 10^3$
during anion exchange	3
Extraction of uranium and plutonium, %	99.9
Degree of regeneration, %	
nitric acid	95
extractant	99.7

Nonaqueous methods of regeneration of spent nuclear fuel (sublimation, pyrometallurgical processes, etc.), although quite well studied, at present have not reached the stage of industrial application.

Processes combining both aqueous and nonaqueous methods, e.g., the aqua-fluor-process, have proved to be interesting. The most important difference in the known variations of the process consists in the precipitation and separation of the valuable components. The aqua-fluor-process, with the extraction cycle for the combined extraction of the actinides into the organic phase and their purification from fission products at the head of the technological scheme, has the advantage over other alternatives [33]. The preliminary separation of the fission fragment elements from uranium and the transuranic elements considerably simplifies the direction and control of the entire process. Control is simplified in the operations for correcting and stabilizing the valence forms of plutonium and neptunium, and the solution of problems of the volatility of ruthenium in the zone of dehydration of the uranium product after removal from it of plutonium and neptunium is not eliminated, but is considerably facilitated. There is no need in the plant for any additional measures due to the buildup in the fluoride and separation zones of fluorides of the main mass of fission products and the origination of heat release as a consequence. The fission products are removed in the aqueous raffinate, which can be subjected to direct thermal concentration. The total purification factor from fission product elements in the extraction cycle amounts to 10^3–10^6.

The aqua-fluor-process permits spent nuclear fuel of any type to be regenerated: metallic (uranium, plutonium, thorium, or their alloys), oxides, carbides, nitrides, silicides, etc. However, the prospects for its industrial utilization are doubtful, because it is inferior to extraction methods in its technological indices and it leads to the formation of additional solid radioactive wastes.

In connection with the planned program of nuclear power generation development, the Soviet Union has worked out the principles for locating the establishments for regenerating the spent fuel from nuclear power stations, storage and transportation of the burnt fuel elements, and protection of the environment [34]. Technological schemes for the combined regeneration of spent fuel elements from nuclear power stations with thermal and fast reactors provide for the use of extraction [34, 35] and sorption operations [36] during regeneration of uranium, plutonium, and also neptunium, americium, and other valuable elements. In this case, considerable attention is being paid to the dissolution, radiation chemistry of aqueous and organic solutions, extraction and ion-exchange separation of macroquantities of plutonium and uranium, and the use of water-soluble neutron absorbers.

Extraction processes have been studied for the extraction, separation, and purification of uranium, plutonium, and neptunium in different valence states, using tertiary aliphatic phosphine and arsine oxides [37-39], amides of carbonaeous and phosphoric acids, phosphazo compounds [40], and phosphazines [41]. Thus, the investigation of the extraction capability of normal and isomeric tertiary aliphatic esters of phosphoric acid showed a higher chemical stability of tri-isobutylphosphate, a thermoselectivity of trialkylphosphates in the extraction of uranium and the transuranic elements from nitric acid solutions, and an inversion of the reaction capability of the transuranic elements in extraction equilibrium states [42].

As a result of a number of improvements, it became possible to achieve high indices in aqueous processes of spent fuel regeneration. Suffice to say, that in a single extraction cycle for regenerating the fuel elements of water-cooled/water-moderated power reactors, almost the complete separation of uranium, plutonium, and neptunium has been achieved, and the purification factor of uranium from fission products amounted to $5 \cdot 10^5$–10^6.

The fluoride method has been improved considerably, which was demonstrated by the experimental regeneration of the spent uranium fuel from the BOR-60 core with a burnup in excess of 10%, and a cooling time of 3-6 months. Stripping of the BOR-60 and BR-5 fuel with alloyed cladding has been carried out; the distribution of uranium, plutonium, and fission product elements throughout the plant of the technological circuit has been studied.

However, despite this, the unique methods of regeneration that have received widespread recognition are extraction using a system based on tri-n-butylphosphate (pyrex-process) and sorption based on strongly basic anionites in the refining.

The main problem at present is the further increase of the economy and effectiveness of technological schemes of operating, under-construction, and planned factories. Scientific-research development should be directed at increasing the purification of the valuable elements from fission products, the choice of the optimum ratio between extraction and sorption operations during regeneration, determination of the resources of the possible operation of extraction and sorption systems without their replacement or regeneration, and also on increasing the purification of plutonium and neptunium in the final refining operations and the use of fire, explosion, and nuclear safety systems. The latter is especially important, since emergency situations [43] in the majority of cases have been determined by the properties of the extraction and sorption systems in operation.

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NUCLEAR SUPERHEATING OF STEAM, RESULTS AND PROSPECTS AT THE PRESENT STAGE

B. B. Baturov, G. A. Zvereva,
Yu. I. Mityaev, and V. I. Mikhan

Testing of the extended operation of the superheating channels (SC) of the Beloyarskaya Atomic Electric Power Plant (BAEPP) has shown convincingly the economy of nuclear superheating of steam. The channels being operated at the BAEPP with a steam temperature up to 565°C at the exit confirmed their high reliability with a fuel depletion of 35 kg/ton and a calendar term of service of 6-7 years. These data allow acceptable economy to be obtained for an atomic electric power plant (AEPP) in comparison with a thermal electric power plant (TEPP), notwithstanding the relatively large number of neutron absorbers in the active zone.

The use of SC with fuel-element rods in which the amount of steel per unit mass of uranium is reduced but the catalyst is excluded from the fuel composition permits improving the engineering–economic characteristics of the channel reactor when nuclear superheating of steam is produced in it.

The results of the operation of the AEPP have been supplied in a report, and the prospects for nuclear superheating have been discussed as an example of the sectional-modular high-power reactor (RBMKP), in the design of which problems of this type in energy reactor construction, which is important from the standpoint of saving uranium and significant reduction of thermal discharge, have been solved most completely.

The idea of obtaining superheated steam directly in a nuclear reactor attracted attention in the very first stages of energy reactor development. Already in 1950 during discussion of possible alternatives to the reactor of the first AEPP in the world (Obninsk) an alternative with nuclear superheating of steam was considered [1], but it was postponed as technically insufficiently prepared. The successful start-up in 1954 and operational test of the reactor of the first AEPP served as the basis for realization of the idea of nuclear superheating of steam having high parameters in the most powerful energy reactors. Great interest in nuclear superheating was exhibited in the USA, West Germany, England, Sweden, and other countries; however, the long-term test of the operation of the I. V. Kurchatov BAEPP is the most impressive in the industrial sense.

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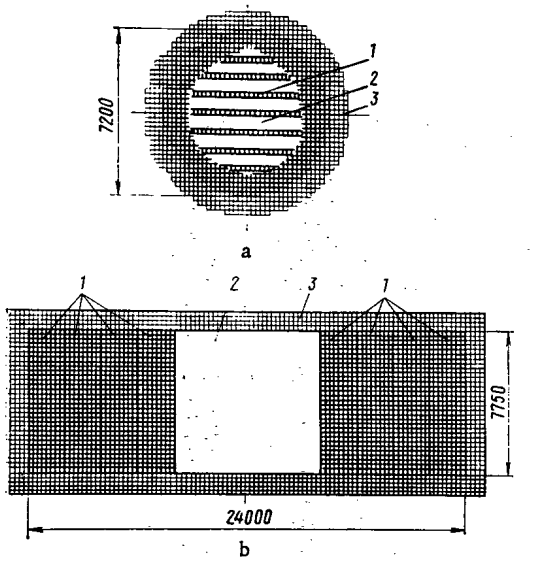

Fig. 1

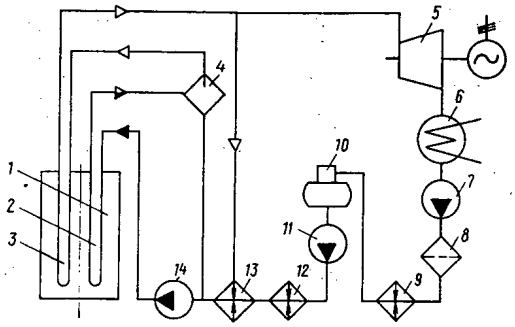

Fig. 2

Fig. 1. Arrangement of the fuel channels in a (a) circular and (b) rectangular active zone: 1, 2) evaporative and superheating sections; 3) reflector units.

Fig. 2. Engineering layout of the unit: 1) reactor; 2) evaporative channel; 3) superheating channel; 4) separator; 5) turbine unit; 6) condenser; 7) condensing pump; 8) condensation purifier; 9) low-pressure preheater; 10) deaerator; 11) feed pump; 12) high-pressure preheater; 13) superheating regulator; and 14) circulation pump.

The most suitable reactor in the constructional sense for obtaining high-parameter superheated steam is the channel-type, in which the separate organization of the evaporative and superheating zones, which should in the general case have different physicostructural characteristics and operating properties, is solved more simply in comparison with reactor vessels. These zones should provide, in particular, the necessary ratio of power to evaporation and steam superheating.

Nuclear superheating in connection with the use of a single-circuit layout with direct supply of steam to a turbine and the operation of thermomechanical equipment on active steam determined the advisability of the use of tubular-type fuel elements as a first step in reactors with nuclear superheating; such elements have already shown reliability in the operation of the reactor of the first AEPP. The standard parameters of traditional power engineering for steam were selected, viz., 510°C and 90 kgf/cm^2.

Construction of the channel-type water–graphite reactor which was adopted for design studies corresponded to the greatest extent to the problem posed, with the past experience and the possible outlook taken into account.

Peculiarities of Nuclear Superheating of Steam. Nuclear superheating of steam has a number of positive qualities. Nuclear superheating, together with the possibility of the use of standard thermomechanical equipment, provides a high thermodynamic efficiency to a facility, which lowers the consumption of nuclear fuel and the discharge of heat per unit of generated electrical energy and reduces the thermal emission into the environment. The latter fact takes on especially important meaning in connection with a significant increase in the total energy production and an increase in the concentration of AEPP in industrially developed regions, in particular in connection with estimating the possible ecological consequences resulting from the effect of the heat discharge on the temperature conditions of the environment. This effect is still difficult to measure in financial terms, but its significance increases in proportion to the growth of the energy supply, and it is impossible to disregard it.

The choice of a water–graphite channel reactor permits providing:

freedom of installation in the reactor of fuel channels of various purposes and differentiated action on the physical and heat-engineering characteristics of the active zone (Fig. 1);

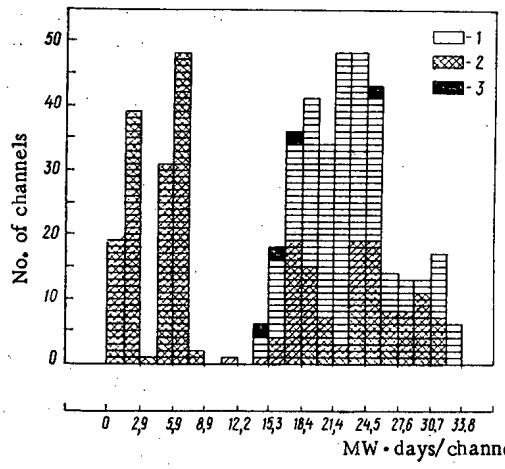

Fig. 3. State of the steam-superheating channels of the second unit on Jan. 1, 1976: 1) channels operating in the reactor; 2) channels removed from the reactor due to the absence of reactivity; and 3) channels removed from the reactor for defects in the fuel elements.

through-channel overloading for more effective use of the fuel in the case of a sufficiently good equalization of power distribution throughout the active zone;

the use of various designs of the fuel (removable and nonremovable) channels, sleeve and rod fuel elements (see Fig. 1);

the use of a progressive single-circuit engineering layout with the input of steam from the reactor to a turbine (Fig. 2); and

enlargement of the individual power capacities of reactors on the basis of standard elements without fundamental restrictions from above both from technical reasons and from the point of view of safety.

The operating possibilities of this type of reactor are distinguished by great flexibility. The output of a reactor with nuclear superheating of steam into the energy cycle can be accomplished without the use of outside heat sources.

The existing objective tendency towards reconsolidation of the energy supply diagram can increase the requirements on the adjustability of the energy units. The engineering and economic characteristics of reactors with nuclear superheating permit considering them as potential semipeak energy sources [2].

The introduction of nuclear superheating is positively expressed in the characteristics of the heat engineering portion of the unit, since the reliability of turbine operation is increased due to the elimination of the possibility of moist steam entering it. In this connection the layout of the turbine unit is also simplified due to rejection of intermediate separators and superheaters. The use of high-speed turbines (3000 rpm) in connection with the enlargement of the individual capacities of the turbine units to 1.2-2.0 million kW, as well as tapping the heat for central heating and industrial needs, has turned out to be theoretically possible.

Principal Problems of Organization of Nuclear Superheating of Steam. The most important scientific-engineering problem in creating a reactor with nuclear superheating is the development of fuel elements which would permit producing steam at a temperature of 500-540°C, a pressure of 90-130 kgf/cm^2, and thermal loads up to $1 \cdot 10^6$ kcal/m$^2 \cdot$ h with acceptable neutron-physics characteristics and an economically practical depletion of the uranium.

The physical problems of creating such a reactor, in addition to providing for uranium depletion (when significant unproductive neutron absorption in the SC is present) acceptable on economic grounds, are included in the maintenance of an equalized energy distribution and the ratio of capacities for producing and superheating steam necessary for a thermal balance. In this connection the physical characteristics of the reactor should provide for safety of the transition and start-up modes, in particular, an acceptable reactivity effect upon conversion of a SC from water cooling to steam, and vice versa.

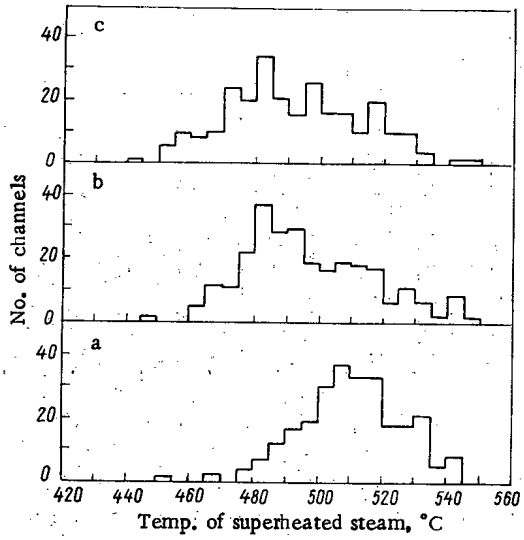

Fig. 4. Temperature of steam at output from the superheating channels:

	a	b	c
Date of measurement	Feb. 27, 1975	Aug. 2, 1972	Sept. 17, 1972
Electrical capacity of the unit, MW	196	170	172
Pressure in steam pipe, kgf/cm^2	75	73	72
Average superheating temp., °C	515	496	497

An important problem is providing for reliable operation of the reactor, fuel channels, and fuel elements in steady-state and transitional operational modes under variable-load conditions as well as for acceptable reliability of the main subassemblies and systems of the reactor based on a 20-30-yr useful life.

Nuclear superheating affiliated with the single-circuit thermal layout has determined the high level of requirements on provision of radiation safety for the staff, in particular, for the machine room when the turbines are operating on radioactive steam.

I. V. Kurchatov Beloyarskaya AEPP (BAEPP). The problems noted for nuclear superheating have essentially been successfully solved in the designs and upon the construction of the first reactors of the BAEPP.

The experimental checking of the most important elements of the reactor, physical characteristics, thermal hydraulic processes, and transitional engineering conditions was conducted on special test stands and in the experimental loops of the Obninsk AEPP [1, 3].

The powering-up of the first reactor with nuclear superheating and an electrical capacity of 100 MW occurred in 1964, followed in 1967 by a second reactor with a capacity of 200 MW; the gross efficiency of both units was 37-38%. The reactors are identical in the structural sense, and they differ only in the capacity and the external engineering layout. Up to now the total working time of both installations amounts to $\approx$21 reactor-years with an acceptable installed capacity usage coefficient of 62-77% and time coefficient of 75-91% [4].

One should note that the supplying of steam (20 Gcal/h) for heating a settlement located several kilometers from the power station is accomplished at the BAEPP along with the production of electrical energy.

Results of Operation of the BAEPP. The experience of extended operation of industrial reactors with nuclear superheating of steam is unique, and the data accumulated during their operation are the basis for creation of the next generation of reactors. Let us note the main results of the operation of the BAEPP reactors. Replaceable SC are used in the BAEPP in whose fuel elements, having stainless-steel jackets, uranium dioxide is used, which is enriched up to 5.0-6.5% in uranium and dispersed in a heat-conducting matrix alloy. The allowable temperature of the fuel element jackets is 630-650°C, which provides for superheating of steam up to 565°C in the channels.

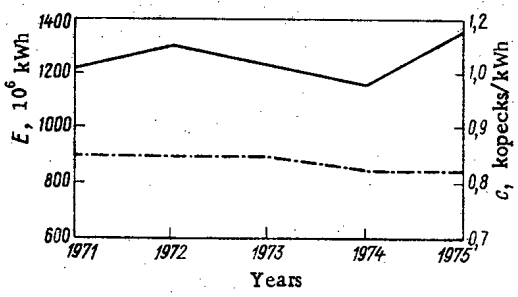

Fig. 5. Production (—) and cost (·--) of electrical energy in the second unit.

From the moment it was started up right up to the present more than 700 SC have been operated in the reactors. The average energy production of the unloaded SC is ≈26 MW · days/kg, and their useful life in the reactor is 5-6 years. However, the characteristics cited are not the limit. A group of channels is operating with an energy production of ~35 MW · days/kg, which it has been decided to bring up to 37-40 MW · days/kg in them.

During the operation 30 SC were extracted prematurely from reactor 1 for monitoring inspections and checks by virtue of putting the channels out of service and for other reasons, and only eight channels were extracted from reactor 2 in connection with disturbances in the operating conditions or for monitoring inspections (Fig. 3). During the entire period of operation of the superheating fuel elements no case of their being put out of service due to radiation impairment and incompatibility of materials was observed [5].

Thanks to the high reliability of the channels, the physical characteristics of the BAEPP reactors (uranium enrichment, reactivity) provide for a satisfactory amount of the fuel component in the cost of the electrical energy, notwithstanding the significant unproductive neutron absorption in the SC.

Evaluation of the fuel component in the cost of electrical energy permits confirming that at an average depletion of 34 MW · days/kg and with maintenance of the existing technology and the cost of preparation of the fuel elements and channels one can expect values of the fuel component of ≈0.3 kopecks/kWh, which makes nuclear superheating competitive in regions with a price level of 20-22 rubles/ton for organic fuel [6].

The operating experience with the Beloyarskaya reactors confirmed a rather stable equalization of the energy distribution. A reduction in the capacity of maximally loaded channels and a practically constant ratio of the total capacities of the evaporative and superheating circuits, as well as a negligible scatter in the temperature of the steam at the output from the SC (Fig. 4), are a consequence of this. Regulation of the temperature of the superheated steam, the average over the reactor and at the output from individual SC, presents no complication. The temperature of the steam at the output from the channels is stable in time, and its oscillations are negligible (2-3°C). Fluctuations on the ratio of power for evaporation and superheating of steam did not exceed 1%. When necessary, e.g., during start-up, one can vary this ratio by altering the radial energy distribution with the regulating rods.

The designers of the BAEPP reactors strived for the minimum effects possible of reactivity associated with variation in the operating conditions of the AEPP, in particular, a variation in the amount of water in the active zone under different operating conditions of the units, especially when starting and stopping them. The operation of both reactors of the BAEPP has confirmed their weak sensitivity to the amount of water in the zone. The greatest effect of reactivity in the BAEPP reactors is connected with emptying or filling the SC with water during the start-ups and shutdowns of the units. This effect changes significantly during the operating process, which is explained, e.g., by its dependence on fuel depletion; however, it does not exceed 0.4% in absolute magnitude. The variation of the reactivity during the start-up of the reactor is easily compensated by the regulation system.

Operational experiments with regard to the physics of channel reactors with nuclear superheating has shown that nuclear-physical characteristics can be selected in this type of reactor which completely satisfy both the nuclear safety requirements and the specific heat engineering requirements for nuclear superheating while simultaneously providing for an acceptable amount of the fuel component, notwithstanding the use of steel in the fuel channels and the additional neutron loss in the SC.

The production and cost of electrical energy during 1971-1975 in the second unit of the BAEPP are shown in Fig. 5.

TABLE 1. Principal Characteristics of Reactors with Nuclear Superheating of Steam

Characteristic	BAEPP 1	BAEPP 2	Supercritical parameters of steam		RBMKP
Electrical capacity, MW	100	200	800	1200	2400
Thermal capacity, MW	286	530	1820	2730	6500
Fuel charge, tons	67	50	59,8	80,4	293,2
Av. depletion (EC/SC), MW·days/kg	13,7/23	13,7/23	33/33	34/38	19/19
Enrichment of uranium, %	3,3	3,4	6		2
No. of evaporative chan. (EC), pieces	730	732	389	429	1920
No. of superheating channels, pieces	268	266	1304	1264	960
Steam temp. before the turbine, °C	505	515	540/540*	540/540*	450
Steam pressure before the turbine, kgf/cm²	90	90	240	240	65

*Turbine with intermediate superheating of steam.

The design of the fuel channels provides for an appreciable reserve with regard to the number of permissible heat-exchange cycles in the channel during the operating period of the fuel with rapid load variation. The number of such cycles during 6 years is about 200, and the actual maximum rate of change in the steam temperature was 20-40 deg C/min and in the pressure, ~0.7 kgf/cm² in 1 min. The reliability of operation of the basic equipment is characterized by the readiness coefficient of the main circulating pumps (0.997-0.999) and the feed pumps (0.993-0.995) [7].

The radiation environment of the AEPP site, and in particular next to the turbine during operation and in connection with the maintenance of the process equipment during shutdown of the units, does not prevent carrying out the maintenance operations. The deposition of radioactive corrosion products on the inner surfaces of the turbine are negligible. The radiation intensity at the high-pressure cylinder is 1.0-10 μR/sec and at the low-pressure cylinder 0.2-8.0 μR/sec. The strength of the radiation doses is 0.05-0.10 μR/sec in continuously occupied places, 0.3-12.0 μR/sec in places occupied part of the time, 15-20 μR/sec next to the equipment of the superheating circuit of the first unit, and 5-50 μR/sec near the equipment of the condenser–feed line of the second unit [8]. The ejection of radioactive products into the atmosphere under conditions of normal operation is less by a factor of 5-10 than the permissible health standards [9].

Prospects for the Development of Nuclear Superheating of Steam. A water–graphite reactor with nuclear superheating to supercritical parameters of steam can be used for the indicated purposes under conditions of the increasing need for energy systems in subpeak energy units and of the need for operation of AEPP according to a dispatcher load diagram. Design studies of such a reactor are being conducted in the USSR. According to the expenditures cited, a specialist atomic unit will be competitive with similar units using organic fuel for a comparable power of 800 MW in the utilization range of installed capacity of 3500-5000 h/yr [2].

The existing tendency of enlarging the individual capacities of reactors and turbogenerators makes the combining of nuclear superheating with the application of low-absorbing construction materials continually more urgent. The design of the RBMKP-2400 reactor, in which the superheating of steam to 450°C at a pressure of 65 kgf/cm² is provided [10], is promising in this direction; the zirconium alloys already mastered in reactor technology are being used, and stainless steel will be used only for the casings of fuel element rods made of uranium dioxide [11].

Prototypes of the superheating channels of the RBMKP-2400 reactor are presently undergoing resource tests on the BAEPP. Improvement of the engineering-economic indices of nuclear superheating is expected in the RBMKP-2400 reactor due to an increase in the specific power of the fuel, the use of more favorable construction materials for the active zone in the neutron physics respect, the application of nonremovable-channel design, etc.

The principle of sectional-modular preparation has been realized in the RBMKP reactor, which improves the engineering-economic indices of the AEPP, simplifies operations of bringing about a reactor on-line, and permits regulation of the temperature of the superheated steam with the help of a system for controlling and regulating the energy distribution. The reactor is discussed in more detail in [10, 11]. The principal technical characteristics of reactors with nuclear superheating of steam are given in Table 1.

CONCLUSIONS

The operating experience of the BAEPP reactors has confirmed the possibility of the industrial realization of nuclear superheating of steam right up to 510-540°C, sufficient reliability, and the safety of reactors of this class.

The introduction of nuclear superheating is economically justified: when steam is superheated to 500°C and higher with the use of stainless steels as the construction material in the active zone and the use of removable and sleeve fuel elements; when zirconium alloys are used in the active zone and the steam temperature is ~450°C, and when rod fuel elements, nonremovable channels, and the appropriate organization of steam in the channel is used.

Reactors with nuclear superheating of steam permit operation under variable conditions and at atomic heat and electric power plants with channeling of the heat to domestic and industrial needs.

Channel-type reactors with nuclear superheating permit enlarging capacity on the basis of standard units and the use of high-speed turbine units having large capacity, and they significantly reduce the thermal emissions into the environment.

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THE PRINCIPAL TECHNICAL PROBLEMS AND PROSPECTS FOR THE CREATION OF GAS-COOLED FAST REACTORS WITH A POWER OF 1200-1500 MW USING A DISSOCIATING COOLANT

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Preliminary engineering-economic characteristics of atomic electric power plants (AEPP) with a fast reactor of 1200-1500-MW electrical capacity were determined on the basis of neutron physics, thermal hydraulic, and engineering calculations and design studies of the reactor and the main equipment of an AEPP in which dissociating nitrogen tetroxide (N_2O_4) is used as the coolant (BRGD-1200-1500).

The advantages of such an AEPP are a decrease in the amount of equipment due to the use of a single-circuit layout of heat conversion and a reduction in the metal content of the equipment by virtue of peculiarities of the thermophysical properties of N_2O_4, as well as high yield rates of secondary nuclear fuel. This permits one to predict the attainment of specific investments in AEPP of the BRGD-1200-1500 type up to the level of investments in AEPP with water coolant.

Reactors of 1000-MW electrical capacity based on N_2O_4 can, according to the computational data, yield up to 500-900 kg/yr of plutonium. These same reactors permit yields of up to 1400 kg/yr when operated as reprocessors [1].

A large number of alternatives were considered in the course of the design studies of fast reactors based on N_2O_4, and they differ among themselves in the gas exit temperature of 280-570°C, the pressure in the circuit of 80-160 bars, the construction of the fuel elements (rod and spherical), and the type of fuel composition (matrix fuel based on uranium dioxide and nitrides in Nichrome or chrome matrices with 30-40% by volume [2]; low-alloy metallic fuel with double protection from possible interactions of N_2O_4 with the fuel; and carbide fuel [3] with a carbon-silicon casing for spherical microfuel elements).

All the alternatives discussed essentially satisfy contemporary requirements on the yield rate of secondary nuclear fuel. Investigations of the fuel cycles of the growing nuclear power show that the consumption of natural uranium in a nuclear power system can be reduced by 45-50% upon the introduction (in 5 years) of fast reactors based on Na and N_2O_4 in comparison with thermal and fast reactors based on Na (the external cycle time is T = 0.5-1 year). In addition, in ~30 years the system under discussion will develop into the mode of providing its own plutonium [8].

The principal thermal hydraulic and physical characteristics of breeder reactors and reprocessors of the BRGD type with a matrix fuel based on uranium dioxide and plutonium in the active zone (1500-MW electrical capacity) are given in Table 1. At a gas exit temperature from the reactor of 450°C and a pressure of 150 bars, a maximum temperature of the fuel-element casings of 650-680°C, and with heating of the gas in the reactor to 230-270°C one can achieve a heat release rate of 800-1000 kW/liter of the active zone, having obtained a doubling time of 5-6 years with a plutonium yield of 500-900 kgf/yr for breeder reactors and up to 1400 kg/yr for reprocessors.

The best characteristics of the BRGD-type reactor are produced by: the high energy release rate; the rigid spectrum of the neutrons (especially in the case of the use of a chrome matrix) $(\alpha)^{239}_{N_2O_4} = 0.248$ and $(\alpha)^{239}_{Na} = 0.260$; and the large contribution of the shields to the reproduction of fuel by virtue of the high leakage of neutrons from the active zone and the use of metallic uranium in the shields as the source material.

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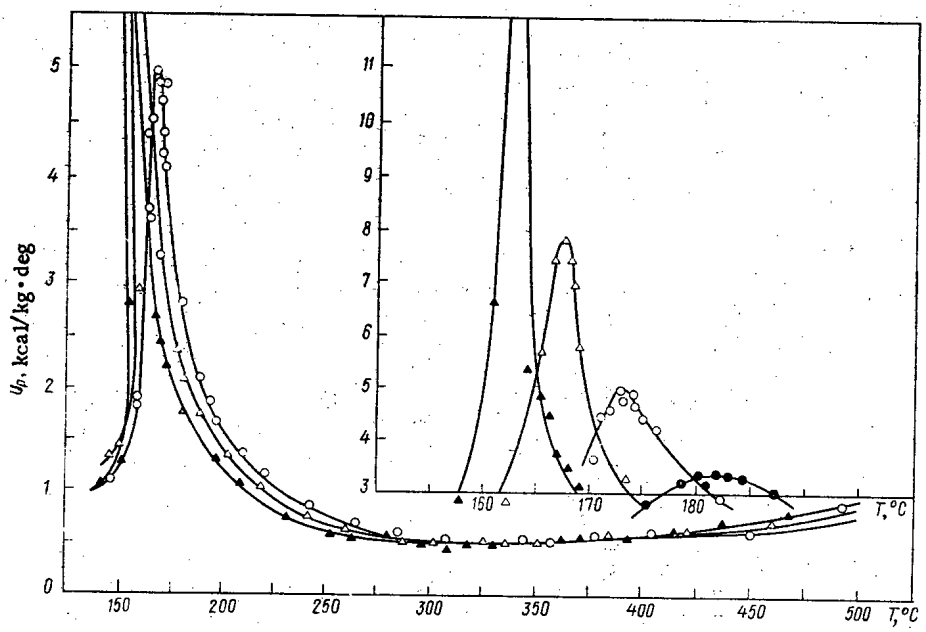

Fig. 1. Experimental data on the specific heat of N_2O_4, kcal/(kg·deg): ▲) 116; △) 130; ○) 150; ●) 170.

Alternatives with matrix fuel based on uranium dioxide were also considered along with the use of metallic uranium in the shields. The possibility of increasing the diameter of the fuel elements in the shields in comparison with metallic uranium serves as a prerequisite for this by virtue of the relatively high thermal conductivity and the higher limiting temperature of the matrix fuel.

Investigations have shown that a significant advantage is maintained in the doubling time of a reactor with metallic uranium in the shields when the matrix content in the fuel of the shields is > 40%. When the matrix content is ~30%, the doubling time of reactors with matrix and metallic fuel in the shields (with optimal shield thickness) is approximately identical.

Double protection of the fuel is advisable for increasing the reliability, excluding contact of the coolant with the fuel, and reducing the outflow of gaseous fission fragments. The use of matrix-type fuel [2], which consists of particles of uranium dioxide and plutonium covered by metal and contained in a casing made of stainless steel, is explained by this increase in reliability. The role of backup protection is filled by the metallic matrix with fuel particles dispersed in it.

The use of matrix fuel reduces, on the one hand, the amount of uranium dioxide and plutonium and increases the contribution of the construction materials, which increases the parasitic capture of neutrons. On the other hand, it is possible to raise the energy release rate of the active zone by a factor of 1.5-2 in comparison with pure uranium dioxide due to the better thermal conductivity of the matrix fuel (higher by approximately a factor of 5-6 than that of pure uranium dioxide). This circumstance allows the creation of a more compact active zone and production of satisfactory physical characteristics. In addition, the matrix material, playing the role of an additional casing, reduces the outflow of gaseous fragments into the circuit.

Microfuel elements based on UO_2 and UC_2 with a coating of pyrocarbon, silicon carbide, and chrome are proving to be very promising for application. Such coverings not only protect the fuel from the corrosive effect of the coolant but also retain the radioactive fragments very effectively [3].

Another distinctive feature of reactors based on N_2O_4 is the comparatively low level of the coolant temperature, which offers the prospect of using metallic uranium in the reactor shields.

The dissociating nitrogen tetroxide proposed for use as a coolant and working medium has the following interesting characteristics.

TABLE 1. Calculated Thermal Hydraulic and Physical Characteristics of Breeder Reactors and Reprocessors Operating on N_2O_4

Reactor characteristic	Rod fuel elements*		Reactors with microfuel elements*		Rod fuel elements†	
	chrome matrix		pressure, bars		chrome matrix	
	30%	40%	150	80	30%	40%
Thermal capacity of the reactor, MW	5100		5100		5100	5910
Temp. (exit/entrance), °C	451/183		451/183		451/183	
Gas pressure (exit/entrance), bars	154/169		150/80		154/169	84/99
Gas consump., kg/sec	5670		5670		5670	7260
Fuel in the active zone	UO_2+PuO_2 + matrix				UO_2 + matrix	
Equivalent diam., m	2,67	2,53	1,8	1,93	2,67	2,53
Height, m	1,02	0,98	1,8	1,92	1,02	0,98
Vol., liters	5,3	4,94	4,6	5,62	5,3	4,94
Av. heat release rate of active zone, kW/liter	864	926	1000	944	864	926
Cassette size, mm	92×2		172×2		92×2	
Triangular lattice step, mm	94		187		94	
Fuel density, g/cm³	10		10		10	
Fuel element diam., mm	6		6		6	
Material of fuel element casing	Stainless steel					
Casing thickness, mm	0,4		0,4		0,4	
Total critical charge of ^{239}Pu and ^{235}U, kg	1900	1780	1537	1794	2600	2460
Av. enrichment, %	12,8	14,5	15,62	15,43	17,6	20
Reproduction coeff. of active zone	0,90	0,80	0,73	0,75	0,45	0,38
Fuel of shields	Metallic uranium					
Yield of excess plutonium per year	690	620	860	940	1400	1380
Doubling time (years) when T = 0.5 year	4,6	5,2	4,0	4,5	—	—
Total reproduction coeff.	1,51	1,40	1,62	1,58	0,92	0,88

*For breeder reactors.

†For reprocessor reactors.

TABLE 2. Values of the Mean Integral Specific Heats of Various Coolants, kcal/kg·deg

T, °C	N_2O_4	CO_2	Na	He
150—280	1,5—1,6	0,28	0,31	1,243
200—500	0,7—0,8	0,275	0,306	1,243

1. The significant size of the thermal effect of the chemical reactions of dissociation upon heating and recombination upon cooling, $N_2O_4 \rightleftharpoons 2NO_2$ (−149 kcal/kg) $\rightleftharpoons 2NO + O_2$ (−293 kcal/kg), permits organizing intense heat removal in the active zone of the reactor and heat exchange equipment.

2. The values of the specific heat are high over a wide range of temperatures and pressures. Experimental data on the specific heat [4] are given in Fig. 1. The comparison of the mean integral specific heat (Table 2) was carried out on an isobar at the identical distance from the critical one ($\pi > P/P_{crit} = 1.1\text{-}1.2$).

3. Obtaining heat-transfer coefficients which are enhanced in comparison with the inert gases [5, 6] due to concentration diffusion; thus, average heat-transfer coefficients of ~90,000 kcal/(m²·h·deg) are obtained in tests for the conditions of the active zone of the BRGD-1500 reactor with respect to temperature, pressure, and thermal flux. This fact permits creating simultaneously a more compact and lower-metal-content heat exchange device.

4. The relatively small amount (~100 kcal/kg) of hidden heat of vaporization and the parameters on the saturation line permit realizing a subcritical gas−liquid cycle in which complete evaporation and superheating of the coolant occur due to regeneration; the reactor is purely gas-cooled in this case.

5. On account of the lower expansion ratio in the cycle (for equal initial parameters and identical cooling conditions of the terminal heat exchange) a turbine operating in N_2O_4 will have first stages significantly larger, and for a density of N_2O_4 behind the turbine 30-40 times larger than that of water vapor, the last stages will be 2-3 times smaller than for a turbine (of same capacity) operating on water vapor. This exerts a favorable effect on the efficiency, and the total number of stages is reduced by a factor of 4-5, which decreases the overall size and metal content of the turbine.

TABLE 3. Main Characteristics of Experimental-Industrial AEPP

Characteristic	Reprocessor	Breeder reactor	
Electrical capacity, MW	300	300	300
Thermal capacity, MW	955	955	955
Cycle		Gas–liquid	
No. of circuits, pieces	1	1	1
No. of reactor cooling loops, pieces	3	3	3
No. of emergency cooling loops, pieces	2	2	2
Main coolant parameters:			
flow rate through reactor, kg/sec	1064	1064	1064
temp. at entrance–exit of reactor, °C	196/480	196/480	196/480
pressure at entrance–exit of reactor, bars	165/152	165/152	165/152
temp. at entrance–exit of condenser, °C	60/31	60/31	60/31
pressure at entrance–exit of condenser, bars	2,25/2	2,25/2	2,25/2
Net eff. of AEPP, %	31,4	31,4	31,4
Fuel comp. in active zone	UO_2+30%Cr	UO_2+PuO_2+30%Cr	
Diam., m	1,3444	1,3444	1,3444
Height, m	0,74	0,74	0,74
Vol., liters	1050	1050	1050
Diam. of fuel elements × casing thickness, mm	6,2×0,4	6,2×0,4	6,2×0,4
Av. energy release rate (max.), kW/liter	835/1250	800/1225	825/1220
^{235}U–^{239}Pu charge, tons	0,935	0,591	0,574
Size of cassette "under key", mm	142	142	142
Operating period, effective years*	0,616	0,602	0,602
Shield material	Umet	Umet	UO_2+30%Cr
Total repro. coeff.	1,020	1,644	1,466
Amt. of excess plutonium unloaded from the reactor, kg/eff·yr.	381	210	160
Doubling time for Pu content (1%) in the unloaded uranium of the reproduction zone and T = 0.5 year	—	6,0	6,8

*Net operating time at nominal capacity.

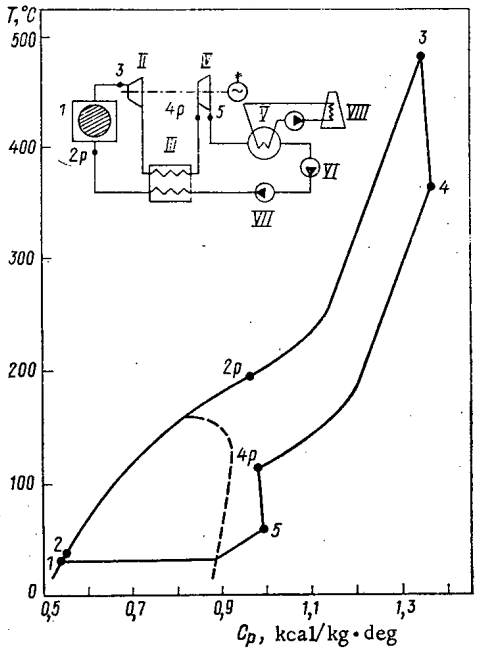

Fig. 2. Basic thermal layout and cycle of the BRGD-1500 AEPP: I) reactor; II) high-pressure turbine; III) regenerator; IV) low-pressure turbine; V) condenser; VI) booster pump; VII) feed pump; VIII) "dry" water-cooling tower.

6. Obtaining high thermal efficiencies of gas–liquid cycles over a wide range of temperatures and pressures.

7. The presence of resistant construction materials in an N_2O_4 medium.

8. The rather high radiation resistance of the coolant. Thus the radiation losses for the BRGD-1500 AEPP amount to 10^{-5}-10^{-6} in the case of a gas temperature at the reactor exit of 450°C and a pressure of 150 bars. Irreversible decomposition is higher by approximately a factor of 1.5 and 2 for H_2O and CO_2 under comparable conditions. The thermal decomposition of N_2O_4 is negligibly small for these parameters.

9. The intrinsic activity of the N_2O_4 coolant is rather low ($5 \cdot 10^{-5}$ Ci/g for N_2O_4 based on ^{16}N, and 10^{-8} Ci/g for that based on ^{17}N).

The corrosion activity amounts to $2 \cdot 10^{-7}$ Ci/g when a decontamination system with a coolant flow rate of 1% the total flow rate is present. The activity of the sodium circuit amounts to 50 Ci/liter as a rule for ^{24}Na.

A single-circuit thermal layout with heat regeneration at an intermediate pressure was selected for the BRGD-1500 AEPP [7]. The thermal layout includes three reactor cooling loops with turbines of 500 MW each. The main equipment of the AEPP also contains heat exchange devices – regenerators, a condenser, and feed pumps.

The basic thermal layout of an AEPP with the cycle is presented in Fig. 2. The coolant is compressed in the liquid phase in the booster VI and feed VII pumps up to the maximum supercritical pressure in the cycle; then it is superheated in the regenerator III to the gaseous state due to the heat of the gas entering from the high-pressure turbine II. The coolant is heated up to the maximum temperature of the cycle in the reactor I, from which the gas enters the high-pressure turbine; the final expansion of the gas takes place after the regenerator III in the low-pressure turbine IV. The cycle is completed in the condenser V, where the coolant is cooled to the minimum parameters of the cycle. Taking account of the coolant parameters on the saturation line, it is possible to use effectively "dry" air water-cooling towers of the Heller type VIII in the condenser (pressure ~2 bars, and entrance– exit temperature is equal to 60-31°C for N_2O_4).

Engineering-economic calculations have shown that effective plutonium yield and electrical energy production at acceptable cost can be accomplished in two economically equivalent optimal areas of maximum gas temperature behind a 430-480°C reactor for an oxide fuel and behind a 250-320°C one for a metallic fuel.

The main characteristics of the BRGD-1500 AEPP are given below for characteristic regions of the parameters.

Thermal capacity, MW	5100
Coolant flow rate, kg/sec	5670
No. of cooling loops, pieces	3
No. of emergency cooling loops, pieces	2
Parameters of coolant prior to the turbine:	
temp., °C	450
pressure, bars	154
Parameters of coolant at the condenser entrance:	
temp., °C	66
pressure, bars	2.36
Parameters of coolant prior to the pump:	
temp., °C	31
pressure, bars	2.02
Net eff., %	34
Cycle	Gas–liquid

The theoretical possibility of creating an AEPP at the exit from a reactor with a subcritical pressure of 80-90 bars in the circuit has also been evaluated at the Nuclear Power Institute of the Beloyarskaya SSR Academy of Sciences. In this case the physical characteristics and thermal-hydraulic characteristics of the equipment deteriorated.

Nitrogen tetroxide is a product of the chemical industry (it is used in the production of nitric acid and belongs to the group of nonexplosive noncombustible materials).

The operating experience accumulated in the chemical industry in connection with the production of N_2O_4 and the extended operation of semi-industrial test stands at the Nuclear Power Institute of the Beloyarskaya SSR Academy of Sciences (from 1965) permit drawing a conclusion as to the mastering of the coolant and the advisability of proceeding to larger-scale tests.

A large complex of experimental-industrial test stands has been created at the Nuclear Power Institute of the Beloyarskaya SSR Academy of Sciences on which tests are conducted of mock-up models of heat-exchange devices, scram system units, the turbine, pumps, and engineering instrumentation, materials study research is conducted, etc.

Countries which are members of the Council for Mutual Economic Aid are participating extensively in a program within the framework of the Program of Scientific-Engineering Cooperation of scientific-research work on mastering the dissociating coolant. Thus, loop and ampoule apparatus for materials study tests of construction and fuel materials are being produced at the Nuclear Research Institute of the Polish People's Republic, and a high-pressure apparatus for conducting thermal-hydraulic tests of mock-up equipment and engineering research is being developed in Hungary at the Budapest Technical University.

The concepts, scientific-engineering data, and basic design characteristics of the BRGD-1200–1500 AEPP outlined above show that the proposed AEPP satisfies contemporary requirements with respect to its engineering and economic characteristics, and the distinctive features of the engineering layout and the thermodynamic cycle (gas–liquid) permit reliably guaranteeing the safe operation of the AEPP. The program of theoretical and experimental research being conducted into the basis of the design and the mastering of the technology of the coolant operation in the required range of parameters provide every reason for posing the problem of creating an experimental-industrial AEPP based on N_2O_4.

The main characteristics worked out for an experimental-industrial AEPP with a fast reactor operating on N_2O_4 are given in Table 3.

The plant is intended for finishing up the research on a gas-cooled fast reactor operating on N_2O_4 and the study of a single-circuit layout for heat conversion, as well as for finishing up all the engineering solutions associated with the specific properties of the coolant being used.

The thermal capacity of 1000 MW was selected from considerations of the possibility of using this reactor after its completion and introduction as an independent power unit capable of providing for the production of low-potential heat and electrical energy together with the production of secondary nuclear fuel.

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PHYSICOTECHNICAL ASPECTS OF NUCLEAR AND CHEMICAL SAFETY OF POWER PLANTS WITH GAS-COOLED FAST N_2O_4 REACTORS

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The provision of safety is one of the most controversial problems in the design of new atomic power plants, especially of those based on fast reactors.

If atomic power plants are to be competitive with conventional plants with respect to safety, the probability of occurrence of serious emergency situations in them must be at least as low as in conventional plants at any given location. The fact that atomic power plants possess better thermoeconomic characteristics than thermal plants is presently not sufficient. Thus, all atomic plants currently being designed or constructed must be provided not only with conventional safety devices used in thermal plants burning organic fuel but also with special devices and systems that ensure security in case of specific accidents involving the danger of reactor core melting or discharge of the coolant into the environment.

By using dissociating nitrogen tetroxide (N_2O_4) as a coolant and process medium, it is possible to design high-power atomic plants with fast reactors having, as indicated by preliminary studies, promising technico-economic characteristics [1-3].

The main advantage of such plants is the possibility of using a single-loop heat-conversion circuit based on the gas–liquid cycle [4]. The heat circuit includes three reactor cooling loops with turbines of 500 MW each [1]. As follows from the description of the heat circuit of such a power plant, its reactor is purely gas cooled.

It is well known that the use of gaseous media for core cooling creates certain safety problems. These problems are associated with the high thermal stresses existing in the reactor core, the low storage capacity of the coolant, and the correspondingly fast rate of development of emergency situations.

The basic condition for safe operation is thus the assurance of reliable circulation of the cooling agent in emergency conditions and during reactor cooldown. The most dangerous emergency situations include the development of leaks at various points of the technological circuit, power cutoff in the entire plant and to the supply pumps, accidental changes of reactivity, etc.

Various possible emergency situations have been analyzed in course of the design of a high-power atomic plant (BRGD-1500) and an experimental industrial 300-MW electric power plant (BRIG-300).

One of the most dangerous failures in multiple-loop atomic power plants with gas or liquid metal coolants is power cutoff either to the entire plant or to the supply pumps and compressors, since in such cases it becomes very difficult to secure continuous circulation of the coolant. Such a continuity can be prolonged for a certain time by means of pressure equalization, provided there is a certain amount of coolant on the high-pressure side.

The use of a gas–liquid thermodynamic cycle has an advantage over purely gas or liquid metal systems because of the large amount of coolant in the loop in proportion to the flow rate, and because of the great difference between maximum and minimum pressure in the loop. With a constant flow rate in the technological circuit, the ratio of the coolant mass M to the flow rate is given by the factor $K_m = (M/G)\int_0^e dl/W$, equal to the time for which the section l can maintain the rated coolant flow rate. Depending on the construction and cycle parameters, this time can vary between 20 and 60 sec from section to section, whereas in purely gas cooled circuits using, e.g., helium, this time does not exceed 2-7 sec [5].

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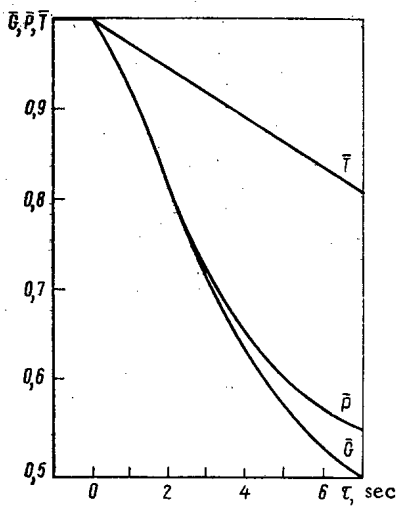

Fig. 1

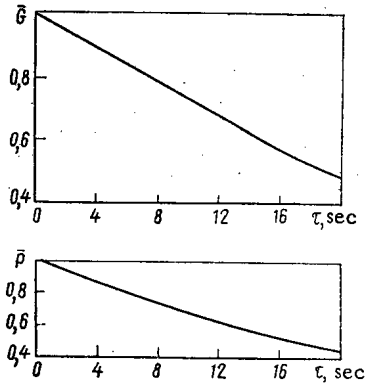

Fig. 2

Fig. 1. Variation of flow rate, pressure, and temperature of coolant at the reactor inlet in case of power cutoff to the main circulation pumps and the entire power plant.

Fig. 2. Variation of coolant pressure and flow rate at the reactor inlet in case a break in the main steam line behind the turbine.

The difference between maximum and minimum pressure in the loop determines the driving forces which cause natural coolant flow. For an atomic power plant with a fast reactor and a gas–liquid cycle using a dissociating coolant, the pressure increase factor amounts to about 70, and the pressure difference to 160 bars. This figure is considerably lower for other gas coolants. These facts offer good conditions for unaided flow of large amounts of coolant in case of power failure and shutdown of the main supply pumps [6]. For example, Fig. 1 shows the changes in coolant flow rate G, pressure P, and temperature T at the inlet to the reactor in case of power failure. It has been assumed that no pump coasting takes place (worst case conditions). Figure 1 indicates that a considerable coolant flow is maintained for a time sufficient to connect emergency power sources (the fuel-element temperature did not exceed the maximum permissible value for nearly 5 sec).

Studies of the loss of containment indicate that a break in the main steam line behind the turbine results in a gradual fall of coolant flow rate from nearly its nominal value (Fig. 2). The rate of fall does not exceed 2.2 bars/sec, the flow rate decreasing to about 50% of its nominal value after 20 sec.

Figure 3 shows the variation of coolant parameters at the reactor inlet in case of a break in the main steam line between the reactor and turbine. The initial rapid rise of the flow rate is followed by a fast drop to about 35% of the nominal flow rate after 20 sec; the temperature of fuel and of the fuel-element jackets initially decreases. In this case it is desirable to provide flow rate restrictors with a diameter one-half as great as the diameter of the break in order to prevent deformation and damage of the structural elements of the reactor.

The most dangerous emergency situation involves the loss of containment in the main loop at the reactor inlet which causes an initial drastic drop of coolant flow through the reactor and reverse circulation. Circulation reversal is practically instantaneous, so that the fluctuations of temperature and pressure taking place in a very short time interval are of great importance. Circulation reversal can be prevented by increasing the number reactor feed lines so that a break in any one line does not cause reversal and an intolarable drop of circulation.

Thus, due to the accumulation of coolant in the reactor circuit, power failure or loss of containment in the main steam lines do not immediately interrupt circulation which continues for a time sufficiently long for restoring power or for shutting down the reactor and starting the cooling procedure.

Of great importance in operating safety are the self-regulation properties of the reactor. A comparison with a sodium-cooled reactor proved that to cause the same power "overshoot," an N_2O_4 reactor needs a reactivity twice as high. The effect of density in large reactors is negligible within the operating range. Even with total removal of coolant from the core and breakdown of the power control system, the available protective devices are capable of shutting down the reactor without increasing the power above the permissible level.

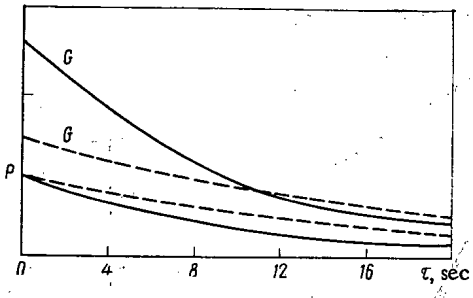

Fig. 3. Variation of coolant pressure and flow rate at the reactor inlet in case of a break in the steam line at the reactor outlet with (---) and without (——) flow restriction.

To improve safety, atomic power plants using N_2O_4 cooling are provided with the following additional measures: duplication of supply pumps, several reactor cooling loops (at least three), turbine bypassing in case of generator power failure, the use of two emergency cooling systems based on two independent and different operating principles.

To improve radiation safety, it has been decided to use double protection of the fuel from interacting with N_2O_4. The matrix fuel consists of uranium dioxide or plutonium particles covered by metal and placed in a stainless-steel jacket. The operation of fuel compositions in N_2O_4 media will be studied in Poland in the loop system of the "Mariya" reactor.

The use of chemically active N_2O_4 as a coolant and active medium of an atomic power plant, as proposed by the Institute of Nuclear Power Engineering of the Academy of Sciences of the Belorussian SSR, places additional demands on safety associated with the fact that any discharge of coolant can cause contamination of the surrounding space with nitrogen oxides.

Nitrogen tetroxide is toxic and strict norms are imposed on the amount of nitrogen oxides and of the products of their chemical interaction released into air. Thus, their total amount must not exceed 5 mg/m^3 in hot laboratories, and 0.085 mg/m^3 in populated areas. Accordingly, atomic power plants using N_2O_4 as a coolant and active medium must be provided, in addition to systems ensuring radiation safety, with systems securing chemical safety in normal operation and in emergency situations.

Nitrogen tetroxide as a process medium is quite familiar in the chemical industry (in the production of nitrogen fertilizers and other similar products). Experimental test stands operating for more than 10 years at the Institute of Nuclear Power Engineering of the Belorussian Academy of Sciences indicate that there is a practical possibility of designing constructions capable of withstanding pressure up to 160 bars and coolant temperatures of 500-600°C, i.e., over the entire range needed in atomic power plants. Accidental leaks of N_2O_4 vapors from the plant can be reduced to a few kilograms per hour and subsequently removed by the ventilation system. Chemical devices for monitoring the presence of nitrogen oxides in air allow early detection of coolant leakage. The points of leakage can be rapidly determined with the aid of special leak detectors as well as by color (yellow) and odor.

Calculations indicate that a ventilation pipe 125 m high together with "dry" cooling tower (Heller tower) can expel into the atmosphere more than one ton of coolant vapors per hour. Dispersal of such large discharges is made possible by a current of warm air 15°C hotter than the environment ascending at a speed of about 5 m/sec and having a diameter of 90 m at the top section of the tower.

Thus, with such a ventilation system the permissible discharge of coolant will be limited not by the sum of purposeful and accidental leaks but the material balance of the coolant and the observation of proper radiation norms.

The sources of prearranged leaks are blowoffs of gases from the main-loop condenser to remove the incondensible products of radiation and thermal decay and from the end seals of the turbine shaft. The major fraction of coolant vapors from these leaks is returned to the main circuit after separation and purification of the accompanying gases.

To localize any significant discharge of coolant vapors the power plant can be enclosed inside a sealed protective cover. The volume of the cover should be large enough so that the largest possible discharge of coolant

vapors does not raise the pressure under the cover above 4 bars. For still tighter localization, the pressure is lowered by means of special technological equipment that condenses the coolant vapors and collects the condensate in special reservoirs. After aging and radiochemical decontamination (if necessary), the condensed coolant is put back into circulation.

Penetration of the coolant to the environment through leaks in the condenser is prevented by using a "dry" cooling tower with an intermediate water-cooling circuit. At the same time, provisions have been made to allow the power plant to operate with a certain leakage from the condenser. The produced nitrous acid is removed from the cooling water by ion exchange and from the coolant, by rectification.

The following conclusions can thus be drawn from the above discussion: the specific properties of the technological scheme and of the gas–liquid cycle of N_2O_4 allow continuous circulation and initial heat removal when the protective system operates in case of emergencies; physicochemical properties of the dissociating coolant ensure that in the cool-down procedure of a gas-cooled reactor the coolant will be in a liquid phase; structural measures applied to fuel elements, control systems, cool-down procedures, power plant cooling, outside cover including devices for localization and elimination of the consequences of accidental discharge of coolant ensure safe operation of atomic power plants using N_2O_4.

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PHYSICAL PROPERTIES OF FAST POWER REACTOR FUELS AND THEIR EFFECT ON THE FUEL CYCLE

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The article discusses the principal properties of spent fuel: isotopic composition, activity of fission products and steel, and residual heat release. Data are cited on the accumulation of transplutonium elements in spent fuel and on the effect of these elements on activity and residual heat release.

The effect of higher plutonium isotopes on the natural activity of fuel and on the radiation environment in handling plutonium fuel is analyzed. The necessary degree of decontamination of fuel of fission products in chemical processing has been determined by analyzing the natural activity of fuel.

The effect of shortening the cooling time of spent fuel on the dynamic changes of fission products activity and residual heat release is investigated. The effect of storage time of fuel between chemical reprocessing and loading into the reactor on the magnitude of residual heat release and neutron activity of the unloaded fuel is estimated.

Engineering problems arising in connection with the reduction of the cooling time of spent fuel in the fuel cycle system are discussed on the basis of the obtained information.

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TABLE 1. Accumulation of Transplutonium Elements as a Function of Storage Time, kg/ton

Element	Storage time, years				
	0	0,5	1	2	3
^{238}Pu	0,9	1	1,05	1,2	1,3
^{241}Am	0,6	0,8	1	1,4	1,8
^{242m}Am	0,03	0,06	0,08	0,12	0,16
^{243}Am	0,5	0,5	0,5	0,5	0,5
^{242}Ku	0,11	0,18	0,24	0,37	0,49
^{243}Ku	0,01	0,02	0,03	0,05	0,07
^{244}Ku	0,1	0,1	0,1	0,1	0,1

TABLE 2. Total Activity of Fission Products

Unit of activity	Cooling time, years				
	0	0,25	0,5	1	2,5
Ci/ton $(U+Pu)O_2$	$7,6\cdot10^7$	$2,2\cdot10^7$	$1,3\cdot10^7$	$7,7\cdot10^6$	$2,7\cdot10^6$
g-eq/ton $(U+Pu)O_2$	$2\cdot10^7$	$3,8\cdot10^6$	$2,2\cdot10^6$	$9\cdot10^5$	$4,1\cdot10^5$

Fast power reactors are called upon to provide the required space of development of future nuclear power engineering. The solution of this problem depends in a considerable measure on fast turnover of spent fuel in the external fuel cycle which involves cooling, transportation, and storage of fuel, chemical reprocessing, preparation of new fuel elements, etc. The external fuel cycle time, together with the conversion ratio and the burnup fraction, is one of the most important factors that determine the breeding rate of nuclear fuel in fast reactors. For example, reduction of the external cycle from two years to one year has the same effect on the rate of plutonium breeding as a twofold increase of allowable fuel burnup.

Most of the external fuel cycle time is occupied by cooling the fuel after its exposure in the reactor. Various specialists give different figures for the optimal cooling time for fast reactor fuel. One of the main reasons for such different estimates is the fact that the properties and specific features of fuel have not been adequately studied. An important stage in such studies is a comprehensive experimental investigation of fuel properties. This stage has not yet received sufficient attention since the experience gained in operation of fast reactors is still insufficient. Nevertheless, quantitative analysis based on experimental nuclear-physics data makes it possible to reveal the basic features of the external fuel cycle which are pertinent for this or that stage of the cycle [5].

The characteristics of the BN-1500 reactor calculated on the basis of a fast sodium-cooled power reactor of 1500 MW (electrical) are listed below.

Thermal power	4000 MW
Fuel	$(U + Pu)O_2$
Pu enrichment	14-18%
Duration of run ($\varphi = 0.8$)	480 days
Average fuel burnup	70 kg/ton fuel
No. of fuel assemblies loaded (unloaded)	
core	320/yr
side breeder blanket	120/yr
No. of elements unloaded from assemblies	
core	90,000/yr
side breeder blanket	7000/yr
Mass of fuel in assemblies	
core $(U + Pu)O_2$	70 kg
end breeder blanket (UO_2)	50 kg
side breeder blanket (UO_2)	190 kg

TABLE 3. Activity of $^{95}(Zr+Nb)$, ^{103}Ru, and ^{106}Ru in Ci/ton $(U + Pu)O_2$

Nuclide	Cooling time, years			
	0,25	0,5	1	2,5
^{95}Zr	$2 \cdot 10^6$	$8 \cdot 10^5$	$1{,}1 \cdot 10^5$	$3{,}2 \cdot 10^2$
^{95}Nb	$3{,}3 \cdot 10^6$	$1{,}5 \cdot 10^6$	$2{,}4 \cdot 10^5$	$7{,}4 \cdot 10^2$
^{103}Ru	$2{,}6 \cdot 10^6$	$5 \cdot 10^5$	$2 \cdot 10^4$	—
^{106}Ru	$2{,}6 \cdot 10^6$	$2{,}2 \cdot 10^6$	$1{,}6 \cdot 10^6$	$6 \cdot 10^5$

Mass of steel in assemblies
- core and end breeder blanket . . . . . 60 kg
- side breeder blanket . . . . . 40 kg

Amount of plutonium dioxide (isotopic mixture) unloaded from:
- core . . . . . 3.5 tons/yr
- end and side breeder blankets . . . . . 0.8 ton/yr

Amount of PuO_2 loaded per fuel cycle as a function of external fuel cycle time:

T, years	0	0.5	1	2	3
PuO_2, tons	5	6.9	8.8	12.6	16.4

Isotopic Composition of Plutonium. Accumulation of Transplutonium Elements

At the first stage of development, fast reactors are intended to use plutonium fuel generated in thermal reactors. Plutonium obtained in thermal reactors under high burnup conditions [1, 2] has approximately the following isotopic compositions: 10^{-5}% ^{236}Pu and 1, 58, 23, 14, and 4% of $^{238-242}Pu$, respectively.

If such fuel were loaded into the BN-1500 reactor, plutonium unloaded from the reactor core would be composed of the nearly same amount of ^{236}Pu, $^{239-240}Pu$, and ^{242}Pu and approximately one-half as much of ^{238}Pu and ^{241}Pu.

The computed isotopic composition of plutonium in fuel unloaded from the different enrichment regions is: $(5\text{-}6) \cdot 10^{-6}$% ^{236}Pu, 0.7-0.5% ^{238}Pu, 62-65% ^{239}Pu, 8-7% ^{241}Pu, and 6-5% ^{242}Pu.

The isotopic composition of the unloaded plutonium is close to the composition of equilibrium plutonium generated in fast breeder reactors with multiple recirculation of fuel and plutonium make-up from the shield. According to the author's estimates the equilibrium composition of such plutonium is $5 \cdot 10^{-6}$, 0.2, 62, 27, and 6% for ^{236}P and $^{238-242}Pu$, respectively.

Transplutonium elements produced in irradiation of plutonium fuel affect both the activity and residual heat release of fuel. The content of transplutonium elements in plutonium fuel in the different enrichment regions of the core is given below in kg per ton $(U + Pu)O_2$:

$^{237}Np \sim 0.2$
$^{241}Am \sim 0.9\text{—}0.4$
$^{242m}Am \sim 0.05\text{—}0.03$
$^{243}Am \sim 0.5$
$^{242}Cm \sim 0.13\text{—}0.1$
$^{243}Cm \sim 0.012\text{—}0.006$
$^{244}Cm \sim 0.1$

These data were obtained under the assumption that the original loaded fuel contains no ^{241}Am.

In addition, irradiation causes production of 7-day ^{237}U with a final content of about 0.01 kg per ton $(U + Pu)O_2$.

The decay of ^{241}Pu in storage after chemical reprocessing causes production of ^{241}Am at the rate of 4.8% of ^{241}Pu content per year of storage, and of ^{237}U which comes into equilibrium with ^{241}Pu after 1 month in storage.

The loss of nuclear fuel in decay of ^{241}Pu and accumulation of ^{241}Am is the greater the longer the external fuel cycle. This and also the "parasitic" capture of ^{241}Am neutrons should be accounted for in calculating the rate of fuel increment or the doubling time.

TABLE 4. Activity of Volatile and Gaseous Fission Products in Ci/ton $(U + Pu)O_2$

Nuclei	Cooling time, years			
	0,25	0,5	1	2,5
^{129}I	0,065	0,065	0,065	0,065
^{131}I	$2,4 \cdot 10^3$	1,1	—	—
^{134}Cs	$1,9 \cdot 10^5$	$1,8 \cdot 10^5$	$1,5 \cdot 10^5$	$9 \cdot 10^4$
^{136}Cs	$3,5 \cdot 10^3$	—	—	—
^{137}Cs	$2,2 \cdot 10^5$	$2,2 \cdot 10^5$	$2,2 \cdot 10^5$	$2,1 \cdot 10^5$
^{85}Kr	$1,21 \cdot 10^4$	$1,19 \cdot 10^4$	$1,16 \cdot 10^4$	$1,05 \cdot 10^4$
^{131}Xe	$4,7 \cdot 10^2$	—	—	—
^{131}Xe	$5,2 \cdot 10^1$	—	—	—

If the regenerated fuel is stored for some time and then irradiated in the reactor, the amount of accumulated transplutonium elements increases as a result of the initial presence of ^{241}Am. Table 1 shows the effect of storage time of regenerated fuel on the accumulation of transplutonium elements in plutonium fuel unloaded from the intermediate enrichment zone in the reactor.

The content of ^{241}Am and ^{242}Ku in the unloaded fuel increases 3 and 5 times, respectively, when the storage time of the regenerated fuel is increased from 0 to 3 years. The increase of ^{242}Ku content will significantly change the neutron and α activities and the residual heat release in the unloaded fuel.

Activity of Fission Products. Residual Heat Release

The activity of fission products must be known for evaluating both the degree of decontamination of fuel in chemical reprocessing, and the radiation environment and residual fuel release in transport and technological operations within the reactor and in the transportation of spent fuel to the reprocessing plant.

Table 2 shows the total activity of solid fission products ($T_{1/2} \geq 5$ days) of spent fuel from the reactor core as a function of cooling time after unloading. After a cooling time of one-half year, most of the fission products activity comes from ($^{95}Zr + {}^{95}Nb$), ^{106}Ru, and ^{144}Ce nuclides. With increasing cooling time, the activity of fission products is governed by ^{106}Ru, ^{144}Ce, ^{147}Pm, ^{134}Cs, and ^{137}Cs nuclides. ^{134}Cs is produced by activation of the stable cesium isotope.

After prolonged storage (>3 years) the activity of fission products is determined chiefly by the nuclides ^{90}Sr and ^{137}Cs.

During the first half year of cooling, the total γ activity of radium fission products decreases by a factor of 10, in the next half year the activity decreases by half so that from this point of view prolonged storage of fuel is not advisable.

To secure nuclear safety, a certain amount of fissionable material is loaded into the dissolver apparatus. As a result, the concentration of fission products in dissolved plutonium fuel of the fast reactor will be after chemical reprocessing lower than in dissolved fuel of thermal reactors.

For cooling times up to 3 years the most difficult to purify in chemical reprocessing are the nuclides $^{95}(Zr + Nb)$, ^{103}Ru, and ^{106}Ru [3]. Table 3 shows the activity of these nuclides as a function of fuel cooling time.

Of considerable importance in the external fuel cycle technology are volatile and gaseous fission products. Table 4 shows the activity of such fission products of plutonium fuel as a function of storage time after unloading from the reactor.

With the exception of 10-yr ^{85}Kr with an activity of 10-12 Ci/ton of fuel and the long-lived ^{129}I, whose activity after 6 months is comparable to the activity of ^{131}I, only an insignificant amount of volatile and gaseous fission products remains after storage of 6 months or more. For cooling times of 6 months or less, the activity of ^{131}I increases drastically and requires a special decontamination system.

In addition, the fuel of fast reactors contains radioactive ^{3}H and ^{14}C. Tritium forms in the fuel elements in the course of irradiation in the reactor; spent fuel contains 0.5-5% of the total amount of tritium producing up to 70 Ci/ton of fuel. A conference of the IAEA established that the removal of tritium does not present special difficulties [4].

TABLE 5. Neutron Activity of Spent Fuel in 10^9 neutrons/sec · ton $(U + Pu)O_2$

Isotope	Cooling time, years					
	0	0,25	0,5	1	2	3
^{242}Ku	2,8	1,9	1,3	0,6	0,1	—
^{244}Ku	1,2	1,2	1,2	1,1	1,1	1,1
Total activity	4	3,1	2,5	1,7	1,2	1,1

^{14}C is formed in the reaction $^{14}N(n, p)^{14}C$, and with a nitrogen concentration of 0.1% in the oxide fuel the activity of ^{14}C amounts to about 10 Ci/ton of fuel. According to [4], the escape of ^{14}C into the atmosphere during reprocessing can act as a source of population irradiation. The problem of ^{14}C is still not clear and requires further studies. The activity of the structural materials of fuel elements and thermal assemblies is governed by ^{54}Mn and $^{58,60}Co$ isotopes and after 6 months of cooling amounts to $3 \cdot 10^5$ g-equiv/ton of steel. The activity of steel drops by a factor of 2.5 after 1 year of storage and by a factor of nearly 15 after 3 years.

Neutron activity of spent fuel is determined chiefly by the content of ^{242}Ku during the first year of storage and by the content of ^{244}Ku afterwards. Table 5 shows data on the neutron activity of fuel unloaded from the intermediate enrichment zone as a function of storage time, the storage time of the regenerated fuel before irradiation being zero.

As the storage time of regenerated fuel increases, the neutron activity of spent fuel increases as a result of increasing ^{242}Ku content. Because of the high neutron activity of spent fuel, the shields of transportation containers should include hydrogenous materials since heavy materials such as steel and lead used as γ-radiation shields are not effective against neutrons.

After a cooling time of 0, 0.25, 0.5, 1, and 2.5 years the residual heat release is 42.5, 5.6, 3.5, 2.0, and 0.7 kW per fuel assembly respectively, decreasing by a factor of 12 after the first 6 months and by a factor less than 2 in each succeeding 6-month period.

The contribution of ^{242}Ku into residual heat release is 2, 11, 12, 10, and 3% after 0, 0.25, 0.5, 1, and 2.5 years of storage, respectively.

After 1 year of storage of regenerated fuel the contribution of ^{242}Ku into residual heat release can reach 25% (6 months cooling time).

Natural Activity of Regenerated Fuel and Radiation Environment in Fuel Production

One of the components of natural activity of plutonium fuel is α activity. According to [1, 2] the total α activity of plutonium fuel with an isotopic composition as obtained in a thermal reactor is about 230 Ci/kg PuO_2 and is determined chiefly by ^{238}Pu. The α activity of fast-reactor plutonium of equilibrium composition is approximately one-half of the α activity of thermal-reactor plutonium. High α activity necessitates hermetically sealed technological equipment for handling unshielded plutonium.

The natural neutron and γ fuel activity governs the degree of exposure of personnel in the course of production of fuel elements and heat-generating assemblies.

Neutron activity of fuel is due to spontaneous fission of ^{238}Pu, ^{240}Pu, ^{242}Pu and to the (α, n) reaction with oxygen.

Neutron activity of thermal-reactor plutonium amounts to about $4 \cdot 10^5$ neutrons/sec · kg of PuO_2 [1, 2]. The contributions of spontaneous fission and of the (α, n) reaction are approximately the same. Neutron activity of equilibrium plutonium from a fast reactor is equal to the neutron activity of plutonium from a thermal reactor. The neutron activity due to the (α, n) reaction in light-element impurities (Be, B, F) is comparable to natural neutron activity of plutonium fuel when the amount of impurities exceeds 10^{-2} mass %.

Gamma activity of plutonium fuel is determined by "soft" characteristic radiation (13-17 keV) and amounts to about 10^{12} γ-quantum · sec/kg PuO_2 for thermal-reactor plutonium and about $4 \cdot 10^{11}$ γ-quantum · sec/kg PuO_2 for equilibrium plutonium from a thermal reactor. The γ radiation intensity of ^{241}Pu (in equilibrium with ^{237}U) and ^{241}Am isotopes exceeds the intensity of soft characteristic radiation.

TABLE 6. Decontamination Factor for Fission Products

Cooling time, years	^{95}(Zr + Nb)	^{103}Ru	^{106}Ru
0,25	$5,3 \cdot 10^7$	$1,3 \cdot 10^6$	$6,5 \cdot 10^6$
0,5	$2,3 \cdot 10^7$	$2,5 \cdot 10^5$	$5,5 \cdot 10^6$
1,0	$3,5 \cdot 10^6$	$1 \cdot 10^4$	$4 \cdot 10^6$
2,5	$1 \cdot 10^4$	—	$1,5 \cdot 10^6$

In the absence of ^{241}Am, the dose rate from unshielded plutonium fuel of a thermal reactor is determined by ^{238}Pu radiation and can reach up to about 3000 μR/sec on the surface of plutonium dioxide powder; the dose rate of equilibrium plutonium of fast reactors is 1000 μR/sec. After 1 year of storage of thermal-reactor plutonium, the accumulation of ^{241}Am raises the dose rate by about 1500 μR/sec.

The dose rate on the surface of thermal-reactor fuel pellets with 18% plutonium enrichment can reach up to ~400 μR/sec and decreases rapidly with distance so that manual handling of such pellets is safe.

The γ-radiation dose rate of fuel elements and heat-generating assemblies is due mainly to decay products of ^{241}Pu, ^{237}U, and ^{241}Am. On the surface of fuel elements (thermal-reactor plutonium) the dose rate is ~ 50 μR/sec and is equal to the maximum permissible dose rate for manual handling at a distance of 5 cm from the surface. The dose rate of equilibrium plutonium of fast reactors is approximately one-half as high. After 1 year of storage the dose rate of regenerated fuel increases by a factor of nearly 2 as a result of ^{241}Am accumulation.

The dose rate of heat-generating assemblies using thermal-reactor plutonium is also determined by γ and neutron radiation and can reach up to 60-80 μ rem/sec on the assembly surface depending on the degree of fuel enrichment. The dose rate is not affected significantly by accumulation of ^{241}Am in regenerated plutonium during storage (5% increase per one storage year). The contribution of ^{236}Pu decay products into the dose rate is small even after 10 years of storage.

To ensure radiation safety in fuel assembly operations, a 15-cm-thick shield of a hydrogenous material should be mounted at the level of the active part of the stack providing free access to the cap and stem of the assembly. The dose rate from an unshielded heat-generating assembly does not exceed the maximum permissible value at a distance 2 m from its surface.

The necessary decontamination factors can be evaluated considering natural activity, the activity of plutonium fuel fission products (see Table 3), and the permissible concentration of fission products which is difficult to purify in chemical reprocessing. Assuming that the contribution of these fission products into the surface dose rate of heat-generating assemblies does not exceed 10% (for each isotope), the permissible content of these nuclides in regenerated plutonium fuel has been calculated as follows: ^{95}(Zr + Nb) 0.1 mCi/kg, ^{103}Ru 2 mCi/kg, and ^{106}Ru 0.4 mCi/kg of $(U + Pu)O_2$.

Table 6 lists the necessary decontamination factors for these nuclides.

CONCLUSIONS

An analysis of fast-reactor fuel reveals certain specific features that are important in planning the entire fuel cycle technology.

The principal drop of activity of spent fuel takes place during the first 6 months.

The problems of gaseous activity must be analyzed if the cooling time is less than 6 years.

The problem of ^{241}Am can have a significant effect on fuel cycle technology. Increasing the duration of the entire fuel cycle causes partial loss of ^{241}Pu as a result of its decay into ^{241}Am. Increasing the duration of the external cycle interval between chemical reprocessing and mounting the heat-generating assemblies in the reactor results in an increased accumulation of ^{242}Ku which increases the neutron activity of fuel and the heat release.

The natural fuel activity should be given much attention at such stages as the preparation of fuel elements and heat-generating assemblies, transportation of "virgin" fuel, and initial inspection in the reactor. Manual operations can be carried out despite significant levels of natural radiations.

The transportation and handling equipment should be capable of unloading and storage of spent fuel after a short cooling time.

High activity and heat release complicate transportation and distribution of spent fuel.

All stages of fuel cycle are interrelated and necessitate a comprehensive technological and economical analysis which should take into account all specific features of fuel discussed above.

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ATMOSPHERIC RELEASE OF VOLATILE FISSION PRODUCTS FROM OPERATION OF NUCLEAR POWER REACTORS AND SPENT FUEL REPROCESSING FACILITIES AND PROSPECTS FOR EXTRACTING THE PRODUCTS

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One of the most important scientific–technical problems in the development of nuclear power is that of ensuring a high level of extraction of radioactive elements in gaseous emissions from reactors and radiochemical reprocessing plants. The importance and urgency of this problem are evident at present, in spite of the fact that, among all types of industrial and agricultural human activity, the safety of the personnel, the population and the surroundings is highest in the atomic and the nuclear power industry. Here the main concern is in removing relatively long-lived volatile fission products from the gaseous emissions, primarily tritium, iodine isotopes and radioactive inert gases (RIG), in addition to collecting aerosols.

As a rule, short-lived isotopes of iodine and RIGs are found in the local radioactive makeup of the atmosphere and constitute a radiation constraint only near nuclear objects. Because of their rapid decay, their contribution to the radiation dosage over large areas is insignificant.

The long-lived nuclides ^{3}He, ^{85}Kr and ^{129}I propagate in the atmosphere on a global scale, and further development of nuclear power requires a comprehensive examination of possible effects associated with contamination of the surrounding medium by these nuclides. The accumulation of gaseous fission products in nuclear fuel depends on the burn-up level and the type of fuel (and reactor). For example, the tritium content in thermal and fast reactor fuel is 200 and 2000 Ci/ton, respectively, for ^{85}Kr it is 5000 and 20,000 Ci/ton, and for ^{129}I it is from 10 to 15 Ci/ton. When one considers that in 2000 years the total reactor power will reach 4000-5000 million kW, one can calculate the expected accumulation at that time of tritium, iodine, and krypton.

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At present in nuclear fuel cycle operations the ^{85}Kr is practically ejected into the atmosphere. The main source for its accumulation in the surrounding medium is reclamation plants for spent nuclear fuel.

Krypton is absorbed to an extremely small degree by dry rock, is absorbed very little by the oceans, and is not assimilated by living organisms. As a result, the chief process whereby ^{85}Kr is eliminated from the atmosphere is its radioactive decay ($T_{1/2} = 10.6$ years).

In the world today most of the ^{85}Kr is distributed mainly in the troposphere of the northern hemisphere. Its concentration in air at ground level is ~20 pCi/m^3. If nuclear power develops at the estimated rate, the radiological capacity of the atmosphere in regard to ^{85}Kr, equal to $1.5 \cdot 10^4$ MCi, will be exceeded before the year 2000; the annual emission of krypton at that time will be $1.7 \cdot 10^3$ MCi, and the specific activity of air in regard to this nuclide will reach 10^{-9} Ci/m^3. Under unfavorable annual conditions the local contamination of the atmosphere in a fuel reprocessing locality may exceed the limiting allowable concentration by a factor of 10-100. Accumulation of ^{85}Kr in the atmosphere leads to deterioration of the environment and is a problem in air-separator plants, where by the year 2000 the radiation dosage received by the workers may be 10-15 mrem/yr: a user of stable krypton will obtain up to 15 mCi of ^{85}Kr with each standard bottle.

While the technical capability presently exists to seal fuel rods and thereby localize the main mass of ^{85}Kr in the spent fuel elements, and to concentrate the removal of gaseous emissions from it at the reprocessing plant, for tritium, however, it is a different matter. Since it has an extremely high permeability, it can diffuse through the fuel envelope during reactor operation and enter the surrounding medium as liquid and gaseous emissions, both at the power station and in the fuel reprocessing plant. Existing data indicate that the residual content of tritium, e.g., in the spent fuel of fast reactors, enclosed in a stainless-steel envelope, constitutes only a few percent of the theoretical accumulation. It follows from this that to prevent the emission of tritium to the atmosphere one must purify the emissions, not only in the reprocessing plant, but also in the power station.

The tritium resulting from power plant operation propagates globally. However, in contrast with RIG it is relatively easily oxidized and forms a compound with water as the HTO molecule. The comparatively rapid removal of tritium from the atmosphere and its scattering in the surrounding medium as liquid debris leads to its nonuniform geographic distribution. For example, with a background concentration of tritium in atmospheric precipitation in 1975, its concentration in water at the lower levels of the Danube was from 130 to 200 TE. In the same period in the surface water of the Black Sea the tritium concentration was 30 TE, and in the water of the Baltic Sea it was 60 TE. In atmospheric precipitation water over the USSR, the tritium content in 1972 was 90-230 TE; in the Leningrad region during 1974-1976 it was 30-60 TE. In spite of the nonuniform distribution of tritium, it is assumed that finally it will accumulate mainly in the mixing layer of the ocean water.

Information on the presence of ^{129}I in the surrounding medium is presently limited mainly to data from the immediate vicinity of nuclear plant and experimental nuclear test ranges. For example, its concentration in air at various points of the U.S.A. ranges from 10^4 to 10^9 atoms/m^3 of air, and in air above the oceans it ranges from 10^5 to $3 \cdot 10^6$ atoms/m^3.

Analysis of the propagation of volatile radionuclides around emission sources shows that, prior to global mixing, they will exist in concentration exceeding the background at a distance up to several hundreds of kilometers from the source. For an expected average distance on the order of 100-120 km between nuclear plants in some regions there will be a field which adds the emissions from individual plants, and thus, there will be a danger of regional contributions to the surrounding medium and the population.

The purification of gaseous emissions is universally recognized to be an inherent facet of the technology of spent fuel reprocessing. Recently, both nationally and internationally, a trend has appeared towards formulating standards for emissions. For example, in the U.S.A. it has been established that the standards for emission of ^{85}Kr in that country may be limited to 50,000 Ci/yr · 1000 MW. This means that a 10-20-fold purification of gaseous emissions from krypton is required.

Many factors determine the fraction of gaseous fission products (GFP) released in different reprocessing operations, and therefore an absolute value for each stage cannot be predetermined for all cases. Practical experience shows that, e.g., in the process of mechanical cutting of fuel elements from water-cooled –water-moderated reactors, about 5-10% of the iodine, up to 20% of the tritium, and 40-50% of the krypton are carried away with the air flow. The remaining part is retained by the fuel, goes into solution with the fuel, and is subsequently distributed between the gaseous and aqueous phases.

The variety of forms in which iodine occurs makes the problem of removing it a very complex one. Usually the following forms are found: molecular iodine, iodides, iodates, and iodo-alkali (as a rule, methyl iodide). Some fraction of the iodine comprising part of the initial solutions can be retained through the extraction operations and is then distributed between the aqueous and the organic phases, thereby setting up conditions for contaminating both the final and the intermediate products. This last fact makes for accumulation of iodine in the washing solutions and subsequent unmonitored emission in the low-activity separation line.

All that has been said confirms that it is desirable to remove iodine prior to the extraction operations. A very effective method of removing iodine from solutions after the fuel is dissolved is to blow it away with a stream of air (both in the molecular form and in the form of iodo-alkalis), and subsequently to absorb it from the gas–air stream by means of solid absorbers or solutions. Many kinds of solid absorber, based on inorganic and organic substances, are well known, containing different impregnators (nitric acid or silver iodide, etc.), which remove iodine efficiently: in element form up to 99.95%, and in metal iodide form up to 99.90%. A negative facet of solid absorbers is their low specific capacity and the difficulty of regeneration (for repeated use), which is particularly important in regenerating fast reactor fuel, containing a high iodine content. In this respect liquid absorbers deserve attention, e.g., solutions of inorganic salts, e.g., mercury nitrate, and organic solvents and have significantly higher specific capacity with respect to iodine. However, they are inferior to solid absorbers in regard to the purification of gases from iodine vapor. Therefore, it is more promising to employ purification systems containing two successive operations: absorption by liquid absorbers and a final cleaning with solids. Any method of cleaning unavoidably meets the problem of safe containment of residues containing ^{129}I.

A no less important task is that of complete purification and subsequent burial of tritium. Its distribution between the liquid and gas phases leads to the situation where one must create different purification systems. While the vapor phase can be collected in a small volume, after condensation in cold traps, liquid tritium-containing products (condensates from chilling the first cycle refined products) are distinguished by having a large volume, and for this reason their burial is difficult. At present intensive studies are in progress to assess possible recycling of tritium-containing condensates (and nitric acid) in the technological process.

A possible method of removing tritium being considered is volumetric oxidation of the fuel (voloxidation), followed by removal of the RIG in a minimum volume. This method is being studied also from the point of view of removing other volatile products, iodine, RIG, and possibly ^{14}C, although the problem of ^{14}C is less clear, both because of the sources of its formation, and also because of the uncertainty in determining it.

The single inert gas isotope which is a radiation hazard is ^{85}Kr. Until now existing plants for reclaiming fuel have not established systems for eliminating krypton, at least systems designed for full-plant power. However, in the not too distant future such systems will apparently be commissioned at all the plants.

The principles which are being followed in setting up facilities for purifying gaseous emissions from RIG at the reactors and in the reclamation plant differ substantially, due to differences in the chemical and isotopic composition of the emissions. The purification systems at the reactors are intended to eliminate short-lived nuclides, which, in turn, allows short-duration gas containment systems to be used as the main technological agent, set up in the emissions process line. A system which would ensure operation of such a process must act for a long period without the need for periodic regeneration of the cleaning agent. Here the short-lived RIG nuclides decay and are localized in the cleaning system itself, without reducing its efficiency of operation. In the practical operation of reactors of water-cooled–water-moderated reactors such systems are based on absorption of RIG from gaseous emissions by activated charcoal. An example of this kind of system is the facility operating at the Kolsk reactor which reduces the RIG activity in the emission gas stream by a factor of 200, a level which fully meets the safety requirements.

To purify gaseous emissions of regeneration plant one must use low-temperature rectification processes, low-temperature absorption, and also a method for selective absorption of fluorocarbons. All the residual well-known methods for RIG absorption drop from consideration, when they are evaluated for an industrial scale.

Low-temperature rectification and absorption processes offer not only efficient cleaning, but also separate out the krypton and xenon. Here it should be noted that the gaseous plant emissions from reclamation of nuclear fuel can be a raw-material source for obtaining stable xenon, enriched with xenon to a factor of about 10^4 greater than air. In a single plant, reprocessing 5 tons/day of nuclear fuel, one can obtain up to 1000 m^3/day of high-cost stable xenon, for which the requirements are continuously increasing. The utilization of stable xenon, obtained from the gaseous emissions, can compensate to some extent for the cost of creating such a plant to clean and separate the inert gases.

Among the methods for retaining ^{85}Kr, removed during purification of gases, it is profitable to consider storing it in special underground vaults (geological beds), or in tanks under pressure, and also converting it to clathrate compounds.

A problem very closely related to that of cleaning RIG is purifying the gas-air emissions from radioactive aerosols, which are formed in great quantity in reprocessing plant. In nature these aerosols may be both condensed and dispersed. The main mass of aerosols takes the form of salt mists, and also mists of acids, and the mass concentration of the dispersed substance usually amounts to tens of milligrams, and in some cases up to several grams, per 1 m^3 of gas. The basic radioactivity of the mists comes from particles of size no less than 1.0 μ.

To shield the surrounding environment and the safety zones of plant from aerosol contamination one needs a reliable continuously acting purifying system, with purification factors suitable for the allowable atmospheric conditions.

Experience of radioactive mist cleaning has shown that the most reliable method is to use fiber-type filters, which give the necessary degree of cleaning and do not require special treatment.

At present fiber self-cleaning filters have been developed and used successfully; one can subdivide them into preliminary and high-purification filters. The process of filter self-cleaning is one where liquid particles precipitating on the fibers migrate to the filter layer under the action of different forces, and are removed as a result. Thus the filter characteristics remain constant with time.

The external and internal filtration mechanisms, both for coarse and for fine fiber filters, including all the changes in the filtering layer, which occur in the presence of liquid in the filter (e. g., formation of secondary drops on ultrafine fibers), have been thoroughly investigated and described in the work of Soviet specialists. On the basis of this work, self-cleaning filters of fine glass fiber have been constructed which do not require force removal of liquid. To overcome capillary forces which oppose the efflux of liquid, the filtering layer is located vertically in the fine filters, and its height is significantly larger than the height of capillary rise of the liquid in the filter layer. Such filters are produced in different modifications (cassette-wedge and cylindrical) with filtering layer area of from 1.5 to 5.6 m^2, and are used as technical equipment for long-term application.

The fuel reprocessing and fuel element manufacture processes include operations accompanied by the formation of a large amount of dry dust, and in some cases this dust is a valuable product, which must be returned to the technical system. In recent years filters based on metallized cloth with a high efficiency for removing solid particles have been increasingly used to purify gases containing dust, and they operate very well at both low and high temperatures (up to 500°C). Metal–cloth filters have high strength and corrosion resistance, they are simple to manufacture, and, in contrast with metal–ceramic materials, have considerably less aerodynamic drag.

In the Soviet Union metal–cloth filters are widely used to extract dry radioactive dust from gaseous emissions. The efficiency of filters under self-filtering conditions, for particles with mean diameter of less than 1 μ, is 95-99%, and it is 99.5-99.9% for particles of 2.5-3 μ. The usual filtering material is a mesh of heat-treated wire (stainless steel) of two sizes: diameter 0.09/0.055 mm and 0.064/0.032 mm. The allowable dust concentration is up to 40-50 g/m^3.

Operating experience indicates that metal–cloth filters can be used satisfactorily as a first stage of gas purification for high dust concentration in the gaseous emissions.

The scientific–engineering level of development achieved at present allows us to predict confidently that a combination of existing systems for purifying gases from aerosols and dust, and complex schemes for removing volatile fission products from the gases will allow one in coming years to reliably protect the surrounding environment and the population from contamination associated with operating reactors, atomic facilities and reactor fuel reprocessing plant, and that this undoubtedly will promote the growth of nuclear energy at the predicted rate.

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PROBLEMS IN TRANSPORTING REPROCESSED NUCLEAR FUEL

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At present the total nuclear reactor power in the USSR is ~7.9 million kW. According to the main trends in the development of the industry in 1976-1980, the total nuclear reactor power in 1980 will be 13-15 million kW. In the future the rate of introduction of new nuclear reactors must increase. Basically nuclear power stations with thermal reactors will be built, and fast-neutron reactors are proposed for construction after 1985, and the rate of construction will depend on the amount of plutonium generated in the thermal reactors. The individual power levels of the main types of reactors in the period 1986-1990 will be 1-1.5 million kW. Some features of the fuel of existing and future reactors are shown in Table 1.

The reprocessed fuel of the reactors constructed in member-countries of the Council of Mutual Economic Aid (CMEA) will be transported for reprocessing to the Soviet Union. The total reactor power in these countries by 1980 will increase to 7.3 million kW, which will require a corresponding increase in the number of transport operations to handle this fuel. Analysis shows that the most convenient and economical transport is by rail, since this form of transport is widely developed in the USSR and the other member-countries of CMEA: there are rail lines at almost all the reactors; and the cost is lower than automobile transportation. A single railway train can carry the annual output of spent reactor fuel from an installed electrical power of 1 million kW.

Specially developed and constructed wagon containers (Fig. 1) are used for rail transport, in which the containers are located vertically or horizontally, depending on the size of the fuel bundles. Bundles less than 3.5 m in length are transported in vertical containers, and the others – in horizontal containers. Considerably more fuel is held in the vertical containers. The wagons have movable sections in the top, which makes it easy to load and unload. The wagon sizes are standardized. In the USSR the maximum height and width of wagons is, respectively, 5300 and 3700 mm, and in member-countries of the CMEA, it is 4650 and 3150 mm. The allowable rail loading for railways in the USSR and CMEA member-countries is 22 and 18 tons per wheel pair. Because the railway gauge in the USSR and other CMEA member-countries differs (1520 and 1435 mm) the container wagons are equipped with both pairs of wheels, which are changed at the border stations.

At present in the USSR single-layer steel containers (Table 2) are being developed and manufactured, lined internally with stainless steel. The outer surface of the container has welded to it fins and special

TABLE 1. Fuel Characteristics of USSR Power Reactors

Reactor	Electrical power, MW	Fuel charge, UO_2, tons	Av. burn-up level, MW·day/ton	No. of bundles	Bundle dimensions, S×L*, mm
VVÉR-210	210	44	13	349	144×3200
VVÉR-365	365	44	28	349	144×3200
VVÉR-440	440	44	28	349	144×3200
VVÉR-1000	1000	72	41	151	238×4665
RBMK-1000	1000	210	18,5	1693	79† ×10065

*S – "end fitting" dimension; L – bundle length.
†Bundle diameter.

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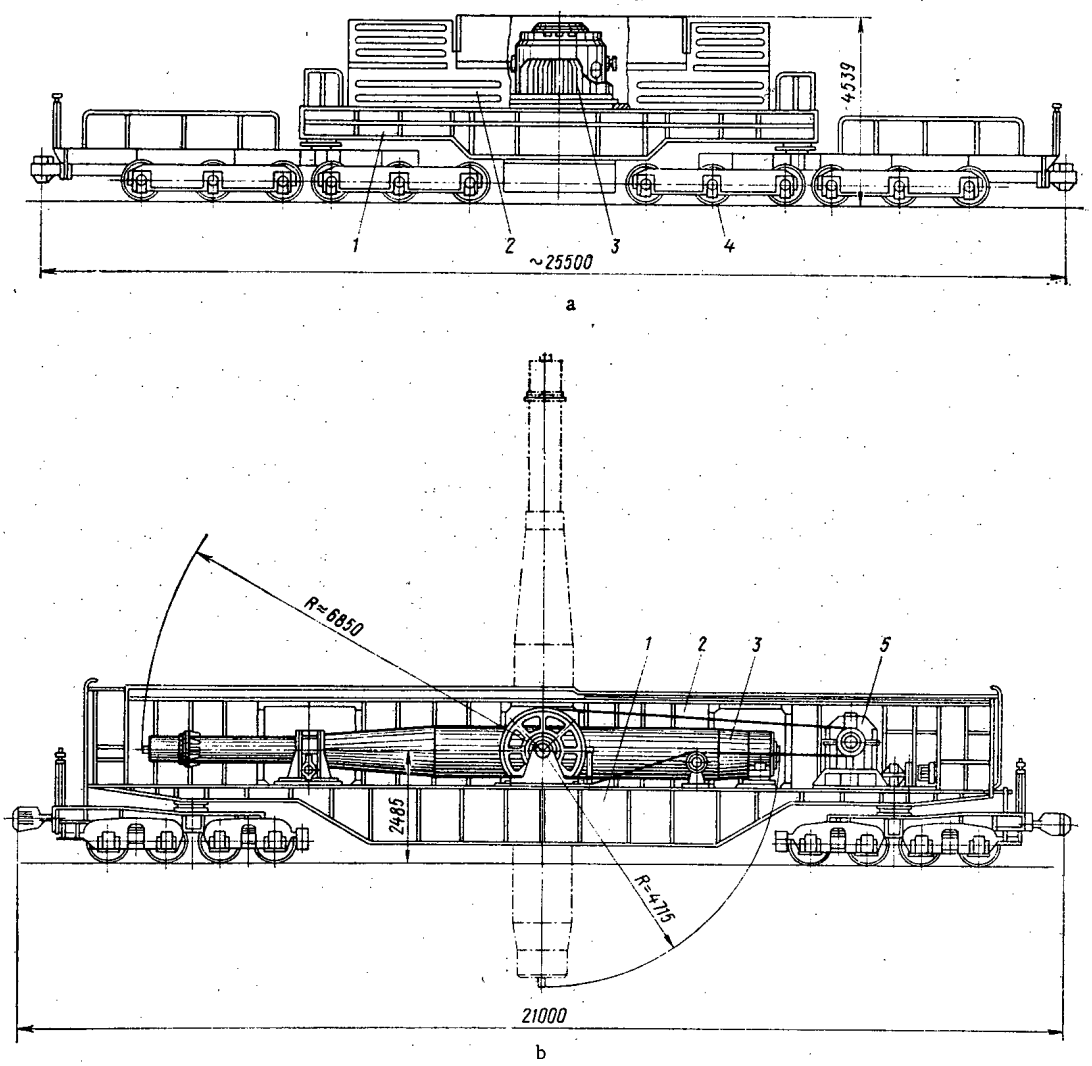

Fig. 1. Rail container wagons for transporting spent reactor fuel from VVÉR-440 (a) and RBMK-1000 reactors (b): 1) transporter; 2) case; 3) container; 4) wheel pairs; 5) mechanism for altering the container from the horizontal to the vertical position.

supports on which the container rests in the wagon (Fig. 2). In the containers there are covers, consisting of a steel envelope with a base and transverse gratings.* The number of gratings depends on the size of the covers, their structural features, and on the number of fuel elements carried in a case. The apertures in the gratings and the identations in the base provide a stable fixed location for panels in the case during loading, unloading and transportation.

In the USSR and other CMEA member-countries there have also been studies of transportation of spent fuel by mixed methods (automobile, water, rail), e.g., for the "Kozlodui" reactor in Bulgaria. It should be noted that mixed transportation is more complex and expensive, since it requires the arrangement of special transfer bases. At present, the CMEA member-countries have begun to develop the technical conditions to handle spent fuel by mixed transportation and the transport guidelines, on an informal basis.

The transportation of spent reactor fuel to the reprocessing plant is one of the most important problems in the nuclear power fuel cycle, and is the governing factor in regard to fuel element downtime. While the costs at the reprocessing plant increase only negligibly when the downtime is decreased from 3 years to 1 year, the costs in transportation are then increased by a factor of 2-3. For a 3-yr downtime there would be considerable simplification in solving a number of technical questions, e.g., possible transportation of VVÉR-440 fuel elements in a gaseous environment. For this reason, and also to account for the fact that in the next 10-15 years the amount of reprocessed fuel will be relatively small, a 3-yr downtime for spent fuel from thermal

*See Fig. 3 in the article by Arkhipovskyi et al. (At. Energ., 39, No. 1, 46 (1975)).

TABLE 2. Some Characteristics of Transportation Containers

Reactor	Form of container	Mass, tons	Dimensions, m	Thickness of steel shield, mm	Fuel mass (UO_2), tons	No. of bundles
VVÉR -440	Vertical cylinder	90	$\varnothing$ 2,3; $H=4,4$	400	3,8	30
VVÉR -1000	Horizontal cylinder	110	$\varnothing$ 2,1; $L=6,1$	410	3,0	6
RBMK -1000	Horizontal cylinder	73	$\varnothing$ 1,2; $L=11,5$	350	1,3	11
AEPP "Reinsberg" (GDR)	Vertical cylinder	80	$\varnothing$ 3; $H=4,3$	350	3,9	30
KS -150 (Czech.)	Horizontal parallelepiped	78	1,68·1,2; $L=5,4$	385	2,7 *	16

*Natural uranium.

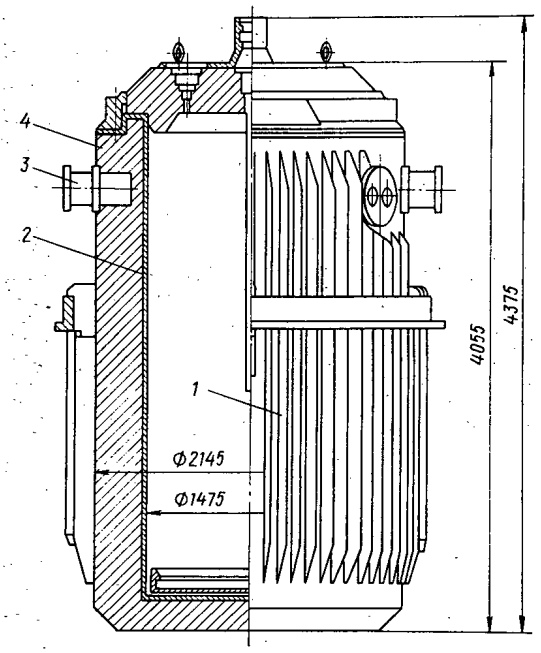

Fig. 2. Transport container for carrying VVÉR-440 spent fuel: 1) fin; 2) internal lining; 3) pivot; 4) case.

reactors is considered to be acceptable. Later the downtime must be decreased to 1 year, which would allow the fuel cycle time to be decreased. The question of downtime of fast reactor fuel is being studied.

The residual heat release and the specific γ activity of the bundles mainly determine the amount of fuel in the container and its biological shielding (Table 3). In the USSR and the CMEA member-countries it is planned to develop multilayer containers using high density materials (uranium, lead, etc.) for biological shielding. Such shielding will allow the container wall thickness to be reduced and the internal volume to be increased for the same permissible outside dimensions.

It will be possible to lower the cost of transporting spent fuel by increasing the amount of fuel transported at one time, by setting up transport systems of greater capacity; by rationalizing the transportation parameters, allowing minimal turn-around time in the transport systems, and also by unifying transport systems as much as possible; by standardizing design and developing technical recordkeeping. To a considerable extent cost reduction depends on decreasing the system turn-around time, which is added to the duration of operations such as loading, establishment of steady-state conditions, dosimetric monitoring and other types of control, off-loading, examination, maintenance, and empty trips of the transport system. Altogether, the time for actual transport of the load constitutes 10-20% of the total time that the transport system is used. An increase in turn-around can be achieved mainly by reducing the duration of nontransport operations.

A key question in solving transportation problems is unification and standardization of transport systems, the objective being to reduce expenditure in developing new types of containers, cases, etc. In the governmental standardization system in the USSR, standards have been adopted for transporting fuel which

require the use of appropriate standards in planning transport systems. For example, there are government standards for treatment, the volume of documentation, size of container wagons, etc.

Safety in transportation is ensured by observance of technical conditions, standards and rules for transporting spent nuclear fuel; in the strength and thermal calculations and in planning transport systems, in calculating nuclear and radiation safety, and also by carrying out bench tests of containers to check that they correspond to the requirements of existing standards and safety rules. In designing a container for strength, the most hazardous situation adopted is to drop the container from a height of 9 m onto a hard surface. It is assumed also that thereafter the container falls on its pivot from a height of 1 m, and then drops into a hearth with an open fire temperature of 800°C for 30 min. Container design has been basically solved, although one meets with difficulties which can be avoided by conducting tests of individual container elements, reduced-scale models, and natural full-scale test specimens. Experience shows that the greatest hazard is in dropping a container from a height of 9 m on the top corner, when the center of gravity passes through this point, and also "flatwise" on its long side and on the pivot pin.

Special attention should be paid in design and testing to welded seams, which are the most vulnerable places in accident situations, since their shock strength is low compared with that of the basic container material. Various damping devices, e.g., ribs welded to the outside of the container body, are used for shock protection in the construction of all types of containers. In choosing the number and size of the ribs, one should take into account not only the increased strength, but also the improvement in the thermal resistance of the container.

Considerable attention is given in the USSR to thermal design of transport containers. The standards and regulations specify very strict requirements as to thermal conditions of containers, both for normal operating conditions, and for emergency situations. Here two sources of heat are considered: residual heat release of the spent fuel bundles, and solar radiation. The limiting allowable temperature of the outside container wall is taken to be 82°C. At this temperature, as evidenced by accumulated experience, ~250 W of heat can be removed per 1 m^2 without the use of artificial cooling or fins on the external container surface. The design methods are based on numerous checks of heat release in models simulating fuel bundles. For future multilayer containers, heat design methods require additional refinement. Further investigations are necessary on models simulating bundles, since present and future containers differ in number, size, location and methods of mounting of the bundles.

Design for nuclear safety must be carried out by means of special methods for normal and emergency conditions.

The results of all types of design are subject to verification in test beds on models and test specimens of containers. A complex test stand includes instruments for conducting the basic tests, in accordance with existing standards and procedures (dropped from a height of 9 m and on the pivot pin from a height of 1 m; drop into an open fire for 30 min; thermal shock; the action of low temperatures, etc.). Following these tests the container is checked for sealing and for damage to the biological shielding. Hoist mechanisms on the test bench allow testing with containers of mass up to 160 tons. The same stand can be used to check specific elements of container structure. The instrumentation of the stand includes a number of laboratories, e. g.,

TABLE 3. Residual Heat Release of Spent Fuel Bundles and Their Specific γ Activity, as a Function of Storage Time

Reactor	Residual heat release of bundle after storage, kW per bundle				Specific γ activity, 10^3 g-eqv. Ra, per bundle			
	storage time, years				storage time, years			
	0,5	1	2	3	0,5	1	2	3
VVÉR-210	1,2	0,7	0,3	0,2	37,0	10,0	4,4	3,5
VVÉR-365	2,1	1,1	0,5	0,3	64,0	18,0	7,5	6,0
VVÉR-440	2,2	1,2	0,6	0,3	67,0	19,0	7,9	6,3
VVÉR-1000	11,1	6,0	2,8	1,7	340,0	96,0	40,0	32,0
RBMK-1000	1,0	0,6	0,3	0,2	28,0	8,6	3,5	2,9
AÉPP "Reinsberg" (GDR)	1,1	0,6	0,3	0,2	34,0	9,7	4,0	3,2

metallographic, radiochemical, etc. By combining transport container design and verification on a test stand one can guarantee reliable and safe transportation.

The transport of spent fuel is constantly linked to the question of deactivation. In the loading/off-loading operations the surface of the transport systems can be contaminated by radioactive materials. The contamination may show up on both internal and external surfaces of transport containers, cases, and wagons. Deactivation methods include fixed systems for washing or individual deactivation devices. Good results are obtained with deactivation by jets and steam-ejection methods, using acid and alkali solutions. Depending on the kind of contamination, the deactivation coefficient per cycle of solutions is from 10 to 100. By reducing the contamination of transport systems as much as possible one can minimize the cost of deactivation, and therefore, reduce the cost of transporting spent fuel.

BOOK REVIEWS

Yu. A. Surkov

GAMMA SPECTROMETRY IN SPACE INVESTIGATIONS*

Reviewed by Yu. V. Sivintsev

The well-known successes in space investigations, particularly in the study of the material composition of the moon and planets of the solar system, would have been impossible without the use of γ-ray spectrometry methods and instruments. This book is the first, and undoubtedly a successful, attempt to correlate material on the use of γ-spectrometry in investigating the moon and planets using spacecraft, and also in the laboratory study of extraterrestrial material, mainly lunar soil returned to earth. The author, who has been studying this subject for many years, has brought together a great deal of material, which, along with the work of other scientists, including Americans, forms the basis of this monograph.

The structure of the book is clear and lucid. It contains three chapters. The first correlates data on radioactivity of bodies of the solar system. It presents information on the potassium–uranium system of natural radioactive elements in very widely distributed earth rocks, on the moon, other planets and meteorites (Figs. 13-16). Of special interest is a comparison of measured results in lunar specimens returned to earth by Soviet and American spacecraft. It should be noted, incidentally, that the excessive compression of the information in this chapter makes it difficult sometimes for the unprepared reader to understand. The special terminology (e.g., regolite, brecchia, chondrite, and many other terms) are used excessively without explanation.

The second chapter describes γ-spectrometry apparatus for investigating extraterrestrial material, both from on board spacecraft, and also under laboratory conditions. Here there is a detailed description of γ-quanta detectors (scintillation units with various scintillators, single-crystal and composite semiconductor detectors), methods and apparatus for activation analysis, functional and electrical diagrams of onboard γ spectrometers, including those used on spacecraft of the Luna, Mars, and Venera series, as well as laboratory apparatus of ultrahigh sensitivity.

The topic of the third chapter, which comprises almost one-half of the book, is γ-spectrometric investigations of the moon, Mars, and Venus by means of spacecraft. This section containing a great deal of factual and generously illustrated material, is written in a lively style and makes an especially strong impression. Unfortunately, some of the detailed information which it should present is missing. For example, important data on the γ-quanta counting rates from decay of natural radionuclides contained in lunar and earth soil (Table 24, p. 157), and on the content of uranium, thorium, and potassium in the soil of Venus and the earth (Table 28, p. 212) should have been supplemented by values of the errors, and the graphical material of Fig. 86 (p. 171) on correlation of natural radioactivity levels with the lunar surface relief should be supplemented by numerical values of the correlation coefficient.

The book also contains other defects, carelessness in terminology (e.g., on p. 84 the term "intensity" should more appropriately be "activity," and on p. 123 "counting rate" has been called "count;" in some cases the unsuitable term "monochromatic" has been used instead of "monoenergetic" radiation, etc.), and there are frequent jargon expressions, and examples of inappropriate word order which obscure the meaning of a phrase. In Figs. 4, 22, and 102 the captions are incomplete or have not been retouched quite successfully.

These defects cannot alter the general high value of the monograph. The Atomizdat publishers have made an outstanding contribution to specialists in the fields of space and applied nuclear physics, by publishing Yu. A. Surkov's interesting book. A new initiative of the publishers merits approval, the book jacket gives a brief account of the scientific interests of the book's author, a well-known Soviet scientist in the space research field.

In conclusion, this review would not be complete without a mention of the splendid jacket by artist G. A. Zhegin.

*Atomizdat, Moscow (1977), 240 pp., price 2 rubles 04 kopecks.

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DEVELOPMENT OF METHODS OF SOLIDIFICATION AND BURIAL OF RADIOACTIVE WASTE FROM FUEL CYCLE

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In all countries with an atomic industry it has been acknowledged that the safest method of "perpetual" burial of liquid waste with both a high and intermediate level of activity is to bury it in a solidified form and, especially in the case of high-activity waste, in the form of vitreous materials. This form of burial requires less maintenance, is safer for the environment, and, according to our calculations, is more economical. Technological processes are being developed in the USSR for obtaining phosphate [1, 2] and silicate [3, 4] vitreous materials. Two of the apparatus and technological flowsheets under development are discussed in the paper; a two-stage process with a crucible for one-time use and a one-stage process without precalcination.

Solidification of High-Activity Waste

Two-Stage Process [5, 6]. This process of vitrification of radioactive waste was developed by employing calcination in an air-fluidized bed in the first stage, followed by fusion of the calcine in a concrete crucible

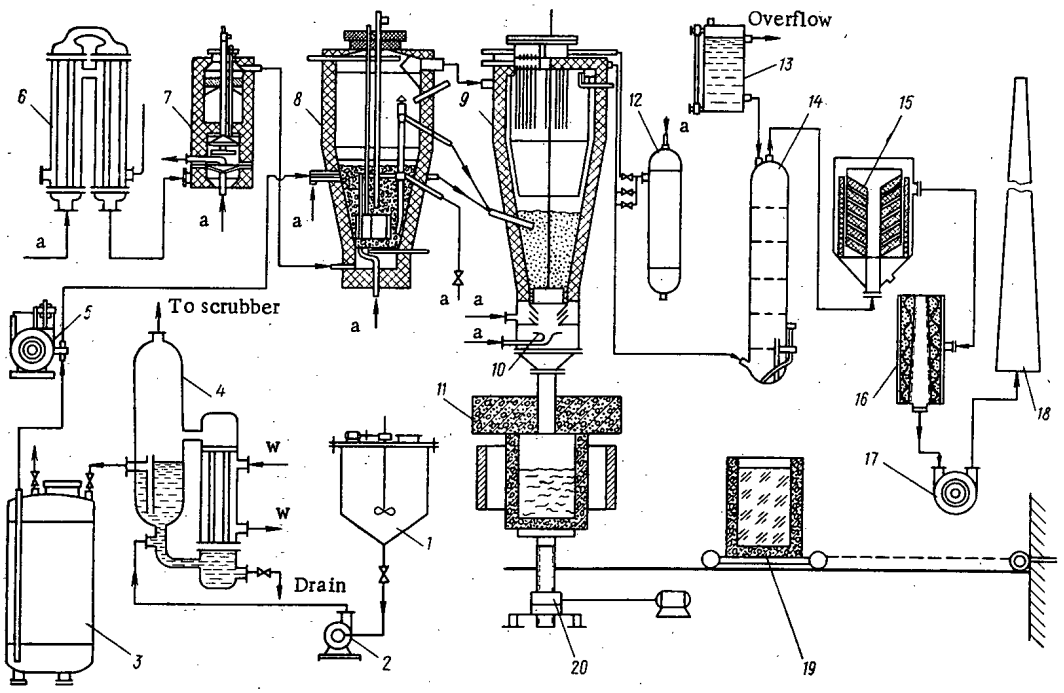

Fig. 1. Flowsheet of the KS-KT-100 installation; 1) monte-jus; 2) pump; 3) reservoir of solution to be evaporated; 4) evaporator; 5) metering pump; 6) tubular heat exchanger; 7) radiator; 8) dryer; 9) MKF filter; 10) pneumatic feed table; 11) crucicible; 12) receiver; 13) tank; 14) bubbler-absorption column; 15, 16) coarse and fine filters; 17) vacuum pump; 18) smokestack; 19) crucible on trolley; 20) hoist; a) air; w) water.

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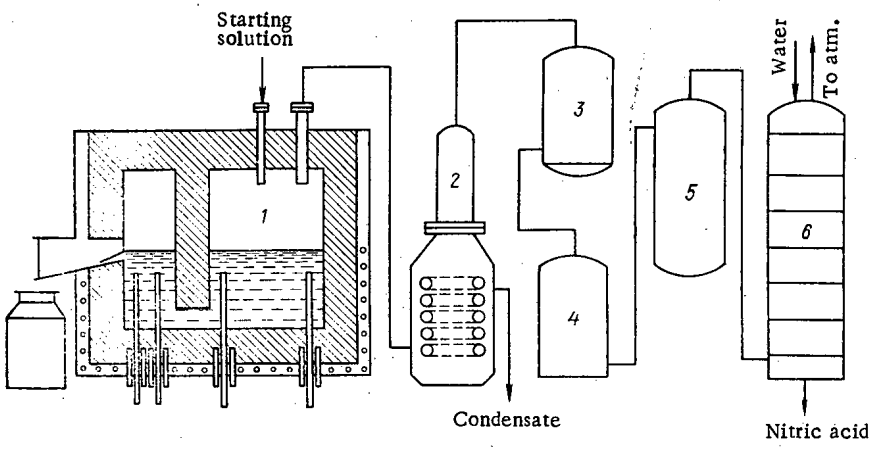

Fig. 2. Flowsheet of waste-vitrifying installation; 1) electric furnace; 2) bubbler-condenser; 3, 4) coarse and fine filters; 5) column with pyrolusite; 6) absorption column.

(second stage). An enlarged KS-KT-100 installation (Fig. 1), which has been developed and tested, comprises units for preparation of the solution, evaporation, calcination, fusion, and gas scrubbing. The starting solution has an aluminum nitrate content of 240 g/liter, sodium nitrate 125 g/liter, orthophosphoric acid 120-130 g/liter, and molasses 90-150 g/liter, the latter for imitating the fluxing of nitrate waste.

Tests showed that the units and apparatus operated satisfactorily. The technological parameters were found for the processes of evaporation, calcination of the solution, and fusion of the calcine. The presence of molasses in the solution (up to 150 g/liter) allowed the solution to be concentrated in the evaporator, to decompose part of the nitrates, and to prevent the formation of ruthenium tetroxide. At a calcination temperature of 350-400°C and a fluidization rate of 1.5 m/sec the capacity of the installation of the starting solution was 100 liters/h, for the evaporated solution 65 liters/h, and for glass 20 kg/h. The efficiency of the gas-scrubbing system lay within the limits 10^7-10^9.

Testing of the apparatus and technological system demonstrated stable operation of the apparatus with an air-fluidized bed of a heat-transfer agent for treating solutions previously fluxed with phosphoric acid.

No difficulties were encountered in obtaining silicate glasses by calcining solutions with no sodium nitrate content [7]. The use of one-time crucibles eases the requirement for the corrosion resistance of the metal and permits use of high-frequency heating which is convenient for remote shaping. Considerable difficulty is presented, however, in putting together apparatus for the unit for dosing powdered high-activity material from the calcination unit into the crucible.

Waste Vitrification without Precalcination [8]. To simplify the apparatus flowsheet, to ensure a high output with uniform heating of the glass mass, and to increase the operating reliability of the main apparatuses a method of vitrification in electric furnaces is being developed and tested in trials; in this method the glass mass is heated by the passage of an alternating current through the melt. In contrast to the process mentioned earlier, a continuous process of fluidization, calcination, and vitrification takes place in a single apparatus (Fig. 2).

Tests were run with a test solution containing sodium nitrate (130-250 g/liter) and aluminum nitrate (150-300 g/liter). When phosphate glass was obtained phosphoric acid was added to the solution in sufficient quantity to yield a high-grade glass mass. Glass melt was present in the electric furnace continuously for two years with no interruption in the electricity supply. The solution feed was interrupted for short periods necessary for improvements in the feed system for the solution and the reducing agent, analysis of the scrubber performance, changes in the composition of the solution, and verification of the monitoring and measuring instrumentation. In the trial operation 1000 m^3 of the test solution were processed and 250 tons (100 m^3) of phosphate glass were obtained. The behavior of cesium, strontium, and ruthenium during vitrification was studied and it was shown that the minimum carryover of radionuclides is observed when molasses is introduced into the test solution: cesium 0.3%, strontium 0.1%, and ruthenium 4.5%. The trials demonstrated the feasibility of remote monitoring of the course of the technological process and control of the process by means of automatic controls and monitoring and control instruments. The system consisting of elastic furnace and gas scrubbers can be recommended for trials with actual high-activity waste.

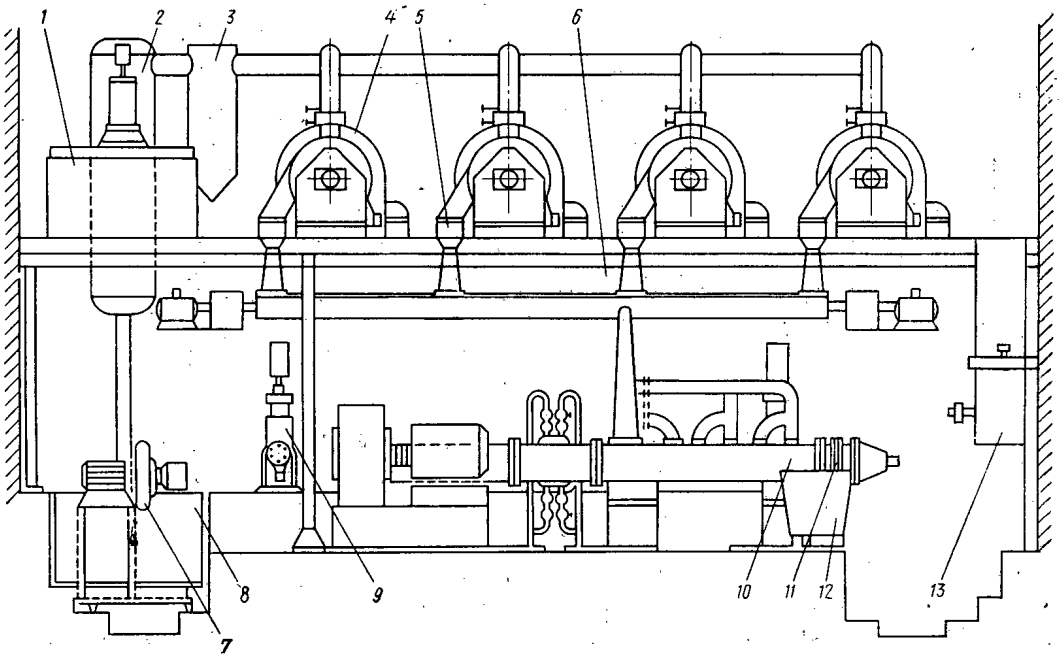

Fig. 3. Schematic diagram of UBD-200 installation: 1) intermediate reservoir; 2) heat exchanger; 3) dust catcher; 4) dryer; 5) screw conveyor for discharging salt from dryer; 6) screw conveyor for feeding salt into mixer; 7) ventilator; 8) condensate collector; 9) circulating pump for heat-transfer agent; 10) mixer; 11) compound extrusion; 12) shaping trolley; 13) bitumen storage.

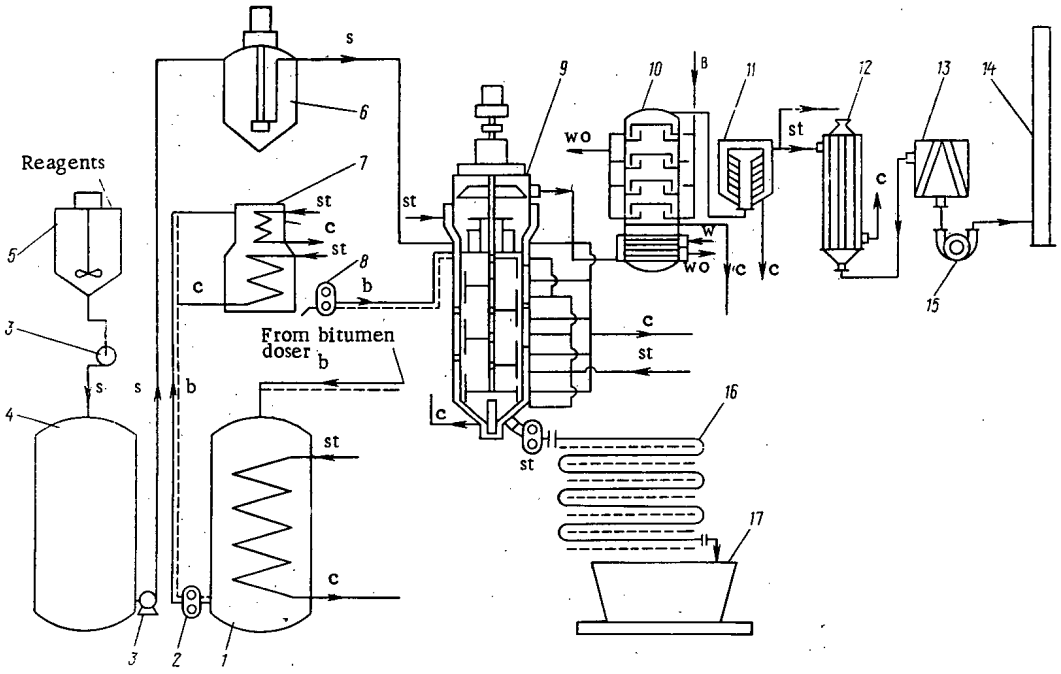

Fig. 4. Installation for bituminizing liquid waste; 1, 4) containers for storing bitumen and solution; 2, 3) bitumen and centrifugal pumps; 5) solution-preparation units; 6) submersible pump; 7) bitumen-dosing reservoir; 8) bitumen-metering pump; 9) RB-1000-12 bitumenator; 10) bubbling column; 11, 13) coarse and fine filters; 12) heat exchangers; 14) discharge tube; 15) vacuum pump; 16) piping for transporting bitumen mixture; 17) container for bitumen mixture; w) water; wo) water overflow; st) steam; c) condenser; s) solution; b) bitumen.

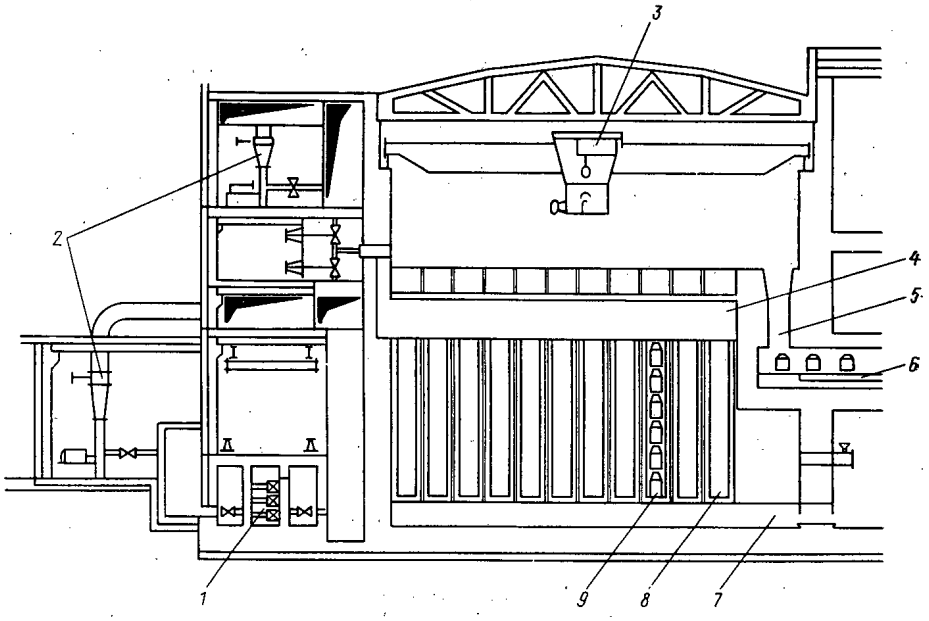

Fig. 5. Air-cooled storage vault for vitrified waste (cross section): 1) filtration station; 2) ventilator; 3) remote-controlled crane; 4) conduits for exhausting air from storage vault; 5) reloading shaft; 6) conveyor; 7) air intake conduits; 8) reinforced concrete wall; 9) container of vitrified mass.

Bitumenization of Radioactive Waste

For safe burial of liquid waste with an intermediate level of activity a method has been developed in the Soviet Union for solidifying it by bitumenization instead of storing it in liquid form in stainless steel containers. The process was developed with a solution with a composition that is most typical of most kinds of intermediate-activity waste: sodium nitrate 500 g/liter; ferrous hydroxide 100 g/liter; calcium sulfate and sulfanol, 25 g/liter each; ion-exchange resin 50 g/liter; and pH of 10-12.

Various grades of bitumens with softening temperatures of 33 to 53°C and penetration of 220-42 (0.1 mm) were tested.

Testing of the prepared bitumen compounds showed them to have adequate moisture resistance (10^{-4}-10^{-5} g/cm^2 · day). From the point of view of radiation resistance the bitumens tested can be used to process liquid waste with a specific activity of at least up to 10 Ci/liter without any significant effect on the water resistance, leachability, and changes in the structure of the bitumen compounds during long storage [9].

Research is under way on improving the properties of the binders and increasing the percentage of inclusions of salts by replacing the bitumen or by introducing additives to it. Good results have been obtained by using asphalt from the deasphalting of tar. When the content of salt filler in this product was increased from 40 to 60% the water resistance of the ready compound was not affected: the filler elution curves differ only as to the starting points. Testing of this material for radiation resistance at a dose rate of $4.3 \cdot 10^4$ rd/h and with a total dose of $1.9 \cdot 10^8$ rd showed that under these conditions there is no increase in the volume of the sample. An advantage of this asphalt is its low cost (it costs one-third as much as bitumen). To obtain soft bitumens, suitable for holding radioactive waste, a variety of diluents for solid bitumens (e.g., BN-4), have been tested, viz., solar oil, fuel oil, green oil, etc., thus considerably extending the base of raw materials for waste solidification [10].

In 1974-1975 the UBD-200 high-output, continuously operating, two-stage installation was built on the basis of standard industrial apparatus [11]. In the first stage, moist salts are obtained in electrically heated roller dryers; these salts are then fed into mixing worms where they are heated gently by a high-temperature organic heat-transfer agent (ditolyl methane), after which they are fed into molten bitumen (Fig. 3). Some alterations were made in the industrial equipment used to ensure airtightness, this pertaining mainly to the unit in which the salts were stripped by knives. In trials the output of the installation was 200 liters/h; a further increase in the output can be attained by setting up more roller dryers. Investigations have shown that splitting the bitumenization process into two stages does not affect the chemical stability of the compounds

obtained. In 1976 the UBD-200 was put into commercial operation; the results will be utilized in the design of atom installations for atomic power plants.

A promising design of an apparatus for bitumenizing liquid radioactive waste is the rotor bitumenator with an abradable film [12]; in it, water is evaporated from a thin film of a mixture of liquid waste and bitumen, flowing down the heated cylindrical wall of the apparatus. Mixing in the film is effected by scrapers mounted on a rotating vertical rotor; the scrapers intensify the processes of heat and mass exchange and simultaneously clean the heating surface. In the lower part of the bitumenator the bitumen–salt mixture is mixed by an anchor stirrer and pumped by a gear pump into containers for burial (Fig. 4). The technical documentation has been drawn up for two types and sizes of bitumenization installations with an output of 180 and 500 liters/h with respect to evaporated water [12].

Storage and Burial of Solidified Waste

Storage Vaults for Vitrified Highly Active Waste. Along with the development of methods for the vitrification of highly active liquid waste, a search is under way in the Soviet Union for methods for reliable and safe storage and burial of such waste.

The following requirements should be taken into account in the development of storage-vault designs: a storage vault is situated on the surface of the earth above the groundwater level, adjoins a waste-vitrification plant, and is connected to it by a transport corridor; containers with vitrified waste are delivered to the storage vault by a remote-controlled crane; as the amount of heat released in the vitrified blocks falls off, section by section the storage vault is converted to a burial ground regime but the possibility of removing reservoirs is preserved.

The storage vault consists of concrete sections (Fig. 5) inside which pipes are arranged with a certain spacing for holding containers with vitrified waste. Each section is designed to be filled within 1 year. The containers are loaded into the wells through hatches which are closed with concrete covers.

Two systems for delivering air to cool the containers were considered. In the first the air is blown through the annular gap between the container and the inner surface of the well; the air, heated to 150°C, is collected in the upper conduits running above the wells and, after purification in filters, is discharged into the atmosphere. In the second system the cooling air flows around the outer walls of the enclosing pipes.

In the first period of operation of the storage vault the air is circulated by means of ventilators in both systems; once the heat release from the waste has dropped it is possible to convert to natural convection (by using an exhaust vent pipe). This conversion can be carried out more quickly for the second system since in this case the heated air does not require filtration [13].

Burial of Radioactive Bitumen Compounds. To choose the optimal conditions for the final burial of radioactive bitumens a study was made of their behavior on a proving ground, in particular, in burial in clayey soil in experimental burial grounds (trenches 2 m deep furnished with a special pipe for sampling water in contact with the blocks). Bitumen materials were buried in the burial ground in the form of blocks weighing 1-2 tons and possessing an activity of $1 \cdot 10^{22}$ to $9 \cdot 10^{-2}$ Ci.

The blocks were prepared with various bitumens containing 27-60% salts. For a long time samples were taken of the water in contact with the blocks. In a period of 900 days the specific activity of the water changed over quite narrow limits; the maximum value was $3 \cdot 10^{-9}$ Ci/liter and the minimum value was $1 \cdot 10^{-10}$ Ci/liter. The greatest elution of activity from the blocks was observed for bitumen material based on BN-4 bitumen and the least, for material based on BN-2 bitumen. For all bitumen materials, however, the eluability of radionuclides proved to be quite low. The low leachability of radionuclides from bitumen materials and the good sorptive properties of clayey soils were responsible for the extremely slow downward spread of activity. When boreholes were sunk in the burial ground 2 years after burial of two blocks with a total activity of $1.5 \cdot 10^{-2}$ Ci the core samples displayed no anomalous activity. The normal background of the formation was established 15 cm from the bitumen blocks.

Thus, the low eluability of radionuclides, established during proving-ground storage of large bitumen blocks obtained by promising concentrates from decontamination plants, opens up the possibility of storing bitumen materials with a specific activity of up to 1 Ci/kg (with an eluability rate of 10^{-4}–10^{-5} g/cm^2 · day) and up to 10 Ci/kg (with an eluability rate of 10^{-6} g/cm^2 · day or less) right in the ground.

Technicoeconomic Comparison. A technicoeconomic evaluation of the vitrification method in comparison with the storage of high-activity liquid waste in containers was carried out for an installation for vitrification of discharge solutions at the rate of 200 liters/h, which corresponds to a conventional plant with a 5-tons/day output of nuclear fuel from water-moderated–water-cooled reactors [14]. Comparison showed that the cost of vitrifying 1 m^3 of high-activity waste is 12% less than that of storing the waste in containers and the reduced capital outlay is less by a factor of 2.2. With respect to capital outlay and annual operating costs, waste processing by the bitumenization method is also more economical than storage of liquid radioactive concentrates in containers [15]. The capital outlay for bitumenization is almost twofold less than for storage in containers.

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PRINCIPAL PREREQUISITES AND PRACTICE OF USING DEEP AQUIFERS FOR BURIAL OF LIQUID RADIOACTIVE WASTES

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Solving the problem of the reliable and safe burial of radioactive waste, especially high-activity waste, is an important and pressing task confronting the nuclear industry today [1, 2]. Bearing in mind the scale and rate of development of nuclear power and construction of plants for the regeneration of spent nuclear fuel, one must deem it extremely urgent to have international cooperation in the search for acceptable methods for the final burial of radioactive waste, methods which would preclude present and future deterious effects of radioactivity on the biosphere [3].

Underground burial of radioactive liquid waste in deep-lying aquifers, reliably isolated from other horizons and from the surface of the ground, has been studied in detail on the experimental and pilot scale for a number of years in the Soviet Union [1, 2, 4, 5] with due regard for the legislation in force in the domain of environment protection and rational use of natural resources [6].

The principal scientific developments and the results of underground burial of low- and intermediate-activity waste were considered in [7, 8]. The present paper discusses some ideas on the substantiation for and organization of underground burial in deep-lying aquifers as well as the first results of extended field tests on the burial of liquid waste.

Principal Prerequisites and Organization of Underground Burial of Radioactive Liquid Waste. Underground burial of such waste is a complex problem whose solution, which is specific to each atomic plant, requires a detailed study of the geological structure and hydrogeological conditions of the region as well as physicochemical, scientific-technical, and sanitary and hygienic investigations. Only a combination of information on these topics will make it possible to substantiate the feasibility and safety of underground burial in one region or another.

First, on the basis of the latest data on geological structure, tectonics, hydrogeology, and useful minerals, organizations of the Ministry of Geology of the USSR provide a regional appraisal of the closed and half closed geological structures (lithosphere "pockets" [9]) containing groundwater not suitable for economic use and securely isolated from the present drainage system and underground water suitable for water supplies or balneological or industrial purposes. Such a preliminary appraisal serves as a basis for the elaboration of a program and methods for the detailed study of the region proposed as a site for an underground storage vault in a deep-lying aquifer.

To bury radioactive liquid waste reliably and safely is taken to mean putting the waste in the interior of the earth in a way so as to eliminate the possibility of the waste coming to the surface of the earth, entering rivers, lakes, and seas, contaminating fresh underground water of value for economic and drinking purposes, mineral water of medical and industrial value, as well as contamination of deposits of useful minerals which are being exploited or hold out promise for development. Therefore, for many regions underground burial is an effective means of averting contamination of usable natural water.

According to the legislation in force [6], underground burial of highly toxic industrial effluents is considered as one way of using the interior of the earth along with the exploitation of deposits of useful minerals, construction of mining enterprises, construction and operation of diverse underground facilities, and underground storage of oil, gas, and other products.

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The sanitary-radiation safety of underground burial of radioactive liquid waste is ensured by the use of a geological structure which should reliably contain the waste within the limits of a collector-stratum for the entire period of decay of the radioactive elements to the maximum permissible concentrations. Therefore, the principal criterion of reliability should be the existence of thick recurrent impervious beds in the cross section, especially in the superface of the collector-stratum.

The collector-stratum in which underground storage vaults are made for such waste should meet the following basic requirements:

lie in a stagnant zone or a zone of slow water exchange;

have a considerable extension in area, thickness, porosity, penetrability, ensuring disposal (absorption) of waste of the calculated volume;

be confined by regionally recurrent water-impervious overlying (and underlying) rock eliminating any hydraulic connection between the collector-stratum and other horizons;

not be used as a source of water supply and not contain useful minerals whose development might be effected by a constructed underground storage vault;

lie beyond the limits of regions of tectonically active regions.

The most favorable in this respect are horizons of large artesian basis of the platform type, lying in a stagnant zone and containing highly mineralized water. With respect to structural conditions for underground burial of liquid waste with a lower density than that of the stratal water, it is advisable to use anticlinal domes and to use synclinal and closed monoclinal structures in the case of higher waste densities.

The decision as to the feasibility of underground burial of radioactive liquid waste is made with account for the results of physicochemical studies which determine the behavior of the waste and its radionuclides under the geological and hydrogeological conditions of the region and predict the rate and possible paths of their propagation in the interior of the earth. Prior to being injected into the earth, the waste as a rule goes through a stage of preparation ensuring long stable operation of the injection wells.

The feasibility of establishing underground storage vaults in deep-lying aquifers is in each case verified by calculation of the reliability and sanitary and hygienic safety. For this purpose an appraisal is made of possible paths of migration by radionuclides in the earth's interior: propagation along the collector-stratum and diffusion through the overlying water-impervious rock. An estimate is also made of the dilution of the waste in the collector-stratum and the sorption of radionuclides by rock formations.

The construction and technological flowsheets of underground and surface facilities of storage vaults are substantiated and designed only from data of detailed geological-surveying work and physicochemical and sanitary-hydrogeological studies. The injection systems designed are closed systems using airtight shutoff and control fixtures which control the instruments and glandless pumps. Such design conditions guarantee sanitary safety in the operation of all facilities of the underground storage vault.

The technological flowsheet for underground burial permits a high degree of automation and monitoring of the principal processes and parameters by means of hydrogeological, industrial-geophysical and physicochemical methods. Once they have performed their function, the wells are eliminated by using high-quality plug-back cements and other reagents ensuring the sealing and isolation of all opened aquifers.

The underground burial method is characterized by the following positive factors:

the useful volume of deep-lying aquifers (collector-strata) in which radioactive liquid wastes are disposed measures millions of cubic meters and, therefore, the radioactive wastes are contained in the earth's interior within a small area for the duration of the operating life of the storage vault (20-30 years);

the slow movement of the wastes over the collector-stratum permits the organization of a system for monitoring the spread of the wastes;

the rock formations of the collector-stratum possess a sorption capacity for practically all radionuclides and as a consequence their spread over the stratum is constrained; the front of toxic and long-lived nuclides lags considerably behind the front of motion of the liquid phase of the wastes;

there is a sharp reduction in, or even complete elimination of, the contamination of open bodies of water and the earth's surface as well as aquifers used for economic, drinking, or industrial purposes;

with favorable geological and hydrogeological conditions there is no dependence on the climatic characteristics of the region;

storage vaults (underground and above-ground) are relatively simple and reliable to operate; the principal technological processes involved in waste burial are easily subjected to automation and telemechanical remote control;

in establishing an underground storage vault, a small area (20-30 ha), as a rule, is used for the construction of the main surface facilities located in the first belt of the sanitary safety zone; no limitations are imposed on the use of land in the second and third belts [10];

with respect to technicoeconomic indicators, underground burial surpasses other methods of radioactive waste decontamination, storage, and disposal.

Studies and calculations [5] have shown that this method is becoming increasingly economical, even when the volume of processed waste is 50-100 m^3/day and the depth of burial is 1000-1500 m, in comparison with such purification methods as concentration by evaporation, ion exchange, electrodialysis, and combined chemical methods of purification. It must be taken into account, however, that when high-activity waste is buried there is a limit on the specific content of activity under the conditions of the stratum, this being due to the maximum permissible temperature of the medium which should not exceed the boiling point of the liquid in the stratum.

Principal Requirements Imposed upon Waste to Be Buried

Large-volume wastes should be neutralized and should have the pH value of a medium close to the pH of the stratal water. For a large number of storage vaults we shall assume a pH interval of 6-9. The pH interval can be extended if the volume is limited and the disposal is short-term. The pH range may also be wider for solutions possessing the properties of buffer mixtures as well as stabilized complexones.

Components that form sediments under stratal conditions should be eliminated from the solutions or converted into a form which is stable in the liquid phase. This condition applies to cations which hydrolyze under the pH conditions of the stratum and to ions which enter into reactions with components of the stratal water and form sediments. This requirement refers primarily to corrosion products or components of deactivation waste. If the sediment-forming components are stabilized, their content is not limited.

The overall salinity of wastes is allowed to reach concentrations close to saturated solutions under the thermal conditions of the collector-stratum. There are definite limits on the content and dispersity of suspended matter, limiting the total slurry capacity of the face zone of the injection wells and their planned service life.

Low-activity discharges (reactor, washing, deactivation, etc. discharges) contain a large quantity of surface-active and complexing compounds. The liquids have a high oxidizability, volume-stabilized suspended and emulsified substances, and a low salinity (1-10 g/liter). The compatibility of such discharges with the stratal conditions is ensured by neutralization (bringing the pH to the working range) and clarification prior to burial.

Two variants, interrelated by apparatus design, have been adopted for the technological preparation of such discharges and in various stages of the burial these variants are simplified by the exclusion of one preparation unit or other from the operation: coagulation and filtration on mechanical filters (the coagulants used are salts of iron, calcium, and magnesium, as well as components of the discharges themselves); with a neutralizing tank for holding the solutions for a long time and diluting the detergents, preliminary coagulation is eliminated, which is an economically advantageous move.

Intermediate-activity waste (saline technological solutions, mainly from deactivation of technological equipment, and low-volume waste from the fuel-regenerating technology) has a high salinity and high contents of corrosion and radiolysis products, as well as displaying chemical activity with respect to the stratum. When such waste interacts with the rock formations of the stratum, aluminum–silicon compounds appear in the composition of the waste and the waste may coagulate in the pore space during dilution. Moreover, because of the sorptive properties, salts accumulate on the surface of the rock formations and the surface activity of the minerals changes. The technological preparation of intermediate-activity waste envisages stabilization of hydrolyzing components, correction of the pH of the medium, maintenance of the necessary salinity level, as well as a particular order in which they are mixed and injected into the wells.

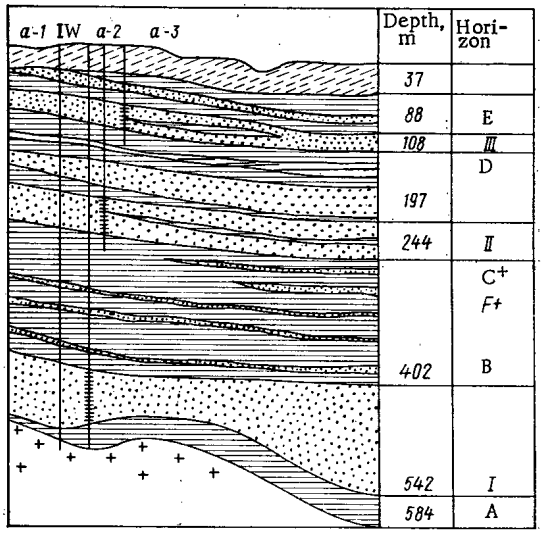

Fig. 1. Geological section: A) argillite and clay; B) clay series with interlayers and lenses of sand, sandstone, and aleurolite; C) clay with calcareous aleurolite, argillaceous aleurite, sand, and sandstone; D) clay with sand interlayers; E) sandy loam passing into loam; IW) injection well; *a*-1, *a*-2, *a*-3) observation wells.

Pilot-scale burial of such waste, carried out in the USSR for more than 13 years, has confirmed the study predictions of the behavior of all components of wastes under the conditions of storage-strata in porous horizons and has made it possible to elaborate under natural conditions systems for monitoring and controlling the circuit for spreading the waste in the stratum. Throughout the entire period of the burial no emergency situations have been detected.

High-Activity Waste. Many years of work with intermediate-activity waste formed the basis of studies on the feasilibity of creating storage-vaults for high-activity waste in a collector-stratum. The investigations [8, 11] were aimed at ensuring the stability of the liquid phase during the process of waste disposal (well intake capacity and uniform distribution of liquid waste in the stratum); establishing the permissible level of specific radioactivity in the stratum for normal heat extraction; developing techniques which would keep the maximum level of specific radioactivity in the stratum from being exceeded at the given conditions; and studying the processes of radiolysis in the heterogeneous "waste-rock" medium and the process of fixation of radionuclides in the solid phase. It was found that in the burial of high-activity waste the stability of the liquid phase in the injection process and, therefore, during preparation for burial can be ensured either by prewashing the well with weakly acidic solutions and creating a transient zone between the stratal water and the waste or by introducing additional reagents (complexes) into the waste and adjusting the pH of the medium.

Build-up of radionuclides can be controlled by diluting the waste, introducing complexes into the waste, or prewashing the well bore with solutions which alter the surface activity of the rock formations (complexones, surface-active substances, etc.). Radionuclide fixation in the rock formation will occur under the effect of ionizing radiation on their sorption parameters in the electrolyte medium. As the absorbed dose increases desorption will become less likely and, therefore, so will migration owing to the structural changes in the primary minerals. This applies especially to strontium and cesium radionuclides which become fixed in rock.

Thus, reliable fixation of radionuclides in the solid phase takes place during burial of high-activity liquid solutions.

Field Trials on the Burial of High-Activity Waste

Trials on the burial of high-activity waste in deep-lying absorptive horizons began in 1972 after geological surveying in the area and experimental investigations. The primary objective of this study has been to develop methods of preparing waste for underground burial; to study the conditions for the transportation and

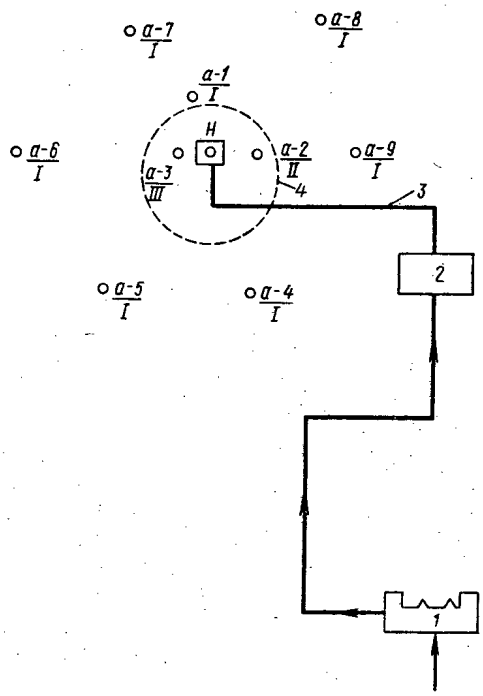

Fig. 2. Flowsheet of test section; 1) installation for preparation of waste prior to burial and for transferring the waste into tanks; 2) pumping station for injection; 3) closed delivery circuits; 4) circuit for distributing waste; O) wells.

injection of high-activity waste into the ground; to check the reliability of wells for various purposes, shutoff and control equipment and monitoring and control instruments in the automatic controls; to study the efficiency of different industrial-geophysical methods of monitoring the burial process; and under field conditions to verify the theoretical and experimental data.

The Geological and Hydrogeological Conditions of the Section. The test section was located in a region of submontane deposits formed on the very edge of a sedimentation basin.

The results of geological surveying on this site confirmed the presence of a considerable (up to 500 m) series of friable deposits (Fig. 1). The collector-stratum chosen for trials on the burial of high-activity waste was an aquifer I (sand–grit patch with interlayers and lenses of clay, aleurolite, and sandstone) in the central part of the depression of a Precambrian base filled with a thick series of Jurassic deposits. The depth of the roof of horizon I within the confines of the test section was 350-410 m, the total thickness was 55-85 m, and the average effective thickness was 25-35 m. In the central part of the depression, where the test section was located, clays of the impervious bed A underlie the collector-stratum. The collector-stratum is separated from the overlying aquifer II (fine- to coarse-grained sand) by the 150-170 m of aleurolite and clay series of the impervious bed B. Because of lithological screening along the periphery horizon I is a comparatively closed water-bearing system.

The collector-stratum consists of various-grained sands, aleurolites with subordinate interlayers and lenses of conglomerates, breccias, gritstones, and clays. The facies inhomogeneity also resulted in its filtration inhomogeneity: the maximum conductivity (25-36 m^2/day) is observed in the central part of the depression; with distance from the periphery the conductivity of the rock formations gradually deteriorates and drops off to 0.6-3.3 m^2/day at the boundaries of the region under consideration. The filtration coefficient of the individual permeable interlayers varies from 0.2 to 1.4 m/day, depending on the content of pulverized and argillaceous fractions. The coefficient of piezoconductivity is more sustained over the area and averages $1.6 \cdot 10^5$ m^2/day. The average effective porosity of the rock formations of horizon I is 7-10%.

According to data from hydrogeological studies (pumping and injection of water), horizon I is reliably isolated from aquifers II and III (sand with loam lenses), as is confirmed by the differences in the static level marks of the horizons, the absence of oscillations in the levels in horizons II and III during long hydrogeological testing of horizon I, and the difference in the chemical composition of the waters of the horizons. The

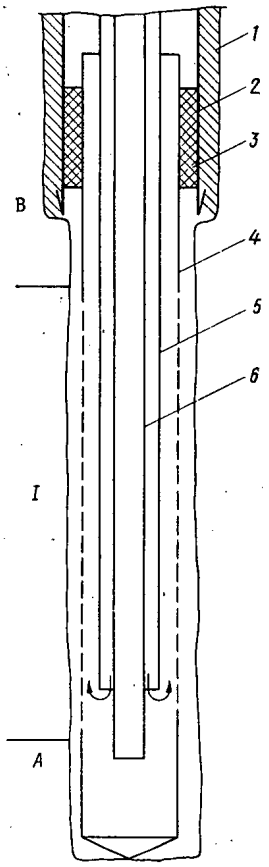

Fig. 3. Construction of injection well (filtration well (filtration zone): 1) cement of space beyond pipe; 2) operational column; 3) packing gland; 4) filtration column; 5) column of pipes for delivery of solutions; 6) measuring leakproof pipe; A, B) impervious beds; I) absorbing horizon.

general direction of the natural flow of underground water in horizon I is south to north and the average rate is 2.2 m/yr. The static level of the underground water of horizon I in the region of the test section is 60 m below the ground surface.

Flowsheet and Desigh Features of the Facilities of the Test Section. The trial section (Fig. 2) is situated close to storage vault of high-activity liquid waste. From the storage vault the waste enters the installation for preburial preparation, consisting mainly of adjustment of the medium, salinity, and radioactivity level. The prepared waste is transferred into the reservoirs of the pumping station for injection through the closed delivery circuit of stainless steel pipes placed in a reinforced-concrete channel, covered with epoxy resin. From the pumping station the waste proceeds at a pressure of 10-12 kgf/cm^2 through the closed delivery circuit to the heat of the injection well and then onto the collector-stratum. All the facilities of the test section give the necessary protection for the operating personnel, provide a capability for washing and deactivating the piping and fittings, and observe a number of other sanitary-radiation safety measures. The principal technological parameters of the operation of the well and section (injection rate and pressure, specific activity and medium of the solution, content of suspended substances, etc.) are remotely monitored and the indications displayed on a control panel in the pumping station where the indications of a temperature sensor and gas counter are also displayed. Secondary indicating instruments are installed in a pavilion above the head of the injection well in order to duplicate and check the measurements and provision has been made for sampling gas which is released from the injection well during the storage of the waste.

Ten wells have been located in the test section: one injection well (IW); six observation wells (a-4 to a-9), covering the collector-stratum and situated roughly 100 m from the IW; two observation wells (a-2 and a-3), covering aquifers II and III (these wells are used to monitor the chemical composition of the stratal liquid and the variations in level in the opened horizons); one "blind" observation well (a-1) lying 50 m from the IW which has been bored to the same depth. This well, which has been provided with a welded column of stainless steel pipes filled with pure water, is employed to monitor the burial process by industrial-geophysical methods.

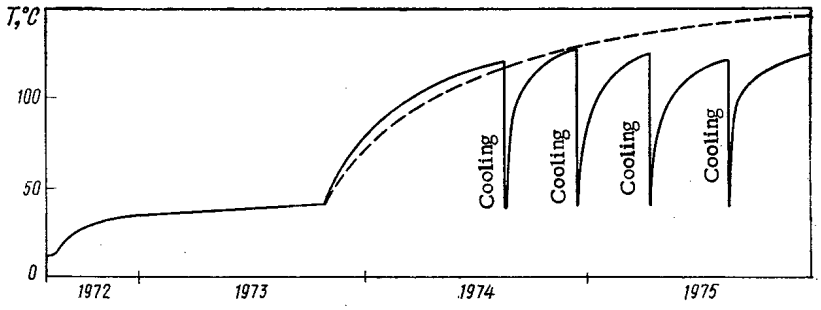

Fig. 4. Variation of temperature in horizon I: ——) observation; and - - - -) calculation.

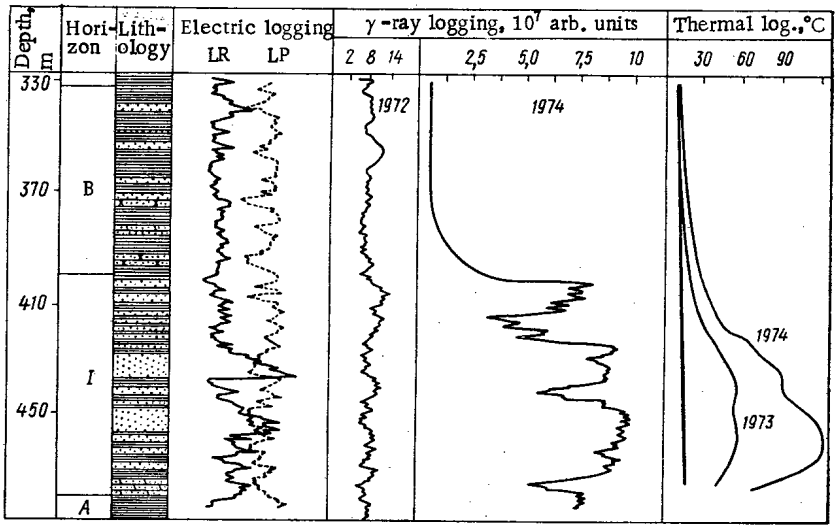

Fig. 5. Results of geophysical monitoring along injection well.

The IW (Fig. 3) has been provided with welded operational and fractional columns of stainless steel pipes. The space beyond the pipe in the operational column has been cemented from the base to the head. Waste is injected through a special lift pipe lowered inside to the base of the collector-stratum. Inside the column through which waste is injected is a blind measuring pipe filled with pure water. This is used to measure the temperature and the γ-ray field directly in the end face zone of the wall. All the other wells are cased in oil-series pipes with filtration columns of stainless steel. Once the observations and monitoring envisaged by the program have been carried out, these wells are eliminated.

Monitoring and Observation in the Test Section. Various forms of monitoring are employed in the test section. In hydrogeological and hydrochemical monitoring the water level in wells a-2 to a-9 and the pressure at the well heads are measured periodically or continuously and samples of the stratal liquid are taken periodically from the monitoring wells for abbreviated and complete chemical and radiochemical analyses. Technological monitoring (in the pumping station) envisages: continuous measurement of the flow rates of the chemical reagents, waste, and the injected liquid; measurement of the specific activity and temperature of the waste being buried and the content of suspended substances (solid phase). In technological and industrial-geophysical monitoring (at the head of the IW) measurements are made of the flow rates (the data of the pumping station are duplicated) and the temperature and γ-ray fields in the end face zone of the well; the quantity and composition of gases released through the IW are determined and samples are taken for chemical and radiochemical analysis. Industrial-geophysical monitoring (through walls a-1 to a-9) envisages γ-ray logging and thermal logging along the well bore.

Principal Results of Trials

At the present time burial of low- and intermediate-activity waste is continuing on a pilot scale. A total of 750,000 m^3 of deactivated and other low-level discharges had been deposited at a depth of 1500 m in the storage vault of the NIIAR by 1976. In the test facility for the burial of intermediate-activity waste ~2.0 million

m^3 of waste with a total activity of 95.0 million Ci had been deposited in the collector-stratum by 1976. The spread of the contour of wastes in the stratum at the SDK level lies within the limit of 400 m from the center of injection and is in keeping with predictions.

The trial burial of high-activity liquid waste was carried out in two stages (in 1972 and 1973 with an interval of 14 months) by the variant with preparation of the stratum forcing back of the waste. The amount of waste disposed of was 450 and 1050 m^3 (7.5 and 53 million Ci), respectively, in the first and second stages. The injection rate was 20-35 m^3/h and the pressure did not exceed 12-14 kgf/cm^2. A slight rise in temperature, to 36-40°C, was observed after the first stage. Observations during the second stage yielded the material shown in Figs. 4 and 5. For 1 year after the second stage the temperature rise was in good agreement with calculation; the temperature gradually stabilized at 120°C (the boiling point in the stratum was 253°C). Periodic cooling by pumping 7-10 m^3 of water at 15-20°C into the well was carried out to ensure that the measuring instruments functioned accurately.

The results of systematic monitoring of the variation in the γ-ray field and thermal logging along the well bore IW (Fig. 5) indicate that the waste was localized within the confines of the collector-stratum, as confirmed by observations in wells a-1 to a-3.

According to data from monitoring measurements and sampling from all observational wells, the migration of radionuclides over the collector-stratum is limited to a radius of 40 m. Observations revealed no signs of the penetration of waste beyond the limits of the stratum or beyond the space beyond the pipe.

Detailed observations established the first release of gas over a period of 11 months (first stage). At the beginning of the second stage the volume of gases liberated was 0.06 normalized m^3/day. A system with a reducer and aerosol filter was provided for the gas discharger. An increase in the volume of the liberated gas was noted after the second stage (up to 0.4 normal m^3/day). However, the process gradually stabilized at a level of 0.1-0.15 normal m^3/day, which is nearly half the predicted value. The composition of the gas was 90-99% nitrogen, the remainder being oxygen, nitrogen oxides, and carbon dioxide; the hydrogen content was $< 0.2\%$. The gas was liberated because of radiolysis and was not active.

Thus, the first results of trial burial of high-activity liquid waste and 3 years of observation of the state of the collector-stratum and the units of the facilities confirmed the technical feasibility, sanitary-radiation safety, and economic expediency of this method of disposing of waste with a specific activity of up to 10-25 Ci/liter. The observations are being continued.

Measures against Failures

To eliminate possible emergency situations and to localize radioactive contamination, provision has been made for preventive and restorative measures which are assigned the paramount role from the point of view of ensuring radiation safety and protection of the environment.

The principal measures with respect to the underground storage vault are:

siting the injection wells on the grounds of the storage vault with allowance for the elimination of interference and injection at the minimum pressure necessary;

monitoring the state of all opened aquifers by means of the observational wells;

remote monitoring of the character of the filling of the collector-stratum;

creation of a closed system for the removal, settling, and injection of the decanted liquid from washing and drainage water

organization of a sanitary safety zone of three belts in the region of the underground storage vault;

formation of water suspensions in a aquifer lying above the collector-stratum.

Other preventive measures which have been elaborated and are being implemented concern external communications for radioactive waste, the pumping stations and reservoirs, and the heads of the injection and observational wells. Implementation of all preventive measures reduces to the minimum the possibility of emergency situations arising with underground burial. In addition to the above, there are restorative measures which are classified as urgent (cutting off the supply of waste, finding the site of the failure, filling the well bore with a high-density solution, etc.) and subsequent liquidating measures – eliminating the cause of the leak of waste, collection and burial of the contaminated soil in a drained place, creation of chemical suspensions (in addition to the water suspension) in the contaminated aquifer in order to fixate the contaminations and

limit their volume, repair-restorative work or liquidation work on the wells, etc. Over the entire period of work on the underground burial of liquid waste in the USSR the preventive measures have ensured the complete sanitary safety of waste disposal in the earth and, therefore, the restorative measures have not been required.

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DEPOSITED ARTICLES

CALCULATION OF PARAMETERS OF WEAK-SIGNAL DETECTION IN MASS AND ELECTRON SPECTROMETERS IN PULSE-COUNTING MODE

M. L. Aleksandrov, M. S. Kobrin, and N. S. Pliss

UDC 621.384.8:519.237.3

In mass- and electron-spectroscopic investigations fluxes of ions (electrons) recorded with special apparatus constitute the sources of primary information. In a number of analyses a weak useful signal is received against the background of a stronger noise signal and the problem arises of detecting the weak signal.

The paper considers a mode of analysis with recording and counting of individual pulses. Two variants are adopted for counting data input into a computer: inputting into the computer pulses recorded by a counter in a given number of successive intervals of time ("direct sampling"); inputting into the computer an interval of time during which a given number of pulses is recorded ("inverse sampling").

The statistical formulation is: a reference noise signal and the signal under study, which constitutes a Poisson stream of pulses of unknown intensity, λ_0 and λ_1, respectively, are accessible to observation.

Hypothesis H_0: $\lambda_0 = \lambda_1$, no useful signal; alternative H_1: $\lambda_1 > \lambda_0$, a useful signal exists.

A solution in favor of H_0 or H_1 should be adopted on the basis of observations.

For both variants the inputting techniques in the paper propose decision rules which give the maximum probability of correct observation, $\beta = P(H_1/H_0)$, at a given probability of erroneous observation, $\alpha = P(H_1/H_0)$, for small signal-to-noise ratios

$$C = \frac{\lambda_1 - \lambda_0}{\lambda_0} = \frac{\lambda_1}{\lambda_0} - 1 = \gamma - 1.$$

The paper gives exact (for low total number t of pulses recorded) and asymptotic (as $t \to \infty$) distributions of criteria, making it possible to choose the appropriate threshold values. For each of the criteria presented formulas are derived for calculating the power functions $\beta(\gamma)$ which determine the probability of correct detection for any signal-to-noise ratio $C = \gamma - 1$ and for a given total number t of pulses recorded.

Tables and graphs are given in the paper for the most characteristic ranges of the arguments. Moreover, the t necessary for ensuring given α and β is found as a function of C. It is shown that for small values of C the dependence of t on C is logarithmically linear.

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0038-531X/78/4402-0179$07.50

CALCULATION OF PARAMETERS OF NEUTRON THERMALIZATION IN LEAD

Sh. Kenzhebaev

UDC 621.039.51.12

It is well known that neutron transport in a heavy gaseous moderator is described by the Wilkins second-order differential equation for eigenfunctions ψ_k with eigenvalues Δ_k:

$$y\psi_k'' + (3-2y^2)\psi_k' - (\Delta_k - 4\lambda y)\psi_k = 0, \quad \lambda = -1/6 \cdot Ml_s^2B^2. \tag{1}$$

The objective of the present paper is to obtain the diffusion and thermalization parameters for neutrons in lead within the framework of the model of a heavy monatomic gas.

In [1] it was shown that

$$\lambda = k + a_{k1}\Delta_k + a_{k2}\Delta_k^2 + a_{k3}\Delta_k^3 + \ldots \quad (k=0, 1, \ldots), \tag{2}$$

and numerical values of the coefficients a_{ki} were found for k = 0, 1, 2, 3, 4, 5 and i = 1, 2, 3, ($k > 0$, $a_{k3} \approx 0$). We considered a cubic (quadratic) equation obtained by retaining the first four (three) terms on the right-hand side of Eq. (2). The root of such an equation was calculated by iteration after which the attenuation constant $\alpha_k' = \Delta_k/2Ml_s$ was found. The eigenvalues Δ_k were also found numerically in the quasiclassical approximation by the WKB (Wentzel–Kramers–Brillouin) method.

The calculation was performed by both methods for the first six eigenvalues Δ_k as a function of the values of B^2 in the interval $0 \leq B^2 \leq 0.015$ cm^{-2} for M = 207.2, l_s = 2.67 cm. Table 1 compares the results of calculations for $B^2 = 0$ with the results of [2].

It is known that

$$\alpha_0' = \bar{D}B^2 - CB^4 + FB^6 + \ldots . \tag{3}$$

We approximated the coefficients of Eq. (3) on the basis of approximation of the two obtained above from the relation $\alpha_0' = f(B^2)$ by the least-squares method as well as with the assumption that $l_{tr} = v/(v + a)$ in accordance with [3, 4].

Table 2 gives the calculated values of diffusion parameters and the results of [2] for comparison. However, for the diffusion cooling coefficient C the perturbation method gave us values of $92.18 \cdot 10^5$ and $86.041 \cdot 10^5$ cm^4/sec, respectively, in the intervals $0 < B^2 \leq 0.085$ and $0 < B^2 \leq 0.015$ cm^{-2}.

The diffusion parameters were also determined by another method: the results of calculations by the perturbation theory and WKB method are considered as "experimental" fluctuations of a point. Then the least-squares method is used to find the diffusion parameters (3). Their values ($\bar{D} = 2.196 \cdot 10^5$ cm^2/sec, C =

TABLE 1. Damping Constants for Neutrons in Lead

α_k', sec^{-1}				τ_{th}, ($\equiv 1/\alpha_1'$), μsec
α_0'	α_1'	α_2'	α_3'	
0	933,42	1972,08	3113,0	1071,33
0	932,53	2034,08	3334,45	1072,35
0	931,22	2010,61	3266,85	1073,86
				1073,85

Note. The first row, calculation by perturbation theory [1]; second row, by WKB method; third row, according to data of [2]; fourth row, in accordance with [3, 4] by a rapid polynomial method.

TABLE 2. Diffusion Parameters of Lead

$\bar{D}$, cm^2/sec	C, cm^4/sec	F, cm^6/sec
$2{,}203 \cdot 10^5$	$86{,}041 \cdot 10^5$	$20{,}565 \cdot 10^7$
$2{,}184 \cdot 10^5$	$89{,}660 \cdot 10^5$	$22{,}310 \cdot 10^7$
$2{,}212 \cdot 10^5$	$90 \cdot 10^5$	—
—	$91{,}170 \cdot 10^5$	$42{,}079 \cdot 10^7$

Note. The first row, calculation by perturbation theory [1]; second row, by WKB method; third row, in accordance with [3, 4] by rapid polynomial method; fourth row, in accordance with data of [2].

$89.115 \cdot 10^5$ cm^4/sec, $F = 22.42 \cdot 10^7$ cm^6/sec) are in good agreement with the results of [2] and with our calculations by the rapid polynomial method [3, 4]. The values of F however, differ from the results of [2].

The first terms of the expansions of Δ_k into a power series in B^2 were found by the least-squares method.

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THERMAL EXPANSION OF URANIUM CARBIDE WITH ADDITIVES IMITATING STABLE FISSION FRAGMENTS IN 8% BURNUP OF HEAVY ATOMS

A. A. Ivanov, V. S. Belevantsev, Z. F. Evkina, V. A. Zelyanin, and S. N. Bashlykov

UDC 621.039.542.32:536.413.2

To evaluate the efficiency of fuel elements it is necessary to know the effect of fission-product elements on the linear thermal expansion coefficient (LTEC) of the fuel. For this purpose imitators of fission-product elements were introduced into carbide fuel in a quantity corresponding to 8-10% burnup. The method of preparation, structure, and phase composition of such an imitator of irradiated fuel were described earlier [1]. The characteristics of the materials studied are given in Table 1.

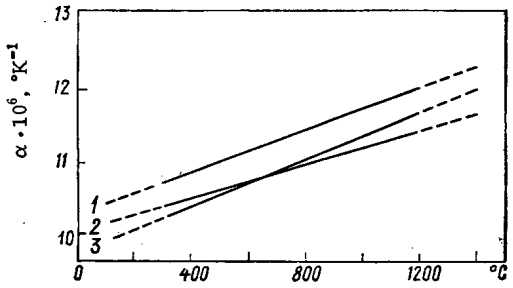

Fig. 1. Linear thermal expansion coefficient of uranium carbide: 1) results of present paper; 2) data of Carnigli [2]; 3) data for imitator of irradiated carbide fuel.

TABLE 1. Characteristics of Specimens

Material	Phase composition*	Phase content, vol.%†	$\bar{\alpha} \cdot 10^{-6}$, $°K^{-1}$
Uranium carbide	UC	90,8	11,2 [2]
	UC_2	9,2	15,1 [2]
Imitator of irradiated carbide fuel	(UZrMo)C	78,38	11,2‡
	$(UREE)_2C_3$	13,55	10,5 [2]
	$UMoC_2$	2,09	—
	$(REE)_2O_3$	3,07	10,3 [3]
	BaOSrO	0,74	13,5 [3]
	$U(RuPd)_3$	2,17	—

*According to data of [1].
†Calculated by proceeding from chemical composition.
‡It was assumed that the dissolution of small amounts of Zr and Mo does not change the LTEC of uranium carbide.

The linear thermal expansion coefficient was measured at temperatures ranging from 20 to 1400°C. Specimens in the form of a stack of pellets were heated in a vacuum of $2 \cdot 10^{-5}$ mm Hg at the rate of 2.5 deg C/min; the LTEC was determined with an error of ±3%.

At a temperature exceeding 1200°C the thermal expansion was observed to slow down somewhat, this evidently being due to CO escaping into the dynamic vacuum and phases of lower density dissolving in the uranium monocarbide. The values of LTEC (Fig. 1) were therefore calculated only up to 1200°C. The average values obtained for the LTEC of hot-pressed uranium carbide are close to those calculated by the equation suggested by Carnigli for solid carbide [2]. The average LTEC of the imitator in the range 20-1200°C was roughly 5% smaller than in uranium carbide.

On the basis of Table 1 by the law of additivity we calculated the LTEC of the uranium carbide specimens studied and the imitator for temperatures ranging from 20 to 1000°C. The experimental and calculated data are in good agreement for uranium carbide, $11.8 \cdot 10^{-6}$ and $11.6 \cdot 10^{-6}$ $°K^{-1}$, and for the imitator, $11.4 \cdot 10^{-6}$ and $11.0 \cdot 10^{-6}$ $°K^{-1}$, respectively.

Calculations and experiment show that the introduction of fission-product imitators into carbide fuel in amounts corresponding to 8% burnup reduces the LTEC by 5%; it may be expected that under natural conditions the LTEC of carbide fuel after a burnup of 8-10% of the heavy atoms will not differ from the LTEC of the original carbide by more than 5%.

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THE $^{27}Al(n, p)^{27}Mg$ CROSS SECTION FOR 14.9-MeV NEUTRONS

V. I. Melent'ev and V. V. Ovechkin

UDC 539.171.015.4

The new value obtained in [1] for $\sigma_{n,p}$ in aluminum for 14-MeV neutrons appreciably exceeds the results of [2]. Accordingly, in the present paper the $^{27}Al(n, p)^{27}Mg$ cross section is measured by activation of aluminum in an NG-150I neutron generator with the induced activity being recorded by a NaI(Tl) detector measuring 150 × 150 mm with a 33 × 70-mm wall. The magnitude of the neutron flux during the activation process was checked with a monitor based on a Si(Li) detector [3] with automatic recording of the readings (Y_i) at intervals of t_i = 30 sec.

The cross section was calculated by the formula

$$\sigma_{n,p}=\sigma_{n,\alpha}\frac{\left[\lambda^{-1}(S_1S_2S_{12}^{-1}+N_0)^{-1}e^{-\lambda\tau}(1-e^{-\lambda t_r})\sum_{i=1}^{n}Y_i(1-e^{-\lambda t_i})e^{-(n-i)\lambda t_i}\right]_{Na}}{\left[\lambda^{-1}S^{-1}(\varepsilon_1\lambda_1+\varepsilon_2\lambda_2)e^{-\lambda\tau}(1-e^{-\lambda t_r})\sum_{i=1}^{n}Y_i(1-e^{-\lambda t_i})e^{-(n-i)\lambda t_i}\right]_{Mg}},$$

where $\sigma_{n,\alpha}$ = 111 ± 4 mb is the cross section of the $^{27}Al(n, \alpha)^{24}Na$ reaction [2], selected as a standard reaction; S_1, S_2, S_{12}, respectively, are the areas under the individual peaks of total absorption of cascade γ rays with energies of 1.37 and 2.74 MeV and under the combined peak with an energy of 4.11 MeV from the ^{24}Na nuclide; N_0 is the number of pulses under the entire ^{24}Na spectrum; τ is the delay time from the end of irradiation to the beginning of recording; t_r is the recording time; S is the total area under the 844- and 1014-keV photopeaks of the ^{27}Mg being determined; and $\varepsilon_1\gamma_1$ and $\varepsilon_2\gamma_2$ are the products of efficiencies of γ-ray outputs for energies of 844 and 1044 keV, respectively.

The value of $\sigma_{n,p}$ = 71 ± 5 mb obtained is consistent with the results of many published papers.

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INTERPOLATION FORMULAS FOR CALCULATING THE INTEGRATED COHERENT AND INCOHERENT SCATTERING CROSS SECTIONS

O. S. Marenkov and B. G. Komkov

UDC 539.12.04

The integrated cross sections for coherent and incoherent γ-ray scattering, σ_{coh} and σ_{incoh}, calculated by integrating the Thomson and Klein–Nishina differential cross sections over the scattering angle with allowance for the atomic form factor and the incoherent scattering function, are usually tabulated [1, 2]. The accuracy of the quantum-mechanical calculations of the form factors and incoherent scattering functions depends on the accuracy of the approximation with which the electron distribution in the atom is described. In [1] form factors and incoherent scattering functions obtained on the basis of the Hartree–Fock–Slater wave functions were used in the calculation of the integrated cross sections. It has been shown [2] the most accurate calculations of the form factors and incoherent scattering functions are based on the use of the Hartree–Fock wave functions. The cross sections σ_{coh} and σ_{incoh} have been tabulated in [2] for 100 elements over the range of energies 10^{-1}–10^5 keV.

For numerical Monte Carlo simulation of the transport of low-energy quanta ($E < m_0c^2$) in matter the functional dependence of the scattering cross section on the photon energy E is advisably and conveniently expressed in simple formulas. In [3, 4] it was shown on the basis of tabulated data [1] that the energy dependence $\sigma_{coh}(E)$ and $\sigma_{incoh}(E)$ is approximated with good accuracy for any element by the expressions:

$$\sigma_{coh}(E) = (\sigma_0 + \sigma_1 E + \sigma_2 E^2)^{-1};$$
$$\sigma_{incoh}(E) = (\sigma'_0/E + \sigma'_1 + \sigma'_2 E)^{-1}.$$

In the present paper, tables present the coefficients σ_i and σ'_i calculated for 100 elements by the least-squares method from the data of [2].

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LETTERS

ONE ERROR OF THE RADIOISOTOPE METHOD OF MEASURING THE CONTINUITY OF A TWO-PHASE FLOW

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UDC 539.106

Besides computational methods of evaluating the gas content of two-phase gas–liquid flows, extensive use is made of methods of direct measurement, especially the radioisotope method, based on the measurement of the γ-ray flux of the medium under study [1-4].

In using this method it is convenient to estimate the true volume gas content in terms of the continuity of the two-phase flow:

$$Q=(\rho_l V_l+\rho_g V_g)/\rho_l V_0, \quad (1)$$

where Q is the continuity of the two-phase flow; ρ_l and ρ_g are the densities of the liquid and gaseous phase; V_l and V_g are the volume contents of the liquid and gaseous phases, respectively, in the monitored volume; and V_0 is the monitored volume.

The total systematic error of the continuity measurement consists of a number of independent components [5-8], which are due to the following factors: the statistical character of the radiation processes, interaction with the monitored medium, and detection of the γ rays; averaging of the radiation characteristics with respect to time; deviation of the radiation spectrum from the monochromatic; and the build-up factor. In the course of measurement of the continuity of a gas–liquid flow the structure of the flow may change and an additional error component may arise owing to the deviation of the flow regime from that assumed during calibration of the instrument; this component may attain considerable values in some cases [4]. This error, in turn, can be split up into two components. The first is due to the nonlinearity of the law of attenuation of γ rays in matter and the second, to the nonuniformity of the radiation field in the volume monitored. The latter component can be eliminated by setting up a uniform radiation field in the monitored volume [9]. A change in the flow regime affects the error due to the build-up factor.

Reference [4] gives a method for calculating the eroor component caused by the nonlinearity of the law of γ-ray attenuation, for laminar and uniform emulsion modes of flow in flow regimes in rectangular piping.

The authors of the present paper propose a method for calculating this error for any given regimes of two-phase flows in piping of any shape.

Let us consider a parallel beam of monochromatic γ radiation perpendicular to the sensitive surface of the radiation detector. The monitored volume is taken to mean the square between the source and the detector, illuminated by γ rays (Fig. 1). Let us break up the monitored volume V_0 into elementary volumes V_{0i} with identical bases ΔS such that in each base the distribution of the radiation flux could be considered to be uniform. In this case the radiation flux Φ_i through the base of the elementary volume, resting on the detector, is given by the following familiar relation:

$$\Phi_i=\Phi_0 \exp[-\sigma(n_l d_{l\,i}+n_g d_{gi})-\mu_p l_{pw\,i}], \quad (2)$$

where Φ_0 is the radiation flux through the base of an elementary volume resting on the source; σ is the atomic cross section for the interaction of the radiation with the medium under study; n_l and n_g are the numbers of atoms per unit volume of the liquid and gaseous phases, respectively, without any flux; d_{li} and d_{gi} are the effective thicknesses of the liquid and gaseous phases in the volume V_{0i}; μ_p is the linear coefficient of attenuation of radiation in the pipe material; $l_{pw\,i}$ is the effective pipe-wall thickness in the volume V_{0i}.

Upon rearranging Eq. (2), we get

$$\Phi_i=\Phi_0 \exp(-\mu_m Q_i d_i-\mu_p l_{pwi}), \quad (3)$$

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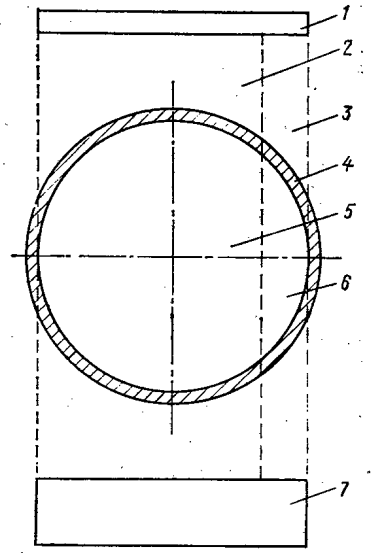

Fig. 1. Relative position of radiation source, detector, and pipe: 1) γ-ray source; 2) monitored volume; 3) elementary monitored volume; 4) pipe with medium under study; 5) interaction volume; 6) elementary interaction volume; 7) radiation detector.

where μ_m is the linear coefficient of radiation attenuation by the medium under study; Q_i is the continuity in the volume V_{0i}; and d_i is the altitude of the volume V_{0i}.

The continuity in the monitored volume is related to the continuity in the elementary volumes by

$$Q=\left(\sum_{i=1}^{n} Q_i V_{0i}\right)/V_0, \tag{4}$$

where n is the number of elementary volumes in the monitored volume. Solving Eq. (3) for Q_i and rearranging, we get

$$Q=S\sum_{i=1}^{n}\left[(\ln \Phi_0/\Phi_i)-\mu_p l_{pw}\right]/\mu_m n V_0, \tag{5}$$

where S = ΔSn is the sensitive area of the detector.

Under real conditions, instead of the fluxes Φ_i we measure the complete flux across the sensitive area of the detector, Φ_{mr}. In this case, the measured value of the continuity $Q_{i\,meas}$ in the volume V_{0i} is given by

$$Q_{i\,meas}=\left[(\ln \Phi_0/\Phi_{i\,mr})-\mu_p l_{pw}\right]/\mu_m d_i, \tag{6}$$

where $\Phi_{imr}=\sum_{i=1}^{n}\Phi_i/n$ is the radiation flux, reduced to an element of the sensitive area of the detector.

Taking Eqs. (3) and (4) into account, we obtain an expression for the measured value of the continuity in the monitored volume V_0:

$$Q_{meas}=S\left\{\ln\left[n/\sum_{i=1}^{n}\exp\left(-\mu_m Q_i d_i-\mu_p l_{pw\,i}\right)\right]-\mu_p\sum_{i=1}^{n} l_{pwi}\right\}/\mu_m V_0. \tag{7}$$

The sought error Δ will be defined as the difference between the measured and true values of the continuity:

$$\Delta=Q_{meas}-Q. \tag{8}$$

In view of the fact that the values of Q and Q_i are known under given flow regimes and shape of pipe cross section, the error is calculated only from Eqs. (7) and (8). The relations obtained are applicable to pipes with any shape of cross section.

The value of the continuity in the interaction volume V_0', i.e., in that part of the monitored volume in which the γ rays interact with the medium being studied, usually is of practical interest. In this case the error arising with deviation from the two-phase flow regime assumed during calibration is best found as the difference between the values of Q'_{meas} calculated from Eq. (7) for the calibration regime and for the reestablished regime. In doing this in Eq. (7) we substitute the values of the V_0' and continuity Q_i' in elementary interaction volumes V'_{0i}.

To achieve satisfactory accuracy in estimating the error under consideration the interaction volume V_0' can be broken up into 5-10 elementary volumes.

The derived relation was verified in the range of intermediate gas contents ($Q \simeq 0.5$) in which the error due to the indeterminacy of the structure of the flow reaches maximum values [4]. Pipes with round and square cross sections were considered. The source used was ^{241}Am with an activity of 0.55 Ci. A NaI(Tl) scintillation crystal served as detector.

The transition from a bubble regime to a dispersion-annular regime of flow was studied in a 38-mm round pipe. The flow regime was imitated with drilled cylindrical segments of Plexiglas which were filled with air. The calculated and experimental values of the error are 0.03 and 0.04, respectively.

In a square pipe measuring 30 mm on a side we studied the variations in a flow regime from laminar (in the direction of irradiation) to ram. The components of the two-phase flow were water and air.

The calculated and experimental values of the error in this case were 0.07 and 0.11, respectively. The relative error in the determination of the continuity in all measurements did not exceed 3%. Moreover, the value of the error calculated by Eq. (8) was confirmed to be in good agreement with the calculated and experimental data of [4].

Thus, the results of the verification demonstrate the practical usefulness of the derived relation. The relations obtained in the present paper can be used in the development of radioisotope instruments for measuring the continuity as well as for evaluating the possibility of employing existing instruments under concrete operating conditions. Furthermore, the results can be used to enhance the accuracy of continuity measurements by the introduction of corrections for the variations in the flow regime. The flow regime could also be identified in the process, for example, by the pulsation of the shear stress in the pipe wall.

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EFFECT OF IMPURITY IN SINTERING OF URANIUM DIOXIDE

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UDC 621.039.54

In ceramic technology the process of sintering a product is sometimes controlled by various oxide additives to the basic material. Most frequently, oxide additives are used to reduce the sintering temperature or to produce the desired microstructure. Uranium dioxide powder is also sensitive to additions of other oxides [1].

Figure 1 illustrates the effect of additives of titanium and niobium oxides in reducing the sintering temperature of uranium dioxide powder. For the investigations we used pressed specimens 15 mm in diameter and 20 mm in height. The specimens were heated in a hydrogen stream and held at the desired temperature for 2 h. As is seen from Fig. 1, several tenths of one percent of titanium or niobium oxide reduce the range of intense sintering of uranium dioxide powder by roughly 150°C and the process of compaction ends in principle at 1300°C, whereas the process of composition of the powder without additives ends at 1600°C. A noticeable compaction of both the pure powder and the powder with additives begins to be felt below 1000°C but the increase in density with a rise in temperature is appreciably greater for powders with the addition of titanium or niobium oxides.

The microstructure of sintered uranium dioxides with the addition of titanium oxide is shown in Fig. 2. Almost all the pores have developed along the grain boundaries and a slight porosity remains in the center of the uranium oxide grains.

It is well known that the increased rate of compaction of uranium oxide powder at a low temperature is due to excess oxygen in the crystal lattice. A change in the oxygen content within narrow limits (from $UO_{2.000}$ to $UO_{2.050}$) increases the diffusion coefficients of the uranium and oxygen ions by several orders of magnitude [2, 3].

It is not always easy to explain the mechanism by which the additives act. The simplest explanation is that a liquid phase appears and that the transport of matter is accelerated by dissolution-precipitation. However, the liquid phase does not appear in the uranium dioxide–titanium oxide until a temperature of 1645°C [1], which is considerably above the range of intense sintering. The drop in the sintering temperature of uranium dioxide in the presence of titanium or niobium oxides can feasibly be associated with the presence of oxygen in the crystal lattice of uranium dioxide. The increased concentration of oxygen may be maintained as the result

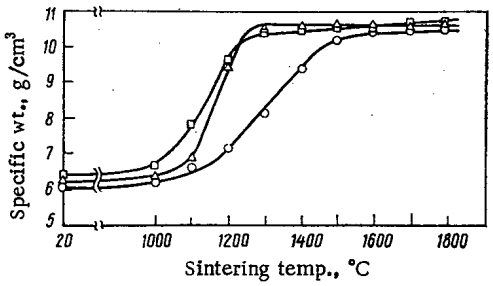

Fig. 1. Sintering curves for powdered uranium dioxide: □, △) with addition of 0.1 and 0.2 wt.% of TiO_2 and NbO_2, respectively; ○) without additives.

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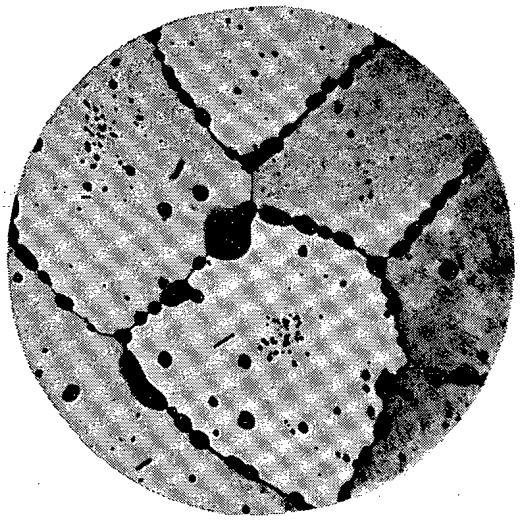

Fig. 2

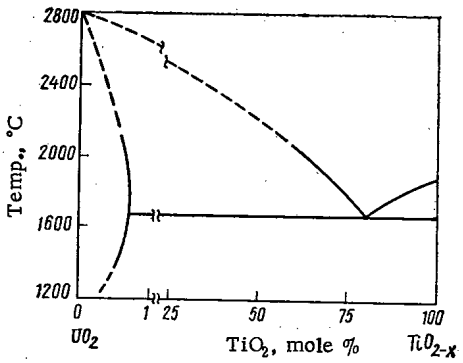

Fig. 3

Fig. 2. Microstructure of sintered uranium dioxide containing 0.1 wt.% TiO_2, specific weight of specimen 10.5 g/cm^3, unetched metallurgical section (×450).

Fig. 3. Refinement of state diagram of uranium dioxide–titanium oxide ($TiO_{1.57}$) system.

of the dissociation of titanium or niobium oxides or because of ions with a valency of less than +4 penetrating into the crystal lattice while at the same time the electric charge is compensated by the penetration of excess oxygen. To determine the degree of dissolution of titanium oxides in uranium oxide the mixture of oxides was heated in hydrogen or purified argon over the range 1400-2000°C and subjected to chemical, microscopic, and x-ray analysis in the MS-46 microanalyzer. By microscopic examination of the metallographic sections in reflected light it was established that the liquid phase appears in the uranium dioxide–titanium oxide system at a temperature of nearly 1650°C, which is in good agreement with the results of [1]. According to the results of x-ray analysis, the eutectic consists of uranium-free titanium oxide; this points to the absence of solid solutions in the system as far as titanium oxide is concerned. The existence of two phases in the eutectic, viz., uranium dioxide and titanium oxide, indicates that under the conditions studied no chemical compounds between uranium dioxide and titanium oxide were detected. Evidently, the chemical compounds between these oxides which were detected in [4, 5] exist under the conditions under which TiO_2 exists. Under the conditions of the experiments titanium dioxide loses oxygen; according to published data, the composition of the oxide is close to $TiO_{1.5}$ [1].

In a specimen with a 0.1 wt.% content of titanium oxide and annealed at 1400°C we detected inclusions of the original titanium oxide; the inclusions had quite distinct boundaries, which indicates the absence of dissolution or a very low rate of dissolution. After heating at 1600 and 2000°C no separate inclusions of titanium oxide were detected; the titanium was distributed uniformly over the entire area of the metallographic section. The titanium concentration in the uranium dioxide crystals, according to data from x-ray spectral analysis, is 0.1 wt.%. By chemical analysis we found 0.13 wt.% titanium oxide in this specimen. With the microanalyzer we established that at 2000°C the solid solution contains 0.16 wt.% (0.54 mole %) titanium oxide. The limiting concentration is probably close to this value since the second phase is clearly visible in a specimen with a 0.55 wt.% titanium oxide content.

Our investigations made it possible to refine the state diagram for the uranium dioxide–titanium oxide system (Fig. 3) published in [1].

According to Fig. 3 it may be assumed that the region of titanium oxide–uranium dioxide solid solution exists below 1200°C but even at 1400°C the dissolution rate of titanium oxide and uranium dioxide, as shown above, is low enough that it is hardly likely that any decrease in the sintering temperature of the uranium dioxide is caused by the penetration of titanium ions into its lattice. Most likely the increased sintering rate of uranium dioxide powder with the addition of titanium oxide at a low temperature is explained by the increase in the oxygen coefficient of the uranium dioxide because of the oxygen released by the titanium dioxide. By calculations it can be shown that the loss of 25% of the oxygen in the titanium dioxide when 0.1 wt.% of the latter is present in uranium dioxide may change the oxygen coefficient of uranium dioxide by 0.01, as a result of

which the coefficients of self-diffusion of the uranium and oxygen ions in the uranium dioxide lattice increase by several orders of magnitude.

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METHOD OF GRAPHICAL CALCULATION OF EXTRACTION PROCESS FOR SYSTEMS WITH TWO EXTRACTABLE MACROCOMPONENTS

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The graphical method of calculating the distribution of substances from the extraction section for systems with two extractable macrocomponents is based [1] on the representation of the extraction isotherms in

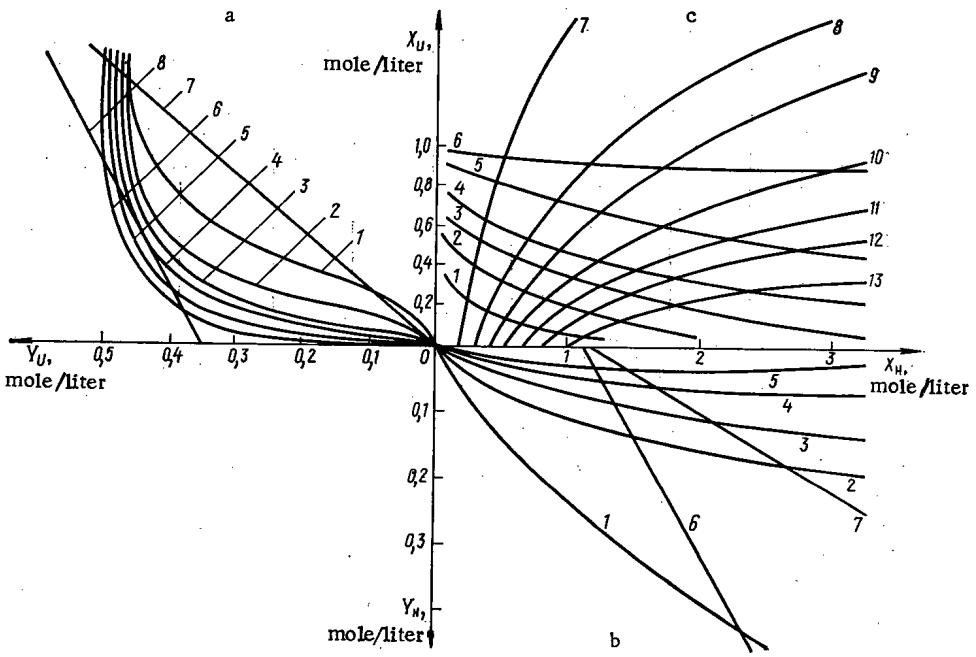

Fig. 1. Y–X diagrams for system uranium (VI)–nitric acid–30% solution TBP; a) lines 1-6 are isotherms for extraction of uranium at constant acid aqueous phase of 0, 0.5, 1.0, 1.5, 2.0, and 3.0 moles/liter; 7,8) are extraction and scrubbing lines for uranium; b) lines 1-5 are isotherms for extraction with nitric acid at uranium contents in the aqueous phase of 0, 0.2, 0.4, 0.6, 0.8 mole/liter; 6-7) extraction and scrubbing operating lines for acid; c) extraction isotherms replotted in X_U–X_H coordinates; 1-6) lines of constant Y_U level equal to 0.1, 0.2, 0.35, 0.4, 0.45, 0.48 mole/liter; 7-13) lines of constant Y_{HNO_3} level equal to 0.01, 0.025, 0.04, 0.06, 0.08, 0.1, 0.15 mole/liter.

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TABLE 1. Calculation of Nominal Extractor Regime

No. of stage	First approx.		Second approx.		Third approx.		Computer calc.	
	X_1	X_2	X_1	X_2	X_1	X_2	X_1	X_2
1_{scr}	0,15	2,5	0,11	2,0	0,10	2,0	0,105	1,96
Shift point	1,09	1,65	1,07	1,42	1,07	1,35	1,072	1,38
1ex	0,1	2,6	0,11	2,2	0,12	2,0	0,122	2,05
2ex	0,0	2,2	0,0	1,9	0,0	1,8	$2,1 \cdot 10^{-3}$	1,82
3ex	0,0	1,9	0,0	1,5	0,0	1,4	$4 \cdot 10^{-5}$	1,43
4ex	0,0	1,13	0,0	1,0	0,0	0,97	$1,7 \cdot 10^{-6}$	0,974

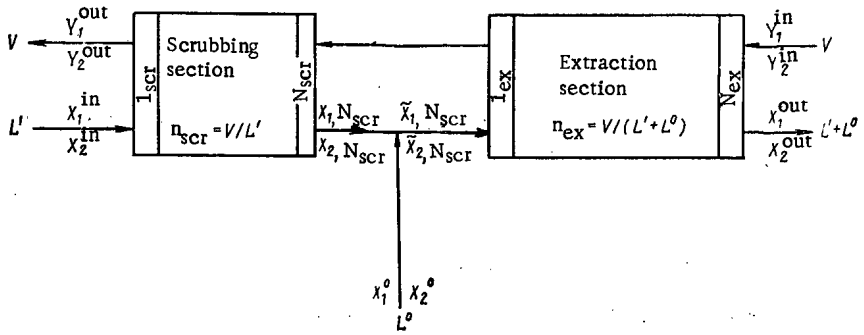

Fig. 2. Extractor diagram and notation for flows and concentrations: L', L^0, and V are the flows of scrubbing, initial aqueous, and organic solutions, liter/h; X and Y are the concentrations of the extractable components in the aqueous and organic phases; N_{scr} and N_{ex} are the numbers of stages of scrubbing and extraction sections.

the form of surfaces described by the equations $Y_1 = f_1(X_1, X_2)$ and $Y_2 = f_2(X_1, X_2)$. The Y−X diagrams used directly in the calculations are projections of the sections of these surfaces by the planes X_1 = const and X_2 = const (Figs. 1a and b, respectively). The method consists in making successive approximations in the Y−X diagrams for every component by using the operating lines and constructing an individual equilibrium line for every degree.

Graphical calculation of the extraction in the system uranyl nitrate−nitric acid−tributyl phosphate (TBP) by this method takes a good deal of time owing to the need of twofold iteration. Iteration is carried out successively over each component by using their Y−X diagrams in succession. Moreover, the general iteration successive approximations are made on the Y−X diagram for nitric acid so as to determine the exact position of the working line.

The Algorithm of the Method. The proposed method of calculation is based on another way of representing the extraction isotherms for a two-component system. Lines of constant Y_1 and Y_2 level are constructed in the X_1−X_2 plane (Fig. 1c). The points for constructing these lines are obtained by sectioning the families of curves $Y_1 = f(X_1, X_2)$ and $Y_2 = f(X_1, X_2)$ where the parameters are X_2 and X_1, respectively, by straight lines parallel to the abscissa (i.e., Y_1 = const and Y_2 = const). With this representation of extraction isotherms, the algorithm of graphical calculation consists of the following (the notation for the flows and concentrations are given in Fig. 2):

1. The value of Y_1^{in} is found from the material balance for the entire apparatus with allowance for the fact that in the prelimiting regimes* $X_1^{out} \approx 0$ and, usually, $Y_1^{in} = 0$:

*A "limiting regime" is one in which the concentration of the macrocomponent at the extraction stage outlet is equal to the equilibrium concentration, the equilibrium curve and the working operating line intersect (the driving force of the mass exchange tends to zero). Although this regime in fact is not realized (it requires an infinite number of stages), it is an important characteristic of the process; with a "prelimiting" regime deep extraction of the component is possible and with a "translimiting" regime the component is inevitably discharged into the raffinate. The concentration of valuable elements in the raffinate is determined in principle by the degree to which the regime approaches the "limiting" regime.

$$Y_1^{out} = (L^0 X_1^0 + L' X_1^{in})/V.$$

2. The value of Y_2^{out} is determined from the material balance for the entire apparatus with allowance for the approximate equality $X_2^{out} \approx X_2^0$ (as experiment shows, this condition is admissible in the first approximation): $Y_2^{out} = [L^0 X_2^0 + L' X_2^{in} - (L' + L^0) X_2^{out}]/V \approx L'(X_2^{in} - X_2^0)/V$.

3. The values of Y_1^{out} and Y_2^{out} are used to find the equilibrium concentrations $X_{1,1}$ and $X_{2,1}$ (see Fig. 1a).

4. Operating lines of the scrubbing part are constructed for both components from points with coordinates (X_1^{in}, Y_1^{out}) and (X_2^{in}, Y_2^{out}): the slope of the operating line, as is known, is found from the condition $\tan\alpha = 1/n_{scr} = L'/V$.

5. The values of $Y_{1,2}$ and $Y_{2,2}$ are determined from the operating lines. Points 3 and 5 are repeated N_{scr} times, i.e., all stages up to feed intake.

6. The values of $X_{1,N_{scr}}$ and $X_{2,N_{scr}}$ are calculated with allowance for the material balance at the feed intake:

$$L' X_{i,\,N_{scr}} + L^0 X_i^0 = \tilde{X}_{i,\,N_{scr}} (L' + L^0), \quad i = 1, 2.$$

7. The operating lines of the extraction part of the apparatus are constructed for both components by the points with the coordinates

$$(\tilde{X}_{1,\,N_{scr}};\ Y_{1,\,N_{scr}}),\ (\tilde{X}_{2,\,N_{scr}};\ Y_{2,\,N_{scr}})$$

$$\text{and } \operatorname{tg}\alpha = \frac{1}{n_{ex}} = \frac{L' + L^0}{V}.$$

8. Points 3 and 5 are repeated N_{ex} times.

9. Upon completion of the calculation, $X_i^{out} \approx 0$.

In nearly limiting regimes, $X_1^{out} \neq 0$ and, therefore, in the calculation of such a regime a particular value of X_1^{out} must be prescribed and the operating lines for the first component and for the second component must be adjusted.

10. If the value of X_2^{out} calculated in point 8 does not agree with the value assumed earlier in point 2, the former is taken for the initial value for a second approximation and the entire calculation of point 2 is repeated.

Thus, in the proposed method successive approximations must be made only in order to determine the position of the operating line from the nitric acid.

Calculation of the Distribution of Uranyl Nitrate and Nitric Acid over the Extraction Stages. As an example, we made calculations for the extractor of a hypothetical Purex process system [2] (see Fig. 2) with one scrubbing stage and four extraction stages ($N_{scr} = 1$, $N_{ex} = 4$).

The initial data for calculation of the nominal regime are: $V = 338$ liters/h, $L' = 56.2$ liters/h, $L^0 = 75$ liters/h, $Y_1^{in} = Y_2^{in} = 0$, $X_1^{in} = 0$, $X_2^{in} = 2$ moles/liter, $X_1^0 = 1.8$ moles/liter, $X_2^0 = 0.95$ moles/liter (subscript 1 corresponds to uranyl nitrate and 2, to nitric acid. The results of graphical calculation and its comparison with computer calculations are presented in Table 1.

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QUANTUM YIELD OF ELECTRONS FROM THE CYLINDRICAL CASING OF AN ISOTOPIC γ-RAY SOURCE

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The stream of secondary electrons from a plane absorber irradiated by a parallel beam of monoenergetic γ rays incident upon the absorber at right angles has been considered in [1-3]. However, such geometry is encountered very rarely. The direct transfer of these results to the case of any geometry, e.g., cylindrical, as is done in [1-3], is inadmissible since the stream of forward-emitted electrons depends on many factors: the spectral composition of the primary γ rays, the irradiation geometry, and the thickness and atomic number of the absorber.

In this paper we present the results of calculation by statistical examination of the spectral distribution and the quantum yield of the secondary electrons from the cylindrical casing of an isotropic ^{60}Co γ-ray source. Such a geometry is observed in coaxial self-powered detectors, in emitters for contact γ-ray therapy, etc.

The radiation source we considered had a diameter of 4 mm and a length of 10 mm and was encased in materials with different atomic numbers. The thickness of the casing in each case is equal to the extrapolated track length of electrons of maximum energy in the given material.

The calculation was carried out in two stages. In the first stage, using the well-known method with random numbers [4] we traced the history of a γ-ray quantum until its exit from the casing of the source or absorption in the source. In these cases we stopped tracing the further history of the γ-ray quantum. If the interaction of the γ-ray quantum as the result of the Compton or photoelectric effect occurred in the casing material, we traced the further history of both the scattered γ quantum and the Compton electrons or photoelectrons up until they are absorbed, reflected into the source, or are emitted from the casing. In doing so, we established the number of emitted γ rays, the number of electrons reflected back into the source of casing, and we found their energy and direction. The number of electrons produced as the result of the Compton and photoelectron effects was calculated separately. The histories of the Compton and photoelectric effects were traced by a well-known method [5]. The angle of deviation of the electron from the original direction was calculated from the distribution considered in [6], with a Bethe correction for large angles. For each casing material we considered the history of $1.5 \cdot 10^5$ γ-ray quanta. The calculation was performed on an M-222 computer with a computing program written in FORTRAN.

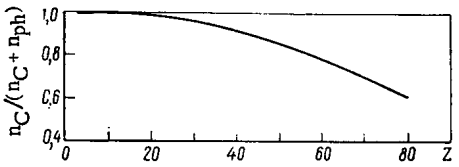

Fig. 1

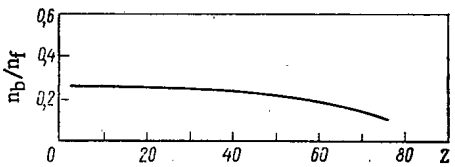

Fig. 2

Fig. 1. Ratio of number of Compton electrons to total stream of electrons as a function of atomic number of casing material.

Fig. 2. Ratio of number of electrons reflected backward (n_b) to number of electrons leaving casing (n_f) as function of atomic number of casing material.

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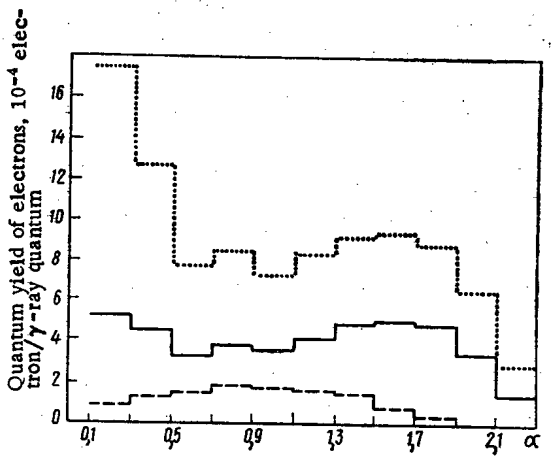

Fig. 3

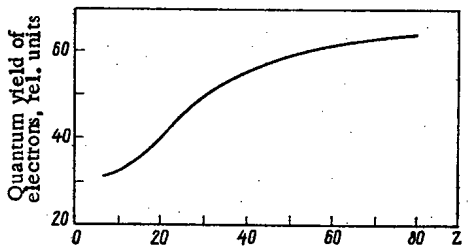

Fig. 4

Fig. 3. Spectral distribution of electrons reflected backward (- - -) and emitted from casing (——); dose distribution of electrons (. . .) leaving source.

Fig. 4. Quantum yield of electrons as function of atomic number of casing material.

It is seen in Fig. 1 that the contribution of the Compton electrons falls off from 100% in low-Z elements to 65% in tungsten.

In the case of a parallel monoenergetic beam of γ rays perpendicularly incident on a flat plate materials with intermediate atomic numbers of 20-40 have the minimum secondary-electron yield [1-3], this being due to the fact that identical streams of secondary electrons are emitted forward and reflected backward.

In Fig. 2 the ratio of the stream of electrons reflected into the source and the casing to the stream of electrons leaving the casing is plotted as a function of the atomic number of the casing material. It is seen that this ratio is practically constant (~ 0.25) for materials with an atomic number of 7-30, although a general increase is observed in the stream of secondary electrons as the atomic number increases. Then this ratio drops to 0.12 for tungsten and the fraction of electrons leaving the casing increases correspondingly.

The spectral distribution of electrons leaving the iron casing and those reflected into the source is shown in Fig. 3. Such spectra were calculated for casings of various materials and were used to obtain the quantum yield of electrons as a function of the atomic number of the casing material, as shown in Fig. 4. It is seen that unlike the case of plane geometry and parallel beam of γ-ray quanta the quantum yield of electrons from the cylindrical casing of an isotropic ^{60}Co γ-ray source increases with the atomic number of the casing material. This is due to the isotropy of the distribution of the radiation from the source, the scattering of γ rays in its material, as well as the cylindrical form of the casing. In view of the isotropy of the radiation from the source the γ-ray quanta move principally in the direction of the axis of the source; this is conducive to an increase in the number of electrons emanating from the casing. As a result of the scattering of γ rays in the material of the source and the casing, low-energy quanta are produced with a larger photoabsorption cross section than that of the primary γ-ray quanta; and because of the cylindrical shape of the casing, in contrast to the case of plane geometry the electrons leaving the casing are in a solid angle greater than 2π.

To verify the correctness of the method of computation and the algorithms used, we compared the results with data obtained by other authors. The plot of the ratio of the number of Compton electrons to the total stream of electrons as a function of the atomic number of the casing material (Fig. 1) match the data of [3] to within ±3%. The character of the relation we obtained giving the total stream of secondary electrons as a function of the atomic number of the casing material (see Fig. 4) is in agreement with the analogous relation given in [7]. The quantum yield of secondary electrons from the casing of a cylindrical source apparently also depends on the energy of the primary γ radiation of the source; this calls for additional investigations.

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EFFICIENCY OF α AND γ RADIATION IN THE FORMATION AND REGENERATION OF E_1' CENTERS IN QUARTZ

L. T. Rakov and B. M. Moiseev

UDC (549:535):549.514.51

Quartz is often used for paleodosimetric determinations associated with prospecting for uranium deposits. The dose of natural radiation is usually estimated from the concentration of E_1' centers. This form of defect consists of an electron captured by an oxygen vacancy. An E_1' center formed in quartz only under the effect of radiation is capable of existing for millions of years. The question of the predominant role of separate forms of ionizing radiation in the formation of E_1' centers in quartz under natural conditions is basic to the use of quartz as a paleodosimeter.

It was noticed earlier that the predominant contribution to the concentration of these defects is usually made by the α-ray background [1]. For limited contact of granulated quartz with radioactive elements, e.g., graphite, the influence of γ-ray background was observed [2]. The present paper is devoted to the study of the efficiency of α and γ radiation in the formation and regeneration of E_1' centers on the basis of the results of experimental irradiation of quartz in the laboratory. The E_1'-concentration $C_{E_1'}$ was measured by the electron-paramagnetic resonance (EPR) technique.

The experiments were carried out on specimens of natural polycrystalline quartz which did not originally have $C_{E_1'}$. These specimens were subjected to artificial α-ray irradiation from an AIP-35 ^{238}Pu source or γ-ray irradiation from ^{60}Co on an "Issledovatel'" (Researcher) facility.

In the case of α-ray irradiation the dose was estimated from the formula

$$D = \int \varepsilon N(\varepsilon)\, d\varepsilon / S\lambda\rho,$$

where N is the number of α particles of energy ε; S is the area of the irradiated surface; λ is the mean depth of α-particle penetration into quartz; and ρ is the density of quartz. The dose of γ-ray irradiation was identified with the exposure from ^{60}Co in the working channel of the facility [3]. The concentration of E_1' centers formed under the effect of the α particles was calculated from

$$C_{E_1'} = n/S\lambda\rho,$$

where n is the number of E_1' centers in the quartz specimen studied. From the relation plotted in Fig. 1 it follows that the values of $C_{E_1'}$ in quartz corresponding to identical α- or γ-ray doses differ appreciably. The efficiency of these radiations in forming E_1' centers can be estimated from the angle of inclination of their dose curves, $C_{E_1'}(D)$ (see Fig. 1). For α rays, $C_{E_1'}/D = n \Big/ \int \varepsilon N(\varepsilon) d\varepsilon \approx 4 \cdot 10^4$ spin/erg. The flux of α particles towards the specimen from the AIP-35 source with an activation of no more than 200 mCi, according to specifications, was $2 \cdot 10^9$ particles/h. For γ radiation, $C_{E_1'}/D \approx 4 \cdot 10^3$ spin/erg. Thus, the values of the efficiency under consideration differ by an order of magnitude. From the estimates given above it follows that α particles with an energy of 4-5 MeV create E_1' centers comparable in number to the number of the particles.

An important question is that of the role of α and γ radiation in the radiative regeneration of these centers. It is known that the process of annealing of E_1' centers takes place in two stages [4]. The first stage is associated with the breakup of the electronic subsystem of the E_1' centers, the loss of unpaired electrons; the second stage, the decay of E_1 precenters, consists of the recombination of oxygen vacancies with oxygen interstitials. The first stage shifts the defects into a stage which cannot be recorded by EPR but does not mean the

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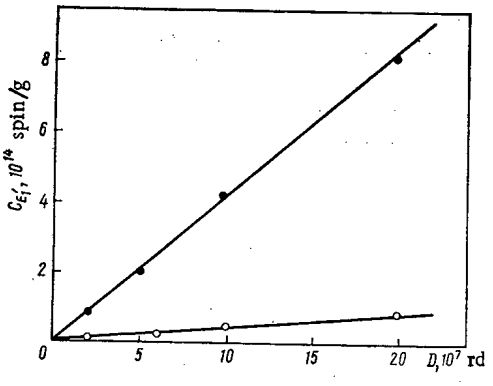

Fig. 1

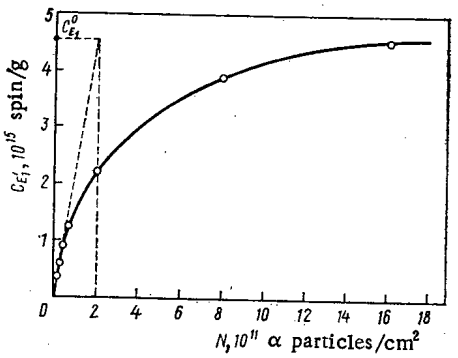

Fig. 2

Fig. 1. Concentration of E_1 centers in quartz as function of dose of α-ray (●) and γ-ray (○) irradiation.

Fig. 2. Concentration of E_1' centers regenerated in quartz annealed at 500°C as function of α-radiation dose: ○) experiment.

total destruction of these defects. In the radiation field, surviving precenters capture electrons and the previous concentration of E_1' centers is restored.

To study the efficiency of α and γ radiation in the regeneration of E_1' centers the quartz specimens with a known concentration of defects were annealed at 450-550°C and subjected to α- or γ-ray irradiation. It was found that regeneration ends at doses of 10^7-$5 \cdot 10^7$ rd under the effect of α particles and 10^5-10^6 rd under γ rays. Probably, in limited regions α particles regenerate E_1' centers less effectively than γ-ray quanta do because of the localization of the energy transferred by these forms of radiation to the crystal. This would explain the high concentration of "unfilled" E_1' precenters in finely dispersed quartz selected from deposits of radioactive element in which quartz had been subjected to the action of α-ray background.

In the present paper we have attempted to experimentally determine the effective cross section σ for the radiative regeneration of E_1' centers of α particles. The region in which one α particle with a free path λ takes defects into the paramagnetic state has a volume of $v = \lambda\sigma$. For N such particles we get $V = N\lambda\sigma$, provided that their regions do not overlap. The latter condition is satisfied at the onset of the regeneration process under the effect of α particles when the number of microvolumes with "filled" E_1 precenters is still small. Then, as this number increases these regions overlap, resulting in a nonlinear dependence of the concentration of regenerated E_1' centers on the radiation dose. Figure 2 illustrates such an experimental relation corresponding to the process whereby defects preserved in the annealed quartz are taken into the paramagnetic state under the effect of α particles. The dose is expressed in terms of the number of α particles which, during the irradiation, are incident upon a unit area of irradiated surface perpendicular to the particle flux. From this relation we can find the minimum number of particles necessary for the regeneration of all E_1' centers in the rectangular volume $V = \lambda$ cm^3 with an irradiated area of 1 cm^2. This number is found by interpolation of the linear segment of the relation to the value $C^0_{E_1}$, the concentration of surviving E_1 precenters (see Fig. 2), and corresponds to idealized irradiation with overlapping regions of regeneration. Therefore, the relation $V = \lambda = N_{min}\lambda\sigma$ is valid. Hence for the minimum number of α particles, roughly equal to $2 \cdot 10^{11}$, the cross section is $\sigma \approx 5 \cdot 10^{-12}$ cm^2. The radius of the effective cross sections is r = 120 Å.

It is interesting to compare the result obtained with l, the mean transverse path traveled in the quartz by electrons produced by the ionization of the surrounding atoms by an α particle. The energy w of these electrons is determined by the energy of the α particles and their angle of emission θ [5]. For ε = 4.4 MeV of an α-ray source which is in the intermediate portion of the spectrum after averaging over θ we obtained $w \approx 1 \cdot 10^3$ eV. According to our estimate, the transverse path length l of electrons with such low energies in quartz is ~ 200 Å [6, 7]. The agreement between l and r as to the order of magnitude can be explained by the high concentration of "free" electrons in the regions of regeneration of E_1 centers, i.e., the high density of ionization caused by the α particle in the track.

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INVERSION PROBES IN GAMMA – GAMMA METHODS

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In the gamma–gamma methods of nuclear geophysics, including isotope-excited fluorescence analysis and the nuclear γ resonance or Mössbauer method, extensive use is made of inversion probes [1-6]. At first the length of these probes was based on experimental investigations and qualitative physical description of the effect [1, 2]. Subsequently, individual aspects of the inversion effect for a homogeneous medium were studied with various mathematical models [3-6].

In the present paper the inversion effect is considered for a laminar medium within a unified mathematical theory; this effect suggests an inversion theory for a homogeneous medium as a special case and the relation between all forms of inversion studied earlier (density [1, 6], geometric [2-5], and q inversion [6]).

Laminar Media. The differential equation for the flux of secondary radiation from a point source into a homogeneous medium (Fig. 1a) can be written as

$$dN = C\frac{\rho}{R_1^2 R_2^2}\exp(-k\rho R/2\cos\varphi)\,dV, \tag{1}$$

where $C \equiv C_{r,s}$ is a constant which is different for x-ray or roentgen (r) radiation and scattered (s) γ rays and does not depend on the properties of the medium; $k \equiv k_0 + k_{r,s}$ is the sum of the mass coefficient of attenuation of the primary (0) and secondary (r, s) radiation; ρ is the density of the medium; and R is the length of the probe.

In the case of a two-layered medium (Fig. 1b) the coefficient k in Eq. (1) must be replaced by its equivalent k_{12} and the density ρ by the density ρ_{12} of the two-layered medium.

Suppose that the primary and secondary radiation is propagated in a bounded solid angle $\Delta\Omega$ (cf. Fig. 1). Then the secondary radiation is produced only in the volume ΔV which is formed by the interaction of two cones. If $R_{1,2} \gg (\Delta V)^{1/3}$, then by the mean-value theorem the solution of Eq. (1) is

$$N = C\frac{\rho_{12}\Delta V}{R_1^2 R_2^2}\exp(-k_{12}\rho_{12}R/2\cos\varphi). \tag{2}$$

The function $N(\rho_{12})$ has a maximum whose coordinates are determined by the equation

$$\partial N(\rho_{12})/\partial\rho_{12} = 0. \tag{3}$$

This is the case of density inversion which has been studied for a homogeneous medium. In the case of a layered medium the coordinates of the inversion point depend on ρ_{12} and k_{12}, i.e., on the properties of the medium and the geometry. The inversion probe $R_{\mu h}$ will be found from Eq. (3):

$$R_{\mu h} = \frac{2\cos\varphi}{\rho_{12}k_{12}} = \frac{2\cos\varphi}{\mu_{12}}, \tag{4}$$

where μ_{12} is the linear coefficient of attenuation of radiation for a two-layered medium.

Using the additivity property of the cross section for the interaction of radiation with matter, we calculate μ_{12} (see Fig. 1b):

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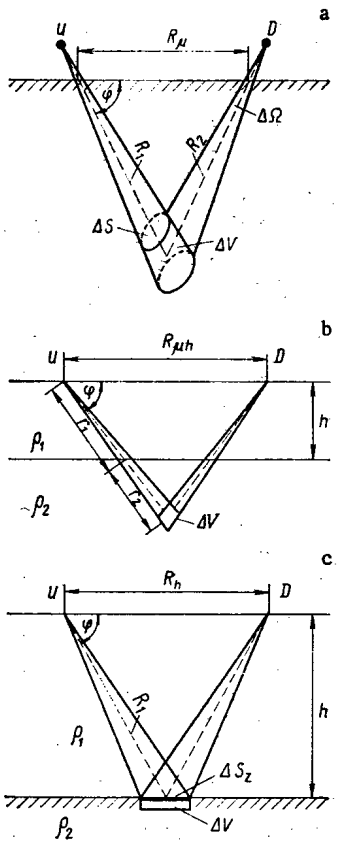

Fig. 1. Geometry: U) source of γ-ray quanta; D) detector.

$$\mu_{12}=\frac{2h}{R}\operatorname{ctg}\varphi(\rho_1k_1-\rho_2k_2)+\rho_2k_2. \tag{5}$$

From Eqs. (4) and (5) we find

$$R_{\mu h}=\frac{2\cos\varphi}{\rho_2k_2}+2h\operatorname{ctg}\varphi\left(1-\frac{\rho_1k_1}{\rho_2k_2}\right) \tag{6}$$

or

$$R_{\mu h}=\frac{2\cos\varphi}{\mu_2}+2h\operatorname{ctg}\varphi\left(1-\frac{\mu_1}{\mu_2}\right), \tag{7}$$

where $\mu_i = \mu_{0i} + \mu_{r,si}$ are the linear coefficients for the respective media.

Equation (7) gives a quantitative description of the effect of inversion through the coordinates of the inversion point. Since the length of the inversion probe with respect to density depends on the values of μ for the media and the geometric h, the general case of inversion may be treated as a (μ, h) inversion.

A two-layered medium can be generalized to the case of a multilayered medium. To do so we rewrite the length of the inversion probe (7) as

$$R_{\mu h}=2L_i\cos\varphi+2hL_i^{\Sigma}\cos\varphi, \tag{8}$$

where $L_i = 1/\mu_i$ is the free path of the quanta in the i-th medium with respect to the combined (primary and secondary) radiation; L_i^{Σ} is the same quantity, but for a layered medium consisting of i layers, and h is total thickness of the upper (i − 1) layers.

The value of L_i^{Σ} can be calculated by the recurrence formula

$$L_i^{\Sigma}=\frac{1}{L_{i-1}^{\Sigma}\sin\varphi}(L_{i-1}^{\Sigma}-L_i). \tag{9}$$

Homogeneous Medium. As h → 0 Eq. (7) gives the inversion coordinates of the considered case of a homogeneous medium:

$$\lim_{h\to 0} R_{\mu h} = R_\mu = 2\cos\varphi/\mu. \tag{10}$$

With a constant composition of the medium, characterized by the effective atomic number Z_{eff}, this case of μ inversion goes over into ρ inversion since for the chosen source of primary radiation $\mu(\rho, Z_{eff})$.

Two-Layered Medium with Weakly Attenuating First Layer. If $\mu_1 \ll \mu_2$ (e.g., $\rho_1 \ll \rho_2$, the case of an air layer between the probe and the medium), then

$$R_{h\mu} \approx 2\cos\varphi/\mu_2 + 2h\operatorname{ctg}\varphi, \tag{11}$$

where the first term on the right-hand side is equal to the inversion probe R_μ for the second medium and the second term expresses the effect of the geometric factor.

With the condition $h \gg L_2 = 1/\mu^2$, when the thickness of the weakly attenuating layer is much greater than the free path of the radiation in the second medium, we have

$$R_{\mu h} \to R_h = 2h\operatorname{ctg}\varphi, \tag{12}$$

i.e., the case of geometric or h inversion.

Equation (12) gives the relation between R_h, h, and φ at the inversion point and is insufficient for determining the quantities which enter into it. To calculate the coordinates of the h inversion we use the equation

$$\partial N/\partial h = 0. \tag{13}$$

The flux N of secondary radiation under the conditions considered here is the solution of Eq. (1) for $k_{12} \approx k_2$, $dV \approx dS\, dr_2$, $R_{1,2} \gg r_2$ (see Fig. 1):

$$N = C\Delta S_Z/R_1^2 R_2^2 k_2, \tag{14}$$

where $\Delta S_Z \approx R_1^2 \Delta\Omega \sin\varphi$ (see Fig. 1c).

Bearing in mind that $\sin\varphi \approx h/R_1$ and $R_1 = R_2 = \sqrt{h^2 + R^2/4}$, from Eq. (13) we find the length of the inversion probe

$$R_h = 2\sqrt{2}\,h = 2.82h. \tag{15}$$

Thus, as follows from comparison of Eqs. (15) and (12), h inversion occurs at a fixed angle:

$$\left.\begin{aligned} \varphi_m &= \operatorname{arcctg}\sqrt{2} = 35°3 \\ \sin\varphi_m &= \frac{1}{\sqrt{3}};\quad \cos\varphi_m = \sqrt{\frac{2}{3}}. \end{aligned}\right\} \tag{16}$$

The solution of Eq. (15) differs from that obtained in [4] because of the differences in the geometric models used for the probe.

q-Inversion. Inversion of the field N_r of x rays has been observed in [6] during variation of the content q of the fluorescing element (q inversion). The study of this case by means of the apparatus developed above shows that q inversion is a special case of μ inversion and is of no independent significance.

Thus, when a field of secondary radiation from a point source is formed in a layered medium a single inversion process is observed and it could be called μ, h inversion.

In the limiting case of a two-layered medium with weakly attenuating upper layer μ, h inversion degenerates into h inversion (geometric inversion). In the other limiting case during the transition from a layered to a homogeneous medium μ, h inversion goes over into μ inversion or ρ, Z_{eff} inversion, of which density ρ inversion is a special case.

The mathematical theory developed enables the coordinates of the inversion point to be determined in both the general case and the special cases considered.

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ACTIVATION OF MOLYBDENUM AND TUNGSTEN IN A CYCLOTRON

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Radioactivation methods of monitoring wear, corrosion, mass transfer, and other processes in products made of molybdenum and tungsten run into some difficulties owing to the lack of natural radioisotopes with long half-lives and high-energy γ rays. It is more convenient to resort to activation with charged particles for this purpose but this does require knowledge of the yield curves for the radioisotopes formed and their distribution over the depth of the active layer [1]. Papers [2-5] have been published on the measurement of the excitation functions of some reactions in these elements, but it is difficult to extract information from them for practical use.

The cyclotron at the Physics and Power Engineering Institute (FÉI, Obninsk), which has a large variety of operating regimes as to type and energy of accelerated particles, was used to measure the yield curves and the depth distribution of the radioisotopes ^{95m}Tc, ^{96}Tc, and ^{97}Ru, as well as ^{183}Re, ^{184}Re, ^{185}W, and ^{185}Os, which are formed when molybdenum and tungsten are irradiated with protons, deuterons, α particles and ^{3}He^{+2} ions with energies of up to 22.5, 23, 46, and 30.2 MeV, respectively.

Specimens and stacks of foil of the metal of natural isotopic composition were irradiated in a special target in an external beam. The irradiation was determined from the activity of ^{65}Zn in calibrated copper monitoring foils with an error of ±10%. The beam current did not exceed 1 μA, and the irradiation time was 1 h. In the case of activation with helium ions, the data on yields were reduced to a current of doubly charged ions.

TABLE 1. Data on Yields on Radioisotopes

Molybdenum activation				Tungsten activation			
particles	particle energy, MeV	isotope	yield, μCi/μA·h	particles	particles energy, MeV	isotope	yield, μ Ci/μA·h
p	22,4±0,2	^{95m}Tc ^{96}Tc	14,0±1,8 660±70	p	22,0±0,2	^{183}Re ^{184}Re	34±5 8±10
d	22,8±0,2	^{95m}Tc ^{96}Tc	7,6±1,0 630±70	d	22,8±0,2	^{183}Re ^{184}Re ^{185}W	45±7 34±5 16±4
α	46,0±0,5	^{95m}Tc ^{96}Tc ^{97}Ru	0,28±0,04 21,5±3,2 105±15	α	46,5±0,5	^{183}Re ^{184}Re ^{185}W ^{185}Os	7,2±1,2 0,32±0,05 2,2±0,5 5,3±0,7
^{3}He^{+2}	30,2±0,3	^{95m}Tc ^{96}Tc ^{97}Ru	0,25±0,03 38,0±4,6 150±20	^{3}He^{+2}	30,2±0,3	^{183}Re ^{184}Re ^{185}W ^{185}Os	2,6±0,4 1,8±0,2 0,6±0,2 0,93±0,13

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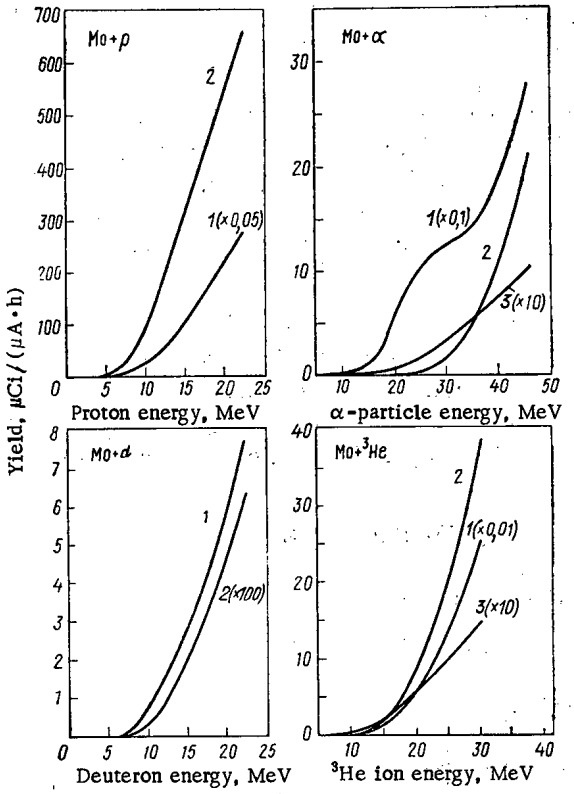

Fig. 1

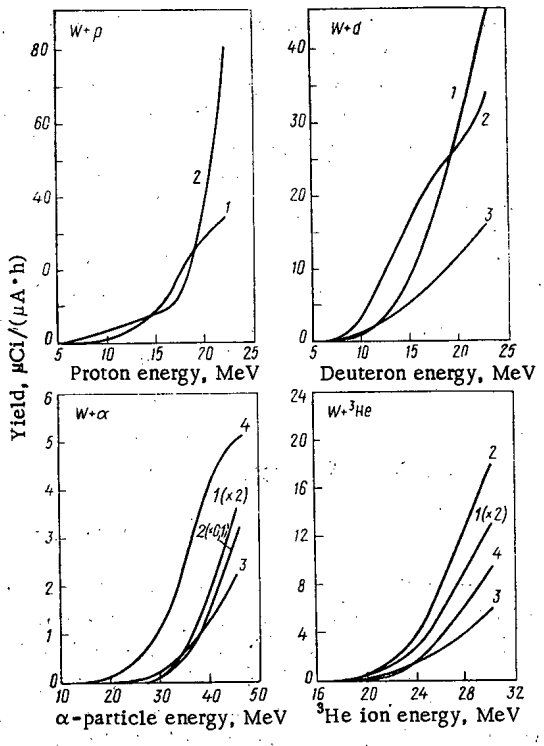

Fig. 2

Fig. 1. Yield of radioisotopes: 1) ^{95m}Tc; 2) ^{96}Tc; and 3) ^{97}Ru from molybdenum as function of particle energy.

Fig. 2. Yield of radioisotopes: 1) ^{183}Re; 2) ^{184}Re; 3) ^{185}W; and 4) ^{185}Os from tungsten as function of particle energy.

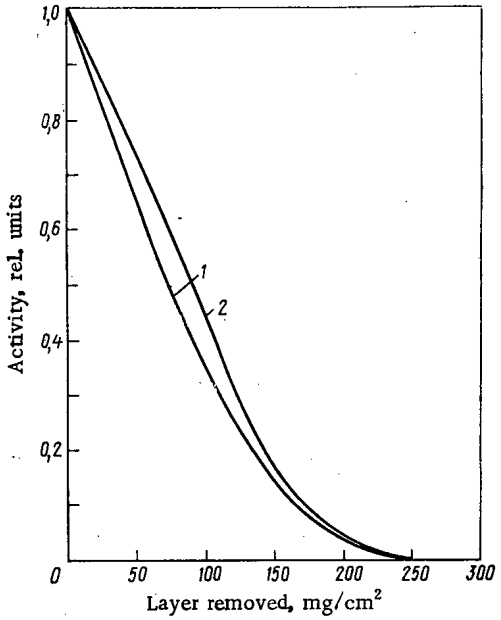

Fig. 3. Distribution of 1) ^{95m}Tc from molybdenum and 2) ^{184}Re from tungsten with depth of materials under activation by 11.5-MeV protons.

The activity was measured with a γ spectroscope, both of the scintillation type based on a 70 × 70-mm NaI(Tl) crystal and one of the semiconductor type with 40 cm^3 of Ge(Li). The multichannel analyzers used were AI-256 and LP-4840 instruments. The relative change in the activity with respect to the stack of foil was measured in a one-channel spectrometer both from the count in the window of a discriminator which cut out the region of the γ-ray spectrum which contained the photopeaks characteristic of the given radioisotope and from the area of these peaks with a correction for the Compton scattering contribution from harder γ radiators. The error of the relative change does not exceed 3%. The data on the quantum yields of characteristic γ-ray lines have been taken from [6]. The activity of ^{185}W was measured in a Protok (Channel) instrument with a 4π-proportional counter and with allowance for the self-attenuation of the β radiation. In this case the error of the absolute data was ±25%.

The results are given in Figs. 1 and 2 and Table 1. The errors given do not take account of the inaccuracies in the decay schemes adopted in handling the results of the measurements. The purpose of this paper was not to make an exhaustive study of the activation of molybdenum and tungsten; therefore, results are given only for long-lived radioisotopes with the largest contributions to the total activity of the radioisotopic mixture. The ^{92}Nb and ^{95}Nb yields from molybdenum and ^{184m}Re yield from tungsten are small and are of no practical interest.

The authors thank L. V. Ognev for his assistance in analyzing the results.

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COMECON CHRONICLES

COOPERATION DIARY

The eighth meeting of the Council of Scientific-Technical Cooperation on Research Reactors of the COMECON Atomic Energy Subcommittee was held from Aug. 29 to Sept. 2, 1977, in Dresden (German Democratic Republic). The delegates discussed a paper by specialists of the Joint Institute for Nuclear Research (JINR, Dubna, USSR) on the IBR-2 periodic research reactor for neutron research and a paper by the Polish delegation on the use of the Marysia research reactor. The delegates considered the participation of research centers of the member-countries concerned in producing experimental equipment for the IBR-2 and the use of the reactors for research on nuclear physics. The meeting also considered the elaboration of regulations for ensuring nuclear safety during work with research reactors and critical assemblies; draft regulations presented by the Soviet delegation were adopted as a basis.

The meeting discussed a report by the German Democratic Republic on the scientific and engineering work done in 1973-1975 to improve the parameters and operation of the RFR reactor; proposals concerning cooperation between COMECON and IAEA in the domain of research reactors were considered. The meeting agreed upon draft recommendations concerning the requirements on the level of training of personnel for reactor engineering and nuclear power engineering. The delegates discussed information about the course of work on some aspects of the "Development and improvement of nuclear research reactors and reactor physics and engineering work on such reactors" and the specifics of plans for cooperation on these subjects were worked out.

The meeting agreed upon a draft plan of work by the COMECON Subcommittee on Atomic Energy in 1978-1979 in the realm of nuclear research reactors.

The 14th meeting of the Council of Scientific-Technical Cooperation on Radiation Technique of the COMECON Atomic Energy Subcommittee, was held in Warsaw (Poland) Sept. 13-16, 1977. It discussed and agreed upon a draft plan of work and a timetable of measures to be taken by the Subcommittee on Atomic Energy in 1978-1979 in the field of radiation energy; drafts of a program of standardization on "Radiation Technique" (1979-1982) and a plan of the Subcommittee's work in this field in 1978; a draft standard on "Static and Rotating Gamma-Therapeutic Apparatuses for Long-Distance Irradiation. General Technical Requirements;" and the agenda for the 15th meeting.

The Council of Scientific-Technical Cooperation on Radiation Technique discussed information and proposals concerning the drafting of methodological recommendations on the dosimetry of radiation-technological facilities with electron accelerators. It also considered specialization and cooperation on the development and use of radiation technique as well as possible areas of application of radiation technique in the national economy of the Republic of Cuba.

The meeting heard information about the preparation for print of the first collection of unified standardizing documents and regulations on the radiation sterilization of medical supplies and also about the results of the work on the Symposium on Radiation-Chemical Modification of Polymer Materials. The Commission approved scientific and technical undertakings in conjunction with the working agencies of other COMECON permanent commissions and requested that the Council of Scientific and Technical Cooperation on the Chemical Industry provide it with ideas about the development of radiation technique for carrying out the processes of radiation-chemical modification of polymer materials.

The tenth meeting of the Council of Scientific-Technical Cooperation on Radiation Safety of the COMECON Subcommittee on Atomic Energy which was held in Warsaw (Poland) Sept. 20-23 (1977) considered proposals of the Republic of Cuba for a plan of accelerated development of science and engineering of the country in the domain of radiation safety and approved a draft program and the dates for an expedition to study the radioactivity of the Danube.

The Council discussed proposals for unified criteria for methods of measuring and monitoring the radioactive contamination of the environment around atomic power plants in boundary zones and near international bodies of water; discussed the procedure for presentation of urgent information to neighboring countries in the

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case of radiation accidents; approved proposals for further cooperation and outlined the order of the plan of work and the timetable of undertakings by the COMECON Subcommittee on Atomic Energy in the domain of radiation safety in 1978-1979.

The 12th meetings of the Council of Scientific and Technical Cooperation on Reprocessing of Irradiated Atomic Power Plant Fuel (of the COMECON Subcommittee on Atomic Energy) and a meeting of specialists on the development of methods and instruments for monitoring and controlling technological processes for reprocessing spent fuel elements were held in Klučinice (Czechoslovakia) on Sept. 20-23, 1977.

The participants in the meeting considered and approved a report by the Council, including the results of the Fourth Symposium on the Reprocessing of Irradiated Fuel and proposals for further collaboration. The Council has successfully coordinated research in Bulgaria, Czechoslovakia, the German Democratic Republic, Hungary, Poland, and Rumania, and the Soviet Union on the reprocessing of irradiated fuel. The form of the cooperation has been improved continually, plans of work have been made specific, and the technical problems for the work done in the various countries have been agreed upon, thus making it possible to concentrate attention on the most important problems and to eliminate parallelism.

Regulations are being drawn up for the safe transportation of spent fuel from atomic power plants of the COMECON member-countries. On the basis of the principles of cooperation between the scientific-research organizations of six countries an experiment is being conducted on the comparison of results of the determination of the components in spent fuel from atomic power plants.

The Council discussed the participation of the countries in work on finding materials for transportation containers, elaborated working plans on the problem for the period up to 1980, and drafted a plan of work by the COMECON Subcommittee on Atomic Energy in 1978 and 1979 on the reprocessing of irradiated fuel from atomic power plants.

The meeting of specialists considered the general principles for the construction of an automated system for analytical monitoring of the technological process of regeneration of atomic power plant fuel, the possibility of using small computers and the CAMAC system for processing analytical monitoring data, the state of the work, and the prospects for producing models of instruments for long-range determination of uranium and discharge solutions from the reprocessing of nuclear fuel from a water-moderated–water-cooled power reactor (VVÉR).

The 25th meeting of the Working Group on Reactor Science and Engineering and Nuclear Engineering of the COMECON Subcommittee on Atomic Energy was held from Sept. 27 to 30, 1977, in Poznan (Poland). The meeting considered the preparation of working plans for cooperation in selected aspects of the topic "Creation and Mastery of Energy Blocks with 100-MW Water-Moderated–Water-Cooled Reactors" in the 1976-1980 plan; the work on these aspects is to be carried out by dividing it up in an agreement.

The Working Group made a favorable assessment of the work done by specialists of the USSR, the German Democratic Republic, and Poland on the preparation and approval of plans and agreements on two topics out of eleven, selected for implementation on the basis of an agreement. Note was taken of the work of specialists of the German Democratic Republic on the preparation of materials for an agreement on joint work on the "Development of Systems for the Acquisition and Processing of Information about the Operating State of a Power Reactor."

The meeting discussed and approved programs presented by Polish and Czechoslovak specialists for the development of technical projects of vertical steam generators for atomic power plants. A plan of work was approved for the COMECON Subcommittee on Atomic Energy for 1978-1979 in the realm of reactor science and engineering and nuclear engineering and plans for collaboration on some topics were made more specific.

The Working Group discussed the programs of a meeting by specialists on the analysis of raw data and the use of mathematical models to predict the development of nuclear power and the seminar "Research on the Thermal Loads in Bundles of Rods under Steady-State and Non-Steady-State Modes of Heat Exchange."

Responsible representatives in the domain of isotope production met in Tbilisi (USSR) on Oct. 3-6, 1977. At their meeting they considered proposals from the Republic of Cuba on the accelerated development of science and technology in the domain of isotopes and tagged compounds, discussed the results of the IAEA–COMECON meeting held in Leipzig (German Democratic Republic) on Feb. 14-18, 1977, and approved a draft recommendation about further specialization by Rumania and the USSR in the production of 62 compounds tagged with ^{13}C.

The national representatives informed the meeting about the course of the fulfillment of the plan of work by the COMECON Subcommittee on Atomic Energy, plans of work on standardization and scientific and technical collaboration on isotopes and tagged compounds in 1977.

The meeting approved proposals to draft plans of work for the COMECON Subcommittee on Atomic Energy for 1978 as well as a program of work on standardization in 1979-1981 (second round) on "Isotopes, Tagged, Compounds, and Closed Sources of Radiation," providing for the elaboration of five COMECON standards during this period. At the same time, the meeting discussed and approved proposals for the draft timetable of work on isotopes in 1978-1980.

The 13th meeting of the Council of Scientific and Technical Cooperation on Radioactive Waste and Deactivation (of the COMECON Subcommittee on Atomic Energy) was held in Moscow on Oct. 4-7, 1977. It considered the characteristics of various forms of liquid and solid radioactive waste formed in atomic power plants with water-moderation—water-cooled power reactors and formulated a tentative classification of liquid wastes according to the features of their separate reprocessing.

The meeting summed up the work done in 1976-1977 to develop methods and means of eliminating radioactive aerosols and gases from atmospheric discharges (including the joint sorption of radionuclides of krypton and xenon, trapping radioactive inert gases with membranes) and to determine processes by which radioactive iodine enters the atmospheric phase, and the behavior of inorganic forms of iodine in the loop of pressurized-water reactors. The meeting discussed the effect the physicochemical characteristics of mist-forming liquids on the resistance of fine-fibered filters in a self-purification mode, unified methods for testing filters, as well as investigations relating to the development of systems for purification of gas and air discharges which arise in processes of denitration, calcination, and solidification of highly active waste. The large body of research done in this area by the COMECON member-countries is of considerable practical interest. It has been recommended that the results of the work be used in the development and design of gas-purifying systems for atomic power plants.

The Council discussed the plan of work for 1978-1979 on radioactive waste and deactivation, filled in details in working plans on scientific and technical cooperation, and considered organizational topics pertaining to the unification of the efforts of the COMECON member-countries on the most important topics.

The fourth meeting of the plenipotentiary parties to the Agreement on Multilateral International Specialization and Cooperation in Isotope Production was held in Tbilisi (USSR) on Oct. 7-8, 1977. The meeting noted that the obligation to meet the demand of the countries for specialized isotope production is being fulfilled in regard to volume and technical parameters. Some countries have stopped making products in which other countries specialize and satisfy their own demand for these products by importation. Work is being continued on increasing the quality of the products and reducing delivery times.

The parties to the agreement approved proposals for extending the nomenclature for specialized products (especially compounds tagged with ^{13}C), refined the technical parameters and characteristics of some products, and exchanged information about the tentative demands for products in 1978-1979.

The 12th meeting of the Council of Scientific—Technical Cooperation on Fast Reactors (of the COMECON Subcommittee on Atomic Energy) was held in Obninsk (USSR) on Oct. 11-14, 1977. The meeting considered the reports of the delegations from the member-countries on the work done in the period from Sept. 1976 to Aug. 1977, according to the provisions of plans for "Research on Fast-Neutron Power Reactors," discussed and approved a plan of work in 1978-1979 on some fast-neutron reactors, and considered the joint construction and use of an experimental base as well as the course of the preparation of a catalog of experimental stands and critical assemblies.

The Council heard information from specialists on the possible participation of countries in work on the construction of a BN-1600 facility and, in conjunction with this, considered and agreed upon a list of immediate tasks. The Council also discussed proposals submitted by the Soviet Union delegation for technical and economic research on individual systems and designs of the BN-1600.

The meeting considered the course of preparations for agreements on some selected aspects of the topic "Scientific Research and Design Work on the Creation and Mastery of High-Power Fast-Neutron Reactors" incorporated in the agreed plan of multilateral integrated undertakings for 1976-1980. It was noted that specialists of the German Democratic Republic had drawn up two draft agreements and sent them to the USSR for approval and the Czechoslovak delegation is at present completing the preparation of another agreement. The Council discussed the interest of countries in the training of specialists for the operation of atomic power plants with fast reactors as well as in work on the technology of sodium coolant.

The Council heard a report by the Czechoslovak delegation on preparation for a meeting on "The Thermal Physics and the Hydrodynamics of the Active Core and Steam Generators for Fast Reactors," which is to be held in Marianske Lazne (Czechoslovakia) in Apr. 1978.

BOOK REVIEWS

P. Zweifel

REACTOR PHYSICS*

Reviewed by V. I. Pushkarev

The book was written by an eminent American specialist on the theory of nuclear reactors on the basis of lecture notes from courses he has given in several universities. Evidently this basis predetermined the exposition of the material, with considerable space being devoted to a physical treatment and the establishment of numerous problems which arise in the design of the physics of a reactor facility and not to the detailed study of any particular problem of reactor theory. It must be noted that the author has succeeded in finding an exact relation between the physical explanation of a phenomenon and its theoretical description so that the exposition is fully balanced. In this combination lies one of the principal merits of the book. Today, when designing the physics of a reactor sometimes boils down to using numerous computer programs whose physical essence is concealed behind involved mathematics, the publication of books of this type should be welcomed.

The book consists of ten chapters and six appendices. Each chapter is provided with a list of references which the reader can use to get more detailed information about a specific topic of interest to him. Review problems suited to the contents of the chapter are also given.

For greater clarity, the discussion of the main part of the material presented in the book is based on one-group theory: in this approximation the first seven chapters describe the distribution of neutrons, the criticality of the reactor, and variations in the reactivity with time. On the whole, reliable results could scarcely be expected from a one-group description. However, it does give an understanding of the principal processes and this makes it easy to go over to the multigroup scheme used in practice. Therefore, the next chapters extend the results to multigroup theory and discuss the methods of determining the constants which should be used in multigroup calculations. The final chapter describes the theory as applied to concrete types of reactors: fast or thermal reactors, reactors with high or low fuel enrichment, and reactors with light-water or heavy-water moderator.

The purpose of the book, according to the author, was to describe physical methods for the calculations carried out to substantiate a reactor project. However, they are considered fully enough to illuminate all problems which arise when designing reactors in general and power reactors in particular. Insufficient emphasis has been put on such aspects as the calculation of the reactivity coefficients, which in many respects determine the operating conditions of reactors, and the stability of the field of energy release in the reactor. Some inaccuracies can also be noted: on p. 168 the author asserts that the maximum xenon poisoning is 2.4%, although in reactors with a high neutron flux this value may be higher.

Notwithstanding the deficiencies mentioned, this book by P. Zweifel is extremely useful for specialists engaged in the design of the physics of reactors and, in particular, may serve as an introductory course to the fundamental book by Bell and Glasstone, "Nuclear Reactor Theory."

*Atomizdat, Moscow (1977).

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CONFERENCES AND MEETINGS

SESSION OF SECTION OF PHYSICOTECHNICAL PROBLEMS OF POWER ENGINEERING, ACADEMY OF SCIENCES OF THE USSR

Yu. Klimov

The Session, which was devoted to the 60th anniversary of the Great October Socialist Revolution, was attended by scientists and specialists in power engineering of related fields and responsible representatives of ministries and departments, in addition to invited heads of institutes and enterprises.

The Session heard two papers: "The principal directions of development of the fuel and energy balance in the USSR," by Academician M. A. Styrikovich and "Nuclear power engineering and the scientific and engineering problems in its development," by Academician N. A. Dollezhal. The paper by M. A. Styrikovich characterized the present state of the fuel and energy situation in the world. Clearly a turning point has been reached, this being explained firstly by a real danger that the reserves of high-grade organic fuel, especially cheap oil, will be exhausted and, secondly, by the heightened attention to protection of the environment and the interaction of the energy industry with the environment. Consequently the energy industry as a whole is compelled to turn to more capital-intensive solutions, as a result of which the cost of the useful energy rises. The important thing is that this rise is apparently irreversible. Once again there is increased interest in coal but its use is being concentrated in power plants operating in the load base; moreover, the coal must be purified from sulfur (for ecological considerations).

Nuclear power has reached the level of economic competitiveness. An important problem is that of heat supplies. The use of atomic energy for this purpose makes it possible to save scarce high-grade fuel. However, the complication arises from the fact that technicoeconomically they are essentially centralized sources of heat whereas supplies of heat to numerous existing and prospective consumers is limited by the possibilities of transporting the heat and the degree of concentration of the heat demand.

In his paper, M. A. Styrikovich pointed out the difficulties encountered in solving the problem of satisfying varying demands for heat and electricity and spoke of some technical possibilities (constructing gas storage facilities, pumping compressed air into underground cavities, etc.). Finally, he took up the ecological problems of power engineering. The most pressing problem at the present time, in his opinion, was not that of heat discharged from generating facilities but the discharges of noxious gases into the atmosphere, especially exhaust fumes from cars in cities. He also spoke of measures and technical solutions to reduce the ecological damage from discharges (e.g., catalytic afterburning of exhaust fumes).

The paper by N. A. Dollezhal discussed the state and features of the present stage, as well as the further development of nuclear power. As is known, the program of construction of atomic power plants in the USSR is based on a limited number of channel- and vessel-type reactors, the design of the former being justifiably referred to as a Soviet, national design.

At the present time, large commercial atomic power plant blocks with a unit power of 1 MW and 1.5 MW, based on reactors of both types, are in operation or under construction. The possibilities for technical and economic improvements in them are far from exhausted. Such possibilities include raising the efficiency, increasing the burn-up fraction, improving control and adjustment, increasing the energy intensity of the reactor core, etc. Some of these, which have come to light in experiments and in the operation of atomic power plants, are already being realized. An example is the transition from the construction of 1-MW blocks with channel-type reactors to 1.5-MW block without increasing the number and size of the channels in the reactor but merely by intensifying the heat removal.

N. A. Dollezhal presented some of the principal ideas concerning the demand for and possibilities of using reactors in a variable load schedule as well as supplies of heat from them. In the case of the construction of atomic boilers and in the operation of atomic power plants in a half-peak regime the problem lies in ensuring the efficiency of the fuel elements, allowing frequent and rapid changes in load. Speaking of the use

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of nuclear reactors to generate heat for energy-consuming technological processes, he noted that in this case it is necessary to attain temperatures of ~1000-1100°C.

N. A. Dollezhal dealt especially with the safety of atomic power plants and the consequences of their reliability. He cast some light on several aspects of this topic. One of these was the quality of the production of equipment of the so-called nuclear class. Another aspect was the reliability which is determined by the operating conditions and by unforeseen operating conditions arising under changing loads. In this case the reliability of the equipment involves ensuring the brittle and plastic fracture strength of the metal. It is equally important to develop methods of early identification of a failure, based on the fact that a failure does not develop instantaneously.

In conclusion, N. A. Dollezhal discussed the ecological and economic aspects and problems associated with the further growth of installed nuclear power capacity. The solution of some problems may be facilitated by the gradual revamping of the strategy for the siting of atomic power plants and by adoption of a fuel cycle that goes over from a dispersed cycle to the construction of large nuclear energy complexes.

N. A. Dollezhal dealt with many topics, at times in the form of a concise exposition of one problem in power engineering or another.

On the whole, the session quite fully reflected the enormous successes of Soviet power industry and its new branch, nuclear power, during the 60 years of Soviet power.

FIRST ALL-UNION CONFERENCE ON THE SCIENTIFIC – ENGINEERING FOUNDATIONS OF WASTE-FREE PRODUCTION

V. N. Senin

In solving the problem of protection of the environment, it is of decisive importance to revamp the traditional technological processes and methods in such a way as to result in the creation of low-waste, and practically waste-free, production with high technical and economic indices and to be conducive to the fullest and most rational use of natural resources.

Individual principles and methods of waste-free technology have found concrete expression in industry for obtaining raw material and processing the materials for the nuclear power industry. A continuous spread of progressive low-waste technological elements has been observed in other branches of industry as well; each branch at present has an adequate scientific and engineering foundation for developing and introducing processes which exclude environmental pollution.

The development of no-waste production was the subject of a conference held June 21-23, 1977, at the Chernogolovka Scientific Center of the Academy of Sciences of the USSR. It was attended by representatives of the Academy of Sciences of the USSR, of the Academies of Sciences of the Union Republics, and of higher schools of learning. A total of 91 papers out of the 156 submitted were read at the plenary meetings in eight sections.

The opening address to the conference of specialists was delivered by Academician N. N. Semenov. He presented the history of the problem of mineral resources, the contribution of Soviet scientists to the development of basic research and applied disciplines for creating industrial systems with minimum waste.

A paper on the scientific–engineering base of waste-free production was read at a plenary session by Academician B. N. Laskorin, who generalized the principal directions in the development of industrial processes eliminating environmental pollution.

As shown by the material of the conference, experience has been accumulated in the Soviet Union in the creation of systems with a minimum effect on the environment and in the rational use of natural and secondary

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resources as well as waste from industrial production. Analysis has shown that there are general principles for creating technological systems for individual factories and industrial regions as a whole; the concrete embodiment of these principles should take account of the specifics of the particular production and the ecological condition of the region, including the climatic and natural features. These include:

understanding of the maximum admissible discharges for each substance and region;

repeated use of substances which are not turned into the finished product but are necessary for the technological processes, e.g., water and air;

creation of energy and technological complexes which made maximum use of all sources of heat (internal and external) and do not exert any influence on the environment;

rational use of constituent components of the natural raw material, including overburden, tailings, etc.

The nuclear industry has made widespread use of technological processes which make it possible to extract from the mineral raw material components which are present in quantities expressed in fractions of one percent and in the final account to obtain a product with a high degree of purity. The separation of isotopes and reprocessing of irradiated materials is no less complicated. It was here that extensive use was first made of such technological procedures as sorption, extraction, and ultrafiltration. Great successes were achieved in the development of high-strength, corrosion-resistant materials and equipment made of them, capable of working in highly corrosive gases, liquids, and melts, often at a high temperature.

Closed-loop water supply systems have some into widespread use in many enterprises of the atomic, chemical, oil, and oil refining industries and ferrous and nonferrous metallurgy. In the future many branches of industry plan to put into service facilities which will operate entirely on a closed-loop basis without any discharges of polluted effluents into bodies of water. The water intake required in this case is determined only by the water consumption for chemical conversion and natural evaporation.

An inseparable element of effluentless systems is local purification and a plant for the extraction and utilization of valuable components from effluents. For example, aluminum plants in operation are equipped with, or are building, regeneration sections and departments to regenerate cryolite from gas scrubbing solutions and carbon "froth" from the electrolyzers; some plants of the atomic and chemical industries and ferrous and nonferrous metallurgy have been successfully using sorption methods for this purpose, especially with ion-exchange resins and fiber filters, extraction, electrodialysis, etc. Many plants have been provided with installations for biological, mechanical, chemical, and physicochemical purification which reduce the content of pollutants in the effluents to practically safe levels before discharge into bodies of water.

In the domain of the protection of the atmosphere against industrial pollution fundamental problems arise in the construction of power complexes for processing natural fuel as well as in production which is accompanied by the formation of various gaseous products such as sulfur compounds, nitrogen oxides, and compounds containing fluorine. The main direction being taken in the development of power engineering is that of creating waste-free production based on the use of low-sulfur fuel, pregasification of liquid fuel, and introduction of highly efficient facilities for trapping and utilizing gaseous waste.

Much attention is being paid to the development of technological systems for the comprehensive processing of a raw material. The creation of such complexes presupposes new principles for constructing technologies under which unusual interrelationships arise between individual plants, ways of obtaining and using energy and mineral resources, including water and air, and the quality and nomenclature of the products.

The waste-free processing of lead–zinc minerals is based on the KIVTsÉT process developed for a wide range of polymetal concentrates and industrial products (copper, copper–zinc, copper–nickel, nickel, copper-tin, etc.).

One of the promising directions for practically waste-free production of aluminum and aluminum alloys, omitting the hydrochemical stage, is that of electrothermal processes based on energy-technological principles. By this method, the reduction of a melt of the starting mixture with a carbon reducing agent in large ore furnaces results in a crude aluminum–silicon alloy from which pure aluminum is then extracted; the silicon residue, containing iron, titanium, vanadium, and chromium, can be used as a composite deoxidizing agent for steel production.

In Soviet national economic practice the concept of "production waste" is giving way increasingly to the concept of "secondary mineral resources" and many branches of industry are switching, on an ever-increasing scale, to production based on such secondary resources. For example, fluorine-containing waste from mineral beneficiation is today generally used to produce fluorine products which are used in the atomic industry,

ferrous metallurgy, glassmaking, etc. In turn, the waste from these industries are used to produce mineral beneficiation sulfuric acid, and other inorganic products.

In a special role in the development of waste-free technology, especially in large-tonnage industry, is played by those technological processes which make it possible, alongside the main production, to process waste into nontoxic materials for long-term use, e.g., construction materials.

Much attention in the work of the conference was paid to the economics, planning, and information, as well as to the socioeconomic aspects of the creation of territorial-industrial complexes on the basis of waste-free technology.

The conference worked out recommendations for the acceleration introduction of low-waste and waste-free technological processes and schemes into the principal branches of the national economy.

FIRST ALL-UNION CONFERENCE ON THE ANALYTICAL CHEMISTRY OF RADIOACTIVE ELEMENTS

B. F. Myasoedov, A. V. Davydov,
and N. P. Molochnikova

The Conference, which took place in Moscow from Sept. 26 to 28, 1977, was attended by some 300 specialists representing scientific research institutes of Academies and Branches, higher schools of learning, and industrial laboratories, as well as scientists from the socialist countries.

The need to organize and hold such a conference stemmed from the broad development of work on monitoring various processes of the nuclear fuel cycle, the need to generalize the results obtained, to point out the problems for further investigation, and to map out ways of solving them. Some 90 papers were read at the plenary sessions and in six sections.

The first plenary session heard review papers on the analytical chemistry of natural radioactive elements: radium (Yu. V. Dubasov), actinium (Z. K. Karalova), protactinium (A. V. Davydov), astatine (V. A. Khalkin), francium and promethium (A. K. Lavrukhina), and technetium (A. F. Kuzina). These papers generalized the results of the development of new methods of extracting, separating, and determining these elements. Considerable advances have been made in the study of technetium and protactinium: many new solid compounds of the elements have been synthesized and their properties studied, efficient extraction methods have been developed, as have been selective, highly sensitive methods of determination. Interesting results have been obtained in the study of the properties of astatine and actinium.

The main trends in the analytical chemistry of radioactive elements are associated with the development of methods for separation and determination of these elements.

Methods of separating radioactive elements were discussed in 26 papers, most of which dealt with the use of extraction, extractive chromatography, and separation in the gaseous phase. Thus, a paper by G. V. Korpusov and Yu. S. Krylov evaluated the possibilities of employing various extraction methods for separating and identifying individual radioactive elements for analytical purposes. Interest was aroused by a paper by I. Krtil, who considered methods of separating some fission products of uranium and plutonium; these methods have been used at the Central Monitoring Laboratory of the Institute of Nuclear Research at Řež (Czechoslovakia) to determine the degree of burn-up of nuclear fuel.

The conditions for the extractive isolation of actinium, americium, and europium from alkaline media with quaternary ammonium bases were considered by V. V. Nekrasov et al. A number of papers described methods of selective isolation and separation of individual radioactive isotopes, including cases when they are in solutions of complex composition: transplutonium elements (V. M. Nokolaev et al.), americium and curium (N. G. Yakovlev et al.), berkelium and cerium (D. A. Malikov et al.), rhodium and palladium (N. Gorski and B. Gorski, German Democratic Republic), and technetium and molybdenum (V. A. Yatsenko and V. V. Bagreev).

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Extractive chromatography has been used successfully to separate actinide elements (V. P. Shvedov et al.), quantitative extraction of thorium, uranium, neptunium, and plutonium from complex radiochemical mixtures (V. N. Ushatskii et al.), concentration of plutonium (N. I. Gusev and E. I. Balashova), etc. Some papers were devoted to the separation of radioactive elements in the gaseous phase, including the isolation of ultramicroquantities of spallogenic radionuclides (M. Adilbish et al., G. Yu. Baier et al.). The conference heard with interest a communication by M. P. Bolynets and M. S. Milyukovoi about the isolation and separation of actinite elements by thin-film chromatography.

Considerable successes have been achieved in the development of new methods for the determination of radioisotope elements, including chemical, physicochemical, and nuclear-physical methods. The basic trend in the development of these methods is that of increasing the accuracy of determination of radioactive elements (0.01%) and sensitivity (fractions of a microgram) while maintaining an acceptable accuracy, which is particularly important in the analysis of especially valuable and highly radioactive substances. Particular attention was devoted to the application of automated systems for analysis and analytical control of various technological processes which, in a number of cases, make it possible to increase the efficiency of control of technological processes and to cut the costs of analytical control. Thus, papers by A. G. Rykov and G. A. Timofeeva reported on analytical methods of control of technological processes of isolation and transplutonium elements, G. A. Akopova et al. dealt with the development of methods of analytical control of extractive technologies for the regeneration of fuel from atomic power plants, and V. Ya. Gabeskiriya et al. reported on the physicochemical determination of the composition of spent fuel from a water-moderated–water-cooled reactor. J. Moravec described the determination of uranium and plutonium in the laboratory of the Institute of Nuclear Research at Ržež.

The application of nuclear-physical methods is especially promising for automated systems of analysis. At its session the conference heard papers about x-ray fluorescence determination of uranium and transuranium elements in highly radioactive solutions (A. A. Gavrish et al. and V. V. Berdikov et al.) and about mass-spectrometric determination of uranium, plutonium, and transuranium and rare-earth elements, as well as the isotopic ratio of plutonium (F. Sus et al., Ržež; V. Ya. Gabeskiriya et al.; G. M. Kukavadze et al.; and V. S. Prokopenko et al.). A large number of papers was devoted to γ-spectrometric analysis. Thus papers by F. E. Hoffman et al. described automated mass-spectrometric laboratory control of the technological process of extractive reprocessing of fuel elements, whereas V. A. Pchelkin et al. discussed an automated system of analysis for radioactive materials. T. Dragnev (Bulgaria) discussed new nuclear-physical methods of determining the isotopic composition and content of thorium, uranium, and plutonium and a γ-spectrometric method of determining plutonium in solid waste.

Electrochemical (especially coulometric) and spectrophotometric methods have been of great importance for the control of technological processes up to the present time. Papers by I. S. Sklyarenko et al. and by A. Ya. Kuperman et al. reported on the determination of actinide elements in nuclear fuel by the coulometric method. A precision potentiometric method of determining milligram quantities of uranium was discussed in a paper by M. V. Ryzhinskii et al. The differential-spectrophotometric determination of uranium, thorium, and plutonium by using arsenazo-III was described in a paper by S. A. Nikitina et al.

To develop new methods of determining radioactive elements and improving existing methods it is important to know the oxidation–reduction properties of the elements, the state of the ions in aqueous and organic solutions, tendency to form complexes, and the directions and rates of radiation-chemical processes. Papers read at two special section meetings showed that such research is being conducted at quite a high theoretical and experimental level.

A paper by A. A. Baranov et al. was devoted to the study of the oxidation–reduction properties of the systems Bk(IV)/Bk(III) and Ce(IV)/Ce(III). Some papers presented the results of research on the conditions of oxidation of Am(III) and stability of Am(IV) and Am(VI) in solutions of potassium phosphorus tungstate (E. A. Erin et al.), stability of Am(V) and Am(VI) in solutions of sodium peroxydisulfate and bromate (V. A. Ermakov and A. A. Frolov), and the effect of ionizing radiation on the stabilization of valence forms of actinide elements in solutions (M. V. Vladimorova et al.). The conference heard with interest reports on the application of NMR to the study of the state of Np(IV) and Np(VI) ions (V. A. Glebov et al.), on new chemiluminescent oxidation–reduction reactions of the uranyl ion (V. P. Kazakov et al.), as well as on the obtaining and properties of astatine and perastatine ions (R. Dreier et al., German Democratic Republic).

The use of the kinetics of dissociation and the oxidation–reduction reactions of transuranium elements with 1,2-diamino cyclohexane tetracetic acid for analytical purposes was considered in a paper by T. P. Makarova et al. The use of heteropolyanions for masking impurities in extraction and photometric determination of hexavalent actinides was reported by V. P. Shilov.

Research on the association of quaternary ammonium and phosphonium salts was the subject of a paper by I. Paligorich and I. Gal (Yugoslavia). A paper by V. I. Spitsyn et al. reported on the study of the properties of the cluster anion $(Tc_2Cl_8)^{3-}$, whereas A. V. Davidov et al. and E. Hermann et al. (German Democratic Republic) reported on research on new volatile fluorine-containing β-diketonates of rare-earth and transplutonium elements.

Data about complex forms of the occurrence of radioactive ruthenium in objects of the environment were presented in a paper by G. M. Varshal and I. A. Koshcheevoi. The mechanism of uranium sorption on cation-exchange resins with complex-forming groups was considered in a paper by É. M. Kap et al.

The papers read at the conference and the discussion which followed testify to the breadth of the scientific research work carried on in the USSR on the analytical chemistry of radioactive elements and to the great successes achieved.

It was noted at the conference that it was necessary to further increase the accuracy, and selectivity of the analytical methods for the determination of radioactive elements; to intensify research in the realm of controlled synthesis of new reagents suitable for extractive isolation of highly radioactive substances, for the study of their properties and their extraction kinetics and mechanism; and to extend research in the domain of analytical chemistry of natural radioactive elements, including modern methods for their isolation and determination.

CONSTRUCTION OF ATOMIC POWER PLANT IN FINLAND*

In the later 1960s and early 1970s the demand for electricity in Finland increased at an annual rate of 9%. However, in recent years it has become stabilized in connection with the increase in the world prices for organic fuel. The solution was to be sought in the construction and development of a national nuclear energy industry. In the opinion of Finnish experts, in the next several years electricity generation will be increased mainly by means of atomic power plants whose power is to be raised to 2100 MW(electrical) by 1980.

*Based on proceedings of Finnish–Soviet Symposium on Nuclear Power held in Moscow, Oct. 12-13, 1977.

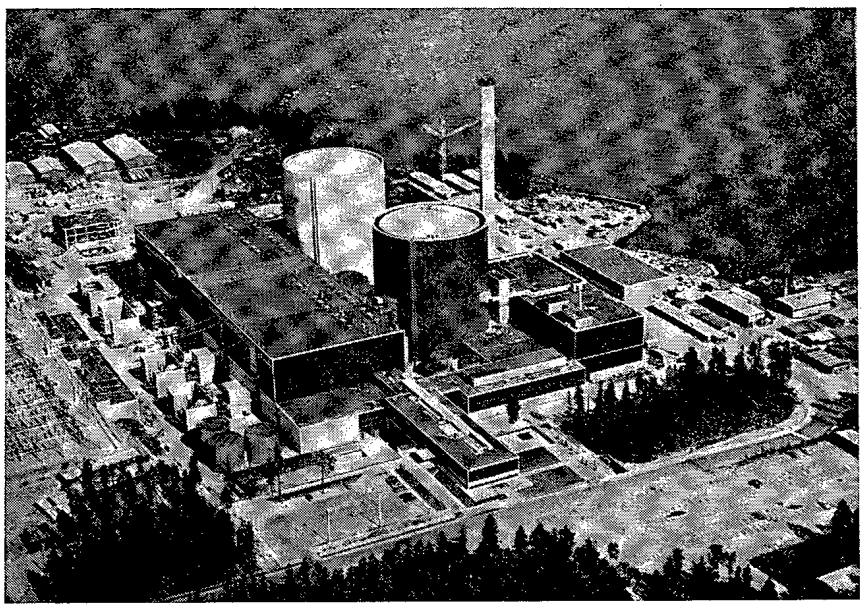

Fig. 1. Lovisa Atomic Power Plant (Finland).

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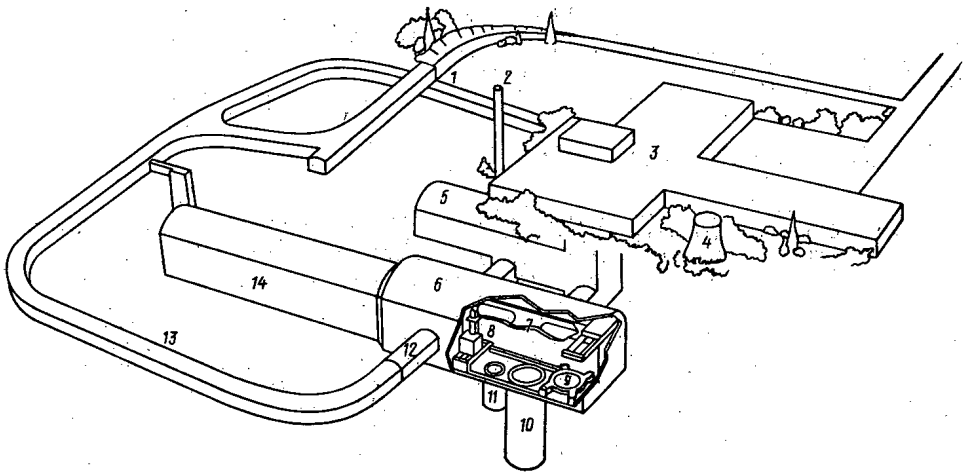

Fig. 2. Atomic boiler house project (Finland): 1) tunnel entrance; 2) ventilation pipe; 3) condensing turbine, control panel, service accommodation; 4) cooling tower; 5) electrical systems; 6) reactor building; 7) heat exchanger; 8) refuelling machine; 9) removable reactor cover; 10) reactor; 11) fuel pond; 12) air locks; 13) transportation tunnel; 14) auxiliary systems.

The development of the nuclear power industry was inaugurated in March 1977 when the first block of the country's first atomic power plant, the Lovisa, built with the participation of the Soviet Union, was started up.

An intergovernmental agreement on the joint construction of two blocks with a VVÉR-440 [water-moderated–water-cooled power reactor] was signed in 1970. At the same time contracts for designing, supplying, and assembly of equipment were signed by the state joint-stock company Imatran Voima and the Soviet foreign trade company Tekhnoproméksport (the commitments of the Soviet party were subsequently assumed by the Soviet foreign trade company Atoménergoéksport).

The Soviet Union supplied the reactors, control and safety system, steam generators, turbines, piping, and equipment for waste storage for the Lovisa plant and did the design, assembly, and start-up and charging work on the main systems and equipment of the power plant. Finnish factories built the main circulating pumps, the refueling machine, cranes for the reactor building, and some other equipment. In their papers at the Moscow symposium Finnish scientists, engineers, and builders expressed a high opinion of the experience gained from the joint work on the construction of the Lovisa Atomic Power Plant praised the quality of the Soviet equipment.

In addition to the Lovisa plant, whose second block is to be put into operation in 1979, two power blocks of 660 MW(electrical) each are to be built at Olkiluoto by the company Teollisuuden Voima in a contract with the Swedish firm ACEA-Atom. The principal power equipment, including a boiling-water reactor, will be supplied by ACEA-Atom. The first block should be put into service in 1978 and the second in 1980.

At the Moscow symposium Finnish specialists spoke of an original design for an atomic facility for supplying heat. They had studied variants of a small atomic boiler house with a power of 100 to 400 MW, suitable for sites near populated areas. Upon consideration it was decided to choose the 200-MW variant which is capable of providing heat for settlements with a population of 50,000-100,000 inhabitants. The peak demand for heat should be covered by boiler houses operating on fossil fuel.

For safety considerations the reactor of the atomic boiler house will be sunk into a rock formation (see Fig. 2); the auxiliary systems which pose no safety hazard will be located on the surface. The reactor shell should be made of prestressed concrete, covered with stainless steel on the inside. Boron reactor control is to be employed and an intermediate loop with a higher loop is to be provided so as to prevent leaks of radioactive water.

The results of economic studies have shown that the projected atomic boiler house will be more efficient than conventional boiler houses because of lower nuclear fuel prices.

The principal characteristics of the heat from the projected facility are:

Temperature, °C	
at reactor outlet	115
at reactor inlet	90
Pressure, bars	7
Temperature of mains water, °C	
at inlet	95
at outlet	60
Flow rate of main water, kg/sec	1360

The main result of the symposium consists in the optimistic assurance that in the process of constructing two atomic power plants Finnish scientists, engineers, and workers acquired good experience in designing, building, and assembling nuclear power facilities and that Finnish industry can produce power equipment of good quality.

SIXTH CONFERENCE ON ENGINEERING ASPECTS OF LASERS AND THEIR APPLICATION

V. V. Aleksandrov and V. Yu. Baranov

The conference took place in Washington, D. C., in June 1977. The more than 1000 delegates from various countries of the world met in 19 sections to discuss 150 papers and 36 review papers. Half of the meetings were devoted to new laser instruments, their operating principles, development of constructions and circuits, as well as improvement of existing solid-state and gaseous lasers. The other papers dealt with the application of lasers in such areas as laser-controlled fusion, communications, measuring methods, isotope separation, photochemistry, etc.

The program for the development of research on laser-controlled thermonuclear research in the United States was presented in a paper by A. Steekly. According to the paper, the laboratories working on this problem have clearly separated responsibilities:

Lawrence Livermore Laboratory – development of superpower neodymium–glass laser systems and their use to attain a high energy yield in experiments on the spherical compression of thermonuclear targets;

Los Alamos Scientific Laboratory – development of superpower CO_2 lasers and their application for experiments on laser fusion;

Rochester University Laboratory – development of medium-sized laser systems and units for laser systems;

KMS Industries Laboratory – use of small neodymium–glass systems and development of technology for the fabrication of targets.

At the present time the Livermore Laboratory is engaged in experiments on the Argus facility, the most powerful laser system in the world, which delivers 3 TW in the laser radiation in two beams of laser radiation suitable for thermonuclear experiments. The experiments on this facility have given the following results: neutron yield 10^9, ion temperature 8 keV, and thermonuclear yield 1%.

A density exceeding five times that of liquid D–T mixture has been attained in compressed fuel in the Janus facility with a spherical system of irradiation. Equicomponent mixtures with an initial density of 2 mg/cm^3 were used in the experiments. A compression ratio of 50-500 has now been achieved.

Further prospects of the Laboratory are bound up with the Shiva facility in which the thermonuclear yield is of the order of units at a fuel density of more than 100 times that of liquid fuel. Finally, in the opinion of specialists, a useful yield of thermonuclear energy will be obtained from the Shiva-Nova facility (1983). Shiva will make it possible to reach a power of more than 20 TW and Shiva-Nova is being designed to produce more than 200 TW of light energy in 1983.

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It should be noted that the Livermore Laboratory has developed radiochemical diagnostical methods which have raised the accuracy of measurement of neutron yield to 15%. The methods involve use of neutron activation with ^{63}Cu and observation of the decay of ^{62}Cu (positron activity) formed during reactions under neutron bombardment. The ion temperature is estimated by the time-of-flight technique to measure the broadening of the spectrum of thermonuclear neutrons. Replacement of D−T fuel by D−^{3}He and measurement of the number of protons formed in D−^{3}He reactions have confirmed the temperature determined by other methods. Measurements with an x-ray electrooptical camera in a slit scanning mode permitted the motion of the compressed cloud to be followed. The velocity found ($\sim 3 \cdot 10^7$ cm/sec) is in agreement with the calculations made in accordance with the Lasnex program.

A high-resolution x-ray spectrometer was used with Janus to measure the broadening of x-ray lines of high-Z materials added inside the target. Analysis of the broadening of these lines confirmed the existence of density ~ 0.5 g/cm^3, two to three times that of liquid D−T fuel.

Of particular interest and importance for understanding thermonuclear experiments are the finite size, shape, and distribution of the zones in which the fusion reaction occurs in the compressed fuel. Some information about the shape and size of the compressed nucleus is obtained by using x rays of quite high energy to photograph the moment when the cloud comes to a stop but this information is problematical and requires involved interpretation.

An important place in the American program is occupied by the search for new laser media with a higher efficiency and acceptable radiation wavelength. The target of the program is to develop, by 1981, a laser medium which radiates at a wavelength of $\sim 0.5\ \mu$, and which has characteristics enabling it to become an acceptable source for laser-controlled thermonuclear fusion.

A considerable number of papers by the Los Alamos Laboratory were devoted to the application of CO_2 lasers in laser-controlled thermonuclear fusion. The scientists in charge of the work in this laboratory are extremely optimistic. D. McCall, one of the program leaders, reported that very hot particles predicted for CO_2 lasers had not been observed in experiments. Thus far the work on the application of CO_2 has been carried out in several stages.

First Stage. A single-beam CO_2 laser system (SBS), delivering infrared radiation of 250 J in a single pulse, was developed. This system permitted experiments to be conducted on the interaction of the radiation with targets at powers of up to $7 \cdot 10^{14}$ W/cm^2. For the first time in the world it was shown that the reflection coefficient at such a density is about 5%.

Second Stage. A two-beam CO_2 laser system (TBS) was built for studying its possible use as a prototype of a two-beam module for an eight-beam system (EBS) and for conduction experiments with a target.

The main effort has been directed at the study of the applicability of various saturable absorbent gases to raise the threshold of self-oscillation (superradiation) in the laser system.

Third Stage. Development of the eight-beam CO_2 laser system (EBS) consisting of four two-beam modules. According to optimistic predictions this system will begin to function in the summer of 1978 (half a year ahead of schedule) and is expected to yield 10^9 n^0. The EBS is to have copper aspherical optics with a threshold resolution of 8-10 J/cm^2. The principal problems are: the high-voltage technology, obtaining and amplifying a multiple-frequency (four lines and P branch) nanosecond pulse, suppression of superradiation, beam quality, operation of a multibeam system, vacuum-control system, position of target in camera with capability for controlled precision displacements of target, focus control, physics of target compression, and protection of system from reflected radiation. All the systems, including the diagnostical apparatus, are computer-controlled.

Fourth Stage. Construction of Antares high-power CO_2 laser. The system will have six beams with 12 sectors in each. According to calculations, the radiation should have the following parameters: power 100-200 TW (100 kJ in 1 nsec, 50 kJ in 0.25 nsec), size of thermonuclear target 400 μ, pulse separation 10 min. Calculations indicate one may expect to obtain a thermonuclear energy equal to the energy input, i.e., one may expect demonstration of the feasibility of thermonuclear combustion. According to D. McCall, the facility will be built in 1982.

An extremely interesting paper from the point of view of the development of new types of lasers was that read by A. Brederlau (Max Planck Institute, German Federal Republic) about the start up of the Asterix III system which constitutes a pulsed laser with a C_3F_7I + Ar mixture. According to Brederlau, the laser has

the following parameters: generation wavelength 1.36 μ, energy 500 J, pulse duration 0.5 nsec, and laser beam divergence $0.9 \cdot 10^{-3}$ rad. The electrical input of the pumping tubes is 40 kJ, i.e., the laser efficiency is ~0.15%. The total length of the laser system is 125 m. The laser pulse has a rather high contrast ($\sim 10^5$-10^6). This laser is one of the most powerful in the world (P ~ 1 TW).

The leading role in the development of new types of layers in the U.S.A. is played by the Livermore Laboratory. New lasers are also developed by Hughes, Sandia, and other companies.

The Los Alamos Laboratory presented a paper on the use of systems for excitation of the following types of lasers by electrical discharges: KrF (2845 Å), ArF (1930 Å), and XeF (3511, 3532 Å).

The U.S. Naval Research Laboratory has studied how the energy of generation of an XeF laser is affected by pressures of up to 5 kgf/cm^2 and the type of gas (argon is poorer than neon). An output energy of up to 2 J/liter has been obtained under electron-beam excitation of the laser.

Great successes in this field have been achieved by a group led by A. Alcock (Canada). By means of a simple electrical circuit and using a mixture with a high helium content (NF_3:Xe:He = 1:3:2700), this group obtained a laser energy of 290 mJ (pulse duration 20 nsec, active volumes 0.18 liter). A linear increase in generation is observed when the pressure is raised to 5 kgf/cm^2. A laser of this kind can be used for experiments on isotope separation.

Researchers engaged in isotope separation place great hope in obtaining considerable peak powers at a wavelength of 16-μ, which coincides with the absorption band of the UF_6 molecule. There are several ways of doing this: retuning the radiation of a pulsed CO_2 laser by means of cells (e.g., NF_4, CF_4) as well as obtaining 16-μ radiation by means of a special scheme for the excitation of CO_2 molecules.

To obtain 16-μ radiation the Westinghouse Research Laboratory used a cooled (140-220°K) discharge tube with a 1.5-m longitudinal discharge (pulse duration 2 sec). The mixture CO_2:N_2:He (1:2:25) at pressure of 8-12 mm Hg was excited by pulsed 9.4-μ radiation from a commercial CO_2 laser. A maximum output energy of 20-50 μJ/pulse was observed with a pulse separation of ~1-1.5 msec.

The seventh conference on engineering aspects of lasers and their applications will also be held in Washington, D.C. in 1978.

CONFERENCE ON RADIOECOLOGY

Yu. B. Kholina

The conference, which was held in July 1977, was attended by 195 delegates from 23 organizations, including representatives of the Academy of Sciences of the USSR, the Ministry of Higher Education, the Ministry of Health, etc. These were 44 papers, including 13 plenary papers, on problems of protection of the environment from radioactive contamination. Section meetings discussed papers on "Animal Radioecology," "Migration of Radionuclides in Biogeocenoses," and "Effect of Ionizing Radiation." During a round-table meeting at the end of the conference leading specialists in radioecology and environmental protection (radioecologists, hygienists, radiobiologists, geneticists, etc.) reported on the problems of radioecology.

At the plenary session, a paper on the main tasks of radioecology in connection with the development of nuclear power paid particular attention to standards on the discharges of radioactive substances and on the purification of discharges to remove the principal dose-producing radionuclides. Estimation of the anthropogenic contamination of the environment with natural and artificial radionuclides was discussed in a paper by A. K. Kruglov et al. The radiation situation as a whole in the biosphere is considered to be satisfactory; the level of radioactivity due to both products from the total fallout and discharges from the nuclear power cycle does not exceed 0.1% of the dose from the natural radioactivity background. At the same time, the local contamination due to discharges of radioactive waste into the atmosphere in the future may be considerable. The greatest radioactive hazard is posed by T, ^{14}C, ^{85}Kr, and ^{129}I which accumulate in the atmosphere. Besides the effective trapping of these radionuclides the problem of the burial of radioactive waste must be solved.

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Much interest was aroused by papers on the migration of radionuclides of iodine and plutonium in the environment. As shown in a paper by B. S. Prister et al., regardless of how they enter the soil, iodine isotopes interact actively with the organic matter in the soil and display a large capacity for migration. The migration in the soil is determined mainly by processes of convective transport and diffusion and removal of iodine by vegetation does not play any significant role in the iodine balance. One of the major factors affecting the radiation situation in the region of a discharge of radioactive iodine is its volatilization, which is determined by the efficiency with which the soil is worked. In the case of natural pastures the main criterion for normalization is the dose rate of the ^{129}I discharge whereas in the case of cultivated pastures the dose rate of the discharge and the accumulation of the isotope in the soil are of roughly the same degree. The behavior of plutonium in soils, as shown in a paper by G. N. Romanov, depends upon the moisture content and the organic matter in the soil. The presence of chelate-like substances considerably increases the mobility of plutonium and irrigation of the soil leads to an increase in the plutonium build-up in vegetation by several orders of magnitude. The author of the paper emphasized that it was incorrect to use the build-up factor for radionuclides without taking account of the form and migrating capacity of these radionuclides in order to evaluate the radioactive contamination of objects in the environment.

The characteristics of tritium migration in the biosphere were discussed in a paper by G. N. Romanov et al. The tritium cycle on the whole corresponds to the hydrogen cycle; a considerable fraction of the tritium in objects of the environment is in a bound form (with water of crystallization). Two tritium components derived from soils with half-lives of 7-27 days and 24-30 months were found. It is assumed that recultivation may accelerate elimination of tritium from soils two- to threefold.

A paper by Yu. B. Kholina and R. M. Aleksakhin gave information about the distribution of radioactive products from total fallout and discharges from enterprises of the nuclear power cycle into the hydrosphere and made a prediction for the next 20 years. The authors of the paper came to the conclusion that the greatest danger will be posed by ^{129}I and transuranium nuclides (^{239}Pu and ^{241}Am), which are efficiently accumulated by hydrobionts. Tritium and ^{14}C enter into biological processes of short duration and do not built up in aqueous organisms above the concentration in the environment. Equilibrium of ^{85}Kr in the aqueous environment will be attained very quickly.

The radiative effect of an atomic power plant on the population was the subject of a paper by researchers of the Biophysics Institute of the Ministry of Health of the USSR. At the present time the mean individual dose of radiation from atomic power plants is an insignificant 0.001 mrem/yr. At the same time the radiation dose as the result of discharges from atomic power plants is several times higher, approaching 0.004 mrem/yr (V. A. Knizhnikov et al.). The population burden (mean individual dose from all sources) reaches 254 mrem/yr. The main contribution comes from the natural background (43%), medical x-ray diagnostic procedures (24%), and irradiation from remaining in stone buildings (23%). The irradiation of the population by fallout products has dropped almost 20-fold since 1963 and is now 2 mrem/yr.

An evaluation of what radiation effect the coolers of atomic power plants have on the population was presented in a paper by D. I. Gusev et al. With allowance for the self-purification of the water, they calculated the individual radiation doses received by the body and tissue irradiated by tritium, $^{134,137}Cs$, $^{131,133}I$, $^{58,60}Co$, ^{51}Cr, ^{54}Mn, ^{144}Ce and ^{90}Sr. The irradiation of the human body owing to liquid discharges from atomic power plants was 0.88 mrem per 1 MW(electrical)·yr. The authors of the papers believe that with existing levels of discharges of reactor water of medium mineralization the radiation effect on the population is insignificant and the body of water can be used for economic purposes.

A paper by N. P. Dubinin and V. A. Shevchenko analyzed the genetic effects observed in chronically irradiated populations. In simulation experiments a study was made of the dynamics of the mutation process in populations of haploid and diploid organisms. An inverse relationship was discovered between the intensity of the radiation and the genetic effects per unit dose. It was found that the dose which doubles the level of the natural mutation process is not constant but depends on the dose rate. The results are certainly of interest for predicting the genetic consequences of radiation acting on the population.

An interesting communication about observations of a forest over a period of 20 years was presented by F. A. Tikhomirov. The results of the observations showed that the forest is characterized by a high retaining capacity with respect to radioactive fallout and a low rate of migration of radionuclides beyond the limits of the contaminated region. Trees (especially conifers) were found to be highly radiosensitive; the radiation effects were due not only to the direct ionizing effect but also to secondary effects caused by the biogenetic bonds being disturbed under the influence of the radiation. A dose of 100 rd should be considered dangerous for the forest.

In a number of cases radiostimulation (acceleration growth and development) of trees was observed under irradiation of 100-300 rd.

A paper on the present tasks of radioecology was read by E. A. Fedorov, who noted that a biogeochemical platform has new been created for the development of scientific and practical recommendations on the protection of the environment from the effects of the nuclear power industry. It is necessary to create models of the events which occur in ecological systems at various levels of radioactive contamination. This can be done by studying the migration of radionuclides in natural biogeocenoses and biological chains.

The round-table discussion touched on the most urgent problems of radioecology today: the elimination of radioactive waste, the combined action of radiation and chemical factors of the environment, and radiation biology and medicine.

In a resolution the conference noted the great contribution of radioecological research to the solution of problems of environment protection. The conference recommended a major expansion of research on how biologically significant radionuclides which are genetically linked with waste from nuclear energy establishments behave in the biosphere, the elaboration of radioecological criteria for the standardization of radionuclide content, and the study of the effects of small doses of radiation on objects of the environment. Much attention was paid to the study of the effect of combined (radiation, chemical, and thermal) factors.

SEMINAR ON THE USE OF LOW-POTENTIAL NUCLEAR HEAT

Yu. I. Tokarev

The seminar, which was organized by American, European, and Finnish nuclear societies, was held in Otaniemi (Finland) on Aug. 21-24, 1977. Some 300 specialists from 23 countries participated, presenting 51 papers. The main topics of the seminar were the possible applications of low-potential nuclear heat with allowance for the economic efficiency and sources from which such heat could be obtained.

The Soviet participants presented three papers on the use of water-moderated–water-cooled boiling reactors for atomic heat and power plants and atomic central heating plants and one paper on experience from the operation of the ARBUS reactor with an organic coolant.

It follows from the proceedings of the seminar that the countries which presented papers have an unlimited demand for low-potential thermal energy. In the most highly developed countries with a moderate or cold climate the demand for such energy is 30-60% of the total energy demand, the bulk of it being required for heating. In such highly developed countries as the U. S. A., the Federal Republic of Germany and Japan the fraction of heat which will be required in the form of industrial steam is expected to be quite considerable (in the U.S.A., e.g., it will reach 16% as against 22% for heating). The highest specific thermal energy demand for heating is expected in the West European countries (e.g., up to 2 kW of the installed capacity per inhabitant in Denmark). In Austria, the Netherlands, Switzerland, the German Democratic Republic, and Hungary the combined heat loads changed only slightly in 1965-1975, whereas in Finland, France, Rumania, and especially in the Federal Republic of Germany, Denmark, Sweden, Poland, and Czechoslovakia these loads are continuing to rise in view of the oil crisis.

The seminar heard reports about existing projects or projects being developed for the use of reactors for supplying heat: New Pickering (Canada), the Brno–Prague region (Czechoslovakia), Greater Helsinki (Finland), Hamburg and the Ruhr basin (Federal Republic of Germany), Barsebekk–Malmö–Lund and Greater Stockholm (Sweden), Bern and Basel (Switzerland) and Minneapolis, St. Paul (U.S.A.). Reactors with a thermal power of 700-2000 MW are to be used in these regions and a 100-MW reactor, in the Saclay region (France). No data of any kind characterizing the demand for nuclear heat for desalinization of salt water were given, although obviously such a demand does exist, mainly in desert and semidesert regions in which primarily developing countries are located. Papers by representatives of the U.S.A. and France told (films were also shown) of the application (on a pilot scale) of the heat from water cooling turbine condensers for heating hothouses (growing

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strawberries in the winter, flower growing, mushroom cultivation), for heating soil, and for heating water in open bodies of water for fish farming (accelerating the growth of fish in warm water). However, no figures were given about the expected scale of such use of heat.

It is proposed to use three groups of reactors, as well as the heat of condenser cooling water, as sources of heat. Most of the propositions are based on obtaining low-potential heat from atomic power and heat plants with reactors with a high electrical power (700-1300 MW in one block) which have performed well in regard to both reliability of operation and economic indicators in atomic power plants. These facilities differ from an atomic power plant only as to turbines with the extraction of steam for central heating. Such reactors are envisaged in 14 out of the 15 central heating projects. The heat is in this case a by-product with an expected cost of 0.2-0.4 cent/kWh, and its transportation over large distances leaves its cost competitive with thermal electric plants operating on fossil fuel. Most of the projects are based on PWR, BWR, and CANDU (Canada) reactors. A paper by Yu. I. Tokarev et al. (USSR) informed of the possible use in an atomic heat and power plant of a water-moderated–water-cooled boiling reactor; the reactor would be housed in a prestressed ferroconcrete vessel, thus eliminating the brittle fracture of the vessel. Consideration was also given to the possibility of obtaining low-potential heat from high-temperature reactors with a helium coolant, the temperature of the gas at the outlet from the reactor being 725-850°C (Federal Republic of Germany, France). However, notwithstanding the quite promising economic effectiveness, no decision has been made to go ahead with the construction of any known project for atomic central heating. The reason probably lies in the growing opposition of public opinion in a number of countries to atomic heat and power plants being sited near (or within) large cities. In this case the plants should have additional (expensive) safety devices, the requirements on which have not yet been formulated in any country. Partially for this reason it is not yet possible to adequately determine what is economically more effective, to build atomic heat and power plants far from citites and on the basis of the usual safety requirements for atomic power plants or to built atomic heat and power plants (or atomic central heating plants) with additional safety systems in the immediate vicinity of the demand. This factor has appreciably retarded the adoption of the final design for the most advanced foreign project for an atomic heat and power plant in Ludwigshafen (Federal Republic of Germany) since the constantly growing safety requirements on such plants are continuously increasing the costs of the project.

The principal advantages of reactors of intermediate thermal power (600-1500 MW) were postulated to be the possibility of the use of the developed technology for reactors used in atomic power plants, reduction of the distance to the consumer, and also better conditions for protection from sabotage. The specific capital intensities and cost of the heat inevitably increase. In comparison with more powerful reactors, the rise in the cost of the heat is ~0.1 cent/kWh for the same length of lines for transportation of the heat. It was pointed out in a number of papers that, with sufficiently large heat loads, facilities with such reactors are all competitive with plants based on fossil fuel. In addition to proposals to use light-water (U.S.A., Britain) and heavy-water (Canada) reactors with developed technology (but decreased power), there were proposals for the use of pressurized-water reactors which have been developed for transportation plant (Federal Republic of Germany, U.S.A., France). It was also stated that the compactness and integrated composition of the facilities, features which are more difficult to realize with large reactors, will cut the cost of the equipment.

The most original suggestions concerning the construction of reactors for atomic central heating plants were presented in joint Swedish–Finnish (SECURE reactor), French (THERMOS reactor), and Soviet papers. The suggestions were based on increased technological and radiation safety requirements which must be satisfied by a reactor in the immediate vicinity of the consumer of the heat. In the SECURE reactor this is achieved by relatively low water parameters at the outlet from the reactor core (115°C; 0.7 MPa), low maximum specific linear power (270 W/cm), absence of mechanical control devices in the core (the reactor is controlled by varying the concentration of boric acid in the primary loop), and the underground location of the atomic central heating plant. The reactor vessel and piping of the primary loop are placed in a prestressed concrete pool of borated water. The systems for the emergency suppression of the reactivity have been designed to operate without the participation of the operator. If the pump stops or if the primary loop piping is ruptured the pressure of the coolant in the reactor core drops and water from the pool, with a high concentration of boric acid, enters. When the coolant in the reactor core boils up, gas is squeezed out of a gas-filled dome above the core. The pressure in the pool rises in the process and this results in the core being filled with borated water from the pool. Suppression of the reactor core for a long time is ensured by the introduction of boron pellets into the fuel assembly cavity. The pool is cooled because of the natural circulation of special loops through a cooling tower on the surface.

In THERMOS, whose core is inside a vessel together with the heat exchangers and pumps, increased safety is attained by putting the entire reactor vessel in a special pool through which the residual energy

release is removed. This pool confines the radioactive discharges if the vessel loses its hermeticity. Moreover, a dome to protect the atomic central heating plant from missiles, shock waves, etc. is erected over the reactor building.

A paper by I. N. Sokolov et al. (USSR) proposes the construction of a water-moderated–water-cooled boiling reactor with heat exchangers built into the vessel. The low parameters of the coolant (190°C and 1.5 MPa) allow the vessel design to be simplified. A ferroconcrete shell is used to confine radioactive discharges.

The exchange of views at the seminar proved to be extremely fruitful and made it possible to get a quite good grasp of the road along which atomic heating will develop in the world as well as to get a good idea of the problems on this road.

LABORATORY APPARATUS WITH β SOURCE FOR RESEARCH ON RADIATION-CHEMISTRY PROCESSES

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The apparatus (Fig. 1) consists of a cylindrical steel container with radiation source and irradiation chamber. The lower half of the container (Fig. 2) has a cylindrical groove whose diameter is equal to half the container diameter. The middle of the groove is displaced relative to the central axis of the container by one-quarter of its diameter. The groove contains the holder with the source which is turned about a vertical axis by means of a handle which passes through the upper half of the container. The section of the lateral surface opposite the holder has a flat area 60 mm wide. In the "store" position the holder is turned so that the source is in the center of the container. In the working position the source is switched by turning the holder through 180° about the vertical axis.

The container is provided with a spring shutoff device (see Fig. 1) which fixes the onset of radiation to within 10^{-2} sec. This is extremely important when the apparatus is used to study fast radiation-chemical reactions by methods which permit the course of the process to be followed directly while the radiation is acting, e.g., by electrochemical methods. The screen of the shutoff device is a 25-mm stainless-steel disk 1 mm thick.

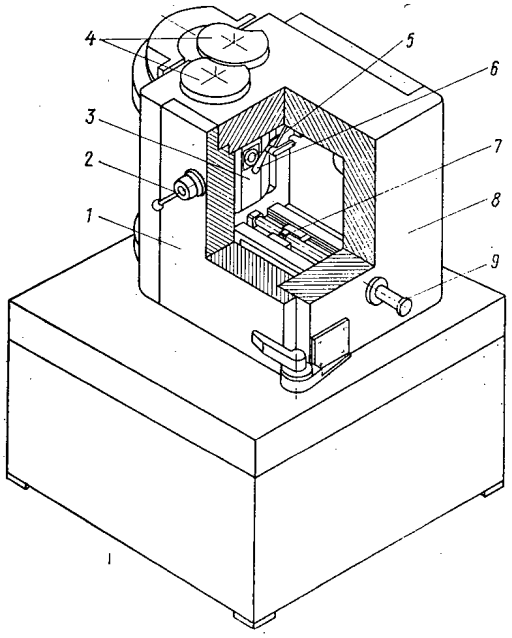

Fig. 1. General view of apparatus with IRUS-1 β source: 1) chamber door; 2) handle with interlock; 3) container; 4) interlocking disks; 5) IRUS radiation source; 6) screen of spring shutoff for source; 7) carriage; 8) irradiation chamber; 9) drive screw for moving carriage with specimens.

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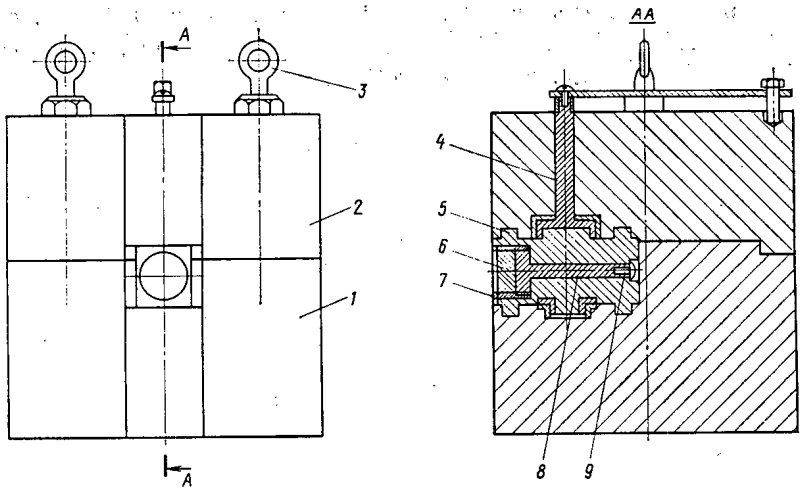

Fig. 2. Container; 1, 2) lower and upper halves; 3) tightening bolts; 4) handle for turning holder; 5) holder; 6) IRUS-1 source; 7) yoke for source; 8) rod; 9) fixing screw.

The source used consists of IRUS-1 preparations with $^{90}Sr + {}^{90}Y$ isotopes with an activity of 18 and 32 Ci.

The shielding of the container against bremsstrahlung generated in the materials of the source itself and the shielding was designed with allowance for high-energy γ-ray quanta in the spectrum of ^{90}Y radiation with an energy of 0.28 pJ and an activity of ~0.02% of the total activity of the source. Such quanta appear especially at the shielding thicknesses required to attenuate the radiation to an allowable level and in practice nothing is gained by using a combined shielding consisting of a low-Z material to reduce the bremsstrahlung yield and a high-Z material to absorb this radiation.

The rectangular irradiation chamber has internal dimensions of 280 × 250 × 350 mm with 90-mm lead walls. It is provided with a carriage on which the specimens to be irradiated are mounted. The carriage, which is propelled by a drive screw, makes it possible to irradiate specimens at various distances from the source and, consequently, at various dose rates. The door of the irradiation chamber, set on journal bearings, opens quite easily without any special devices. Mechanical interlocking with two disks, one of which is attached rigidly to the door handle while the other is attached to the source-turning handle, prevents access to the irradiation chamber when the source is inserted and insertion of the source when the door is open.

The container and the irradiation chamber are set up on a special working table. Their positions relative to each other are shown in Fig. 1. The sources are charged into the container in chambers equipped with manipulators. To begin with, the source is put into the circular yoke (see Fig. 2), and the yoke together with the source is then inserted into the holder of the earlier assembled container. The holder is then turned to the "storage" position and the yoke is fastened with the fixing screw. If necessary, the apparatus can also be charged with BITs, BIS, and other sources of appropriate activity.

The apparatus was dosimetered with a 0.024-mm cellophane film, dyed with thiazine red. The dosimetric films were calibrated with a ^{60}Co source according to the indications of a calorimeter and a ferrosulfate dosimeter. The absorbed dose was found by the decrease in their optical density at a wavelength of 515 nm.

The depth distribution of the dose rate in polymethacrylate is satisfactorily described by

$$P = P_0 e^{-kx},$$

where P is the dose rate at a distance x from the surface of the source; P_0 is the dose rate near the surface of the source; and k is a constant.

With a 32-Ci source $P_0 = 25.1$ W/kg and $k = 8.45$ cm^{-1}.

The measured dose rate near the surface somewhat exceeds the observed rate in present powerful γ-ray apparatuses.

The apparatus described above is simple to operate, permits radiation-chemical research to be conducted in materials at dose rates of up to 25 W/kg. Special equipment makes it possible to study the kinetics of fast radiation-chemical reactions.

The mean energy of the ^{90}Y spectrum (E = 0.15 pJ) is close to the energy of electrons from accelerators which have found most widespread application in industry. It is therefore becoming possible to work out some elements of radiation technology under laboratory conditions.

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