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SOVIET ATOMIC ENERGY

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The Russian press date (podpisano k pechati) of this issue was 1/23/1976. Publication therefore did not occur prior to this date, but must be assumed to have taken place reasonably soon thereafter.

PEACEFUL USE OF NUCLEAR ENERGY AND THE PROBLEM OF NONPROLIFERATION OF NUCLEAR WEAPONS

I. D. Morokhov, K. V. Myasnikov, and V. M. Shmelev

UDC 621.039/623.454.8

An analysis of the population increase of the world and of energy requirements per capita shows that as much energy will be consumed in the world from 1970 to 2000 as was consumed in the last 20 centuries. The energy demand at the end of the century will be approximately three times greater than present levels. Even today the most highly developed capitalist countries are feeling the lack of energy sources. This circumstance has created great interest in the use of a new energy source – nuclear power.

The advantages of nuclear power from the economic and ecological aspects are extremely encouraging. The economic indicators for large nuclear power installations and future improvement in nuclear technology, the development of breeders for example, convincingly demonstrate that the traditional energy sources (coal, oil, gas) will be increasingly supplemented and displaced by nuclear fuel. The International Atomic Energy Agency (IAEA) estimates that by the year 2000, 50% of the total energy balance of the world will consist of electrical power and about 50% of all electrical power will be produced by means of nuclear-power installations (Figs. 1, 2).

The energy crisis has significantly increased the competitiveness of nuclear power in comparison with the usual sources of electrical power. Today, not much more than 20 years after the start-up of the first nuclear-power station in Obninsk, there are more than 100 nuclear-power stations operating in the world with a total installed power close to 40 million kW, and according to predictions the power from nuclear-power stations will be about three billion kW by the year 2000. As is clear from Fig. 2, nuclear power should occupy a leading position in the energy picture of developing countries even in the next few decades. Among these countries, there are those that will consider it more economic to construct nuclear-power stations and not consume their own oil and gas reserves for power purposes. It is precisely nuclear power that makes it possible to free humanity from the threat of a power shortage and to provide power for a developing civilization.

In addition to power, nuclear technology and nuclear methods are becoming indispensable in industry, agriculture, medicine, and geology for monitoring atmospheric contamination and in other fields.

The swift progress of the peaceful use of nuclear energy brings special acuteness to the problem of preventing the proliferation of nuclear weapons. The development of nuclear technology leads to widespread "creeping" of nuclear materials over the entire planet which can give rise to the preconditions for the proliferation of nuclear weapons. Nuclear materials intended for peaceful uses can be diverted into military channels for the production of nuclear weapons or other explosive devices. The possibility of such diversion exists when thermal and fast reactors are used where plutonium, which can be used for military production, is produced as the result of irradiation of uranium. Enriched uranium itself can be used for the manufacture of nuclear weapons.

According to the IAEA, the amount of natural uranium in power reactors will rise to 15,000 tons by 1985 and the amount of enriched uranium to 70,000 tons (Fig. 3). According to the same data, 700 tons of plutonium will be in storage, and the amount of plutonium in the fuel of thermal and fast reactors will be

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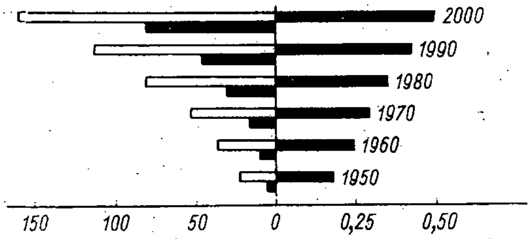

Fig. 1

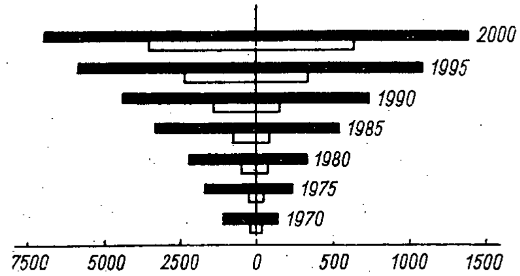

Fig. 2

Fig. 1. On the left, worldwide power production (trillions of kWh); on the right, fraction used in the form of electrical power: □) total; ■) electrical power.

Fig. 2. Electrical power production (thousands of MW): on the left, total installed power for world's electrical power stations; on the right, installed power for electrical power stations in developing countries:□) all stations;) nuclear power stations.

more than 500 tons (Fig. 4). The fact that even by 1980 the peaceful production of plutonium and enriched uranium will be sufficient for the manufacture of tens of atomic bombs per day is clear evidence of the acuteness of the problem. At the same time, if even a small portion of the stored plutonium is used in accordance with existing international agreements for the production of explosives for peaceful purposes, humanity will obtain a huge and inexpensive source of energy.

Many studies on the peaceful use of underground nuclear explosions have been carried out over the past decade in the USSR, USA, England, France, and several other countries. At the Geneva conference in May 1975, to consider the effect of the Treaty on Nonproliferation of Nuclear Weapons, the IAEA presented a review of its activities in accordance with Article V of the treaty. The report presented the conclusions of four international conferences held by the agency in 1970, 1971, 1972, and 1975 with respect to two basic problems: the technical and economic feasibility, and the safety of underground nuclear explosions. Because of the exceptional compactness of the explosive devices and the low cost of the energy produced, peaceful nuclear explosions make it possible to create new, efficient technical procedures, to carry out large-scale civil-engineering projects, and to provide new scientific opportunities. In the report, peaceful nuclear explosions were divided into three groups according to the degree of investigation of the technology:

1. Recognized industrial forms of application of peaceful explosions.

2. Large-scale experiments under field conditions.

3. Laboratory development and theoretical studies.

In the first group are such examples of the use of peaceful nuclear explosions as the elimination of accidental gas blowouts, the intensification of oil extraction, the creation of underground storage cavities in rock-salt masses, and the creation of open reservoirs for water storage.

In the second group are the experiments performed in the USA in low-permeability gas formations for the purpose of developing a commercial technology for gas extraction. Calculations by American scientists show that the national commercial gas reserves will be doubled or tripled by a successfully developed technology. This group also includes experimental nuclear explosions with removal of soil for the development of a technology for the construction of large industrial works – canals: Panama, Pechora-Kolva (USSR), Orinoco-Rio Negro (Venezuela), and Kra (Thailand). The use of underground nuclear explosions should reduce expense and construction time significantly. In the construction of the Kra canal, e.g., the saving would be two billion dollars out of a total construction cost of six billion dollars.

In the third group are the most complex areas for the use of peaceful nuclear explosions such as the extraction of oil by distillation from oil-bearing shale, leaching of copper at the point of deposit, production of geothermal energy, and disposal of radioactive wastes. These are projects which provide for the use of peaceful explosions for scientific purposes.

Considerable experience and experimental data have accumulated in recent years which make it possible to predict the seismic and radiation consequences of underground nuclear explosions with a definite engineering approximation. Specially constructed elements are used for the reduction of radiation consequences. In the report on specific examples, it was shown that the potential irradiation of the population

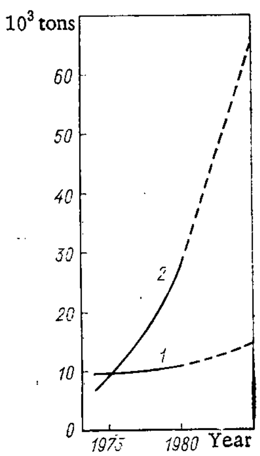

Fig. 3. Amount of natural (1) and enriched (2) uranium loaded into power reactors.

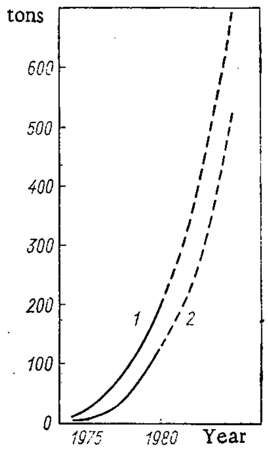

Fig. 4. Amount of stored plutonium (1) and amount of plutonium in reactor fuel (2).

from contained nuclear explosions may be only an insignificant fraction of the dose level established by the International Commission on Radiation Protection. For explosions with soil removal, it may appear necessary in some cases to evacuate the local population temporarily in order to reduce irradiation dose levels.

The seismic consequences of explosions can be predicted with a sufficiently high degree of accuracy including possible costs for reconstruction. By their very nature, the seismic consequences belong more with the factors which determine the economic efficiency of peaceful explosions since the safety of the population can be assured with a high degree of reliability by the working out of special measures.

The prevention of the proliferation of nuclear and thermonuclear weapons of mass destruction represents an urgent problem for mankind. Therefore, a need arose for the development of methods guaranteeing that materials used in peaceful nuclear activities would not be diverted into military channels. A system of such guarantees, and of methods and means for accountability and inspection of nuclear materials, has been developed and used by the IAEA for more than 15 years.

The Treaty on Nonproliferation of Nuclear Weapons, which became effective in 1970, created a juridical basis for the use of international guarantees. By the middle of 1975, 43 governments had concluded inspection agreements with IAEA and had put them into operation; about 30 agreements are in the stage of development or legal formulation. On June 30, 1975, there was under the control of IAEA 47 nuclear power stations, 115 reactors, 29 plants for the processing, manufacture, and reprocessing of fuel, and 195 other separate areas of nuclear-material accounting. In a single year (from July 1974 through June 1975), the agency performed 502 inspections without finding any cases of treaty violations. Experience in the application of IAEA guarantees clearly showed that its control mechanisms functioned reliably to ensure satisfaction by all governments of their obligations.

The treaty facilitates the widespread introduction of nuclear power into the economy of developing countries. The technical assistance extended by IAEA to developing countries in the use of nuclear power is expanding year by year. In the last five years, the total amount of such help was 23.8 million dollars in comparison with 23.5 million dollars in the preceding eleven years.

The agency has performed preparatory work in providing service for the use of peaceful nuclear explosions. It consisted of four technical meetings on this problem which made it possible for nonnuclear countries to evaluate the potential merits of peaceful nuclear explosions, developed principles and procedures for international inspection after their occurrence, and created a special subsection for providing services and further study of this problem.

The Treaty on Nonproliferation of Nuclear Weapons, being an international instrument creating serious barriers to the proliferation of nuclear weapons, not only is not an obstacle in the path of extensive application of nuclear power for peaceful purposes, but facilitates such application and creates favorable conditions for international cooperation in this field.

An important event in strengthening the means for nonproliferation of nuclear weapons was the conference to consider the effect of the treaty during the past five years. Experience during the five years showed that all signatories to the treaty strictly observed its principles. The most important political outcome of the conference was the further expansion of the group of treaty signatories on the eve of the conference, or during the course of the conference, to include West Germany, Italy, Belgium, the Netherlands, Luxembourg, Gambia, Ruanda, and Libya. At the same time, a considerable group of "near-nuclear governments" and two nuclear powers remain outside the framework of the treaty.

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The conference emphasized the importance of a system of accounting and inspection for nuclear material from the viewpoint of the responsibility of treaty signatories and of the importance of collaboration with IAEA for cooperation in achieving the guarantees in accordance with Article III of the treaty. The conference expressed vigorous support for effective guarantees by the IAEA on the application of this inspection to all peaceful nuclear activities of treaty nonsignatories that import nuclear materials and special technical equipment and for recommendations for strengthening the protection of nuclear materials against theft, which were adopted to a considerable extent on the initiative of the USSR.

The conference confirmed the responsibility under Article IV for all treaty signatories to facilitate as far as possible the most complete interchange of equipment, materials, and scientific and technical information, pointing out that the treaty provides favorable conditions for the expansion of international collaboration in this field. The conference adopted recommendations on further development of international collaboration, in particular on studies of the creation of regional centers for the nuclear fuel cycle, which are extremely important for the accomplishment of control.

The conference devoted considerable attention to Article V concerning peaceful nuclear explosions, which provides that treaty signatories not having nuclear weapons will obtain the benefits from peaceful use of nuclear explosions "through the appropriate international organ in which governments not possessing nuclear weapons are properly represented." On the initiative of the socialist countries, the conference adopted a resolution that the IAEA will be just such an international organ. The conference called for a continuance of the work in this field and entrusted the agency with a central role in problems associated with the provision of services relative to peaceful nuclear explosions. Of great value is the proposal adopted by the conference on the recommendation of the socialist countries that the potential benefits of peaceful nuclear explosions be accessible to governments not possessing nuclear weapons and not treaty signatories under appropriate international inspection and by means of procedures developed by IAEA. These proposals, accepted by the conference, eliminate the need for the construction of nuclear devices by nonnuclear governments for peaceful purposes, which is very important from the viewpoint of the nonproliferation of nuclear weapons.

An important position in the treaty is occupied by Article VI which specifies the responsibility of signatory governments to conduct negotiations to halt the nuclear arms race and also to achieve universal and total disarmament.

Important international agreements have been developed in this field in recent years with the active participation of the Soviet Union and other socialist countries. Of outstanding value for the cause of peace and international security are the Soviet–American agreements signed in 1972 and 1973 on the prevention of nuclear war, on the limitation of systems for defence against ballistic missiles, and on certain measures in the field of limitation of strategic offensive weapons. During the third Soviet–American high-level meeting in the summer of 1974, new important agreements were reached, including a treaty on the limitation of underground tests of nuclear weapons. In accordance with this treaty, the USSR and USA are conducting negotiations on peaceful nuclear explosions. This is making a significant contribution to the matter of general prohibition of nuclear-weapon testing.

The conference stressed that adherence to the Treaty on Nonproliferation of Nuclear Weapons by nonnuclear governments was the test method for mutual assurance of the renunciation of nuclear weapons and an effective measure for strengthening their safety. Treaty signatories confirmed their great interest in the prevention of further proliferation of nuclear weapons. They confirmed their vigorous support of the treaty, their unshaken devotion to its principles and purposes, and their responsibility to carry out its regulations completely and more effectively.

The time which has passed since the day the Treaty on Nonproliferation of Nuclear Weapons became effective has demonstrated its widespread international recognition as an effective instrument to check the proliferation of weapons of mass destruction and to promote the peaceful use of nuclear energy.

ATOMIC SCIENCE AND TECHNOLOGY IN THE NATIONAL ECONOMY OF THE USSR

A. K. Kruglov

UDC 621.039/338

Little more than 30 years have passed since the first nuclear reactor in the world first went critical, yet now the nuclear industry in various countries is an independent field occupying a leading place in the national economy.

Atomic science in the USSR not only promotes the creation of the new production needed for nuclear power which has been successfully developed, but is also involved in the introduction of its achievements in the national economy. The energy developed by nuclear fission is employed in a number of new directions.

Radioactive isotopes are used for creating small atomic batteries to supply energy to apparatus and instruments, and also to power artificial organs in man. They are widely used in medicine, industry, and agriculture not just for marking atoms (radioactive indicators), but also as sources of ionizing radiation for changing the material properties.

The nuclear explosives which are produced enable us to use the energy of the atom in order to construct canals, reservoirs, underground storage chambers for various liquids and gases, and also to create new methods of extracting useful minerals.

The achievements of nuclear physics, the science that provides the basis for the atomic industry, have lead to the creation of charged-particle accelerators of various types and designs, which are widely employed in the national economy, giving boosts to the accelerated development of various branches of technology.

We shall briefly list the achievements of atomic science and technology in the USSR.

Reactors and Nuclear Fuel

The development of nuclear power engineering in the USSR, already enables the nuclear power stations to generate energy at a cost per kilowatt-hour which is lower than can be achieved on thermal-power stations, particularly in regions far removed from sources of fossil fuels [1].

TABLE 1. Experimental Values of Depth of Burn-Up of the Nuclear Fuel at the Novovorenezh Nuclear-Power Station, kg/ton U

Burn-up	Unit I	Unit II
^{235}U	12,5	14,3
^{239}Pu	6,4	8,3
Total	18,9	22,6

Improvements in the cost effectiveness of the nuclear-power stations are being achieved both by increasing the unit powers of the reactors and also by improving the technology of reactor construction and the production of nuclear fuel. In a relatively short period of time (about 20 years) since the commissioning of the world's first nuclear-power station at Obninsk, the national economy is benefiting from nuclear reactors which have 200 times the unit power of the reactor in the first nuclear-power station. Two channel-type uranium–graphite reactors (type RBMK)

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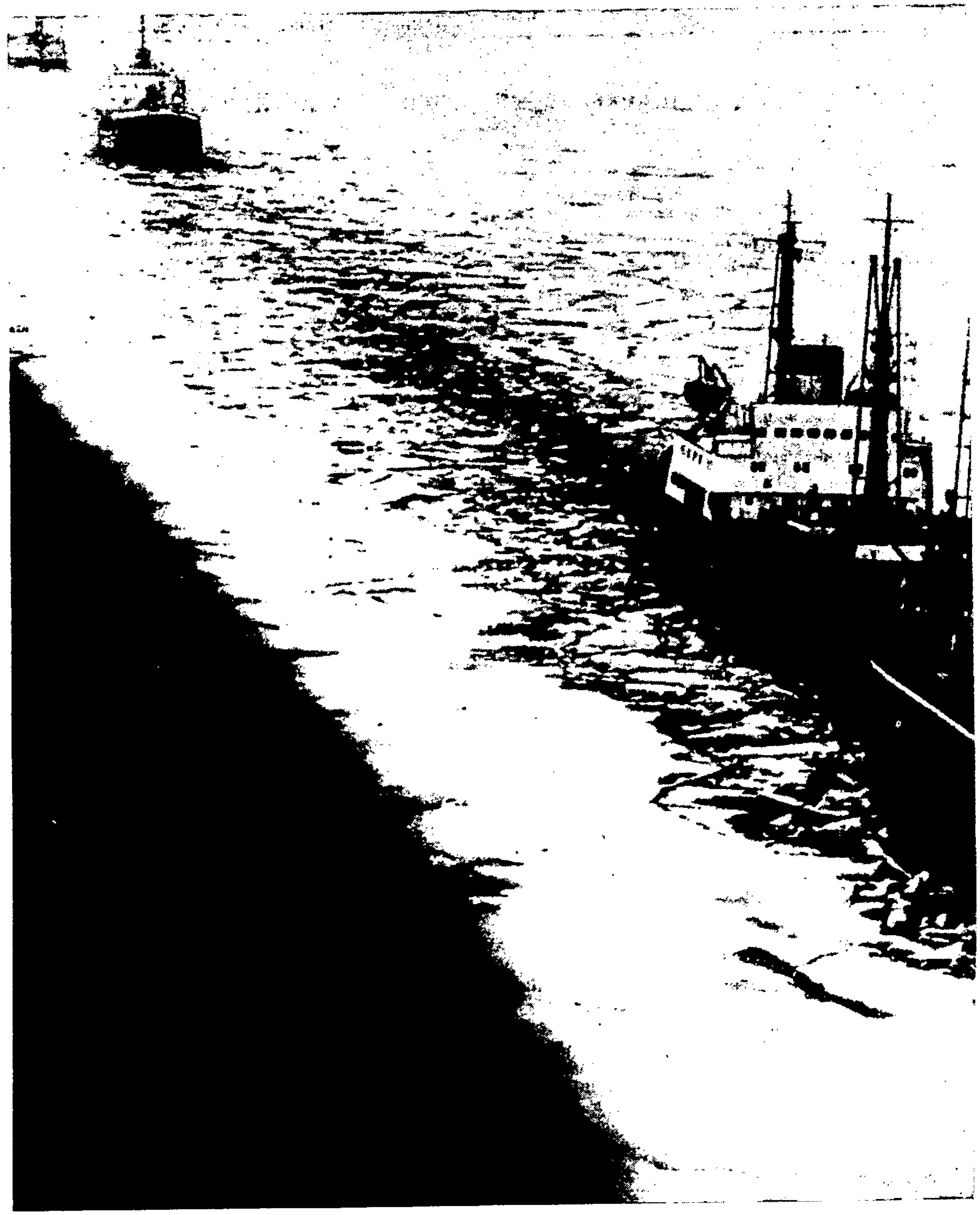
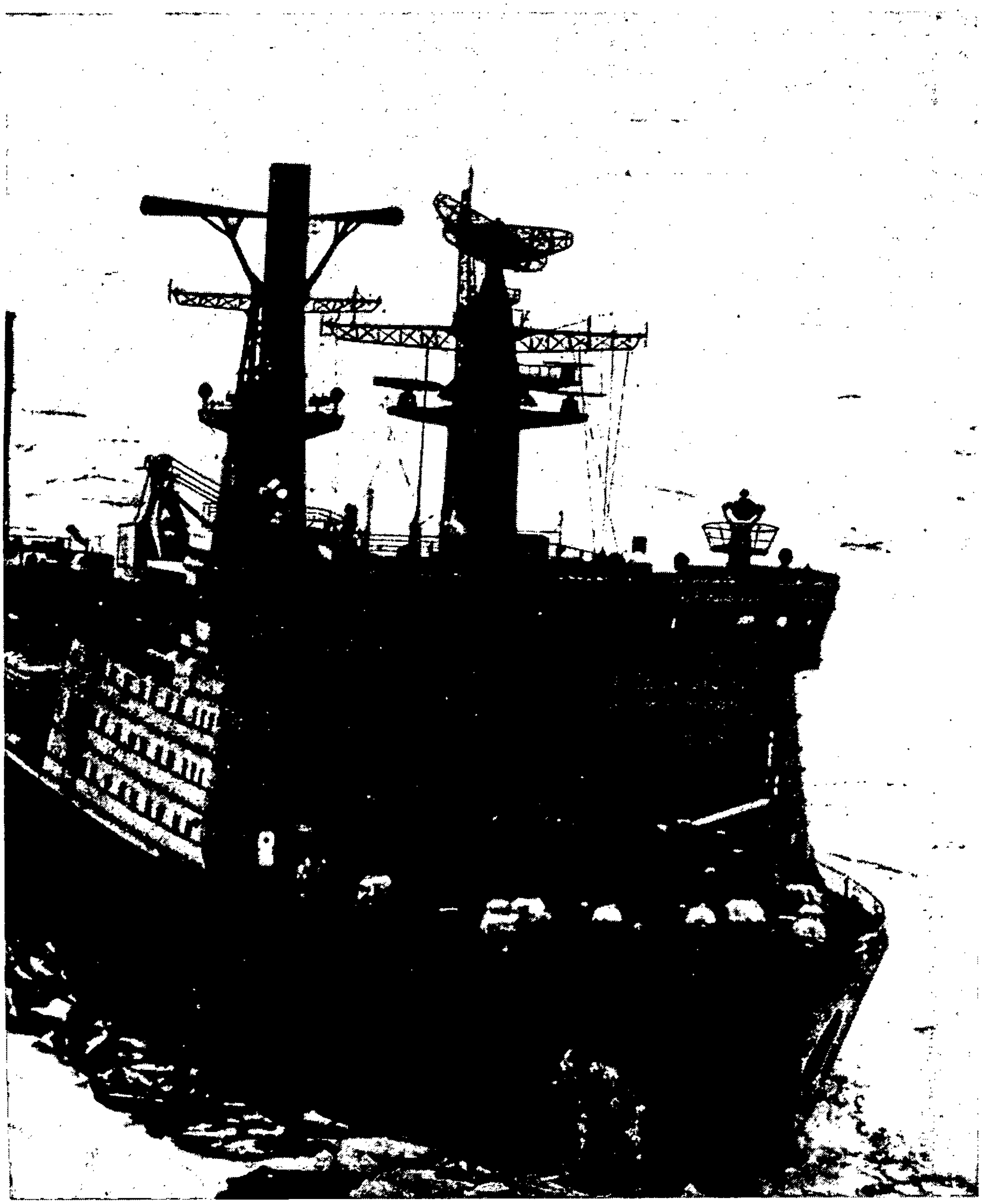


Fig. 1. The nuclear-powered



vessel Artika in polar passage.

TABLE 2. Concentration of Actinides (g/ton) in Irradiated Fuel from Thermal Reactors with Normal Water and Fast Breeder Reactors [5]

Isotopes	Thermal reactor. burn-up 33,000 MW days/ton	Fast reactor. burn-up 80,000 MW days/ton
^{242}Cm	8	42,3
^{244}Cm	31	41,9
^{241}Am	50	1 460
^{243}Am	92	711
^{237}Np	450	180
^{238}Pu	168	1 840
^{239}Pu	5 300	117 000
^{240}Pu	2 140	52 400
^{241}Pu	1 100	14 400
^{242}Pu	340	9 020
^{235}U	8 000	7
^{236}U	4 520	10
^{238}U	940 000	719 000

have been commissioned at the V. I. Lenin nuclear-power station at Leningrad, each with an installed power of 1 GW, while the water-moderated water-cooled vessel reactor (type VVÉR) at the "50th Anniversary of the USSR" nuclear-power station at Novovoronezh has been running successfully for 10 years [2]. The power of the four units on this station comprised 1455 MW at the beginning of 1975. Five units with 1-MW type VVÉR reactors are actively under construction.

The second unit at the Kola nuclear-power station was commissioned in 1974 with a type VVÉR-440 reactor, the total power of the first stage of this station being 880 MW. Construction has now started on the second stage, involving two type VVÉR-440 reactors.

Armyansk, Kursk, Chernoby'sk, Smolensk, and Kalinin nuclear-power stations (this is a far from complete list of the stations constructed with types VVÉR and RBMK reactors) constitute the basis for the development of nuclear-power engineering in the USSR in the near future [3]. During the 20-year period of development of nuclear technology, there has been a significant improvement in the cost effectiveness of both nuclear-power stations and nuclear-power engineering which, as we have already noted, has become competitive with conventional power engineering. At a conference on the planning, adjustment, and operation of nuclear-power stations, data were given on the high cost effectiveness of the stations. For example, the cost of electrical energy at Novovoronezh was 0.752 kopecks per kWh in 1973 and 0.655 in 1974, which is below the cost of the electrical energy generated by the modern steam power stations in the European part of the USSR [3].

Successful operation of the types VVÉR and RBMK reactors enables us to set about creating more economical thermal reactors with unit powers of about 1.5 GW [1].

Experience with the operation of the Novovoronezh station has illustrated the reliability of the nuclear fuel, ensuring an average depth of burn-up of $33.3 \cdot 10^3$ MW days/ton U from the time of loading (1974). It should, however, be noted that this energy output arises not just from ^{235}U; a significant contribution towards the output of energy from a thermal reactor is made by a secondary fuel, plutonium, which is formed from ^{238}U during the operation of the reactor.

Table 1 gives the results of tests to determine the fuel burn-up in the elements of the two reactors at the Novovoronezh nuclear-power station [2]. These results were obtained by the gamma-spectrometer method without disrupting the fuel elements themselves. The errors in determining burn-up did not exceed ±15%.

Any increase in the duration of neutron irradiation of the nuclear fuel in a power-station reactor not only leads to an increase in the total energy generation (burn-up of the original fuel), but also increases the relative role of plutonium in the total generation of energy, i.e., reduces the specific expenditure of ^{235}U per unit power.

If we compare the effectiveness of nuclear fuel and fossil fuel, and assume that 1 ton of conventional fuel = $7000 \cdot 10^3$ kcal, then with complete burn-up of nuclear fuel in thermal reactors [(35 to 40) $\cdot 10^3$ MW days/ton of original slightly enriched uranium], a ton of fuel elements generates energy equivalent to 100-120 $\cdot 10^3$ tons of conventional fuel. With existing energy intensities of the fuel, such a burn-up would be achieved within three years on a power station, an increased burn-up being limited by the cost of the fuel elements.

In fast reactors, which employ other forms of coolant, the specific energy intensity is significantly higher, and a considerably greater burn-up can be achieved more rapidly. In thermal reactors, even with very high burn-ups of uranium and plutonium, the energy output due to fission of the plutonium nucleus does not exceed the energy developed by fission of the nucleus of natural uranium (Table 2), whereas in fast reactors, the breeding of nuclear fuel is possible. Scientists of the FÉI, under the direction of A. I. Leipunskii, have already shown in 1948-1949 that it is possible to involve not just 1 or 2% of mined uranium in the power-generating program, but ten times this, as by a combined use of uranium and plutonium it would be possible to convert practically the whole of the ^{238}U into fissionable plutonium. A study into the creation of

TABLE 3. Characteristics of Isotope Energy Sources

Isotopes	Specific energy, W/g	Half life, yr	Service life, yr	Total energy per decay event, Me	Activity/unit thermal power, curies/W
^{210}Po	144,0	0,378	0,5	5,401	31,4
^{242}Cm	122,5	0,445	0,5	6,213	27.6
^{192}Ir	59,7	0,204	0,2	1,1	154,0
^{144}Ce	26.7	0,78	1	1,409	120,0
^{60}Co	17,5	5.26	5	2,607	65,0
^{227}Ac	14.85	21.7	10	34,332	4,94
^{170}Tm	11,35	0,354	0,4	0,321	525,0
^{232}U	5,0	73,6	10	40.174	4.2
^{244}Cm	2,89	17,9	10	5,895	28,6
^{90}Sr	0.936	27.7	10	1,1	154,0
^{238}Pu	0,58	86,4	10	5,59	30,3
^{137}Cs	0,411	29,7	10	0,786	215,0
^{147}Pm	0,338	2,65	3	0,062	2725,0
^{3}H	0,36	12,26	10	0.019	$2,7 \cdot 10^4$

a physical and engineering basis for fast reactors was carried out in 1954, starting with the BR-1 reactor, followed by a series of experimental reactors: BR-2, BR-3, and BR-5. In order to work out a design for the active zone of the reactors and study the economics of the fuel elements, an experimental reactor type BOR-60 was built at Dimitrovgrad, on which experiments were successfully carried out in 1969.

At the present time, a nuclear-power station is being built at Shevchenko, equipped with a fast reactor type VN-350, designed to generate 150 MW of electrical power and desalinate 120,000 tons of sea water a day. A fast reactor type BN-600 is being built. Experience gained in operating these reactors should define the advantages and drawbacks of the chosen equipment schemes and active zones, provide a more-accurate breeding factor, and supply all the data needed to create more-economical fast reactors, the main type of reactor for meeting future power demands [4, 1].

The book "From Scientific Research to an Atomic Industry" [6] and the jubilee handbook, published in connection with the 20th anniversary of the commissioning in the Soviet Union of the world's first nuclear-power station [7], give many physical and engineering characteristics of practically all types of power reactors in the USSR, whether existing, under construction, or still in the planning stage. Successful operation of the icebreakers Leningrad and Arktika (Fig. 1) illustrates the wide possibilities of employing nuclear reactors with the fleet, in transport roles [7].

Scientists and engineers involved in the manufacture and reprocessing of fuel elements (after removal from reactors) solve problems which are concerned not just with radioactive wastes, but also with the fission products that are produced in large quantities and are used in nuclear engineering as isotope sources.

As can be seen from Table 2, for each unit weight of fuel element submitted for radiochemical processing from fast reactors with a burn-up of 80,000 MW days/ton, the concentration of plutonium isotopes is increased by a factor of ten, while the concentration of the transplutonic elements is increased multifold.

The competition of the original fuel for the fuel elements used in fast reactors includes significant concentrations of the isotopes of plutonium in addition to the ^{238}U. As an example of this, we could note that the concentrations shown in Fig. 2 of actinides produced during irradiation of fuel in a fast reactor include for each ton of uranium and plutonium, in addition to the 130 kg of ^{239}Pu, the following amounts of short-lived plutonium isotopes (in g/ton): ^{238}Pu = 2590; ^{240}Pu = 51,800; ^{241}Pu = 26,000. Therefore, when manufacturing fuel elements for fast reactors, the total activity of the original fuel can be very high indeed.

The design data given in Table 2 has been obtained under the conditions that exist in thermal reactors, in which the original enrichment of ^{235}U comprised 3.3%, while the average specific power comprised 30 MW/ton. The amount of fuel recovered in the cycle comprises 1/3 of the annual fuel loads. Fast reactors use ^{238}U (78%) as fuel together with isotopes of plutonium; the average specific power is 148 MW/ton, while 1/3 of the fuel load is transferred after 153 days.

The use of plutonium in nuclear-power engineering and the consequent radiation safety problems that arise have formed the subject of several articles [8], so that there is no point in dwelling on these topics in the present article.

The increased rate of growth of nuclear-power engineering also accelerates the rates of mining and processing uranium ores, which at the present time are the basic source of nuclear fuel.

For processing these ores, Soviet scientists and engineers have developed technical processes which enable us to extract other elements in addition to uranium, such as phosphorus, molybdenum, the rare earth elements and other minerals that are valuable to the national economy [9]. Soviet scientists have

TABLE 4. Characteristics of Nuclear Explosions in Rock Salt [17]

Yield, kton	Depth of charge, m	Radius of chamber formed, m	Distribution of fissuring above chamber, m
1,1	161	13,0	83
3,4	365	18,7	60
5,3	828	17,4	64
25,0	600	32,0	—

presented papers on the complete utilization of uranium ores, not just to Soviet conferences, but also to international conferences [10]. Complete utilization of lean uranium ores enables the cost of the uranium to be brought down and, in the case of uranium–phosphorus ores for example, enables us to extract a larger amount of the rare earth elements which are used in many fields of industry, together with valuable phosphorus fertilizers. The content of foodstuffs in fertilizers is 40-50%. The rare earth elements are used as catalysts for cracking oil and as alloying agents for cast iron and steel, while such elements as europium and yttrium are used in the manufacture of color picture tubes.

The main requirements for the material of the active zone of a reactor have compelled specialists in the atomic industry not only to create a series of new special alloys, but also to arrange their mass production. Such materials as zirconium and its alloys, special materials containing neutron absorbers, and others are widely used at the present time in nuclear-power engineering and other fields of the national economy.

Nuclear fuels are supplied for the development of nuclear power in various countries, and in the form of fuel elements to the COMECON countries and Finland.

Isotopes in the National Economy of the USSR

Not all the radioactive products separated after radiochemical treatment of the spent nuclear fuel are disposed of by reliable burial. Some of them can be used in the national economy. This applies not only to the fission fragments, radioactive products such as ^{90}Sr, ^{137}Cs, ^{144}Ce, ^{147}Pm, but also to isotopes specially obtained by neutron bombardment, such as ^{60}Co, ^{99}Mo, ^{170}Tm, and ^{210}Po when, as a rule, lighter isotopes of these same elements are used as targets. The transuranic elements comprise a separate group of radioactive elements.

The use of radioactive isotopes in the national economy of the Soviet Union is expanding every year. It is now difficult to name any branch of science, technology, industry, agriculture, medicine, etc. in which these true friends of man do not operate. The technology and reliable production capability for the majority of radioisotopes and stable isotopes of practical use have now been developed in the Soviet Union.

The production list of the Isotop All-Union Combine exceeded 3300 in 1975, including:

Compounds of radioactive tracer isotopes....................1034
Compounds with stable isotopes.............................. 718
Sources of α, β, γ, and neutron radiation.....................1622

This product list is based on 156 radioactive and 240 stable isotopes [11, 12].

By employing such α emitters as ^{210}Po, ^{238}Pu, and ^{239}Pu a large number of sources can be produced for use as radioisotope static neutralizers in the textile, printing, rubber, photographic-film, and other industries. Radioisotope signal fire-fighting equipment is widely employed in the USSR, in which isotopes of plutonium are used in indicator amounts. The total output of these devices in 1975 was 300,000 units. There are 246 types of β-radiation sources based on 11 isotopes, including tritium, ^{14}C, ^{60}Co, ^{85}Kr, ^{106}Ru, ^{144}Ce, ^{147}Pm, and ^{204}Tl. These sources are manufactured with a wide range of activities from 0.015 microcurie to 300 curies.

There is a large assortment of sources of γ and x radiation varying both in radiation power and physical dimensions.

The isotopes of plutonium, ^{210}Po and ^{252}Cf, are widely employed in the manufacture of neutron sources. Plutonium–beryllium sources have an intensity up to $5 \cdot 10^7$, polonium–beryllium up to $4 \cdot 10^8$, and californium from $1.5 \cdot 10^7$ to $1 \cdot 10^9$ neutrons/sec.

The use of radioactive isotopes can develop along three main directions:

1. As radioactive indicators (the tracer-atom method).

2. As sources of penetrating radiation for use in radioisotope instruments for automatic monitoring and regulation of industrial processes and as radioactive fuel.

3. As powerful sources of ionizing radiation for acting directly on materials to initiate various reactions and changes in structure.

The tracer-atom method is a powerful research tool which can enable us to uncover many complex processes taking place in metallurgy, in the chemical and oil industries, and machine manufacturing, to determine the nature of the activity of plants, animals, and human beings, to obtain a picture of the movements of ground water and rivers, and to define the regional distribution of agricultural pests. The method is widely used in both science and technology, in which it can be used, e.g., to determine the rate of wear in machines. There are 740 compounds used as radioactive indicators, marked by various radioactive isotopes, including ^{14}C (300 compounds), ^{3}H (125 compounds), ^{32}P (36 compounds), ^{35}S (50 compounds), and ^{36}Cl (20 compounds).

Radioisotope instruments are extensively employed in the national economy. One of their main advantages in relation to monitoring and measuring instruments based on other principles lies in the absence of any contact with the material or agent being tested. Radioisotope instruments possess high sensitivity and a high speed of reaction, they are indispensable for determining the characteristics of explosion-proof and flame-proof materials, chemically aggressive liquids and gases, and viscous or friable materials.

In the chemical and oil refining industries, radioisotope instruments enable us to exercise direct monitoring and regulation of the level of acids and alkalis in nontransparent reservoirs, to follow processes at high temperatures and pressures, to determine the concentrations of materials in solution, etc.

In metallurgy, radioisotope instruments have been successfully used to automate the process of charging blast furnaces, check the level and overflow of metals, simultaneously observe the wear in the linings of open-hearth furnaces, and provide direct monitoring of the thickness of rolled sections.

In the machine-manufacturing industry, radioisotope instruments make it possible to monitor the flow of products, ensure the interlocking of plants, etc.

In the building industry, hydraulic engineering, and agriculture, radioisotope instruments are extensively employed for measuring the density and moisture content of various materials and determining their uniformity.

As we have already seen, the introduction of radioisotope neutralizers of static electricity has solved the problem of dumping static charges that arise in various industrial processes and of preventing fires during the transfer of inflammable liquids. Data exist to illustrate their high cost-effectiveness. The annual saving achieved by the introduction of one neutralizer in the textiles, printing, and rubber industries is from 1200 to 4000 rubles. The gamma flaw-detection method is used in several fields of industry, enabling us to monitor the quality of welded joints in high-pressure boilers and lines, and to detect fractures in the reinforcement rods of ferroconcrete structures, blowholes and cracks in metal parts and castings.

The percentage distribution of radioisotope instruments by branches of industry is as follows.

Machine manufacturing	16.6
Foodstuffs	13.4
Metallurgical	12.8
Mining	11.7
Chemical	11.1
Light industry	9.0
Building	8.5
Other fields of industry	16.9

Certainly, these figures might change due to renewal of the instrument catalog; they do, however, reflect clearly the range of uses of this form of atomic technology in the national economy.

When selecting isotopes for radiation sources, it is advisable to consider not only the type of decay and the radiant energy, but also characteristics such as specific generation, projected period of service, and the chances of obtaining large quantities of the isotope.

Table 3 gives some of the characteristics of the basis isotopes which are suitable for use as radioactive fuel and other purposes. Bearing in mind the fact that isotopes with half lives from 100 days to 100

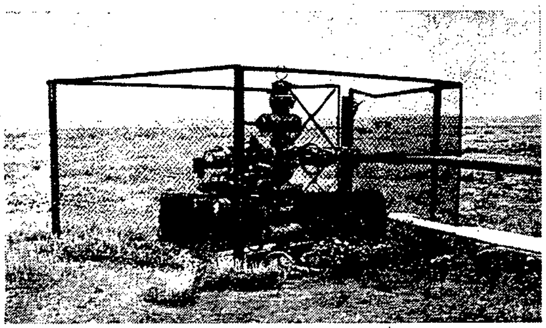


Fig. 2. Surface equipment for 50,000 m^3 condenser created by an underground nuclear explosion.

yr are used as energy sources, only 14 isotopes are given, all of which are available in large quantities [12].

When choosing a suitable operating life for the isotope sources, it is necessary to take into account the radiation stability and thermal stability of the chemical compounds, together with those of the fuel ampoules, and the actual service life to be expected of the device itself.

Table 3 gives data on ^{60}Co, ^{137}Cs, ^{192}Ir, ^{170}Tm, as well as ^{75}Se, ^{241}Am, and others which are used for monitoring the quality of products made of steel and other materials by the gamma flaw-detection method. This method has become one of the principal methods of nondestructive testing of materials and manufactured products, and is suitable for a wide range of thicknesses, from 0.5 to 200-250 mm steel equivalent.

In recent years, radioactive isotopes have found wider application as powerful sources of ionizing radiation in direct action on various processes and materials for the purposes of improving manufacturing techniques and obtaining new materials and compounds to meet the needs of modern technology.

The heat resistance of polymer materials and products made from rubber can be greatly improved by irradiation; the strength of wood can be increased and the quality of cotton fabric improved. The first industrial radiation-chemical plants have been set up and operated successfully at Volgograd, Kazan, Grozny, Podol'ski, and other industrial centers.

In agriculture and the food industry, ionizing radiation offers the possibility of increasing yields, accelerating the breeding of new types of plant, and increasing the storage life of food products. Tests carried out under production conditions using presowing radiation methods and radiation treatment of products have given encouraging results at a number of installations.

The use of manufacturing methods based on radioactive isotopes enable us to achieve significant economies each year in many fields of the national economy.

In medicine, radioactive isotopes and nuclear radiation have been successfully used in the diagnosis of complex illnesses, for the treatment of malignant tumors, for the radiation sterilization of materials, instruments, and medicinal preparations.

The Izotop combine is currently supplying more than 35 radiopharmaceutical preparations to medical institutes in the Soviet Union and abroad, while more than 37 preparations are undergoing medical trials. The isotopes ^{131}I, ^{198}Au, ^{32}P, ^{133}Xe, and ^{99}Tc are of greatest use in radioisotope diagnostics and therapy. About 650 institutes and medical establishments in the Soviet Union employ tracer preparations based on radioactive isotopes, designed for medical or medicobiological purposes.

Stable isotopes such as ^{2}H, ^{15}N, ^{13}C, and ^{18}O are also employed in chemistry and biology for studying the mechanisms of chemical reaction and the replacement processes of materials and living organisms.

Atomic science and technology enable us to create full-scale production of isotopes in the Soviet Union by various methods and to expand the production base of radioactive and stable isotopes, tracer compounds, instruments, and equipment using isotope radiation sources. Isotopes produced in the USSR are exported to 32 countries.

Employing the Energy of Peaceful Atomic Explosions

In recent years, several countries, including the USSR, USA, UK, and France, have studied the possibility of using nuclear blasts for peaceful purposes. This problem has been studied at four conferences, organized by MAGATE in 1970-1975 [13]. Specialists have showed conclusively that peaceful nuclear explosions, having exceptionally high concentrations and low specific energy costs, can be used to create new processes in the construction and mining industries, and also for scientific investigations. These would not present any danger to the population or the environment in the sense of seismic or radiation hazards, provided certain rules were observed [14].

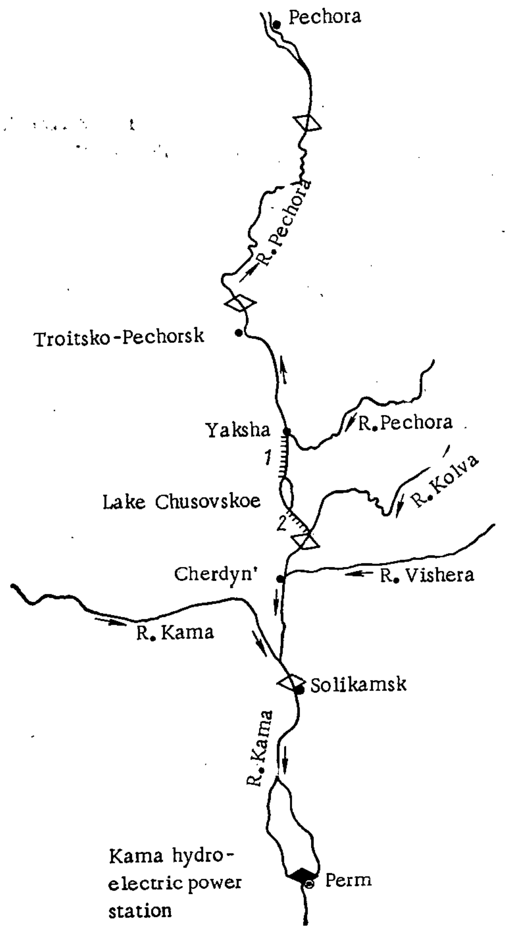


Fig. 3. Approximate arrangement of the main works for transferring part of the flow of the northern rivers into the Volga basin [21]:) Perchoro-Kolvinsk canal: 1) northern section (River Pechora to Lake Chusovskoe); 2) southern section (Lake Chusovskoe to River Kolba); ◇) planned sites of dams.

Considerable interest exists within the USSR in the future possibilities of using peaceful nuclear explosions for carrying out large civil-engineering works or developing useful mineral resources spread over vast expanses of territory.

Depending on the degree of study involved, the technology of peaceful nuclear explosions can be divided into three groups: industrial, experimental, and research.

The first group relates to such examples of industrial or experimental-industrial uses of explosions as the elimination of dangerous gushers of natural gas, increasing the rate of oil extraction, the creation of subterranean cavity reservoirs in solid rock salt and open reservoirs for storing water.

Underground nuclear explosions have been successfully used in the USSR to stop-up accidental natural-gas blowholes. The flow from one of these reached 12 million m^3/day, and for a long time it was impossible to stop this by conventional methods. A 30-kton nuclear explosion at a depth of 1550 m permanently closed off the shaft of the blowhole [15, 16].

Experimental explosions have been set off to speed up the production of oil. A long period of operation of the site after three explosions had been set off showed that the production rate was 27-60%greater than expected. due to the creation of artificial fissuring [15].

An artificial reservoir for water, with a total capacity of about 20 million m^3, was formed by an excavating nuclear explosion with a yield of over 100 kton; the visible crater had a total volume of about 7 million m^3.

The demand for reservoirs has greatly increased over the last ten years due to the rapid development of the gas, oil-refining, and other industries. The search for more-effective methods of creating reservoirs has lead to trials with camouflet nuclear explosions in solid rock salt. Due to the elastoplastic properties of rock salt, it is possible in this way to create large stable chambers (Table 4).

Figure 2 shows the surface equipment for a condenser reservoir created in the solid rock salt at a depth of 1140 m by a nuclear explosion of about 15 kton. The chamber has a volume of about 50,000 m^3 and operates under a pressure of about 80 atm, created by gas from the gas line, the condensate being extracted at the surface [17]. Successful operation of this condenser reservoir illustrates its advantages compared with metal reservoirs built on the surface: the cost of construction is less by a factor of 3-5 than a conventional reservoir, the cost in metal is lower by a factor of 10-20, the construction time is several times less, and the industrial land use is lower by a factor of hundreds. Furthermore, the cost of ensuring safe operation of the reservoir is greatly reduced [18].

Starting with a defined yield of explosion, this method is more advantageous, and at the same time more universally applicable, than the method by which chambers are washed out of the solid rock salt. Constructing reservoirs by means of nuclear explosions is more effective in regions of new industrial development before the actual construction stage is reached, as in these cases more powerful explosions can be used and the reservoir can be sited near to the factories themselves.

The second group relates to explosions which can be used in the construction of large hydraulic works (canals) and for stripping areas of their useful minerals. Plans of this nature are known to include the construction of the Pechoro-Kolvinsk canal in the USSR, the Orinoko-Rio Negro canal in Venezuela,and the Kra in Thailand. It has been calculated that the use of underground nuclear explosions during the construction of these canals would considerably reduce the cost and the time needed for construction [19].

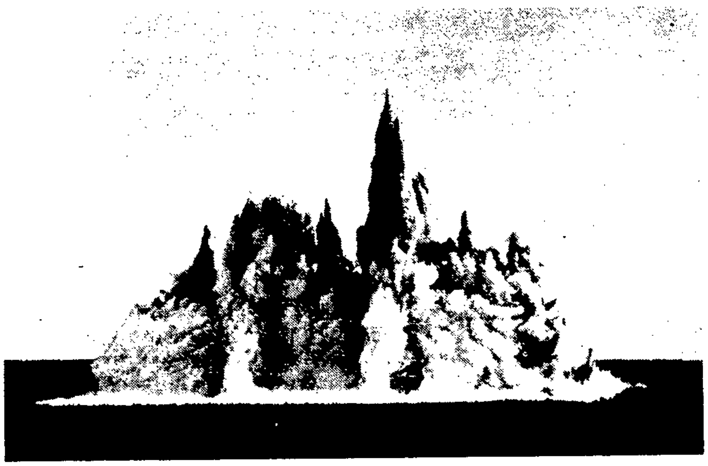


Fig. 4

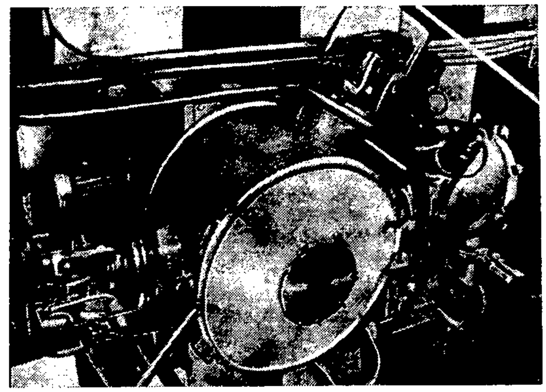


Fig. 5

Fig. 4. Photograph of the development of group underground nuclear blasts on the route of the Pechoro-Kolvinsk canal.

Fig. 5. "Bulat" equipment for producing wear-resistant coatings.

TABLE 5. Basic Characteristics of Mobile Equipment for Neutron-Activation Analysis [24]

Type	Accelerating voltage, kV	Pulse duration, μsec	Neutron output, neutrons pulses	Av. neutron flux, neutrons/sec	Recurrence freq., Hz
NGI-1, 2	80—140	1,5—2	$3\cdot10^7$	$3\cdot10^8$	1—10
NGI-4	110	1—1,5	$0,8\cdot10^7$	$1,5\cdot10^8$	1—33
NGI-5	150	1,5	$0,6\cdot10^8$	$6\cdot10^8$	1—10

The article "Rational utilization and protection of water resources" [20] reports on large-scale works that have been carried out in the USSR to redistribute the pattern of river flows by transferring water to regions with a deficit on their water-economy balance. The advantages are stressed of transferring part of the flow of northern rivers into the Volga basin, to the extent of 20-25 km^3/yr (40-50 km^3/yr in the future), which would be equivalent to increasing the flow of the Volga by 15-20%. The creation of the 112.5-km Pechoro-Kolvinsk canal (Fig. 3) basically reflects this problem, and envisages the growing demand for water in the central and southern regions of the European part of the USSR being met, the level of the Caspian Sea being stabilized, and the output of electricity from the Volga hydroelectric cascade being increased. It is proposed to construct the northern section of the canal (65 km long, useful cross-sectional area 3000 m^2) by means of nuclear blasts. About half of this section passes through a zone of flooded alluvial deposits. No experience in the creation of channels in this type of ground is available, so there was considerable doubt as to mechanical effects of an explosion, the stability of the banks of any channel formed, and the seismic and radiation effects. In order to study these problems, an exceptional blast was set off under similar geological conditions [13, 22]. Three charges of 15 kton were used in the experiment, in three boreholes at a depth of about 128 m. The distances between the boreholes were 163.1 and 167.5 m. Figure 4 shows a general view of the development of a blast 5 sec after the charges were detonated. The blast formed a channel 700 m long, 340 m wide, and from 10 to 15 m deep. The sides of the channel were formed at an incline of 8-10% and these have shown practically no alteration with time. Thus, for the first time it was shown to be possible to form large channels with sufficiently stable banks in a thick cover of weakly flooded ground [22].

A project has been developed for the intensification of mining for useful minerals using nuclear blasts. In the project, large deposits of light metals on the northeast frontier of the Soviet Union can, according to preliminary estimates, be stripped by open-cut mining techniques of up to 900 million m^3 of rocky ground [15].

The third group contains the most complex fields of use of nuclear blasts: the distillation of oil from shales, the leaching of copper from deposits, the release of geothermal energy, the creation of reservoirs for hazardous and radioactive waste products, etc. Plans exist for the utilization of blasts for scientific purposes [13].

Peaceful nuclear explosions represent a new field of application of atomic energy, requiring further, more detailed investigation. There are grounds to suppose that their potential is neither fully realized nor exhausted.

TABLE 6. General Characteristics of Accelerators for Practical Uses

Type	Class	Energy of accelerated electrons, MeV	Power in beam, kW		Output of radiation at 1 m from target, R/min	Diameter of beam leaving tartet, mm	Size, m	Weight, tons
			av.	impulse				
RTD-1	Resonant transformer	1,0	3	18	60	0,25	⌀ 0,0 × 1,5	1,9
Élektron-1	Transformer	0,7	7	7	—	—	⌀ 0,7 × 3,0	1,0
ÉLIT-500	The same	0,5	1	700	18	—	⌀ 0,3 × 0,5	0,04
ÉLIT-1	» »	1,0	8	10 000	360	—	⌀ 0,4 × 0,6	0.12
ÉLIT-3	» »	2,5	10	40 000	—	—	⌀ 1,0 × 1,3	0,8
ÉLT-2	» »	1,5	25	215	—	—	⌀ 1,3 × 2,4	7,0
KGÉ-2.5	Cascade generator	2,5	20	20	—	—	⌀ 3,0 × 6,2	32,0
LUÉ-8-5V (emitter)	Linear accelerator	8,0	5	3 500	—	—	5,0 × 0,7 × 1,75	2,0
LUÉ-13-9 (emitter)	The same	13,0	9	11 000	—	—	5,5 × 1,5 × 2,35	5,0
LUÉ-10-1 (emitter)	» »	10,0	1	1 000	2 000	1,5	2,75 × 1,0 × 0,8	2.0
LUÉ-15-1.5 (emitter)	» »	15,0	1,5	1 500	10 000	2,0	4,5 × 1,5 × 2,0	5,5
B-25	Betatron	25,0	—	—	40	—	—	2,5
B-35	The same	35,0	—	—	250	—	—	5,0
B-50	» »	50,0	—	—	800	—	—	20,0

Fig. 6. Implantation equipment type ILU-4 for surface alloying.

The problem of disposing of radioactive wastes is of great significance to the development of the atomic industry. One of the most promising ways of solving this problem, according to the results of many years experience, is by the disposal of solutions containing radioactive matter in underground absorbent beds in the Dimitrovgrad region [23]. The results of this trial were put to partial use on factories of the USSR Ministry of the Chemical Industry.

Absorbent strata have been found in the vicinity of several chemical combines and turned, without the use of nuclear blasts, into proving grounds for underground burial of concentrates from chemical treatment of toxic materials. Table 4 gives the volumes of underground reservoirs produced by means of experimental nuclear blasts. These give us reason to suppose that in regions where there are no underground absorbent strata it would be possible to create underground chambers by means of nuclear blasts for burying dangerous and radioactive wastes. This supposition stems from hydrogeological and engineering-geological questions concerning the radiation possibilities of the underground vaults used for industrial waste products.

Nuclear Physics and Plasma Physics in the National Economy

The development of nuclear physics, plasma physics, solid-state physics, and atomic materials has not merely created a scientific foundation for atomic engineering, it has also actively facilitated the introduction of the so-called "fall-out products" or by-products of these branches of science in the national economy. For example, until quite recently, charged-particle accelerators have been used only for research into nuclear physics. As their design improved, however, it became clear that accelerators could be employed in various branches of the national economy. In industry, agriculture, and medicine charged-

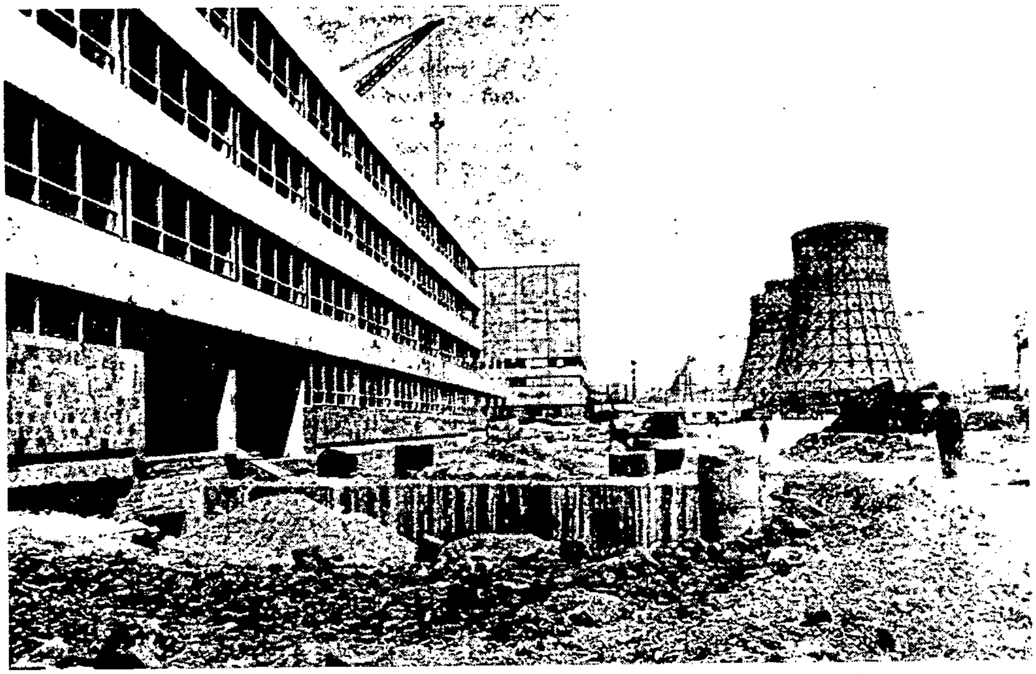


Fig. 7. An atomic building in the USSR. A general view of the construction of the Armyansk nuclear power station. Photograph by V. Bratchikov.

particle accelerators are finding even wider applications, the accelerator developed by the D. V. Efremov NIIÉFA, for example.

Activation analysis is widely used in geological surveying and metallurgy for determining the contents of various elements. Neutron-activation methods employing accelerators have been used to analyze contaminants during the production of very-pure structural materials. The test material is irradiated by neutrons formed by bombarding special targets with accelerated electrons. The neutrons activate contaminants present in the test material, forming short-lived isotopes, by the study of which it is possible quickly to determine the presence of certain elements with a sensitivity measured in hundred-thousandths of a percent. The time needed for analysis is greatly reduced and fewer laboratory technicians are needed. Table 5 gives the characteristics of a number of devices used in the national economy.

Neutron generators of type NG-150 are used in stationary laboratories. These have a flux of about 10^{11} neutrons/sec.

In addition to neutron generators, γ-radiation sources, hydrogen-ion, deuterium-ion, and helium-ion accelerators are also used for activation analysis.

A great deal of attention has been devoted during the last ten years to the problems of designing accelerators for use in flaw detectors and radiation engineering processes, and also in medicine. Such accelerators have a number of advantages over cobalt sources:

a. They enable us to obtain bremsstrahlung having a higher penetration and with doses in the hundreds of rads/min · m at a distance of up to 1 m from the target.

b. Besides brehmsstrahlung, we can obtain beams of electrons, protons, mesons, etc. with energies ranging from hundreds of kiloelectrovolts to hundreds of megaelectronvolts.

c. They enable us to form uniform dosage fields with the clean boundaries needed for therapy, for example.

d. They ensure radiation safety as far as auxiliary operations are concerned; e.g., there is no need to periodically reload and bury spent highly radioactive charges.

Radiologists throughout the world are especially interested in the linear electron accelerator, as this is very simple in design compared with other types of accelerator, the introduction and extraction of particles is very simple, and the energy and power of the radiation doses can be regulated. They can produce powerful doses (not just at high energies but over the whole energy range) in large highly uniform fields.

Linear accelerators have been created for flaw detection at energies of 6-9 MeV with bremsstrahlung intensities of 300-1500 R/min and an energy of 15 MeV with intensities of up to 10,000 R/min (types LUÉ-10-1, LUÉ-15-1.5). Flaw detector accelerators are able to penetrate steel products to depths of more than 400 mm, due to their high power and the penetrating ability of the x rays they generate. Linear accelerators for flaw detectors with energies of 6-9 MeV have been successfully operated at the Izhorsk factory at Kolpino, enabling the test time to be reduced by a factor of 10-12 compared to the cobalt equipment used previously, and increasing the visibility of flaws in product thicknesses of more than 150 mm.

Linear accelerators have been manufactured for sterilizing medical instruments, enabling the degree of sterilization to be increased; expendible medical instruments to be put into mass production, ready for immediate use (hypodermic syringes, catheters, blood-transfusion systems, etc.); cheap plastics, which are unable to withstand thermal sterilization, to be used for the manufacture of instruments; antibiotics to be sterilized which are unable to withstand any method of sterilization other than radiation. Two linear accelerators have been built for the medicinal preparations factory at Kurgan.

Linear accelerators are now being built with energies in the 5-30 MeV range, for treating cancers by braking (x ray) radiation and electron beams. Such accelerators have been successfully used in medical-research institutes at Moscow, Kiev, Minsk, and Obninsk. The use of linear accelerators for treating cancer shows that they possess greater biological effectiveness than cobalt radiation sources.

Experience of operation (the accelerator at the Central Institute for the Improvement of Medicine has been operating since 1966) and the achievements of accelerator technology have enabled a new improved therapeutic linear accelerator to be built with an energy of 15 MeV, which has good characteristics, is easy to control, and is designed for quantity production [24].

The Tire Industry Research Institute has operated a linear electron accelerator manufactured by the NIIÉFA with an energy of 7-8 MeV and a power of 3.5-5.0 kW. The accelerator is designed for studying the technology of radiation vulcanization of rubber tires with the aim of increasing the service life of automobile tires. Table 6 gives the basic types of accelerators used in various fields of the national economy.

Furthermore, intensive beams of heavy ions can be used to produce materials with new properties, study radiation damage in materials, and carry out tests on structural materials relatively quickly.

By irradiating thin films with heavy ions it is possible to produce nuclear filters having apertures with diameters from 20-40 Å up to several hundreds of microns. The Joint Institute of Nuclear Research (JINR) has developed the technology for producing these films and for creating special equipment. Nuclear filters are now being tried out in a number of organizations, as their fields of application are diverse.

Thermonuclear investigations being carried out at the I. V. Kurchatov Institute of Atomic Energy (IAE) and the KhFTI have also found applications in the national economy. The KhFTI have developed a plasma method of obtaining materials with new properties by condensation of matter from the vapor phase in a vacuum while at the same time being bombarded with ions. This method is based on the properties of a low-voltage electric arc in a vacuum. As the arc burns, the material of the cathode is vaporized. At the same time, the cathode effect creates a rapid flow of the plasma formed during ionization of the cathode.

The substrate (component, instrument, or material), on which it is desired to deposit the coating, is held at a negative potential, so that the plasma ions bombard a layer of condensate on the substrate. The KhFTI has developed on industrial equipment type "Bulat" (Damask steel) which produces high-temperature wear-resistant coatings and materials with good physical, technical, and mechanical properties. Tests have shown that durability of a cutting tool reinforced in this way is 2.5 to 5 times greater on the average. At present, the technology of reinforcing and hardening metal-cutting tools by the ion-bombardment-in-a-vacuum method is being introduced on a number of factories in various fields of industry [25].

In the I. V. Kurchatov Institute of Atomic Energy implantation equipment type ILU-4 has been developed for the surface alloying of semiconductors and other materials by the ion-bombardment method. The equipment is in quantity production and is being used for semiconductor studies and investigation of radiation effects in solid bodies [26]. About 50 such devices are now in successful operation, some of them in Bulgaria, Hungary, and the German Democratic Republic.

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EXPERIENCE IN THE CONSTRUCTION OF LARGE POWER REACTORS IN THE USSR

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In the directives of the 24th Congress of the Communist Party of the Soviet Union, among the problems related to the development of power generation in the country, reference was made to the need to put into operation during the 1970-1975 period the Leningrad Atomic Power Station (LAPS), which will have a capacity of 2,000,000 kW. This means a very substantial introduction of nuclear-fission energy into electrical power generation in the coming years. The task assinged by the directives was fulfilled: The first reactor, with an electrical power of 1,000,000 kW, began operation in December 1973 and achieved full power by the 57th anniversary of the October Revolution; the second reactor, identical with the first, went into operation in August 1975, and by December 1975 its power output was approximately 900 MW. The second reactor will also undoubtedly achieve full designed power in the near future. The LAPS, named for the great Lenin, is equipped with channel-type uranium–graphite reactors (RBMK), which have been described repeatedly in the literature [1-4]. The theoretical and engineering principles of this reactor were developed and tested in practice in the Soviet Union, and therefore it can rightfully be regarded as a Soviet type of reactor. The construction and introduction into operation of the LAPS reactors means that one more important landmark has been passed in the process of improving and developing this type of reactor, the conception of which dates from the late 1940's, when the first such reactors were constructed, including the reactor of the world's first atomic power station at Obninsk. The next landmark was the start-up in 1958 of the reactor of the Siberian Atomic Power Station, a film of which was shown, in particular, to the participants in the 2nd International Conference on the Peaceful Uses of Atomic Energy, held at Geneva in the same year. After this, in 1964, the channel-type uranium–graphite reactor of the I. V. Kurchatov Atomic Power Station at Beloyarsk was put into operation. In 1967 a second reactor, with an electrical power of 200,000 kW, was started up at Beloyarsk. These reactors, based essentially on the same technological idea, differ in principle in that, first of all, their fuel channels are cooled by boiling water and, second of all, the steam generated is superheated in special channels in the same reactor. More than 12 years of operation of the reactors of the Beloyarsk APS have confirmed the viability of such a solution. It should be noted that such solutions, carried to the point of satisfactory results, do not exist in any other country in the world, despite many attempts that have been made. Nuclear superheating of steam before it enters the turbine is a very tempting idea, since it not only prevents the danger of wet steam entering the turbine but also makes it possible to do without the intermediate moisture separators and without superheating of the steam between turbine stages, which in turn makes it possible to simplify the production of steam in the reactor. For turbines with large power, e.g., over 800 MW, which may be required in the construction of large atomic power stations, the initial superheating of the steam makes it practical to use a speed of 3000 rpm instead of the 1500 rpm now used in operation with saturated steam. Another significant fact is that the efficiency of the entire installation is improved.

The reactors of the LAPS, like those of a number of other APS now under construction, have no channels for the superheating of steam and produce dry steam, obtained in channels with boiling water in a single-loop scheme and transmitted directly to the turbine. As a consequence of this, it is necessary to include between the stages of the turbine a number of moisture separators and steam superheaters. The next stage in the improvement of channel-type uranium–graphite reactors will undoubtedly be the introduction into the active zone of channels for the superheating of the steam. This will come with the next generation of reactors. They will have an electrical power of 2-3 million kW; the reliable operation of such reactors will be based on the experience obtained in the operation of reactors now under construction, with the complex physics of their large active zones and with the use of modern computers for detecting at the proper

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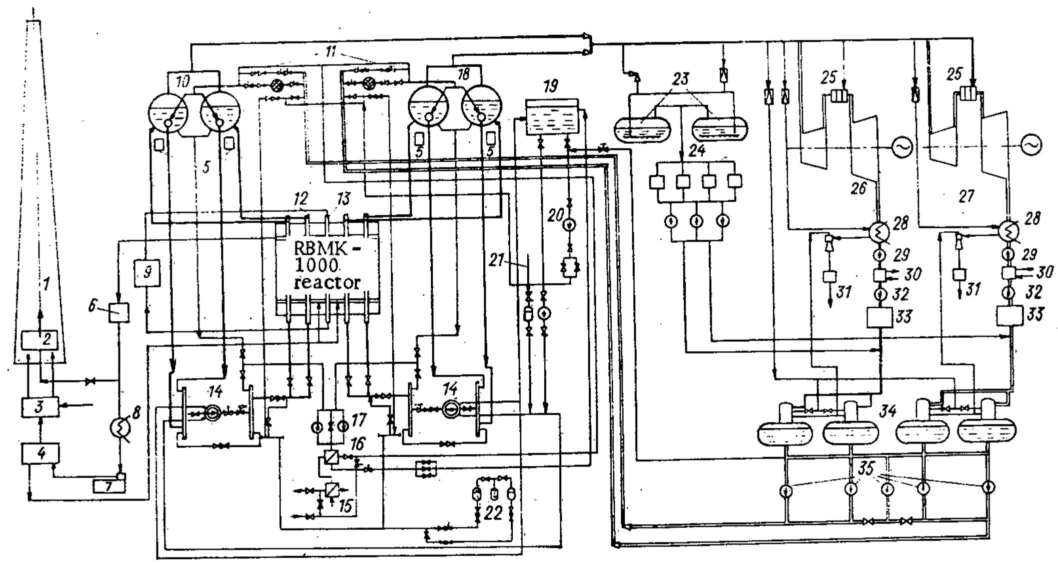

Fig. 1. Theoretical thermal scheme of the LAPS: 1) ventilation stack; 2) set gasholder; 3) storing gasholder; 4) helium-cleaning unit; 5) KGO system; 6) monitoring of integrity of technological channels; 7) compressor; 8) gas-loop condenser; 9) SUZ pump-heat exchange installation; 10) separators; 11) regulating assemblies; 12) fuel channel; 13) SUZ channel; 14) TsVN-7 pump (4 pumps); 15) precooler; 16) regenerators; 17) cooling pumps; 18) separators; 19) APN tanks; 20) emergency feed pump; 21) compressed air; 22) emergency cooling system of the reactor; 23) bubblers; 24) technological condensers; 25) separator-superheaters; 26) TG-1 turbogenerator; 27) TG-2 cleaning; 28) condensers; 29) KN-I condensate pumps; 30) condensate cleaning; 31) units for ignition of explosive mixture; 32) KN-II condensate pumps; 33) low-pressure preheaters; 34) deaerators (7 atm); 35) electrical feed pumps.

time the phenomena going on in the active zones and the effects on the automatic control devices. We must believe that scientific and technical progress in the next few years, particularly in the fields of metallurgy, physical chemistry, and instrument design, will lead to the realization of these ideas as early as the next 5-10 years.

The theoretical thermal scheme of the V. I. Lenin APS is shown in Fig. 1. It is a single-loop scheme, which differs from the known schemes for boiling reactors. The difference lies only in the design of the reactor, in the present case an RBNK-1000 channel-type uranium–graphite reactor. In estimating the advantages of this type of reactor, the following considerations are weighed: the existence of extensive experience in the construction and operation of such reactors; the absence of any specific and new technological processes, so that it is possible to get orders filled by the machine-construction industry without unduly great expense for the retooling of factories, and consequently, without requiring very long delivery times; the possibility of constructing reactors of any dimension by using mass-produced elements and assemblies, i.e., there are practically no limitations on the increase of the unit power of the channel reactors; the structural separation of the moderator and the coolant, making possible a fairly flexible choice of the substances and materials used for them, thereby ensuring effective heat removal in the active zone, with good neutron balance; the possibility of recharging an operator reactor with fuel without reducing the power, thus improving the economic indicators of the atomic power station, since in this case there is practically no need of a reactivity excess for fuel burnup; the simplification of the system for monitoring the condition of each channel, and the possibility of operational replacement of fuel assemblies which have developed leaks; the fact that the cooling loop of the reactor consists of many smaller loops of small-diameter pipe, improving the safety of the installation; the possibility of easily adapting the reactors to the conditions of the fuel market; the possibility of continuously introducing new structural elements and assemblies, with the use of the most modern advances in nuclear-fuel and reactor-material production technology; the convenience and simplicity of introducing nuclear superheating of steam into the scheme of the APS.

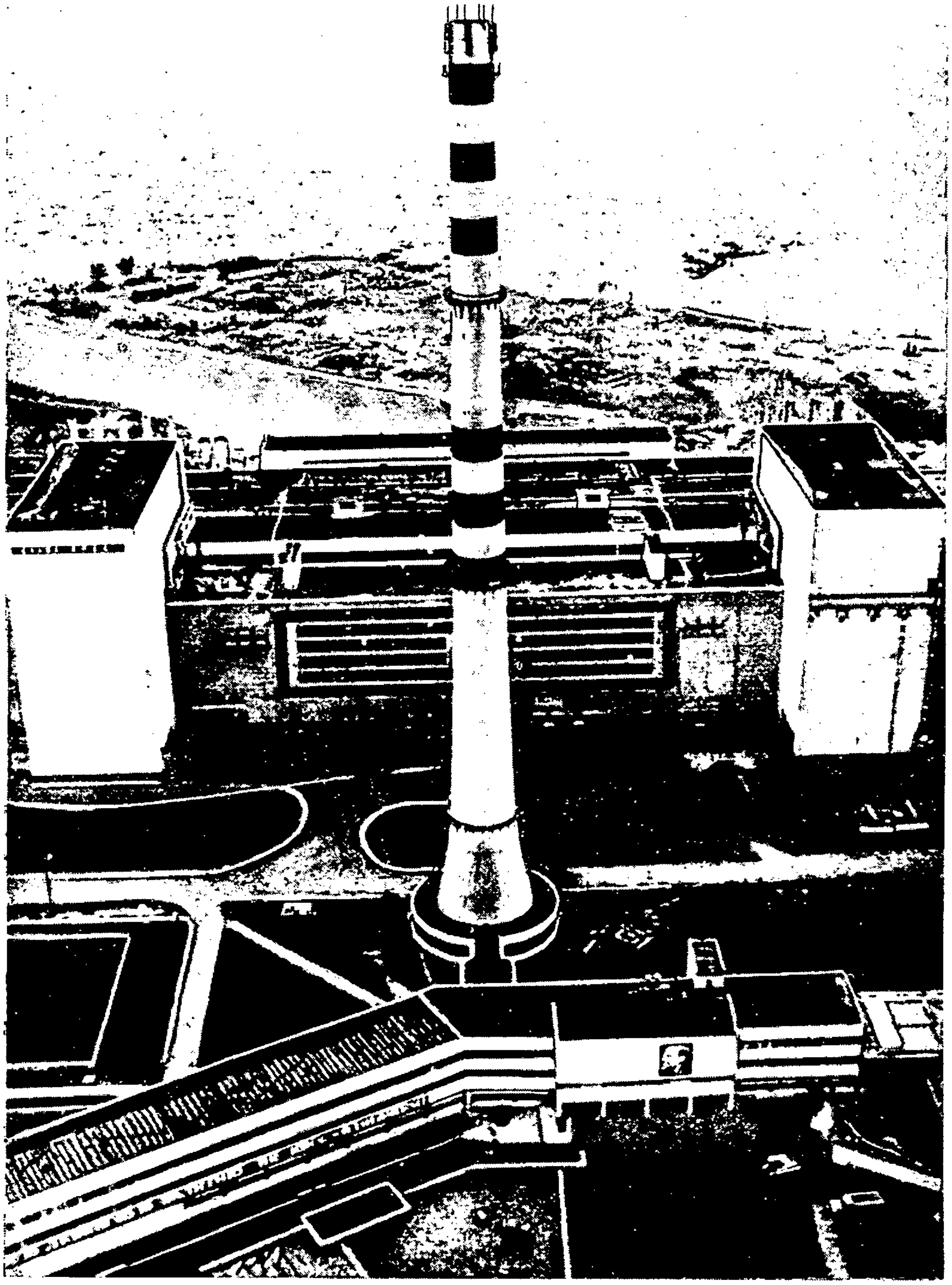

Fig. 2. Overall view of the LAPS in operation. Photo by V. Volkov.

The development of channel-type uranium–graphite reactors is inextricably connected with progress in the technology of reactor materials. The problem that had to be solved was that of constructing a power reactor with a satisfactory fuel cycle and, at the same time, a fairly satisfactory efficiency for the power station as a whole. This required, above all, new structural materials which would retain their strength to high temperatures and would have a low cross section of neutron absorption. Such materials – zirconium-based alloys for the structure of the channels – were produced. As in the case of the Beloyarsk APS, a one-loop thermal scheme was adopted for the RBMK reactors, and ordinary boiling water was selected as the coolant. This solution was based on many years of experience with the operation of boiling reactors.

The cylindrical stacking of an RBMK reactor consists of individual graphite columns with axial cavities which contain the fuel channels and the SUZ channels. A fuel channel is a tubular construction whose

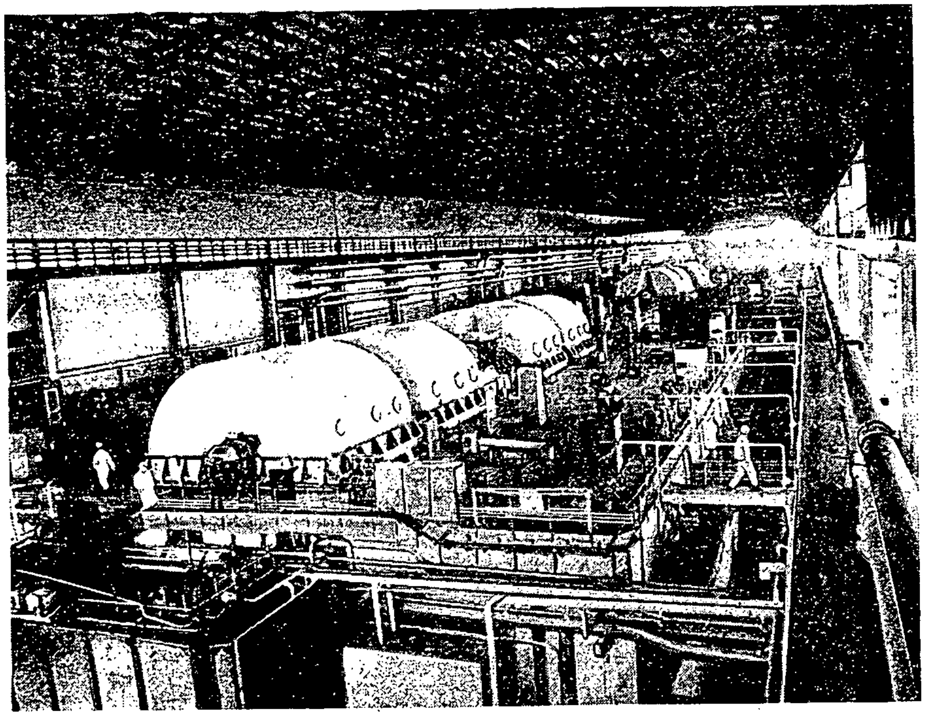

Fig. 3. Machine room of the LAPS.

central position, situated in the active zone and made of a zirconium alloy, is joined to the upper and lower parts, which are made of stainless steel, by means of special adapters. The fuel channel contains a cassette with two heat-generating assemblies, each of which consists of 18 fuel elements. A fuel element is a zirconium-alloy tube measuring 13.5 × 0.9 mm and filled with pellets of uranium dioxide. The active zone, 11.7 m in diameter and 7 m high, contains about 1,700 fuel channels with 195 SUZ channels.

The coolant is water introduced from below into each channel, heated to the boiling point, and partly vaporized in the active zone. The resulting steam–water mixture is removed from each channel into the separators. The saturated steam at a pressure of 70 atm is directed to two turbines, rated at 500,000 kW each, and the separated water, mixing with the feed water, is delivered by the main circulation pumps to the inlets of the channels through a system of distributing collectors. The reactor is equipped with: a control and protection system which, on predetermined signals, takes the reactor to different power levels, until it has reached the subcritical state; a system of physical monitoring of the distribution of energy generation with respect to the height and radius of the active zone; a system for monitoring the tightness of the seal of the fuel-element jackets; a system of channel-by-channel monitoring and regulation of the coolant flow rate; a system for monitoring the integrity of the channels in the reactor. Because there are so many parameters to be monitored, an automatic system of centralized monitoring is used, making it possible to measure and record the parameters of each block. The system includes a digital computer for processing the information and for the operational calculation of a number of parameters that are important in the running of the reactor.

The working design for the RBMK reactor was completed in 1969, and the factories began to construct the reactor in the same year. Construction work on the site was begun in 1968, and the installation of the equipment began in March, 1971. On September 10, 1973 the first heat-generating assembly was charged into the reactor, and physical startup was begun. The construction of the world's largest channel-type nuclear reactor in such a short time was possible because the engineers and technicians had behind them more than 20 years of experience in the successful operation of such reactors. The physical start-up process included charging of the channels with fuel assemblies and rods of additional absorbers. At a number of specified intermediate states, the reactor was brought to criticality in order to carry out experiments in determining its neutron-physics characteristics. As a result, the initial charge of the active zone was formed, the reactivity effects and the effectiveness of the control rods were determined, and recommendations were worked out for the manner in which the control rods should be withdrawn when the reactor was brought up to power.

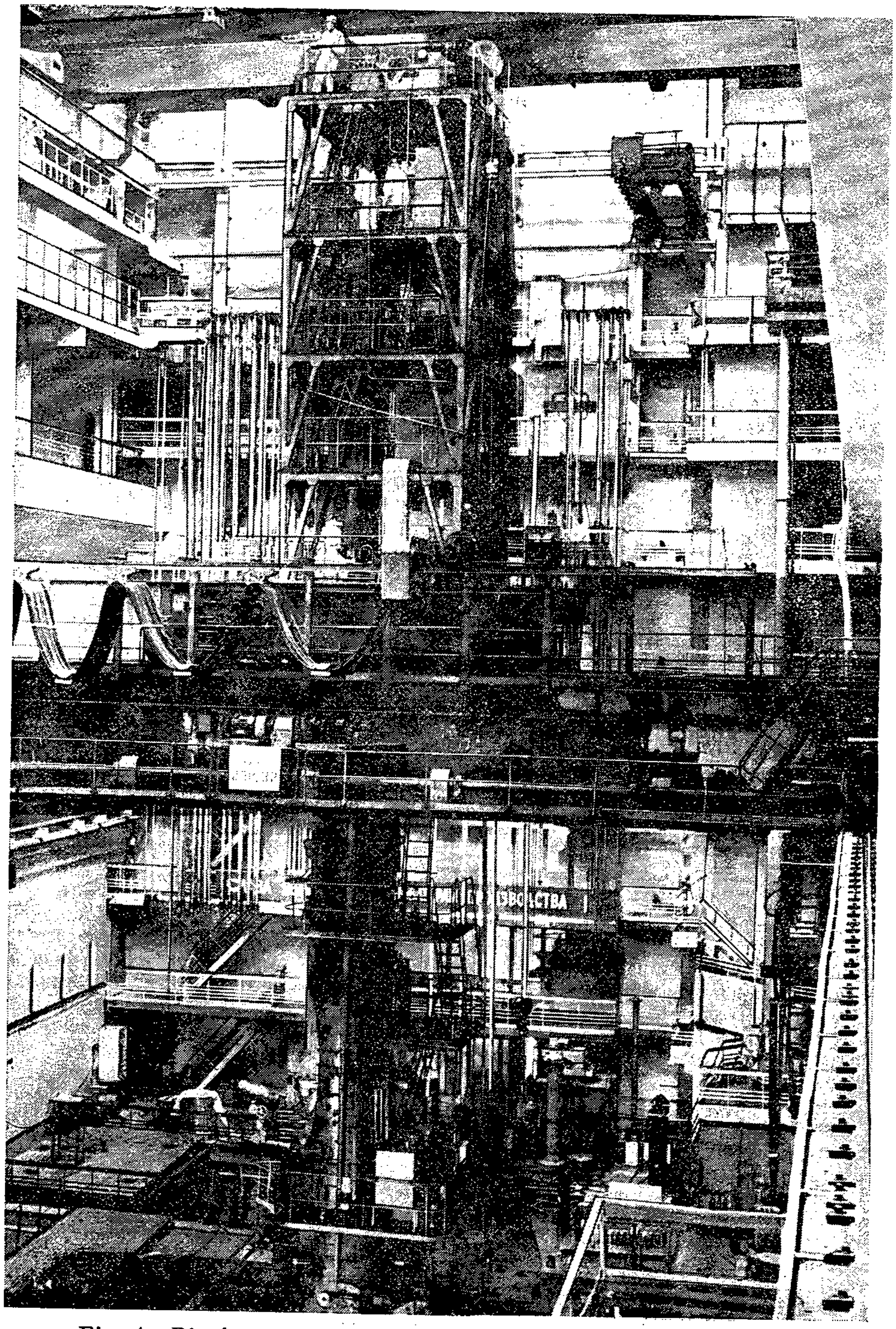

Fig. 4. Discharging and charging machine for the RBMK-1000.

From November 14 to December 21, 1973 power start-up on the first block of the LAPS was carried out and the process of bringing it up to rated power was begun. In the earliest stages, the criterion for safe operation of the active zone was the condition that the possibility of a heat-exchange crisis had to be precluded in the channel with maximum power and minimum water flow rate. The power of the block was increased by consecutive stages to 500 MW, and in the spring of 1974, after a second turbogenerator was connected, the power was increased to 600 MW. During this period special attention was paid to the investigation of the energy generation fields in their active zone and to their equalization and stabilization. For monitoring the state of the active zone before the regular system was brought into operation, the LAPS staff used a complex of programs of physical and heat-engineering calculations, prepared for an external computer but using as the initial data the readings of the energy-generation monitoring sensors, flow-rate meters, control-rod position indicators, etc.

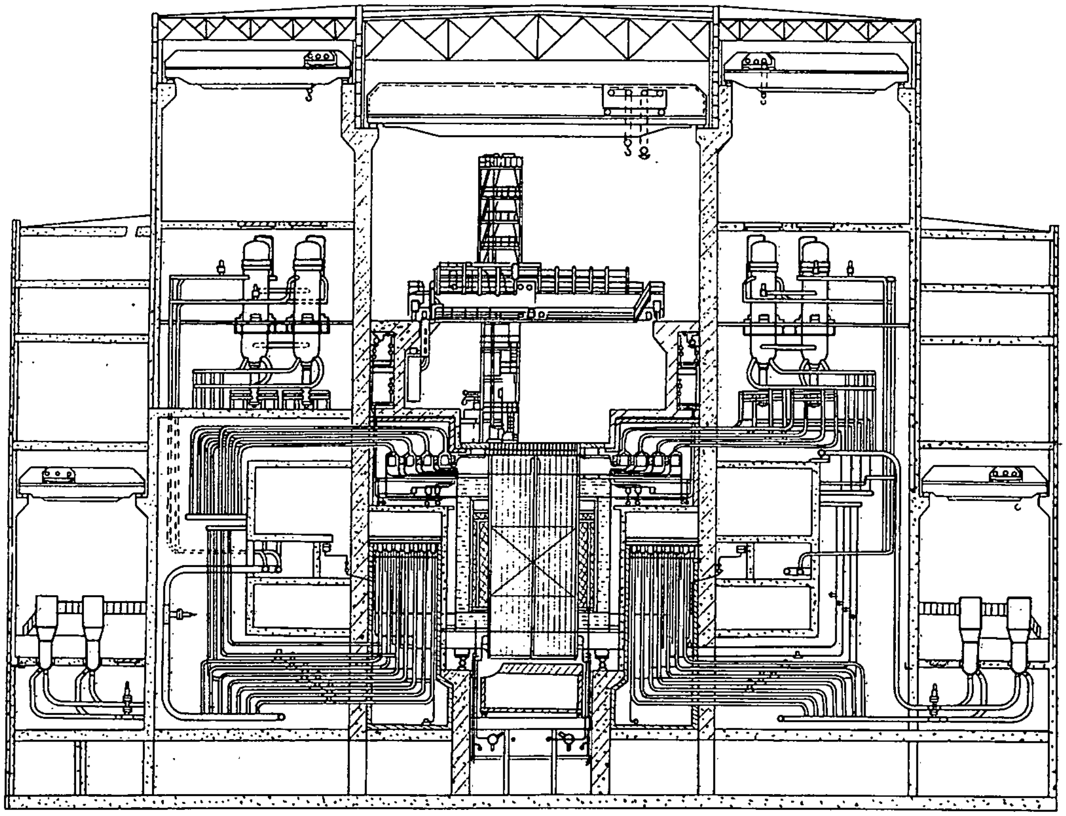

Fig. 5. Arrangement of equipment in the RBMKP-2000 (cross section).

The calculations were used in determining the distribution of power and excess values before a crisis in the reactor channels; the coolant was distributed in the channels in accordance with the power values. In July 1974 the block was raised to a power of about 800 MW. At this power level the staff finally adjusted and put into operation the regular system of operational monitoring of the state of the active zone, making use of design algorithms for the calculation of excesses before a crisis and the calculation of heat-engineering reliability. On November 1, 1974 the leading block with the RBMK reactor was brought to the nominal power value of 1 million kW.

All the basic parameters of the reactor and the block as a whole agreed with the design values.

The equipment of the second block of the LAPS was installed two years later. In May 1975 charging of the reactor began and physical start-up took place, and the block was brought to a power level of about 800 MW by October. Figure 2 shows an overall view of the V. I. Lenin APS at Leningrad in operation. Figure 3 shows an overall view of the machine room, with four K-500-65 turbines. The experience obtained in the process of starting up the first block made it possible to reduce to less than half the length of time required for the main stages of the process of bringing the second block up to power. One may expect that the period of start-up and adjustment operations can be shortened even more. For this it will be necessary to formulate typical programs optimized on the basis of the results obtained in the adjustment and startup of the first blocks, as was done, for example, in the program entitled "Physical startup of an APS with reactors of the RBMK type." The amount of start-up and adjustment work must be determined by tests of the installed equipment and complex testing programs, and obviously only those of the investigative operations should be used which, during the work on the previous blocks, yielded results which are unacceptable for one reason or another in the given case.

The first year of operation of the LAPS has confirmed the high efficiency of the reactor and the main equipment of the station. On the basis of the results obtained in the start-up and adjustment operations, necessary changes were made in the design of some assemblies, the technological scheme, and the regimes of operation, corrections are being made in the design materials, and steps designed to improve the characteristics of subsequent blocks using the RBMK reactor are being developed and carried out. During the period of operation, reactor shutdowns were due mainly to the need for removing from the reactor the additional absorbers at the proper time and charging additional fuel into the active zone. After the process

TABLE 1. Basic Characteristics of High-Power Channel Reactors

Characteristic	RBMK-1000	RBMK-1500	RBMKP-2000
Electrical power, MW	1000	1500	2000
Thermal power of the reactor, MW	3200	4800	5400
Efficiency, %	31.3	31.3	37.0
Active-zone dimensions, m:			
height	7	7	6
diameter (or width and length)	11.8	11.8	7.75 × 24
No. of channels:			
vaporizing	1693	1661	1744
steam-superheating	–	–	872
Uranium charge, tons	192	189	226
Enrichment, %	1.8	1.8	1.8/2.2
Av. uranium burnup in discharged channels, MW days/kg			
vaporizing channel	18.1	18.1	20.2
steam-superheating channel	–	–	18.9
Dimensions of fuel-element jackets (diameter × thickness), mm:			
vaporizing channel	13.5 × 0.9	13.5 × 0.9	13.5 × 0.9
steam-superheating channel	–	–	10 × 0.3
Material of fuel-element jackets:			
vaporizing channel	Zirconium alloy	Zirconium alloy	Zirconium alloy
steam-superheating channel	–	–	Stainless steel
Water flow rate through reactor, tons/h	37,500	29,000	39,300
Pressure in separators, atm	70	70	85
Steam capacity of reactor, tons/h	5800	8800	8580
Steam flow rate of turbine, tons/h	5400	8200	7580
Parameters of steam before turbines:			
pressure, atm	65	65	65
temperature, °C	280	280	450

of recharging the channels on the operating reactor by means of a recharging machine (Fig. 4) has been set up, the number of shutdowns will be determined by the graph for conducting planned preventive overhauls.

To sum up, we can list the main problems which were successfully solved in the process of constructing the RBMK channel-type uranium–graphite reactor:

1. Zirconium alloys – the principal structural material for the active zone – were developed and tested under reactor conditions.

2. A tightly sealed connection between the stainless steel and the zirconium alloy was constructed.

3. A design was worked out for a heat-generating assembly operating in a stable manner in a boiling coolant.

4. Sensors for monitoring the energy generation, to be used inside the zone, were developed, and on the basis of these sensors, systems for the monitoring, control, and stabilization of the energy-generation fields were set up.

5. A system and a set of algorithms for the centralized monitoring and operational estimation of the state of the heat-generating assemblies was worked out by means of a computer.

6. Conditions were worked out for the effective removal of heat from the graphite stack to the coolant.

7. The operating regimes of high-powered nuclear reactors in a complex with a 500-MW turbine using saturated steam were tested.

The LAPS is the first of a series of APS with reactors of this type which are being constructed in the USSR. At the present time, work is being completed on the installation of equipment on the first block of the Kursk APS, and intensive preparations are in progress for startup of the reactor. Installation of the reactor on the first block of the Chernobyl APS is in progress, and the construction of the Smolensk APS

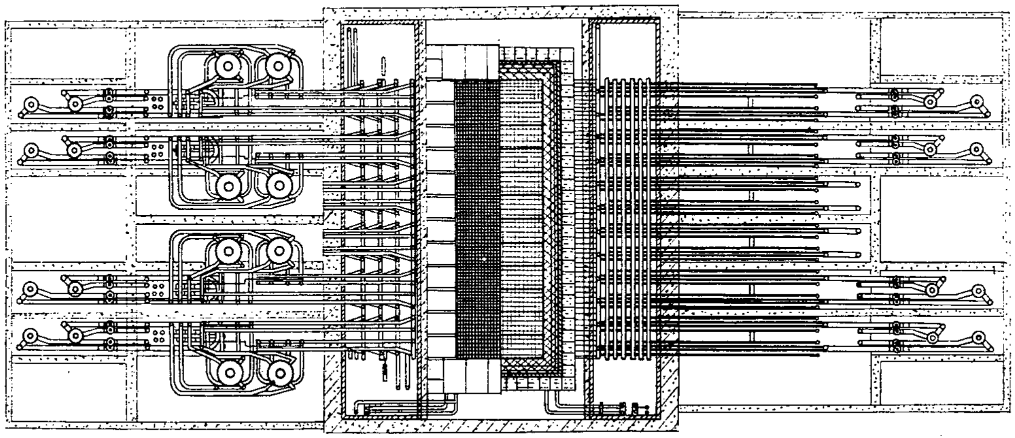

Fig. 6. Arrangement of equipment in the RBMKP-2000 (ground plan).

has begun. Each of these stations will include four blocks with RBMK reactors having an electrical power of 1000 MW each. The decision has been taken to construct a second two-block complex for the LAPS and a number of other APS with RBMK reactors.

Soviet channel-type uranium–graphite reactors of the RBMK type, with a unit power of 1000 MW, represent a step forward in the development of channel reactors. An analysis of the reactor characteristics after the nominal thermal power of 3200 MW is reached showed that there are considerable reserves in the design of the reactor. A number of parameters determining the limiting power of the reactor, such as the temperature of the metal structure and the graphite stacking, actually proved to be somewhat lower than the calculated values. Therefore, the question that naturally arose was whether the power could be increased by making minimal changes in the design of individual assemblies. The idea of such changes was supported by the fact that the main circulation pumps, of the TsVN-7 type, have the necessary reserve in the pressure they develop. The designers of the RBMK reactor enthusiastically tackled the task of investigations involving design, calculation, and experimentation to confirm the technical feasibility of this idea and to determine the allowable limit of the additional power. The most important problem was to increase the critical power of the fuel channel, i.e., the power at which there occurs at the surface of the fuel elements a heat-exchange crisis accompanied by an unacceptable rise in the temperature of the zirconium jacket. This problem was successfully solved by introducing heat-exchange intensifiers into the regular heat-generating assembly. Tests were conducted on a number of variants of intensifier design; the optimal variant was one which used lattice intensifiers with an axial twist in the coolant flow. Such lattices are set up, with a pitch of 80 mm, only on the 3.5-m-long upper heat-generating assembly. Tests on a test stand indicated that the critical power of an RBMK channel with heat-exchange intensifiers is about 1.5 times what it would be without them. After less than one year, in July 1975, the technical design for the RBMK-1500 reactor was brought out; this showed the technical feasibility of increasing the useful power of the RBMK reactor to 1500 MW by intensifying the heat exchange in the fuel channels while keeping unchanged the structure of the reactor as a whole. A number of questions will undoubtedly require further research (e.g., vibration-wear tests of the heat-generating assemblies with intensifiers), but the most important questions were successfully resolved and technical justification for the decisions was established. The technical design for the RBMK-1500 reactor has been approved, and a decision has been taken to construct APS with reactors of this type.

The intensive development of nuclear power and the trend toward increasing the unit power of reactors confronts specialists with the problem of working out a design that will make it possible to construct reactors from unified and standardized assemblies, i.e., without restructuring the machine-construction industry base and complicating the installation of the reactor. The possibilities of uranium–graphite channel reactors have made it possible to find ways of solving the problem. The first step in this direction is the design for the RBMKP-2000 section-block channel reactor, with an electrical power of 2000 MW. The distinguishing feature of the design of the RBMKP-2000 is that its shape is not the traditional cylindrical one but that of a rectangular parallelepiped consisting of separate sections (Figs. 5 and 6). Through the use of sections of uniform type, it becomes possible to set up a reactor of almost arbitrarily large power using identical arrangements, both for the reactor and for the structural elements of the building [5]. Each section includes the necessary equipment and control and monitoring instruments and consists of separate transportable blocks. Special attention should be drawn to the fact that nuclear superheating of steam can be conveniently arranged in a section-block reactor [6]. The vaporizing and steam-superheating sections are uniform in design; the difference lies in the design of the heat-generating assemblies of the vaporizing and the steam-superheating channels and also in the fact that the vaporizing sections include circulation pumps and separators. The sections may be regarded as independent zones of the reactor, in which it is possible to make controlled changes in power within the required limits and there is a certain degree of independence with regard to the cooling-water and steam loops. All of this creates favorable conditions for localizing breakdowns and repairing some equipment without shutting down the reactor.

The RBMKP-2000 reactor consists of eight vaporizing sections, four steam-superheating sections, and two end-face sections. Each section has upper and lower blocks with separation of the pipes, lateral blocks, supports, graphite stacking, and vaporizing or steam-superheating channels. The lateral sections serve essentially as neutron reflectors and are equipped with special cooling channels. In working out the design, maximum use was made of the experience obtained in the construction and operation of the RBMK reactor. The vaporizing and steam-superheating channels of the RBMKP-2000 are practically identical with the channels of the RBMK-1000, i.e., are of tubular construction, with the central portion made of a zirconium alloy. This central portion is connected to the stainless-steel upper and lower portions with similar steel–zirconium adapters. The required temperature conditions of the zirconium tubes of the steam-superheating channels are ensured by slightly superheated steam passing through the annular gap between the shell of the heat-generating assembly and the pressure pipe. The vaporizing and steam-superheating heat-generating assemblies are identical in construction, but the fuel elements of the steam-superheating channels are different from the fuel elements of the vaporizing channels: instead of a zirconium-alloy jacket measuring 13.5×0.9 mm, they use a stainless-steel jacket with an external diameter of 10 mm and a wall thickness of 0.3 mm. Unlike the RBMK reactor, in which the upper and lower supporting and shielding metal structures are filled with a serpentinite charge with a low thermal conductivity, in the RBMKP-2000 reactor the thermostating is achieved by filling these structures with water.

The coolant circulation is achieved as follows. From the deaerator the feed water is fed into the downcomers of the separators, mixed with saturated water, and fed by the circulation pumps into the vaporizing channels of the reactor. From the channels the steam–water mixture enters the separators. The saturated steam is fed into steam-superheating channels, heated to 450°C, and fed at a pressure of 65 atm through steam ducts to two turbines having a power of 1000 MW each. Table 1 shows the main characteristics of the RBMK-1000 and RBMK-1500 high-power uranium–graphite channel reactors with a boiling coolant and of the RBMKP-2000 reactor with nuclear superheating of the steam.

The special features of channel reactors make it possible to modernize them steadily and continuously, and therefore the technical and economic indicators of APS with such reactors will be improved. This is clearly demonstrated by the design for the RBMKP-2000 reactor, which uses the progressive section-block principle of reactor construction, a principle whose possibilities would be difficult to overestimate. The successful solution of a number of problems listed below will make possible further improvements in channel reactors:

1. The development of reactor-materials technology, including the production of high-temperature zirconium alloys which will make it possible to improve the parameters of the vaporizing loop of the reactor and will be usable in steam-superheating channels.

2. Further research on the intensification of heat exchange in channels with a boiling coolant, the development of various designs for intensifiers, and the experimental verification of their operating capacity.

3. The improvement of the design of channels and heat-generating assemblies with superheating of the steam inside the reactor.

4. The development of means of effective heat removal from the graphite stacking in order to reduce the temperature of the graphite and use nitrogen instead of helium for filling the stacking.

5. The improvement of safety measures at reactor installations as a way to increase the number of circulation loops and reduce the diameter of the pipes, including the development of more effective systems for emergency cooling of the active zone and the localization of coolant leaks.

6. The investigation of possibilities of regulating the reactor by using a liquid absorber.

The high reliability of uranium–graphite channel reactors, the relative simplicity of their construction, the possibilities of achieving high safety levels in the event of damage to the pipes of the cooling loop, and the practically unlimited possibilities of increasing power, the possibility of recharging fuel while the reactor is in operation, the flexibility of the fuel cycle, the convenience of introducing nuclear superheating of steam, and a number of other advantages make this Soviet type of reactor one of the most important in the country's large-scale power industry and open favorable prospects for the further improvement and development of these reactors.

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PHYSICAL STARTUP OF THE RBMK-REACTOR* OF THE SECOND UNIT OF THE V. I. LENIN NUCLEAR POWER STATION, LENINGRAD

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In accordance with the program for the development of nuclear power generation in the Soviet Union, in May to June 1975 the physical start-up was achieved at the reactor of the second unit of the V. I. Lenin Nuclear Power Station, Leningrad (LNPS). The physical start-up program for the reactor of the second unit of the LNPS was based on the results of the physical start-up of the first reactor [1] and provided for a number of comparative experiments during charging of the reactor. Charging of the reactor with fuel assemblies FA and with auxiliary absorbers AA was carried out with dry multiple forced circulation loops MFC and cooling of the rods of the control and safety system CSS. Although the charged reactor with dry channels, intended for the insertion of fuel assemblies and auxiliary absorbers, does not have the greatest reactivity, this charging principle allowed the multiple forced circulation loop to be prepared for a power start-up, simultaneously with charging of the fuel assemblies.

For reliable control over the core and for ensuring safety during charging, a temporary control and safety system was used together with the regular control and safety system. It effected control of the neutron flux, the reactivity and emergency shutdown, and it comprised six emergency shutdown rods (scram rods), four manual controls, and also the neutron source actuator with a control switch and a position indicator.

The physical start-up program consisted of the following main stages:

1. Composition of the minimum critical charge without auxiliary absorbers and standard control and safety rods (charge No. 1).

2. Completion of zone up to the maximum number of identical polycells, the so-called periodicity cells (loading No. 2, Fig. 1).

3. Additional charging of the reactor up to 1437 fuel assemblies and 239 auxiliary absorbers.

4. Shaping of the initial charge of the core, taking account of the operating experience from the reactor of the first unit.

5. Estimation of the reserve of reactivity of the initial charge, and ensuring the required duration of operation before the first fuel recharging.

6. Determination of the reactivity effects with dry multiple forced circulation loops and with cooling of the control and safety rods.

*Water-cooled/water-moderated channel-type reactor (high-powered).

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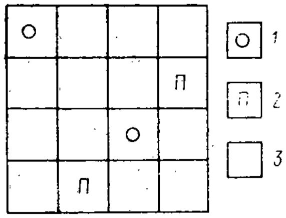

Fig. 1. Periodicity cell: 1, 2, 3) cells with control and safety rods; auxiliary absorbers; and fuel assemblies, respectively.

7. Measurement of the energy-release fields in the cold unpoisoned reactor.

8. Bringing of the reactor to the minimum level of power, controlled by the standard control and safety system.

The necessity for experiments (in comparison with the reactor of the first unit) originates by the difference in the number of technological parameters affecting the physics of the reactor. In particular, to these parameters may be referred the density of the graphite (1.67 g/cm^3 in comparison with 1.73 g/cm^3 in the first unit), the average charge with respect to ^{235}U in the fuel assemblies and the difference in the boron content in the auxiliary absorbers, etc. Comparative experiments during loading, even at the initial stage of the physical startup, permitted those changes to be forceast which must be carried out in the total reactor charge, in order to ensure the required reserve of reactivity and distribution of the energy-release field.

Preparation of the Reactor for Start-up

Before starting to charge the fuel assemblies and auxiliary absorbers, the following operations were carried out:

The multiple forced circulation loops and the control and safety rods were flushed and pressurized.

Running-in of all main circulatory pumps (MCP) and the pumps of the control and safety rod loop.

During operation of all the main circulatory pumps, the multiple forced circulation loops and the graphite brickwork of the reactor were heated up to 150°C over two days, and after heating up the graphite brickwork was cooled to room temperature.

The monitoring system for the integrity of the technological channel (MITC) was put into operation.

The regular and temporary control and safety rods were put into operation.

The through-channel water-flooding system was prepared.

The loudspeaker connection between the central hall and the modular control panel (MCP) was made operational.

The system for filling the multiple forced-circulation loop with water from the emergency feed pump (EFP) tank was flushed and prepared for operation.

The drainage reservoirs were prepared for receiving water.

The general exchange and special ventilation systems were brought into operation.

The entire assembly of auxiliary absorbers was installed, with a ratio of inserts of boron steel and stainless steel of 3 : 1 in the central section, of length 500 cm and 1 : 2 at the end sections with a length of up to 100 cm.

A complete set of fuel assemblies with openings below the interzone sensors and 100 fuel assemblies were installed.

Charging of the Reactor. Comparative Experiments

During charging of the reactor, its control and safety systems were implemented, just as in the reactor of the first unit, with instruments of the temporary control and safety rods in the presence of a neutron source in the zone.

The first critical charge without auxiliary absorbers and the rods of the regular control and safety system in the absence of water in the multiple forced-circulation and the control and safety loops, contained 24 fuel assemblies (Fig. 2) and with the temporary control and safety rods withdrawn K_{eff} was 1.00096 (23 fuel assemblies and K_{eff} = 1.0050 for the reactor of the first unit). Further charging of the reactor was carried out with respect to the periodicity cells (12 fuel assemblies, two auxiliary absorbers, and two control and safety rods). After charging 77 periodicity cells (916 fuel assemblies and 154 auxiliary absorbers), charge No. 2 was brought to the critical state. Further, the critical state was recorded for charges containing 1437 fuel assemblies and 239 auxiliary absorbers, and 1452 fuel assemblies plus 239 auxiliary

TABLE 1. Some Results of Comparison Experiments during Physical Start-Up of the Reactors of the First and Second Units of the Leningrad Nuclear Power Station

charge No.	Reactor condition: FA quantity pieces	FA presence of water in channels	AA quantity, pieces	AA presence of water in channels	presence of water in the control and safety loop in both units	No. of control and safety rods inserted in core: OR	MR	AR	SRA	K_{eff}	Difference in Keff with identical compensation, %
1	23; 24 †	No.	—	No	No	—	—	—	—	1,0050; 1,00096	—1,0
2	916	»	154	»	»	8; —	56; 56	4; —	—	1,00000; 1,00064	+1,1
3	1437	Yes	239	Yes	»	13; 8	89; 89	12; 12	20; 20	1,00034; 1,00016	—0,33
4	1452	»	239	»	Yes	10; 1	89; 85	9; 12	20; 20	1,00000; 1,0032	—0,5

*OR) overcompensation rods; MR) manual control rods; AR) automatic control rods; SRA) shortened rod-absorbers.
†Here, and in future, the first and second figures are for the first and second units, respectively.

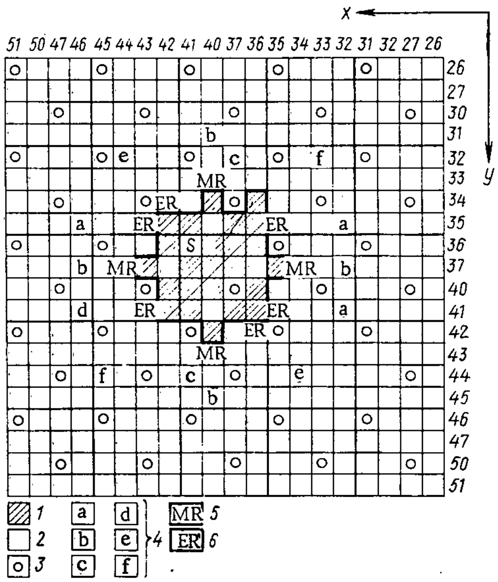

Fig. 2. No. 1 charge and diagram of the disposition of the sensors and the temporary control and safety rods: 1, 2) cells with charged fuel assemblies and uncharged channels; 3) cells with regular control and safety rods; 4) cells with sensors: a) galvanometers (G1, G2, and G3), b) reactimeters (PIR-1 and PIR-2), c) counter-trigger devices (SPU-1 and SPU-2), d) power pen recorder (ÉPPV), e) scram rod boosters (UA-9-1 and UA-9-2), f) scram-rod velocity instruments (UZS-1 and UZS-2); 5, 6) cells under temporary control and safety rods; S) neutron sources; MR) manual control rods; ER) emergency shutdown rods.

absorbers. Similar charges were brought to the critical state in the reactor of the first unit. For the charges containing 1437 fuel assemblies and 239 auxiliary absorbers, the effect of reactivity on the filling with water of the multiple forced-circulation loop was measured by means of a reactimeter; as in the reactor of the first unit, this was found to be $+1.9\beta$.* The results of the comparative experiments obtained during the physical startup of the reactors of the first and second units are shown in Table 1. The difference in the effective multiplication factor K_{eff} was determined in the following way. An identical sequence for withdrawing the control and safety rods was adopted for both reactors on reaching the critical state. By measuring the efficiency of the control and safety rods, amounting to the difference in the compensation position, the difference in K_{eff} was determined.

It follows from Table 1 that all the charges investigated for the reactor of the second unit have a lower reactivity. The difference in K_{eff} varies from 0.33 to 1.1%.

Effect of Various Parameters on the Reactivity

The core structure of the reactor of the second unit, in accordance with the physical start-up program, was specified by the identical core of the first unit. The differences which appeared during the physical start-up of the second reactor necessitated calculations to be carried out in order to assess the effect of deviations of the various parameters on the multiplication properties. The results of the calculations are shown in Table 2.

Analysis of the deviations from the nominal values of the mass characteristics of the fuel in the fuel assemblies, the graphite purity, and the content of boron in the auxiliary absorbers, showed that all these factors can be eliminated from those significantly affecting the multiplication properties. On the contrary, a change of graphite density affects not only the multiplication properties but also the diffusion length L and the lifetime τ of the neutrons, which define both the neutron leakage from the reactor and

*Effective fraction of delayed neutrons.

TABLE 2. Effect of Deviations of Various Parameters on the Multiplication Properties of the RBMK Reactor Core

Parameter	Nominal value of parameter	Devia. from nominal value assumed in calculation	$\Delta K_\infty/K_\infty$, %*
Uranium enrichment	1.787%	+0.01%	+0.18
Fuel density	9.30 g/cm^3	+0.1 g/cm^3	+1.32
Graphite density	1.67 g/cm^3	+0.1 g/cm^3	−0.31 (−7.1 for $\Delta M^2/M^2$)†
Absorption cross section of graphite	4.2 mbar	+0.1 mbar	−0.13
Boron content of auxiliary absorbers	2.0%	+0.1%	−0.02

*Values of $\Delta K_\infty/K_\infty$ are given for the reactor, with water in the multiple forced-circulation loop.
†$M^2 = L^2 + \tau$.

also their overflow to the control and safety rods and the auxiliary absorber rods.

Calculations by the QUAM-2 program showed that a reduction of the graphite density led to the following losses of reactivity for the charges being compared (see Table 1):

Charge No. 1 − 0.96% (− 1.0%);
Charge No. 2 − 0.91% (− 1.1%);
Charge No. 3 − 0.33% (− 0.33%);
Charge No. 4 − 0.31% (− 0.50%).

The experimental data are shown in the brackets. Thus, the calculations confirm that a reduction of reactivity in the reactor of the second unit is mainly due to the reduction of the graphite density. However, according to the calculations, this does not lead to a noticeable change of the depth of burnup in view of the increased plutonium production.

Formation of the Initial Reactor Charge

The reduction of reactivity which appears in the reactor of the second unit is compensated mainly by substituting 9 auxiliary absorbers by fuel assemblies. Moreover, the interchange of several peripheral auxiliary absorbers was effected, which gave rise to certain difficulties in the case of rechargings during operation of the reactor of the first unit. In contrast from the first unit, auxiliary absorbers were installed on the periphery in the lattice of the control and safety rods which, in this region are disposed approximately twice as sparsely as at the center of the core. It was decided not to load eight channels on the periphery of the reactor as, according to calculations, the installation of fuel assemblies in them leads to an increase of nonuniformity of the power release.

As a result of rearrangements and transfers, the initial charge for the reactor of the second unit was defined: 1455 fuel assemblies, 230 auxiliary absorbers, and 8 uncharged channels. The critical state of the initial charge (the multiple forced-circulation loop and the control and safety loop filled with water) was achieved by the insertion in the reactor of 89 manual control rods, 12 automatic control rods, 21 shortened rod-absorbers, and 10 overcompensation rods. 21 emergency shutdown rods and 26 overcompensation rods were withdrawn. With this situation of the control and safety rods, $K_{eff} = 1.00077$, the temperature of all core elements was ~20°C and the reactor power was ~1.4 kW.

Experiments on the Initial Reactor Charge

One of the problems of the physical start-up is to determine the basic physical characteristics of the reactor, necessary for its future operation. For this purpose, in the initial charging of the reactor the effects of reactivity were measured with dry cooling loops of the control and safety rods, fuel assembly channels (estimation of the "steam" effect of reactivity in the cold state), the multiple forced-circulation loop, and with installation of the interzone monitoring sensors.

At the same time, the total efficiences of the inserted control and safety rods (estimation of the reserve of reactivity of the cold unpoisoned reactor) and of the withdrawn control and safety rods were determined.

All negative reactivity effects were measured with a reactimeter during the introduction of reactivity into the critical reactor. The efficiency of the inserted control and safety rods was measured by their successive withdrawal from the critical reactor. If the efficiency of a single rod exceeded 0.3β, then the

TABLE 3. Experimental and Calculated Data for the Initial Reactor Charge

Quantity determined	Experiment	Calculation
Effect of reactivity with:		
dry channels with fuel assemblies	−0.42%	−0.43%
dry channels with auxiliary absorbers	−1.60%	−1.19%
dry multiple forced-circulation loop	−2.02%	−1.62%
dry control and safety loop	Compensated by the insertion of 13 control and safety rods	–
installation of 117 sensors for monitoring the radial neutron field	−0.006%	
Total efficiency:		
of inserted control and safety rods	8.9%	7.3%
of withdrawn control and safety rods	1.9%	1.6%
Nonuniformity factor:		
of radial neutron field	2.04*	1.94*
of neutron height field	1.37	2.45%

*Obtained on fuel assemblies in which measurements were carried out by fission chambers.
†For all fuel assemblies of the reactor.

measurements were carried out by the overcompensation method. Dehydration of the control and safety loop was carried out in the subcritical state.

The relative power release field in the initial reactor charge was measured with small-sized fission chambers. At the same time, five independent measurement channels, in the corresponding way to the commuted channels, participated in the measurements. The measurements were made at eight points with respect to height in 144 fuel assemblies, having at the center dry channels for the fission chambers. The quality of the relative measurements, carried out twice at several points, has a mean-square error of 1.6%. The absolute thermal neutron flux Φ_T was determined by the activation of gold foils in and without cadmium. The absolute power of the fuel assemblies, in which the absolute thermal neutron flux was measured, was determined from the relation

$$W_T = \frac{\Phi_T \sqrt{T_0/T}\sqrt{\pi/4}\,\sigma_{0f}^5 N^5 k_r k_1}{3.1\cdot 10^{10} k_T},$$

where T is the neutron temperature at the point of location of the indicator; T_0 = 293°K; σ_{0f}^5 is the fission cross section of ^{235}U when $T = T_0$; N^5 is the number of ^{235}U nuclei in the fuel assemblies; k_r is a factor which takes into account fission by resonance neutrons; k_1 is the deviation of the neutron flux measured by the fission chambers and averaged over the height, from the neutron flux at the site of irradiation of the indicator; and k_T is the ratio of the neutron flux at the point of measurement to the average neutron flux in the fuel.

The reactor power was determined from the formula

$$W_p = \frac{W_T}{Q_T^e}\,\frac{\sum_{i=1}^{n} Q_i^e}{\sum_{i=1}^{n} Q_i^p}\sum_{i=1}^{m} Q_i^p,$$

where Q_T^e is the relative power of a fuel assembly, measured by the fission chamber, the absolute power of which was determined by gold activation; $\sum_{i=1}^{n} Q_i^e$ is the summed relative power of the fuel assemblies, measured by the fission chambers; $\sum_{i=1}^{n} Q_i^p$ and $\sum_{i=1}^{m} Q_i^p$ are the total relative powers, calculated for the fuel assemblies n in which the fission chamber measurements were made, and all fuel assemblies m respectively.

All critical charges, and also the measured power release fields, were computed by the BOKR-COB and QUAM-2 programs, which describe channel-wise the structure of the core. Moreover, the experimental efficiency of the control and safety rods was computed by the BOKR-COB program.

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The BOKR-COB program is a development of a program [2, 3] based on the solution of the diffusion equations of a reactor by a finite-difference method in x–y geometry (for the cross section of the reactor). In the program, the two-group diffusion equations of a reactor consisting of heterogeneous square cells are solved. The nodes of the reference mesh coincide with the centers of the channels. It was shown by the calculations of the experiments carried out on critical assemblies, and also on the reactor of the first unit of the Leningrad Nuclear Power Station, that such an arrangement of the reference nodes is more preferable than in the angles of elementary cells. The nonuniform poisoning of the fuel by xenon, as a function of the designed distribution of the power-release field and the fuel burnup, are taken into account in the program. The presence in the core of the control rods and other breeder channels is taken into account by assigning the appropriate homogenized properties of the cells in which these absorbers and channels are located. Partially inserted rods are replaced by completely inserted rods of equivalent efficiency. For the operational calculations, a modification of the program is used – the BOKR-COBZ program in which, in order to take account of the partially inserted control and safety rods, experimental measurements are used of the height neutron field by the sensors of the physical control system.

The QUAM-2 program achieves a new method of calculating heterogeneous reactors [4]. The reactor is represented in the form of a finite lattice of channels (in x–y geometry) in an infinite moderator. The transport of neutrons in the moderator is described by two-group diffusion equations of the Galanin–Feinberg type [5, 6], which are transformed to the so-called quasialbedo form similar to the finite difference form, and which are solved by an iterration method. As a result, computer time in solving the equations is shortened by a factor of 15-20 in comparison with the traditional heterogeneous method and amounts to ~1.5 min for the calculation of a single reactor state. The QUAM-2 program enables K_{eff} to be calculated and also the power distribution over the reactor channels with a specified position of the completely or partially inserted control and safety rods. The possibility is provided for taking into account the steady-state poisoning by xenon in the uranium burnup, individually for each channel. In calculating K_{eff}, a correction is made for the axial leakage of neutrons and the nonuniformity of properties over the height of the reactor, and also a correction which takes account of the processes caused by moderated neutrons and due to the presence of nonbreeding channels.

By means of the QUAM-2 program and a system of supporting programs, calculations of about 70 critical states (cold and hot poisonings and with uranium burnup) have been carried out for the reactors of the first and second units of the Leningrad Nuclear Power Station. The mean-square error in determining K_{eff} amounts to 0.5% and the maximum deviation does not exceed 1%.

Comparison of the calculations by the BOKR-COB and BOKR-COBZ programs with the experimental data, shows that the calculations predict satisfactorily the criticality of the various states of the reactor. For the system of neutron-physical constants assumed, the discrepancy in K_{eff} does not exceed 0.9%. For a complete charge, it does not exceed 0.5% and, taking into account the height field of the neutrons (the BOKR-COBZ program), it amounts in all to 0.2%. It is shown that the height distribution of the neutrons has a marked effect on the calculated value of K_{eff}. In the calculations with sinusoidal and measured neutron distributions, the difference in K_{eff} amounted to 0.3%. Therefore, for a more accurate calculation of K_{eff} by the BOKR-COBZ program, it is necessary to take account of the actual height distribution of the neutrons. The efficiencies of different groups of control and safety rods, calculated for different charges, mainly coincide well with the experimental data. The results of the experiments and calculations of certain effects of reactivity and neutron distributions over the core, carried out for the initial reactor charge, are shown in Table 3.

Comparison of the experimental and calculated power-release fields along the radius of the reactor, obtained by the BOKR-COBZ and QUAM-2 programs, showed agreement at the location of the field maximum; the mean-square error in determining the power of the fuel assemblies by both programs is identical and amounts to 9.7%.

After completion of the experiments on the initial charge, the power start-up of the second unit of the Leningrad Nuclear Power Station was effected in July-August 1975. The power of the unit was built up gradually in accordance with the readiness of the turbogenerators. Initially, at a total power of 500 MW, the third turbogenerator was cut-in and then the fourth turbogenerator was brought into operation. On September 30, 1975 the State Commission authorized the handover of the second unit of the V. I. Lenin Nuclear Power Station, Leningrad to commercial operation. On October 10, 1975 in accordance with the start-up program, the electric power output of the second unit amounted to 750 MW.

During the power start-up, the basic decisions taken according to the results of the physical start-up were checked and confirmed. In particular, according to the results of measurements of the power-release fields, the rate of decrease of reactivity as a result of poisoning and burnup of the uranium, the validity of the specification for the initial charge, and the creation of the necessary reserve of reactivity were confirmed.

CONCLUSIONS

During the physical start-up of the reactor of the second unit of the Leningrad Nuclear Power Station, the initial reactor charge was specified, experiments and calculations were carried out which permitted a comparison to be made with the results obtained during the start-up of the reactor of the first unit:

1. The initial reactor charge consists of 1455 fuel assemblies and 230 auxiliary absorbers (eight channels remain uncharged).

2. There are certain differences between the reactors of the first and second units, in that similar charges in the reactor of the second unit have a lower reactivity. The decrease of reactivity of the total reactor charge (charge No. 4) amounted to 0.5%. This difference is explained mainly by the lower density of the graphite in the reactor of the second unit.

3. Dehydration of the 1455 channels with fuel assemblies and the 230 channels with auxiliary absorbers reduces the reactivity by 2%.

4. The effect of reactivity during dehydration of the control and safety loop is positive and is compensated by the insertion of 13 shutdown rods.

5. The reactor has the greatest reactivity when the circulation loop is filled with water and the control and safety loop is dry.

6. Dehydration of the channels with 1455 fuel assemblies leads to a reduction of the reactivity by 0.42%, which exceeds by 0.3% the similar effect in the reactor of the first unit and confirms the more negative steam effect of reactivity on the initial stage of operation of the reactor.

7. The total efficiency of the inserted control and safety rods amounts to 8.9%.

8. In order to bring the cold, unpoisoned reactor with water in the multiple forced-circulation and the control and safety loops to the critical state, 47 shutdown rods must be withdrawn, the total efficiency of which amounts to 1.9%.

9. A comparison of the experimental and calculated power-release fields along the radius of the reactor, obtained by the BOKR-COBZ and QUAM-2 programs, showed that the mean-square deviation between the measured and the calculated powers of the fuel assemblies by both programs is identical and amounts to 9.7%.

The physical startup of the reactors of the first and second units of the Leningrad Nuclear Power Station permitted considerable experimental data to be accumulated concerning the startup of RBMK-type reactors. By taking into account the possibility of variation of certain technological parameters of the core materials, it will be advantageous during startup of the next units to carry out comparative experiments in order to correct the initial charge and to refine the operating characteristics of the reactor.

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HIGH-TEMPERATURE REACTORS AS A FACTOR IN SCIENTIFIC PROGRESS IN POWER GENERATION*

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The successes in power-generating reactor construction have led to the creation of an extremely large branch of power production – nuclear-power generation, which has a complex but essential assembly of scientific-technical means for ensuring its development.

At the present time, there remains a very limited number of nuclear reactor types in the whole world for future utilization and modernization. Reactors with water cooling in three structural modifications occupy the dominant position (Table 1); the vessel type (pressurized water and boiling water) and the channel types which, in particular, have undergone long development in the Soviet Union. Over the last 10-15 years, there has taken place in nuclear-power generation, if it can be so expressed, a natural selection of reactor types according to the extent of their technicoeconomic capabilities and conformity to problems of large-scale electric power production. The limited number of reactor types and modifications as a basis of standardization and unification is an essential condition for the observed intensive growth of nuclear-power generating capacities. However, over recent years and especially in view of the energy generation crisis, a marked interest has been shown toward high-temperature nuclear reactors.

With the well-known convention, the factors stimulating the development of high-temperature nuclear power generation can be divided into two groups.

1. Factors arising from the internal evolution of the development of nuclear electric-power generation and affecting its economy.

2. External factors, either arising from the evolution of the interrelation between nuclear-power generation and the habitable environment, or dictating the ever-increasing necessity for extracting organic fuel from energy-containing thermal processes and the necessity for conversion of the energy carriers.

Among the first group, the efficiency of utilization of nuclear fuel both in thermal neutron reactors and also because of the problem of fast breeder reactors is of the most decisive importance; the conversion factor for the heat from the core into useful power, the specific capital investment in nuclear power stations, and also problems of safety. First and foremost, it should be emphasized that high-temperature nuclear-power generation at the present-day level of scientific knowledge is related with reactors in which the coolant is an inert gas – helium. However, in conformity with the mainly nonelectric power industrial problems, other methods also are possible for removing heat from the core, based on other ideas in contrast from convective heat exchange, inherent with a gaseous coolant.

The distinguishing features of a high-temperature reactor with helium coolant are not only a higher temperature level at the outlet from the core than in all the well-known types of reactors, but also a considerably better neutron balance in the core and a smaller requirement for structural materials, which ensures a greater efficiency of utilization of nuclear fuel per unit of electric power sent out (Table 2).

These properties of a high-temperature reactor are due to an extraordinarily favorable combination of materials or media in the core – ceramic fuel, graphite as the moderator, and helium as the coolant (unactivated gas, having an almost zero neutron absorption cross section). Owing to this, high-efficiency

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TABLE 1. Trend of Development of Reactor Construction in Certain Countries

Country	Type of reactor, accepted for nuclear power stations under construction	Type of reactor, considered as long-term	Type of reactor, having been developed to the stage of commercial experiment
USA	Pressure vessel: with water under pressure, dual circuit: with boiling water, single circuit	High-temperature reactor with helium coolant (HTGR): for the production of electric power (dual circuit); for the production of high-potential heat and electric power, single and dual circuit; for the production of electric power and central heating, single circuit, with helium turbine. Fast neutron breeder reactors: with sodium coolant, triple circuit; with helium coolant, dual circuit	Reactor with organic coolant and moderator.* Pressure vessel reactor with nuclear super-heating.† Sodium-graphite reactor‡
Canada	Channel, heavy-water: with heavy-water under pressure, dual-circuit, CANDU-PHWR. Boiling, with heavy-water moderator and light water as coolant, single-circuit, CANDU-BLW	Channel, heavy-water, dual-circuit	Channel, with organic coolant (development discontinued in 1973)
France	Pressure vessel: with water under pressure, dual circuit: with boiling water, single circuit	Pressure vessel, with water under pressure, dual-circuit. Fast neutron breeder reactor with sodium coolant	Gas-graphite, with CO_2 as coolant. Heavy-water with CO_2 as coolant
Great Britain	Advanced gas-graphite (AGR)	Channel, with heavy-water moderator and light boiling water as coolant (SGHWR), single circuit. High-temperature (HTGR) and with very high temperature of helium at the outlet from the reactor (VHTR)	Gas-graphite, with CO_2 as coolant.
Italy	Pressure vessel: with water under pressure, dual circuits with boiling water, single circuit	Channel, with heavy-water moderator and light boiling water as coolant (CIRENE), single circuit	Gas-graphite with CO_2 as coolant. Reactor with organic coolant and moderator (ORGEL, PRO, ROVI)
Federal German Republic	Pressure vessel: with water under pressure, dual circuits with boiling water, single circuit	High-temperature reactor with thorium fiel cycle and helium coolant	

*OMRE, PIQUA.
†BONUS, PATHFINDER.
‡HALLAM.

utilization of the neutrons in the core and the significantly higher (than in the well-known reactors with water cooling) thermal cycle efficiency (41-43% against 31-33%) is ensured.

The relatively low pressure of the gas coolant (40-60 atm) and its other properties as a medium, makes it the most advantageous and feasible to use a reinforced concrete vessel, in which all the main plant of the primary loop are grouped (integral grouping). The feasibility of making the vessel on the floor of the nuclear power station eliminates serious weight and size problems of transportation and gets rid of the necessity for designing and developing mechanical engineering facilites and workshops for the manufacture of such vessels. The construction of reinforced-concrete vessels is technically feasible at a unit electric capacity of 1.5 to 2.0 million kW.

These, and other circumstances, finally will undoubtedly favor a reduction of the specific capital investment in nuclear-power stations with high-temperature gas reactors and will imply also a reduction of the cost of useful electric power sent out. Particularly attractive in this respect is the direct gas-turbine cycle, which can be achieved even with helium temperatures of 750-850°C at the outlet from the reactor.

Attention should be paid to the extremely important question, which is of an independent nature and which is associated with the future structure of the nuclear power-generation system, i.e., with the optimum ratio between nuclear-power stations with thermal and fast reactors. The optimum dynamic structure of a nuclear power-generation system containing fast reactors is determined to a considerable degree by their efficiency margin, expressing the difference between the inherent and contractural costs in the

TABLE 2. Requirement on Natural Uranium for the Production of Electric Power, with Different Types of Thermal Reactors using Low-enrichment Uranium

Type of reactor	Consumption of natural uranium (for different modifications), g/MW · d
Pressure vessel with water coolant	25-34
Channel with graphite moderator and water coolant	20-25
High-temperature with graphite moderator and helium coolant	14-18

system. In its turn, this margin depends on the ratios both of the fuel component and the magnitude of the specific capital investment of nuclear-power stations with fast and thermal reactors.

The magnitude of the specific capital investment in nuclear-power stations with thermal reactors has been determined adequately. At the same time, for fast reactors this quantity has a considerable indeterminacy, mainly in consequence of the experimental-industrial destination of the isolated fast breeder reactors being constructed. Because of this, in a number of papers on the system analysis of nuclear-power generation, the effect was investigated of the possible excess of the specific capital investments of nuclear-power stations with fast reactors, in comparison with thermal reactors, on the efficiency of fast reactors.

There is still no significant experience of the construction of large-scale nuclear-power stations with fast sodium reactors. However, the experience in planning and building the first nuclear-power stations with these reactors is extremely alarming in relation to capital investments. For example, the specific cost of a fast sodium reactor with a capacity of 350 MW (el.) at Clinch River (USA), marked for completion of construction in 1982, has proved to be very high. There are grounds for fearing that the excess of the specific cost of reactors with liquid-metal coolant over the specific cost of thermal reactors will be extreme, and this will bring fast reactors beyond the limit of permissible economy (according to estimates, it is probably no higher than 50-60%) [1].

The main reason for the relatively high cost of nuclear-power stations with breeder reactors with sodium coolant lies in the complex triple-circuit heat-removal system and conversion of the heat into electric power (sodium – sodium – water), in the listed multiplicity of plant, and in the complexity and importance of the systems for filling, heating up, and monitoring of the liquid-metal loops. Therefore, as alternatives fast reactors with a gaseous coolant are being considered in a number of countries. Design and planned operations in both the USSR and abroad show that these reactors have approximately equal or somewhat better indexes with respect to breeding and nuclear fuel utilization, the fuel cycle and heat conversion efficiency than reactors with sodium coolant. At the same time, the absence of negative qualities which are characteristic of a liquid-metal coolant ensure a much simpler and, consequently, a cheaper heat conversion scheme. Fast reactors with a gas coolant, therefore, give the basis for expectation for obtaining an equal or somewhat larger value of the specific capital investments than for thermal reactors. The construction of nuclear-power stations with fast gas-cooled reactors may be considered as an economically promising problem and a desirable alternative for solving the general problem of fast reactors.

From the scientific and engineering aspect of the problem of building fast and thermal gas-cooled reactors, there are a number of general features, such as material problems, gas-circuit plant and the construction of its elements and units, gas economy schemes and auxiliary services (safety, decontamination, dosimetry, etc.), grouping, and other general plant problems. Therefore, work and results of investigations on thermal and fast reactors with helium coolant are of a mutually beneficial nature.

As concerns problems of safety, these problems as applied to thermal gas-cooled reactors, in view of the qualities which are inherent to these reactors, are solved no more intricately than for other types of reactors:

1. Integral grouping of the primary circuit plant in a reinforced-concrete vessel and a significantly lower coolant pressure (40-60 atm) than in pressurized water-cooled reactors almost eliminates rupture of the primary circuit.

2. Even in the low-probability case, rupture of the primary circuit is not accompanied by a heavy formation of steam from superheated water, which is inevitable for pressurized water reactors.
3. Problems of corrosion of the primary circuit and the deposition in it of radioactive deposits are solved more easily.
4. In view of the very small contribution of the gas coolant to the reactivity of the reactor and the high storage capability of graphite, the loss of coolant is not linked with an abrupt and significant jump in reactivity of the reactor.
5. Melting of the core or its elements is eliminated, as it consists entirely of ceramic materials with a very high melting or volatilization point.

These, as an example, are the basic factors ensuring from the intrinsic evolution of the development of electric power reactor construction, stimulating the development of work on the construction of high-temperature nuclear reactors.

Let us consider the outside factors. Certain of them were shaped only recently, and the cause and effect of their existence is still far from being explained and understood. Therefore, at the present level of our knowledge, only the principal trends leading to their onset can be noted.

Among this group of factors, the question of the connection between a power installation and the surrounding environment recently has occupied an important place. This connection appears more strongly in terms of the efficiency of conversion of the primary heat released in the installation, into useful power. An increase of the conversion efficiency, all conditions being equal, is related to factors which raise the efficiency strictly of nuclear-power-generating plants and improve the efficiency of the nuclear fuel utilization. Finally, since recently, attention to problems of ecology associated with nuclear-power production, in particular with the discharge of heat into the habitable environment, has increased. Therefore, the question of the efficiency of a power-generating plant is now becoming so much broader. It concerns not only the economics of power-generating plants but also problems of preservation of the essential qualities of our habitable environment. High-temperature reactors possess a significantly higher efficiency (41-43%) and conform better to this problem. If we add, that by a further increase of the gas temperature at the outlet from the reactor up to 1000-1100°C, which gas-cooled reactors will allow, there is a possibility in principle of building a competitive modification of a nuclear-power station with a triple thermodynamic heat conversion cycle and with an efficiency of 55-60%. In one of these proposals [2], potassium, diphenyl, and steam of different parameters are used as the working media. Table 3 shows the important possibilities of electric-power generation which could be achieved by the use of high-temperature reactors.

In connection with what has been said above, the question may arise: If it is necessary to develop high-temperature electric power-generating reactors, what should be the attitude toward water-cooled reactors which, as is well-known, at the present time occupy the predominant position in nuclear-power generation? This question can be answered thus. Modern nuclear-power generation as a branch of electric-power production has completed successfully the circle of its possibilities lying, if it can be so expressed, as yet only on the surface of the reactor power-generation phenomenon. With a certain paradox it can be considered even that a nuclear reactor is, in principle, a new heat-generating agent, operating on a steam turbine cycle, the parameters of which will lag very much behind the parameters achieved in thermal power generation until right up to the appearance of the power-generating reactor. In the thermal-power generation sense, the parameters of this cycle to a certain extent are even a step backwards. Nevertheless, electric-power-generation problems, standing before nuclear-power generation, are so vast and the demand on them is so great that reactors with water cooling, obviously, for quite a long time still will successfully satisfy the problems of the electric-power generating industry.

However, the evolution and logic of the development of power-generating reactor construction, and also the possibilities of nuclear-power generation by this time require a gradual transition to a new, higher level with more improved specific branches and, chiefly, with systemic indexes. High-temperature reactors reveal this possibility. Moreover, in recent years important power generation problems have started to be shaped, the solution of which lies beyond the limits of possibility of water-cooled reactors. The sole reason is the low temperature level obtained by means of these reactors.

We shall pass on to a short account of these problems.

TABLE 3. Possible Characteristics of Nuclear-Power Stations with a Triple Production Cycle

Cycle	Limits of cycle, °C/bar		Efficiency, %
	upper	lower	
Potassium	890/3,0	477/0,027	29,1
Diphenyl	455/20,9	287/2,0	16,9
Water (functioning in the condensation cycle)	270/55 (Intermediate superheating 270/8.9)	33/0,051	33,6

The fact that, as a rule, they are found to be outside the sphere of electric power production is general for them. The necessity for their solution is connected mainly with the limitation of organic fuel reserves and the everywhere increasing deficiency of its most important forms – petroleum and gas. Related with these problems are the extraction by nuclear fuel from energy-rich high-temperature processes, of valuable forms of organic fuel (for example, in ferrous and nonferrous metallurgy, and chemistry) and the conversion of energy carriers, effected at high temperatures (gasification of low-grade coals, production of hydrogen, etc.).

The first group of problems concerns branches of the national economy, which over a long time have become more complicated. There is a certain advantage in this, because the special features of power requirements in these fields can be specified with greater certainty, and the demands on the nuclear reactor and its parameters can be defined. But there is also a difficulty there, as the power-technological experience, having been involved in the field, sometimes leads to the onset of branch conservatism which mainly is manifested in a tendency to consider the nuclear reactor as a source of heat which adequately replaces the heat produced by burning organic fuel. However, the specific characteristics of a nuclear reactor for producing the necessary economic effect, in many cases requires a reorganization of the technology of the process, a departure from the power-technological concepts and, as a consequence, changes of the customary constructive solutions. A nuclear reactor in these industrial environments is represented as a factor which affects the trend of technical progress.

The second group of problems is of a systemic multified nature, as it concerns the preparation of a product for multipurpose utilization. This is a new group of all-round complicated problems, in which the question of the construction of a nuclear reactor occurs in a number of other, no less complicated, problems. Here, there is still the greatest indeterminacy of all.

High-temperature reactors might be used in ferrous metallurgy, where there is an extremely large demand for oganic fuel. For example, in the Soviet Union up to 20% of the volume of organic fuel required for the entire industry of the country is expended on ferrous metallurgy requirements. One further reason for the marked interest in the use of the heat from nuclear reactors in this field is the poisoning of the environment during the carrying out of classical metallurgical processes.

At the present time, ferrous metallurgy is based on the use of coking, with the production of the metal, for example by the "blast furnace–converter–rolling mill" system. Together with the future modernization of blast furnace production, it is proposed to introduce cokeless metallurgy processes, with the final product being produced by the process "direct reduction of iron ore–electromelted steel product – rolled product." Cokeless metallurgy processes at the present time are at the initial stage of utilization. Obviously, cokeless metallurgy is very promising and it can be economically competitive with the blast furnace process for the production of metals. Among the several cokeless metallurgy processes are the well-known processes of solid-phase reduction of nodulized iron ore concentrate by converted natural gas, with the production of pig iron, which is carried out at temperatures of 950-1250°C.

Investigations carried out in the Soviet Union and abroad have shown that in this case, the use of high-temperature reactors may prove to be promising. If the natural gas which is expended on heating the reducing gases in this section of the metallurgical cycle up to 950-1250°C is replaced by the heat from a nuclear reactor, then up to 50-55% of the natural gas will be saved. As Soviet and foreign investigations have shown, an economic effect can be expected to be produced. As contractual costs of organic fuel have a tendency toward increasing, this effect in presepective can be significant.

The economical promise and the economical need for the introduction of high-temperature nuclear reactors in ferrous metal production processes, however, is accompanied by technical difficulties. These are due, first of all, to the high level of temperatures required and, secondly, to the considerable difference in pressure between the nuclear-reactor coolant (40-50 atm) and the reducing gases in the direct reduction plant units – the shaft furnaces (6-8 atm). Therefore, in order to reliably prevent radiation contamination of the metallurgical product (pig iron), it will be necessary to ensure directivity of the pressure gradient from the nuclear-reactor coolant to the reducing gas. By using a high-temperature reactor with helium coolant, this can be achieved merely by the construction of a shielded intermediate circuit with a higher pressure than in the reactor circuit.

TABLE 4. Chemical Cycles for the Production of Hydrogen

Name of cycle	Chemical cycle	Reaction temp., °C
Mark-1	$CaBr_2 + 2H_2O \rightarrow Ca(OH)_2 + 2HBr$	750
	$2HBr + Hg \rightarrow HgBr_2 + H_2$	200
	$HgBr_2 + Ca(OH)_2 \rightarrow BaBr_2 + HgO + H_2O$	200
	$HgO \rightarrow Hg + \frac{1}{2} O_2$	600
Mark-9	$6FeCl_2 + 8H_2O \rightarrow 2Fe_3O_4 + 12HCl + 2H_2$	650—750
	$2Fe_3O_4 + 3Cl_2 + 12HCl \rightarrow 6H_2O + 6Fe_3Cl_3 + O_2$	
	$6FeCl_3 \rightarrow 6FeCl_2 + 3Cl_3$	
Hat	$K_2O_2 + H_2O \rightarrow 2KOH + \frac{1}{2} O_2$	Exothermic at 150°C
	$2KOH + 2K \rightarrow 2K_2O + H_2$	725
	$2K_2O \rightarrow K_2O_2 + 2K$	1000

In connection with this, the complicated problem arises concerning the steam pipes and heat-exchangers for the transport and transfer of heat in the range 1000-1300°C. There are still no acceptable solutions or suggestions for this. In general, the complex nature of the nuclear metallurgical problem should be emphasized. It will be necessary not only to find acceptable solutions to the amalgamation of a high-temperature reactor with the conventional metallurgical production plants, but also to find technical solutions for both the high-temperature reactor and the metallurgical production plant units.

Another promising trend for the utilization of the energy of high-temperature nuclear reactors is their use in the chemical industry for the production of ammonia and methanol. These products are the basic raw materials for the production of nitrogenous fertilizers, the demands for which will increase strongly in the near future. Modern industrial technology for the production of ammonia and methanol is based on the large-scale requirement for natural gas, used as a technological raw material and energy-producing fuel. In this case, the fraction of natural gas consumed as an energy-producing fuel amounts to ~45% of the total requirement; large volumes of the combustion products of natural gas are discharged into the surrounding medium; in particular, not more than 15-20% of the heat released in the combustion of natural gas is used in the technological process and it is used at a temperature of 1000°C and higher; the utilization of the heat of the high-temperature, low-pressure waste gases (1 atm) presents a serious technical problem and a large quantity of low-temperature heat at a temperature of 300°C and below is discharged into the surrounding medium.

Thus, the effective increase of production of ammonia and methanol in the volumes necessary to satisfy the increasing demands of the national economy in these products and their reprocessed products encounters considerable difficulties, which are aggravated by the fact that the limitations on the use of natural gas as an energy-producing fuel are being intensified, and the demands are being increased for reducing the discharge of toxic substances and heat into the surrounding medium.

The high-temperature heat from nuclear reactors can be used mainly in the heat-consuming section of the overall technological cycle for the production of ammonia – during the carrying out of the endothermic reaction of the steam–water or carbon dioxide conversion of methane. By the existing technology, these reactions take place at a temperature of 800-900°C in the presence of a catalyst. At the same time, the construction of nuclear reactors, experience in planning, and also prospective scientific ideas and technical development, will allow confidence in the capability of building high-temperature reactor systems which, in the level of working temperatures and also in technical solutions, will satisfy the conditions for their efficient utilization as sources of power for high-temperature technological processes of the chemical industry. As a result of this, the requirement for the chemical production of natural gas will be cut down by up to 45%; the discharge into the surrounding medium of the combustion products of natural gas will be eliminated; the thermal efficiency of the entire technological cycle will be increased; and the production cost of ammonia and methane can be reduced markedly.

To the group of more general power-generation problems, which can be solved by means of high-temperature nuclear reactors, are related the gasification of lignite and coal and other forms of low-grade organic fuel. Work in this direction has been progressing quite broadly during a number of years in the Federal German Republic, the USA, and Great Britain. Additional impetus toward the development of the task was given in view of the energy crisis in the western countries in 1973-1974 and by the proposed future increase of prices of petroleum and natural gas. Several processes are being considered for the

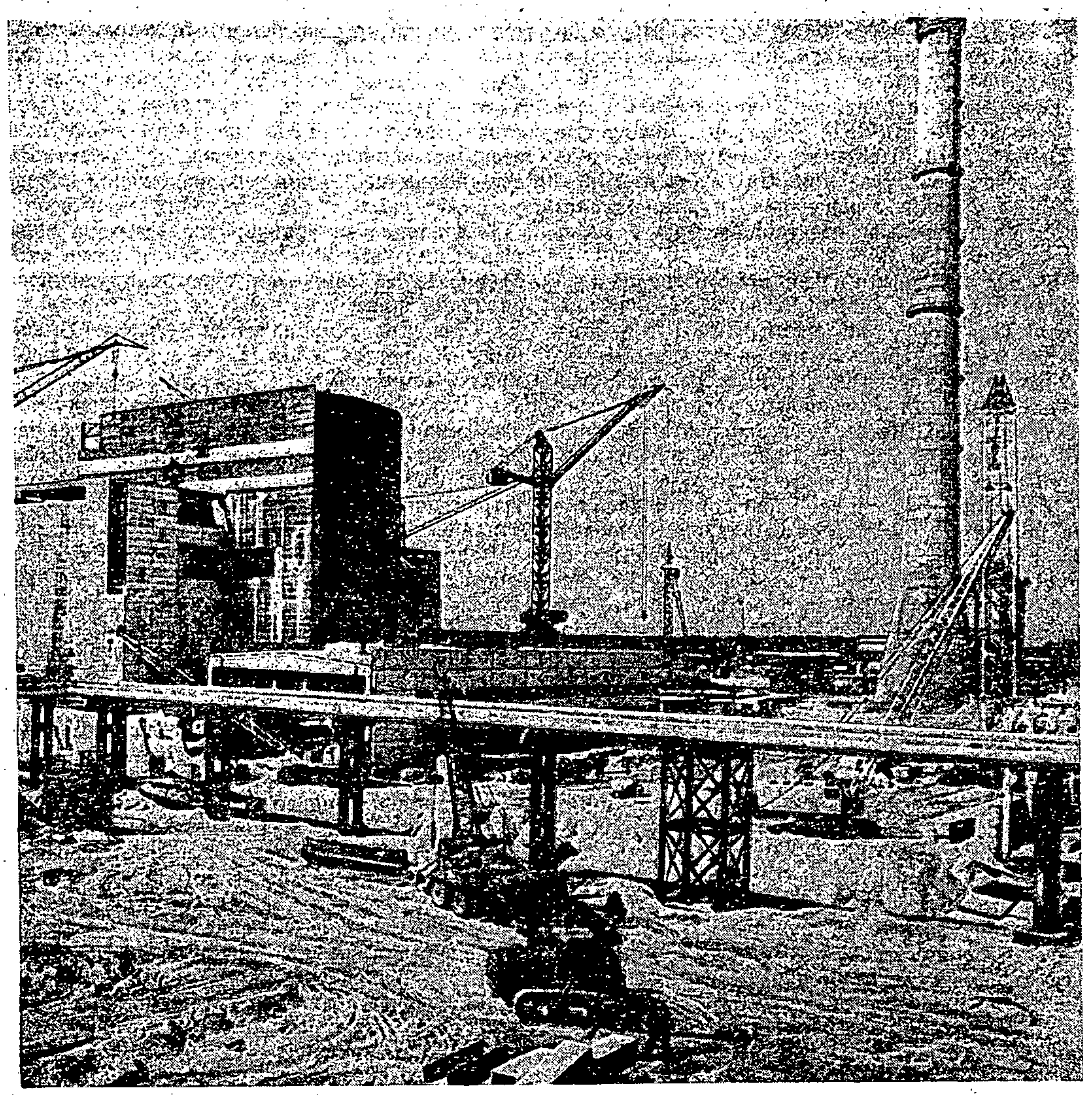

Fig. 1. In nuclear buildings of the USSR. Chernobyl'sk Nuclear-Power Station. General view of construction. Photo by V. Bratchikova.

gasification of coal and lignite, using the heat from nuclear reactors. However, preference is being given to two methods: hydrogenization and steam—water gasification [3, 4]. In comparison with conventional gasification processes, which use as heat the energy from the combustion of organic fuel, it is expected that the use of nuclear heat will give the following advantages:

1. The gas generated will be cheaper.

2. Coal resources will be utilized more efficiently, which is equivalent to a relative increase of these resources.

3. Costly gas can be produced from deposits at a low cost of extraction, which is particularly important when using cheap lignites.

4. As nuclear fuel is a unique form of fuel which is classified according to the relative cost of extraction, the increase of cost of raw material (coal) will have less effect on the cost of the gas produced by comparison with conventional gasification [3].

5. Contamination of the surrounding medium will be reduced significantly, e.g., the discharge of carbon dioxide by 60% and sulfur by 40%.

Fig. 2. In nuclear buildings of the USSR. Chernobyl'sk Nuclear-Power Station. Construction of the reactor hall. Photo by V. Bratchikova.

Recently, attention has been drawn to yet another problem of the conversion of energy-carriers: to obtain hydrogen, which is being designated often as the fuel of the future, by means of the heat from nuclear reactors. It is well-known that the heat from a nuclear reactor can be utilized for the production of hydrogen by means of the electric power produced in a nuclear-power station, which then is used for the conventional electrolysis of water. However, in this scheme the additional stage of conversion of the energy-carrier leads to a low efficiency of the whole conversion cycle. Therefore, the thermochemical decomposition of water is considered energywise to be more efficient, in which the heat from a nuclear reactor is used directly for the production of hydrogen. The power generation efficiency of this conversion amounts to 50%.

TABLE 5. Energy-Generating and Energy-Technological Courses for the Application of High-Temperature Reactors

Course of application	Parameters of process	
	necessary temp. level, °C	pressure level, atm
Improvement of nuclear-fuel utilization, increase of efficiency of nuclear-power station up to 41-43%:		
steam-turbine cycle	550-560	240
gas-turbine cycle up to 60-61%	850-900	40-60
triple cycle	890-900	Up to 55
Production of ferrous metals	850-1250	5-6
Production of ammonia and methanol	750-900	Up to 30
Gasification of coal:		
hydrogenation	700-800	25-50
steam-water	800-950	25-50
Production of hydrogen by thermochemical decomposition of water	730-1000	Up to 10
Transport of latent heat in a cycle with reverse chemical reactions	850-900	Up to 10

Several cycles have been proposed [5] for the thermochemical decomposition of water. Common for all is the maintenance of the quantitative balance of the intermediate chemical elements participating in the reaction cycle (Table 4). The technological schemes for these cycles are quite complex, and the chemical compounds participating in the reaction and the elements are corrosion-active and even toxic. Therefore, their practical achievement is not a simple problem. The storage, transportation,and distribution of the final product – hydrogen – also are complicated.

The technological scheme for the transference of heat in a closed cycle with reversible chemical reactions is interesting [6]. For one of them (steam–water conversion of methane), heat supply is necessary but in the other (by the exothermic reaction of the products produced) heat can be liberated. This peculiarity of the process can be used for the transportation of latent energy to considerable distances [7].

A high-temperature nuclear reactor supplies a chemical converter with heat, in which an endothermic reaction takes place with the formation of H_2 and CO_2. Then these gases separately are transported to the heat consumer. The gases enter a chemical reactor in which the chemically combined energy, in consequence of the exothermic reaction process between H_2 and CO_2, is liberated and is fed to the consumer. No loss of energy during transportation of these gases in the cold state occurs (with the exception, obviously, of the expenditure of energy in transporting the gas).

In all energy-conversion processes, it is essential to ensure a pressure gradient directed from the primary reactor circuit to the chemical production circuit under irradiation safety conditions, which requires the construction of an intermediate coolant circuit between the reactor circuit (if a gas-cooled reactor is used) and the technological plant. The problem of using high-temperature reactors for the production of hydrogen is also a complex problem, falling outside the bounds of solution of the problem merely of the generation of heat by a high-temperature nuclear reactor. Despite all the intricacy and the strict complexity of these problems, their solution at present is urgent.

Courses for the possible application of high-temperature reactors, the temperature levels and pressures of the processes of these courses which, at the level of our knowledge may be assumed to be formulated, are generalized in Table 5.

Thus, in terms of the future requirements of the national economy, in the protection of the environment, and also taking account of the everywhere increasing intensity of the fuel-energy generation balance, it can be assumed that the attraction for high-temperature nuclear reactors for solving on an economic basis the problems facing the national economy will be justified in the near future. To what extent will the status and prospective parameters of high-temperature reactor construction answer these problems?

The first project for a high-temperature reactor was accompanied in England by the startup of the Dragon reactor at Winfrith in 1964. In April 1966, its thermal power was raised to the design power of 20 MW. A large series of tests on various types of fuel have been carried out in this experimental reactor.

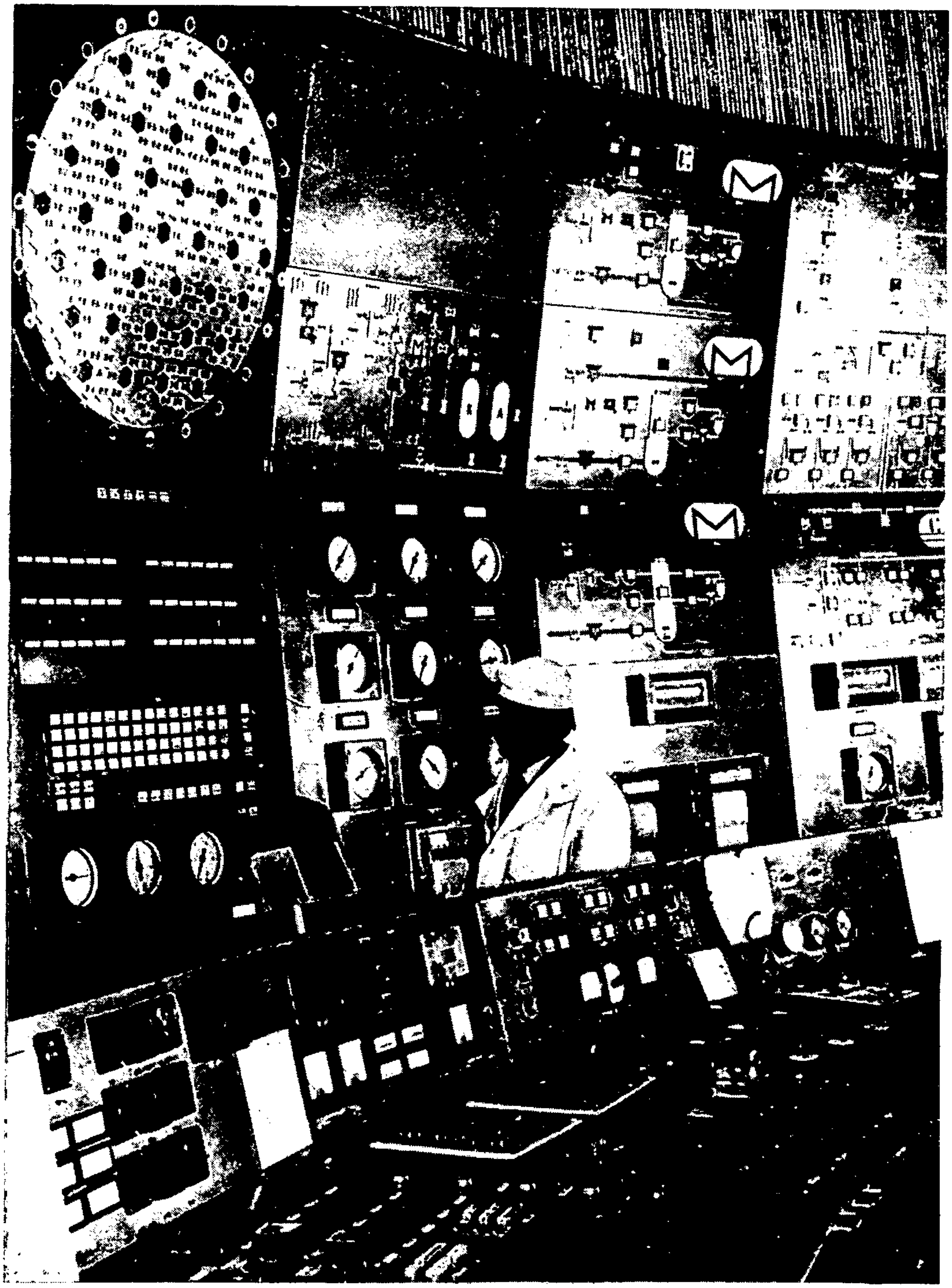

Fig. 3. In nuclear buildings of the USSR. Kol'sk Nuclear-Power Station. Unitized control board. Photo by V. Bratchikova.

Fuel elements were studied under irradiation with direct measurement of the discharge of gaseous fission products and discharged fuel assemblies. At present, according to the statements of the supervisors of the Dragon reactor, there are prospects for obtaining a coolant temperature at the outlet from the active zone of up to 1250°C and, in light of promising experiments with a fuel of uranium nitride particles, it is permissible to consider the possibilities of obtaining even higher temperatures (up to 1450°C). In the USA, the experimental Peach Bottom nuclear-power station with a high-temperature gas-cooled reactor (HTGR) was brought to its operating power at 42 MW in 1967. In the Federal German Republic (Julich), in 1967, an experimental nuclear-power station with a HTGR of electrical capacity 15.6 MW was also constructed. The next stage in the development of HTGR reactors was the construction in the USA of the Fort St. Vrain nuclear-power station. In January 1974 charging of the reactor with fuel was completed and the firm "Public Service of Colorado" obtained a license for its commercial operation. The electrical capacity of the nuclear-power station is 342 MW and the efficiency is 39.2%. At the present time, work is taking place to eliminate troubles which have developed, in particular helium leakage.

It is planned to construct a large-scale nuclear-power station with a HTGR in the Federal German Republic (Wentropp) with a capacity of 300 MW (el). The helium temperature at the outlet from this reactor will be 750-800°C. So much for the prospects and possibilities of further increasing the temperature in HTGH-type reactors.

Recently, reports have appeared concerning the cancellation of contracts on the construction of large-scale nuclear-power stations with HTGR reactors, due to the increase of prices of plant and materials, and the necessity for revising plans in the course of increasing the unit capacities of the reactors, which will put back by several years their periods of construction. These reasons are of a situational and not a scientific-technical nature.

At present, all efforts directed at the production of a high temperature are based on the utilization of fuel elements with microparticles of fuel (uranium dioxide or dicarbide) and with a multilayered coating of pyrolytic graphite and silica. The particle sizes, dispersed in the graphite matrix, are 200-2000 μ. These fuel elements are characterized by a high degree of retention of fission products (up to 1300-1400°C). The production technology is being developed and improved.

Fuel nuclei of uranium dicarbide do not react chemically with the pyrolytic graphite cladding and remain structurally stable up to 3000°C. However, carbides have a relatively high diffusion coefficient through graphite. The upper limit of the working temperature for extended operation of microfuel elements with a nucleus of uranium dicarbide coated with pyrolytic graphite is approximately 1700°C. The use of uranium dioxide as the nucleus of a microfuel element allows contamination of the cladding of pyrographite to be reduced, as the diffusion coefficient of uranium from the dioxide in graphite is less than from the dicarbide in graphite. Complex fuel compositions based on nitride compounds are being considered, which will permit the working temperature of microfuel to be raised to 1700-1800°C. The possibility of designing a ventilzated fuel element is also being studied, for the purpose of equalizing the internal pressure in the fuel element with the pressure of the coolant, in order to reduce thereby the stress in the fuel element cladding.

The idea is noteworthy for removing heat from the reactor core and transferring it to a technological requirement by means of radiative heat exchange with a coarse element solid coolant – graphitic elements – the circulation of which is affected with a servomechanism. Radiative heat exchange and the application of a kinematic servomechanism imparts a new quality to the reactor, which permits confidence in its efficient utilization for the provision of heat for high-temperature technological processes:

1. The pressure inside the reactor vessel can be maintained both above and below atmospheric without reducing the power intensity of the core.
2. The solid coolant (graphite) can be returned to the core at any temperature level.
3. The heat transfer surface between the solid coolant and the technological raw material does not experience mechanical stresses and can be made of heat-resistant but not heat-proof materials, including nonmetallic materials.

Because of these qualities, the operating conditions of thermotechnical plant and the shielding conditions of the technological raw material from radiation contamination are significantly eased, and also the transfer of heat from the core to the technological process is ensured over a narrow range of high temperatures. Moreover, the feasibility emerges in principle for building on the basis of a reactor of this type nuclear assemblies incorporating in a single vessel a nuclear source of high-temperature heat and a technological consumer. According to preliminary estimates, the achievement of these possibilities will permit the production of a tangible economic effect to be calculated.

The study and working out of other ideas is also possible, for example, making the helium coolant dusty with finely dispersed graphite in order to reduce the pressure in the reactor circuit and improving the heat transfer at the same time with convective and radiant heat exchange.

From all that has been said above, the conclusion can be drawn that the feasibility of constructing a high-temperature economical nuclear reactor is an important factor in the future development of technical progress, not only in the field of electric-power generation but also in a number of national economical fields, where fuel plays the influencing role. Moreover, extension of the spheres of application of the heat from the fission of heavy nuclei, undoubtedly, will contribute to the solution of a number of important power-economical problems.

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METHODS OF MATHEMATICAL MODELING AND OPTIMIZATION OF NUCLEAR POWER PLANTS

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The exceptional complexity of the technological structure of the contemporary national power industry and the interconnections between individual power units predestined its division into a number of systems. The unified national power system standing at the upper hierarchical level is divided into five basic specialized systems which together guarantee the national economy all forms of energy and fuel: an electric power system, oil, gas, and coal supply systems, and a nuclear power system presently being formed. These systems are divided into appropriate systems of individual regions of the country. The ultimate elements of large power systems are businesses – power producers and consumers [1]. In turn each power producer is a complex system which includes a large number of units of various types of power plants joined by physical, technical, and transport ties. It is expedient to represent each such complex system as a series of hierarchically coordinated systems.

Four hierarchical levels are generally distinguished in thermal power systems: thermal power plants, power units, groups of plant elements, and plant elements.

The final elements of the hierarchical structure of a thermal power system – the plant elements – must be detailed further for the study of individual phenomena, processes, and structures. These investigations are performed at lower hierarchical levels, i.e., at the level of physicotechnical systems of parts of plant elements when the problems of the development of thermal power become problems of mechanics, thermal physics, physical metallurgy, and other related disciplines [2, 3]. The importance and complexity of the problem of the optimum design and long-term development of thermal power systems of various types are obvious. Nuclear power plants are no exception. Figure 1 shows the system of informational interconnections of a nuclear power plant arranged according to the hierarchical principle.

The external input data result partly from optimizing the power and economic systems at a higher level; the nuclear power plant is one element. The external input data are obtained partly from predictions and expert estimates. The internal input data include a descritpion of the laws and characteristics of the flow of technological processes, physical properties of the working media and coolants, and characteristics of different kinds of plant structures. Descriptions are given of the constraints imposed on the parameters and characteristics of the structures. The input data also include lists of design layout types for power plant equipment and variants of the form of its process flow diagram (or the conditions of their formation).

The information obtained by solving the optimization problem includes integral and discrete characteristics of the form of the process flow diagram of the power plant, continuously variable thermodynamic and flow rate parameters of the energy carriers, discrete characteristics of the types of structures of units and basic plant elements, and continuously or discretely varying structural parameters of plant units and elements. In addition to its direct use to establish optimum parameters and characteristics of a nuclear power plant this information also appears as internal and external feedback data. The internal feedback data determine the direction of further development and improvement of investigations leading to a lower step on the hierarchical ladder, i.e., to the level of physicotechnical models of individual phenomena, processes, and structures. The external feedback data link includes optimum technical and economic indices of the plant for various conditions of its use, technological, weight, and structural characteristics of units and elements of the equipment of an optimized plant, characteristics of surges, industrial wastes, heat release, and other factors which describe the effect of the power plant on external systems and the environment.

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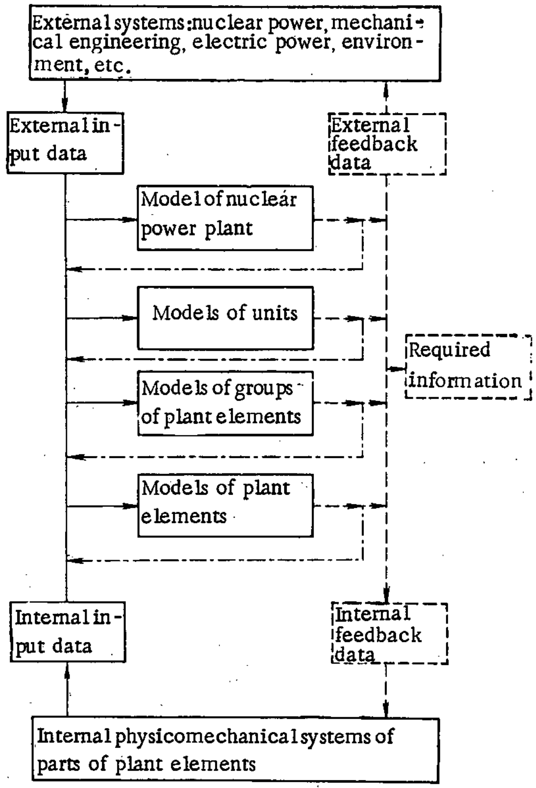

Fig. 1. System of mathematical models and exchange of information for the optimization of a nuclear power plant and its equipment; —; ---; -·-·) input, required, and intermediate data respectively.

In addition to the indicated streams of external and internal data there are direct and feedback streams of so-called intermediate data circulating within the system of mathematical models considered. These data streams contain information on optimum values of parameters obtained by solving optimization problems at various hierarchical levels within the power plant.

This hierarchical structure of intermediate data makes it possible to organize a two-step optimization cycle – downward and upward. In this case there is taken into account the decreasing influence of more detailed information on the refinement of overall optimum solutions, and hence the possibility of decreasing the volume of data exchanged.

The intermediate and sought for data can also be used to investigate the content of the models and the structure of the interconnections between them, i.e., to construct an optimum system of mathematical models. In this case supplementary cycles of interconnections arise between the system of models and the sources of internal input data.

At the present time a qualitative solution of the problem of optimizing the values of the parameters, the form of the arrangement, and the design of an atomic power plant are impossible without the extensive use of mathematical modeling and a computer, leading to a solution whose suitability depends on the extent to which the time of completion and the capital outlay are taken into account.

Studies of the problem under consideration have been directed toward developing a theory and methods of a comprehensive thermodynamic, technical, and economic analysis and complex optimization of a nuclear power plant based on mathematical modeling methods and the solution of complex experimental problems and the use of a computer.

So far the following [2-5] have been developed: theoretical bases for the construction of mathematical models of various types of nuclear power plants for complex calculational studies; methodical bases for the use of nonlinear mathematical programming and computers for solving the power plant optimization problem; practical ways of applying the methods of mathematical modeling, nonlinear mathematical programming, and computers to determine ways of increasing the economy of various types of atomic power plants by the choice of optimum neutron, physical, thermodynamic, cost, and structural parameters, and also an efficient form of the process flow diagram.

The mathematical formulation of the problem of the complex technical and economic calculation of a nuclear power plant of a given form can be written as follows:

$$\mathbf{Z}=\mathbf{Z}[\mathbf{X}_g, \mathbf{Y}_g(\mathbf{X}_g), \mathbf{G}]_{\mathbf{E}_0} \tag{1}$$

for

$$\Phi_g[\mathbf{X}_g\ \mathbf{Y}_g(\mathbf{X}_g)]_{\mathbf{E}_0}=0; \tag{2}$$

$$\mathbf{Y}_g^{*}\leqslant\mathbf{Y}_g[\mathbf{X}_g, \mathbf{Y}_g(\mathbf{X}_g)]_{\mathbf{E}_0}\leqslant\mathbf{Y}_g^{**}; \tag{3}$$

$$\mathbf{F}^{*}\leqslant\mathbf{F}[\mathbf{X}_g, \mathbf{Y}_g(\mathbf{X}_g)]_{\mathbf{E}_0}\leqslant\mathbf{F}_g^{**}; \tag{4}$$

$$\mathbf{X}_g^{*}\leqslant\mathbf{X}_g\leqslant\mathbf{X}_g^{**}, \tag{5}$$

where Z is the target function, $\mathbf{X}_g$ and Y_g are sets of independent and dependent plant parameters; G is a set of parameters depending on the form of the plant design; $\mathbf{E}_0$ is a set of characteristics of given external factors; Φ_g is a set of balance equations for all plant units; $\mathbf{F}_g$ is a set of technological characteristics of the plant units describing the restrictive conditions; one and two asterisks denote respectively minimum and maximum admissible values.

As follows from Eqs. (1)-(5) it is required not only to determine the values of the dependent parameters Y_g and to calculate the value of the target function Z, but also to choose values of the independent parameters X_g which satisfy the set of technical constraints (2)-(5). If the assumed initial values of the set of independent parameters X_g do not satisfy the indicated technical constraints they must be changed to the admissible range.

Appropriate mathematical models of a nuclear power plant and of individual plant elements provide a method for analyzing and searching for the most valid design solutions. In particular by using these models we can:

1. Investigate the character of the interconnections of the plant parameters and analyze their effect on the neutron, physical, thermodynamic, weight, cost, and other plant indices.
2. Investigate the effect of external conditions of plant utilization on the relation of its parameters and on the neutron, physical, thermodynamic, technical, and economic indices.
3. Estimate the numerical values of the subsidiary metal contents, the lowering of the efficiency, the changes in design costs, and other plant indices in case of failure to satisfy the optimum values of the parameters because of technical constraints.
4. Achieve a compatible optimization of all plant parameters.

Among the new possibilities discovered by mathematical modeling is finding the process flow diagram of the plant and calculating the elements of the plant equipment in a single iteration. This achieves a refinement of the heat balances, the thermal and strength calculations of the plant elements, calculations of hydraulic and aerodynamic losses in plant elements and along the coolant channels, values of the efficiency of the basic and auxiliary units, the flow rates of coolants and working media, and thermodynamic parameters of the unit. Such a mutual refinement was not accomplished earlier or was performed very approximately in view of the complexity and laboriousness of the multiple repetitions of calculations of the process flow diagram and plant elements which naturally affected the accuracy of determining the thermal economy of the plant and the technical and economic indices of the installation.

The greatest improvement of nuclear power plant indices is achieved by the compatible optimization of the values of all the basic parameters of the process flow diagram and of the plant elements, i.e., in the complex optimization of the installation.

The goal of the complex optimization of a nuclear power plant is to choose the neutron and physical parameters of the reactor, the parameters of the thermodynamic cycle and thermal design, and the structural arrangement characteristics which correspond to the minimum calculated expenditure for the plant. The method of complex optimization of parameters of a nuclear power plant consists in making compatible changes in the initial set of values of the parameters so as to decrease the value of the efficiency criterion to a minimum with a small number of intermediate variants.

The mathematical problem of the complex optimization of parameters and of the design of a nuclear power plant can be formulated as follows: It is required to minimize a target function (1) with constraints of the form (2), inequalities of the form (3), (4), and constraints on the independent parameters (5). The target function (1) and the constraints (2)-(4) are nonlinear and the parameters to be optimized may be either continuous or discrete. Thus we have a complex problem of nonlinear continuous-discrete programming.

Complex optimization not only determines the best values of the thermodynamic parameters, but also optimizes the distribution of temperature gradients, thermal heads, flow rates, velocities, and pressure drops over the elements of the installation and the plant connections. All this provides a new qualitative effect – the achievement of optimum proportions in the distribution of capital investment between individual elements of the plant equipment and the establishment of the optimum relation between the consumption of nuclear fuel and the capital investment in the plant. In other words an optimization of the internal structure of the nuclear power plant is achieved.

Using the mathematical models developed for a nuclear power plant and programs employing the methods of nonlinear programming, studies were made of the choice of optimum parameters for atomic power plants of various types, their process flow diagrams, and elements of the installation.

The results of theoretical and practical investigations in the form of worked out principles and procedures for constructing mathematical models and methods of optimization in the form of completed algorithms and programs and recommendations for choosing the parameters and design of a nuclear power plant have begun to be used by branches of scientific-research institutes, design institutes, and the construction departments of power machinery construction plants.

Mathematical models of the thermal power part of an atomic power plant permitted the investigation of a number of systems: a single-circuit graphite moderated boiling water reactor, a two-circuit pressurized water reactor, a system with the intermediate separation of moisture and intermediate superheating of steam, and a water-cooled graphite moderated reactor with nuclear superheating. These models used the method of synthesis of optimum process flow diagrams [6]. Using these diagrams the number and interrelation of the plant elements were found as functions of the composition and the values of the optimized parameters.

The mathematical model of the water-cooled graphite moderated reactor with saturated steam, programmed in ALGOL-60, included individual blocks for calculating the thermophysical, thermotechnical, hydraulic, and neutron-physical characteristics of the reactor and the economic indices of the reactor part of the atomic power plant. For a reactor with nuclear superheating the mathematical model was supplemented by similar blocks for calculating the steam superheater part.

Accumulated experience in solving the problem of optimizing atomic power plants of various types shows that it is expedient to use the method of decomposition to construct a mathematical model of an atomic power plant as a complex of models of individual parts related by overall external conditions and internal parameters. For acceptable model dimensions this approach permits a sufficiently accurate account of the external relations and the internal features of each part of the atomic power plant. Thus, in particular, a complex optimization was performed of the parameters and the layout of single-circuit atomic power plants using water-cooled reactors with saturated and superheated steam.

A procedure and a system of mathematical models have been developed for a low-potential atomic power plant complex including the low-pressure turbine, the condensing apparatus, and water-supply and water-cooling systems of various types. This system of models was used to determine ranges of the efficient use of turbines with various exhaust cross sections, and optimum values of plant and structure parameters of a low-potential atomic power plant complex for various water-supply systems in various distribution regions.

A large amount of research has been performed on fast reactors cooled by a liquid metal or a dissociating gas. Techniques and programs have been developed for calculating the physical properties of coolants, including a dissociating gas, taking account of the kinetics of the chemical reactions; effective mathematical models of the reactor and thermal power parts of an atomic power plant have been created; calculations have been performed to determine efficient process flow diagrams and optimum parameters of atomic power plants of the types indicated.

Because mathematical models offer the possibility of a rapid, accurate, repeated solution of a problem using various premises, the rapid introduction of changes in the calculational techniques, etc., they have been widely used in the design and development of nuclear power plants. At the present time mathematical models realized on a computer offer the most effective method for seeking optimum power plant designs and parameters.

However, the potential possibilities of the methods of mathematical modeling and complex optimization of nuclear power plants are as yet little used. Most solved problems are individual isolated problems meeting the requirements of various scientific-research institutes, construction bureaus, and design institutes. Their role in the overall process of design and development of nuclear power plants is relatively small.

Qualitative changes in the design and development of an atomic power plant can be achieved only by a broad application of mathematical models and a computer. But the problem now consists in the transition from development and the use of individual mathematical models to the creation and use of interconnected systems of mathematical models describing all levels of the technological and time hierarchy of the system of optimum design of atomic power plants of various types. For each type of power plant such a system must be realized in the form of a single complex of algorithms and programs taking account of the part played in the design and development process by scientific-research institutes, design offices, and design institutes of various departments. In other words a complex approach to the design and development of atomic power plants is really necessary but is not presently being employed.

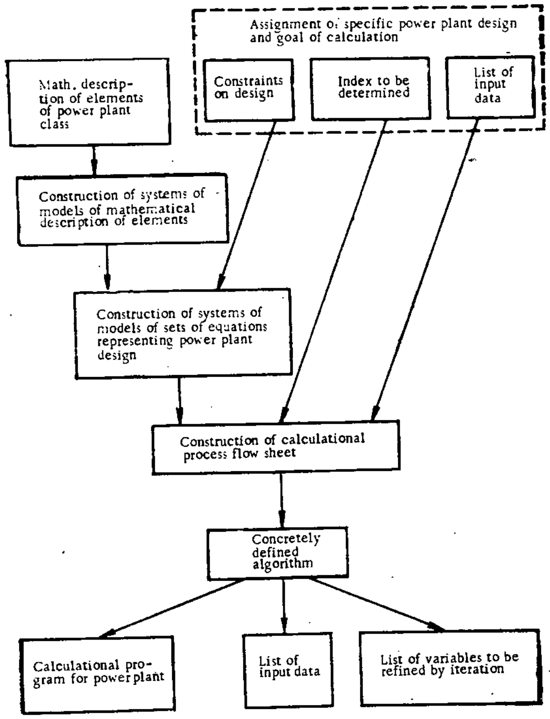

Fig. 2. Flow sheet for computer construction of programs for power plant calculations.

The creation of such a system leads to the best design solution with a significant shortening of design and development times for atomic power plants and an increase in the quality of the design. We note that the designer (builder) is freed from performing ordinary routine work and can concentrate his attention on an analysis of the solutions obtained, on the development of more accurate methods of calculation, and on a search for new structural-arrangement solutions. The rapid performing of design operations makes it possible to check and to incorporate the results of new scientific research into atomic power plant designs.

The creation of a system of mathematical models for optimum design of atomic power plants of various types requires solving a number of scientific, technical, and organizational problems. We consider only certain scientific and methodical aspects of this complex problem.

Among the most important problems which must be solved in developing such a system must be included investigations of the overall structure of the system of mathematical models of atomic power plants of one type or another, and of the principles of the interaction of individual parts of it realized in various scientific-research and structural design organizations.

In developing a system of mathematical models for one type of nuclear power plant or another considerable attention should be paid to the formulation of the requirements for the accuracy of their construction and the flexibility of the algorithms, and to ensuring that each mathematical model or a modification of it can be used to solve problems at various hierarchical levels and in any combinations with other models.

The accuracy requirements of mathematical modeling of any object or process are determined mainly by the goal set and the volumes and error of the input and required information; they are significantly different in considering the object or process at various levels of the technological and time hierarchy.

These requirements can be satisfied by using a number of methodical procedures for constructing mathematical models: 1) methods of separating factors which are important and unimportant for the object or process under investigation (this determined the optimum size of the mathematical model); 2) equivalence methods, i.e., the transformation of one mathematical model to a simpler model approximating the original; 3) decomposition methods, i.e., breaking up the problem into a series of subproblems, studying each subproblem independently, and then coordinating them.

As applied to the specifics of the construction of a system of mathematical models of a nuclear power plant certain of the methodical procedures mentioned above are still insufficiently developed. One of the problems requiring solution is the development of constructional algorithms for the choice of important factors, equivalence, and decomposition.

The widespread incorporation of the method of mathematical modeling into engineering calculations has caused a bottleneck because of the large amount of work required of highly qualified programmers in writing calculational programs. In addition a calculational program developed "by hand" is not optimum since its form and order are based on subjective considerations of the programmers.

The Siberian Power Institute has proposed a more refined automated method of constructing mathematical models of thermal power plants ensuring both the automation of most processes of compiling subsidiary procedures by a description of individual technological processes and plant elements, and the automatic formulation of a mathematical model of a thermal power plant for a given structure of its process flow diagram. Thus it appears possible to mechanize very time-consuming processes by creating mathematical models of thermal power plants and reducing their development time by a factor of ten [7].

The general sequence of the automated construction of calculational programs for thermal power plants is shown in Fig. 2. A number of stages can be distinguished; the mathematical description of the elements of the process flow diagram of each class of power plant; the assignment of a specific power plant design whose calculational program must be obtained, the goal of the calculation; the algorithm of the abstraction, i.e., the transition from the specific power content of the process flow diagram to the system of models considered; the construction of a plan of the calculational process at the level of the system of models; the concretely defined algorithm, i.e., the saturation of the power content of the abstract flow sheet of the calculational process obtained; the obtaining of a calculational program of the specific power plant flow sheet. It is obvious that the approaches which have developed to the automation of the construction of mathematical models of thermal power plants are not unique. Research in this exceptionally important direction should be continued intensively.

Very complex problems will have to be solved for the automation of the process of mathematical modeling of the structures of plant elements. The specifics of these problems are due to the difficulties of representing information on the geometry of the plant elements. Many structural problems which do not have a formal character are difficult to subject to algorithmization.

Special attention should be paid to the information system encompassing all levels of the technological and time hierarchy in creating automated systems of design and development of atomic power plants. The system must ensure rapid and frequent circulation of information between scientific-research and design and construction organizations solving individual problems in the overall system of problems of optimum design and development of atomic power plants. Within the framewok of this system a strict control of the volume of input and outgoing information is necessary. The elements of such an information system originating in individual organizations must in addition record, process, and store information and must provide the mathematical models with the necessary input data.

The most important part of an information system is the data bank designed for storing ordered information. The general principles of constructing data systems are being intensively developed at the present time as applied to various problems of the control of the national economy on various hierarchical levels, and evidently will be basically valid also for the class of problems under consideration, although taking account of specifics requires the development of additional methods and procedures.

The closely interconnected questions of the nature of the input data and the methods of obtaining solutions are very important. At the present time it is recognized that a complex set of random factors has an important effect on the validity of objective laws of the development of the national productive forces, particularly on the trend of the development of the fuel-power industry [1]. This effect makes it impossible to obtain rigorously defined and accurate information on the prospective development of the national fuel-power industry. The size and nature of the error in a large part of the information used in the design and development of atomic power plants are unknown. The uncertainty of the information used does not permit one to obtain a unique optimum solution. It is possible only to determine the region within which each solution is optimum for one or another set of values of the components of the input data. Such a region is called a region of indeterminacy of optimum solutions [1, 8].

The following methods have been developed at the Siberian Power Institute for obtaining a solution when the input data are indefinite; obtaining input data with minimum possible indeterminacy; determining the effect of the input data on the information sought, i.e., finding the region of indeterminacy of the solution; analysis of the region of indeterminacy of the solution for the choice of the variant of the power plant recommended, etc. [9, 10].

The indeterminacy of the input data gives rise to further difficulties in formulating and solving problems of optimizing nuclear power plants. In its most general form the problem can be considered as follows: It is required to determine the minimum of the cost function

$$Z = Z(X, B) \tag{6}$$

under the constraints

$$\Phi(X, C) = 0; \tag{7}$$

$$F^* \leqslant F(X, D) \leqslant F^{**}; \tag{8}$$

$$X^* \leqslant X \leqslant X^{**}, \tag{9}$$

where B, C, and D are vectors of indefinitely specified and frequently interconnected indices of external and internal forces. The presence in constraints (7) and (8) of the nonunique indices C and D requires refinement of the formulation of the problem. Here it is customary to distinguish rigorous and nonrigorous statements of the problem. Other approaches to the mathematical formulation of the problem of optimizing an atomic power plant under indefinite conditions are possible also. Taking account of the indefiniteness of the information used can affect the formulation of the problem of optimizing an atomic power plant, or more accurately can permit the choice of its most reasonable variant. We recall that various approaches are employed to reduce the atomic power plant variants being compared to the same power effect. We consider two extremes:

1. The useful power of a unit is assumed constant. The nuclear fuel requirement and the production of secondary nuclear fuel must be balanced by the introduction of the fixed cost indices for nuclear fuel.

2. The burnup of nuclear fuel per unit is assumed constant. In this case the cost indices for fixed electrical energy and secondary nuclear fuel must be introduced.

For a determinate formulation of the problem both approaches are equally valuable. If the indeterminancy of the input data is taken into account this is by no means so. In this case in each specific problem it is necessary to select the method ensuring the introduction into the solution of the problem of fixed cost indices having the smallest indeterminacy.

CONCLUSIONS

Theoretical research and practical experience indicate that there are large and still far from completely disclosed possibilities of the method of mathematical modeling as applied to technical and economic analyses and the choice of optimum solutions in the development of prospective types of nuclear power plants.

From accumulated favorable experience the use of the method of mathematical modeling and a computer can be recommended for solving most problems of this class.

A further increase in the effectiveness of the method of mathematical modeling requires the development of scientific research directed toward improving methods of the study, processing, and predicting information, the development of methods of obtaining solutions for incompletely specified data, the automation of the construction of mathematical models, and the creation of an automated system of development and optimum design of nuclear power plants.

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OPERATIVE MONITORING SYSTEM FOR THE ENERGY-LIBERATION FIELDS OF THE REACTORS IN THE BELOYARSK NUCLEAR-POWER STATION

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Following the extensive development of nuclear power which has taken place in recent years more and more attention is being devoted to monitoring and control systems for the active zones of nuclear reactors; the provision of such systems ensures the least hazardous and most economical use of nuclear-power installations [1]. Channel-type reactor-control systems operating inside the reactor were described earlier [2, 3].

The second unit of the Beloyarsk nuclear-power station employs an internal reactor-control system [3] using emission-type direct-charge sensors [4]; as secondary instruments for measuring and recording the currents of these it uses an automatic electronic potentiometer of the ÉPP-09MZ type with an amplifier and commutator.

For measuring the currents of the direct-charge sensors in the first unit of the Beloyarsk nuclear power station, an F116 microammeter was used together with a tumbler switch system. Normally the sensor was short-circuited to a common busbar through the switch; in the measuring state it was disconnected from ground and connected to the input of the secondary measuring and recording apparatus. This arrangement lacked the operative character required of nuclear-power installations and it was therefore modernized so as to allow operative control of the energy-liberation fields of both reactors in the Beloyarsk nuclear power station.

In order to provide the desired operative characteristics it was decided to analyze the readings of the direct-charge sensors in the "Karat" data-processing system based on the computer already in service in the second unit of the power station [5]. The direct-charge sensors of the first and second units were directly connected to the computer so as to carry out all the necessary analysis and calculations in accordance with a specified algorithm. The results of the analysis pass to a digital printer, and the deviation signals to the memory display of the reactors in the first and second units. The system allows up to 100 sensors to be connected for each unit and excecutes the following functions:

recording and printing of the currents in all the direct-charge sensors, allowing for a calibration factor;

calculation and printing of the mean current in all the active sensors, and also the current averaged over particular groups;

printing of the currents in all the sensors when placed in a standard or reference field, for which a complex of thermophysical calculations may be carried out in an external computer;

calculation of the deviations (ΔI_i) of the sensor currents from the reference values on the basis of the equation

$$\Delta I_i = \left(\frac{I_i}{I_i^0}\frac{\overline{I}^0}{\overline{I}} - 1\right) 100\%,$$

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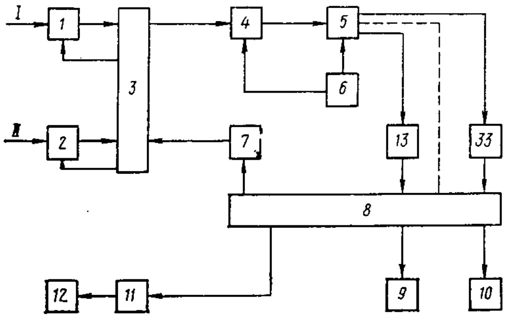

Fig. 1. Structural arrangement of the system of operative control for the energy-evolution fields of the reactors of the Beloyarsk nuclear-power station: 1, 2) switching units; 3) coupling unit; 4, 5) input and output units; 6) address unit; 7) control unit; 8) "Karat" data-processing system; 9, 10) memory displays and digital printer of the "Karat" system; 11) unit coupling with the memory display of the reactor in the first unit; 12) memory display of the reactor in the first unit; 13-33) normalizing converters; I, II) coupling lines from the sensors of the first and second units.

where I_i is the current of the i-th sensor, μA; $\overline{I}$ is the mean current of all the active sensors, μA; I_i^0 is the reference value of the current in the i-th sensor (when placed in a neutron field close to the optimum), μA; $\overline{I}^0$ is the average current of all the active sensors in the reference neutron field, μA;

printing the ΔI values of all the sensors, simultaneously comparing with the set value ("greater than" or "less than") stored in the computer memory, which is capable of being altered if necessary by the operator. Deviations above the setting are printed in red on the recording paper, with a corresponding sign (plus or minus);

producing signals for recording any deviations (ΔI_i) exceeding the set value on the memory display of the reactor.

The structural arrangement of the system is illustrated in Fig. 1. The sensor signals of the first and second reactors, combined into groups with a maximum of 20 sensors in each, pass to the inputs of the switching units and then through the coupling system to the amplification input. The output unit contains memory cells from the output of which signals pass to the norm-setting converters; a unified current signal of 0-5 mA is then introduced into the computer for analysis. After analysis in accordance with the algorithm, the results pass to the digital printer and the memory display of the first reactor (via the coupling unit) and the second reactor of the power station. The serial numbers of the reactors and groups of sensors pass from the computer to the control unit, which executes the required switching processes in the switching and telecommunications units. The address unit ensures synchronous switching of the channels in the input and output units within the selected group of sensors. For monitoring the amplifying tract common to the whole system a standard signal is applied to its input, and after amplification this is analyzed in the computer.

The program in the computer memory consists of three parts. The program for data collection and sensor control collects information from the direct-charge sensors and fills the corresponding positions in the computer memory; it tests the working order of the sensors and the common amplifying tract as well as any errors in the latter. The analysis program analyzes the sensor currents in accordance with the algorithm, and also sends deviation signals to the memory display of the reactors in the first and second units. When instructed by the operator, the printing program sends the results of the sensor current analysis to the digital printer.

The working order of the sensors is checked by comparison with zero and also with the average sensor current over the reactor. The computer sorts out sensors with readings of under 2% and sensors showing a reading which differs by more than ±30% from the mean current. As already indicated [3], the majority of the sensors in the reactor of the second unit (72 items) are placed in the central apertures of the steam-heating channels, and despite the high working temperature (up to 750°C) prove quite reliable in service.

Some of the direct-charge sensors (up to 20) are placed in the evaporation channels on the periphery of the reactor, at which there are no steam-heating channels but at which up to 50% of all the channels occur. These sensors were less reliable, as they lay in the high-pressure circuit and required special sealing into the evaporation channels; they also worked under conditions of considerable vibration in the down pipes, through which the coolant was fed at 5-10 m/sec to cool the fuel elements. In addition to this, the sensors in the evaporating channels required further calibration, i.e., matching to the readings of the sensors in the steam-heating channels. In view of all these difficulties, the sensors of the second unit were modernized, for which purpose 14 steam-heating channels were moved from the edge of the steam-heating zone into the evaporating zone at the reactor periphery in 1973-1974 and direct-charge sensors were placed inside them. The interchanged steam-heating and evaporating channels were connected to the previous pipelines (through which the coolant was introduced and carried away) by means of specially shaped channels. The unification of the sensor system (sensors placed solely in the steam-heating channels) greatly simplified the system for monitoring the energy-liberation fields and increased its reliability. The question as to the transfer of steam-heating channels into the evaporation zone in the first unit of the power station is still being considered.

All the sensors in the reactor of the second unit are divided into five groups (up to 20 in each). The order of arranging the reactor sensors in each group and measuring their currents was chosen in the following way. In the first and third groups the sensors were arranged along two mutually perpendicular diameters. The sensors in the second, fourth, and fifth groups were arranged in rings in the plane of the reactor, with radii of 1.9, 2.5, and 3.0 m from its center. By obtaining the average sensor current over the whole reactor and over each group, as well as the maximum and minimum sensor currents of each group, the nonuniformity and symmetry of the energy-evolution field along the rings of three groups (second, fourth, and fifth) and its equalization along two diameters in the steam-heating zone (first and third groups) may be characterized operatively.

In the reactor of the first unit the sensors are divided into four groups (up to 20 in each). In three groups the sensors are arranged along rings in the plane of the reactor with radii of 1.8, 2.4, and 3.1 m, and in one group uniformly in the central part of the reactor lying in the evaporating channels.

For the operative measurement of the absolute or relative power of the reactor and a rapid determination of any changes in this power, the average current of the sensors uniformly distributed over the active zone of the reactor may be used; the sum of the currents in these gives a fairly accurate picture of the total power of all the fuel channels.

According to the metrological characteristics of the direct-charge sensors, their nonlinearity is 0.5-1.0%. The total power of the fuel channels at reactor power levels of 5-100% N_{nom} may be determined with an error of 1.0-1.5% if the average current in the sensors at nominal power has been calibrated to an appropriate accuracy. Experience in the use of the second reactor of the Beloyarsk nuclear-power station shows that this may be achieved by the repeated measurement of the thermal balance of the reactor, using various thermotechnical methods, with the simultaneous measurement of the mean sensor current. Then

$$\Delta N_r = K_A (\bar{I}_2 - \bar{I}_1),$$

where ΔN_r is the change in reactor power, MW; $\bar{I}_1$, $\bar{I}_2$ are the mean sensor currents in the reference and new fields, μA; K_A is the calibration coefficient averaged on the basis of repeated measurements of the nominal thermal power of the reactor by different methods, MW/μA.

$$K_A = \frac{1}{n} \sum_n \left(\frac{Q_{nom}}{\bar{I}_{nom}} \right)_n,$$

where Q_{nom} is the nominal thermal power of the reactor, MW; $\bar{I}_{nom}$ is the average sensor current at the instant of determining the nominal thermal power, μA; n is the number of measurements.

The quantities N_r, ΔN_r, N_r (% N_{nom}) and ΔN% are calculated in the computer (with respect to the average reactor sensor current at a specified instant of time and at nominal power).

Before direct-charge sensors were introduced for monitoring and controlling the fields of energy evolution in the Beloyarsk nuclear-power station, computing and "rod" methods were employed [6]; the error of such methods has greatly increased at the present time owing to the high fuel burn-ups in the fuel channels and the burn-up of the absorbent material in the control rods. Hence the use of direct-charge sensors in addition to these methods has reduced the error in calculating the power of all the fuel channels in the reactor of the second unit by 5-6% and slightly increased the power of the whole unit.

Whereas in the previous two or three years the stable maximum thermal power of the second reactor was 480-500 MW, as a result of the foregoing complex of measures this was increased to 530 MW in April 1975. In this connection the introduction of the operative sensor system exerted a major influence, all the more so in view of the fact that, after reaching maximum power, the neutron field in the reactor is required to be maintained constant, subject to the condition that the field of energy evolution should not be distorted beyond certain specific limits during the burn-up of fuel between one charging and another and during the extraction of the control rods.

The direct-charge-sensor operative system enables the established neutron field to be controlled and maintained with a relative error of no more than 3-4%. Subsequent work in regard to the perfection of the operative monitoring and control system for the energy-liberation fields (based on direct-charge sensors) will be directed at reducing the error in the creation of near-optimum fields, and also reducing the error in maintaining the established field, with due allowance for the possible instability of the neutron distribution due to a further increase in fuel burn-up and a change to fuel channels containing a slightly greater relative amount of coolant.

Work is also being carried out at the present time as to the further development of the system and its algorithms, so as to excecute more complicated physical calculations as to the determination of the permissible and limiting power of the fuel channels, to estimate the reliability of the apparatus, and to make recommendations to the operator as to the movement of the control and emergency rods with a view to maintaining the optimum neutron field. For this purpose the most complicated physical calculations will be executed in the general computer of the power station, appropriate communications with the latter being provided for the data-processing system of the second unit [5]. This will enable the maximum available power of the unit to be maintained with due allowance for the necessary reserves relative to the limiting power loads in all the fuel channels. The creation and testing of such a system will also lead to a gain in experience which should prove useful in developing automated monitoring and control systems for other nuclear-power stations.

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STRUCTURAL-GEOLOGICAL FEATURES OF URANIUM DEPOSITS IN COLLAPSE CALDERAS

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UDC 553.495

A large amount of data on hydrothermal uranium deposits which are associated with ring faults has now been accumulated. Deposits of uranium–molybdenum formations in collapse are of particular interest. Structures of this type received the attention of several researchers but some problems of the internal structure of calderas, their geological development history, the origin of the faults interesecting the calderas, and the laws governing the distribution of uranium deposits and other core bodies in the calderas, as well as the history of their mineralization, require additional investigations. The present article is an attempt to fill these gaps to some extent.

Two types of collapse calderas are distinguished as far as the features of their internal structure and their geological development history are concerned. The internal structure of the first type is basically given by volcanite covers; the caldera fault developed in a late stage of the volcanism; the uranium deposits are located in the inner parts of these calderas. Calderas of the second type are composed mainly of extrusive cupolae and subvolcanic intrusions; the ring fault on the edge of these calderas developed at the very beginning of the volcanic activity; the uranium deposits are situated in the area of the caldera edge fault.

Calderas of the First Type. These calderas are known in activated central massifs of folded regions and young activated platforms [1-5]. These calderas are usually confined to the intersections of depth faults. The calderas developed as volcanotectonic depressions in which stratified volcanic covers with subordinated strata of volcanogenic–sedimentary rocks piled up while a section of the crust subsided and a partial step-like collapse took place along internal half-ring shaped faults. In the final stage of the volcanism, when acid lavas discharged, the annular block over the magma focus collapsed and the caldera was formed because the subsurface magma supply source had been depleted.

The base of these calderas consists mainly of old granites and, to a lesser degree, of hornfels and schists and is characterized by a complicated surface dissected into individual troughs and protrusions by inner ring faults and inner tectonic faults. The calderas are filled with basic and intermediate volcanogenic rocks and also with more recent volcanites of acid composition. The total thickness of the rocks is 450-500-800-1000 m. Thin horizons of volcanogenic–sedimentary rocks are of lesser importance in the profiles. The bedding of the covers is inclined at angles of 5 to 10-15° toward the center of the calderas. The horizons dip at angles of 40-80° and more in the area of the depth faults around central volcanic vents and close to the caldera edge fault.

The volcanogenic strata of calderas are intersected by numerous dikes some of which are the root sections of overlying strata, whereas others are more recent subvolcanic bodies which sometimes transform upwards into laccoliths. Subvolcanic half-ring-shaped dikes of quartz porphyries, granite porphyries, and granosyenite porphyries are noted in the caldera edge faults (Fig. 1a).

The youngest postvolcanic dikes of regional occurrence (quartz syenite-porphyries, quartz porphyries, granite porphyries, lamprophyres, young lamprophyres, and diabases) which contribute to the structure of the huge dike belts are of the second group.

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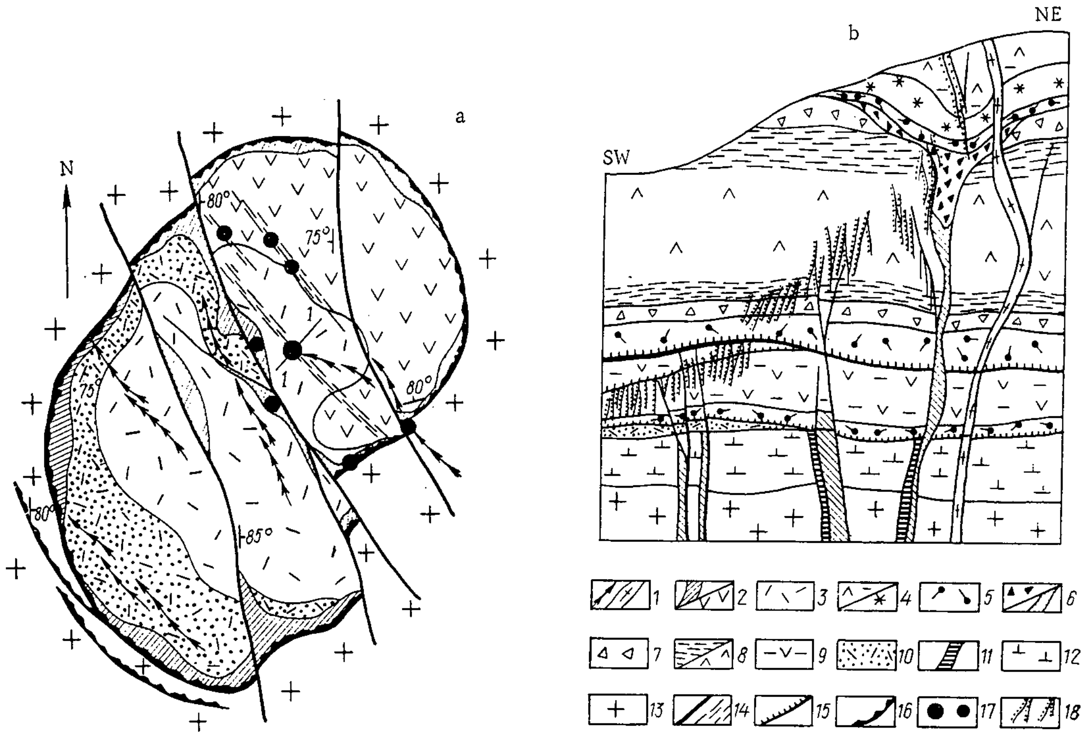

Fig. 1. a) Scheme of the geological structure of calderas of the first type and position of uranium deposits and ore manifestations; b) geological profile of the uranium deposit: 1) most recent postvolcanic dikes of acid and basic composition; 2) subvolcanic facies and vent facies of liparites, their ignimbrites, tuff breccias, and tuffs of the upper formation; 3) various liparites and their fractured facies of the first intermediate formation, 4) striated and spherolitic ignimbrites of felsites; 5) horizons of tuff-sandstones; tuffites, and tuff-breccias; 6) subvolcanic and vent facies (breccias of felsites and various felsites); 7) tuff breccias of quartz porphyries; 8) striated, spherolitic, and solid ignimbrites of quartz porphyries; 9) various ignimbrites and fused tuffs of quartz porphyries; 10) tuff-sandstones, andesite-dacites, and their tuffs and breccias of the second intermediate formation; 11) bosses of andesite basalt and breccias of the lower formation; 13) granitoids of the caldera base; 14) steeply dipping huge transverse and hidden tectonic disturbances; 15) slanting interlayer disruptions; 16) ring faults and late quartz porphyries and breccias filling the ring faults; 17) uranium deposits and ore manifestations; 18) ore zones.

When the structural details of calderas are determined, the complicated intermeshing of ring faults and linear tectonic faults is of great importance. The caldera edge fault with subvolcanic half-ring bodies and extrusive cupolae is of greatest importance among the ring faults. One can distinguish ring faults inside the caldera with an extensively collapsed block in the ring; arched faults in the exocontact of calderas are also known.

The tectonic faults of calderas are divided into transverse (concentrated) faults, which are clearly visible on the contemporary surface and which sometimes intersect and displace the caldera edge faults, and hidden faults (Fig. 1b), which can be recognized in the base but which are strongly masked (scattered) in the volcanic cover layers, particularly in their upper horizons.

Sloping interlayer disruptions which developed mainly along the contact of layers and which are characterized by sharply differing physical and mechanical properties are of great importance for the structure of calderas.

Hydrothermal uranium deposits are located within the volcanic strata and occur in some calderas also in the granites of the base or in the inner part of the caldera (see Fig. 1a). The distribution of the uranium deposits in the calderas is given by the zones of hidden faults (particularly typical is the position of ore bodies in the decay fan of individual upward fault seams), by protrusions of the rocks of the base, by intersections and conjugation areas of linear tectonic faults.

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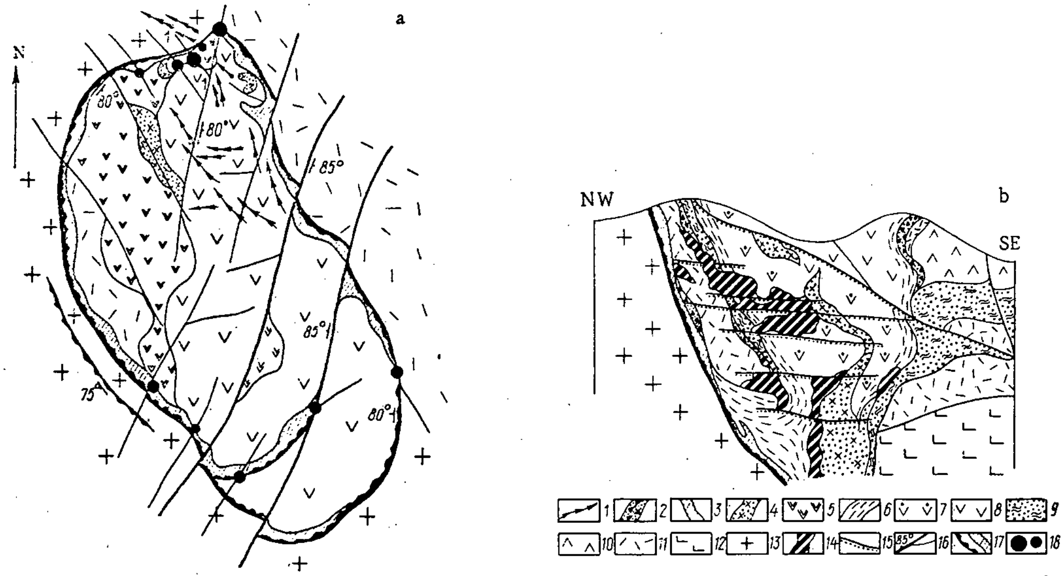

Fig. 2. a) Scheme of the geological structure of a caldera of the second type and positions of the main uranium deposits and ore manifestations; b) geological profile of one of the uranium deposits: 1) most recent postvolcanic dikes of acid and basic composition; 2) late clastic lavas of liparites (explosion pipes); 3) late subvolcanic intrusions of liparitic porphyries and granite porphyries; 4) subvolcanic intrusions of granosyenite-porphyries and granite porphyries; 5) clastic lavas of liparites, felsites, liparitic porphyries; granite porphyries of late volcanic cupolae; 6) clastic lavas of liparites, felsites; 7) liparitic porphyries and granite porphyries; 8) felsites, clastic liparites, liparitic porphyries, and granite porphyries of an earlier volcanic cupola; 9) felsites and clastic liparites; 10) liparitic porphyries and granite porphyries; 11) volcanic covers of the upper formation; 12) volcanic covers of the intermediate formation; 13) granites of the base; 14) ore bodies; 15) gently sloping tectonic faults; 16) steeply dipping tectonic faults; 17) ring faults and subvolcanic intrusions of liparitic porphyries and granite porphyries which are confined to the ring faults; 18) uranium deposits and ore manifestations.

The ore bodies of the deposits are formed by veins or vein-like beds, by stockwork-like bodies of greatly different shapes, among them macrostockworks, and by sloping irregular interlayer streaks and impregnation beds. The minerals are very irregularly distributed in the deposits. The physical and mechanical properties of the enclosing volcanic rocks are of greatest importance. Volcanites of low strength, low values of the modulus of elasticity and the Poisson coefficient, and increased porosity favor the distribution of the minerals. Interspersed layers of volcanogenic–sedimentary rocks and horizons of dense lava with low porosity act as shields. The vertical spread of the mineralization varies between 600–650 m and 1 km and more in this type of caldera.

As far as their ore content is concerned, these calderas are far from similar. Only very small ore deposits are known in some of them, one or two deposits and a series of ore manifestations occur in others, and several deposits are known in a third group of these calderas. Only tentative conclusions concerning the reasons for these differences can be drawn. The differences in the ore content of calderas can be caused by the details of the base which, when it is very hard, furthers intensive mineralization. We can also conclude that an increased ore content results from deeper ore-bearing foci of high yield. Volcanites of basic and intermediate composition, which are characterized by increased alkalinity (trachybasalts, trachydacite, and trachyandesite) are of increased importance in these calderas.

Calderas of the Second Type. The caldera shown in Fig. 2a is an example of this caldera type. This caldera is situated within an activated central massif and confined to the conjugation area of a depth fault with northeastern extension and tectonic faults of northwestern direction [5–7].

The caldera is built up by a huge extrusion dome of acid rocks; the root of the dome is located mainly along an annular edge fault and linear depth faults and, in some parts, along arched internal faults and linear tectonic faults. The cupola is composed of a series of subvolcanic bodies and shallow extrusive domes which cut through the main cupola. Ancient granitoids with intersecting volcanites are disposed at the base of the cupola and around it. The surface of the base is characterized by local protrusions and depressions. The cupola has zonal structure. The peripheral parts of the cupola are composed of felsite porphyries which are in direct contact with their elastic varities. Toward the center the felsite porphyries are replaced first by quartz porphyries and then by granite porphyries. In a vertical cut the copola has a thickness of as much as 1500 m and of about 2500 m when the eroded portion is included in the consideration. The rocks composing the arched section of the cupola are almost horizontal and slightly slope from the cupola's center.

In the zone of the caldera edge fault and along the huge tectonic faults, the basic cupola is intersected by several small, younger dome-shaped volcanoes situated in areas in which the caldera is cut by linear tectonic disturbances. In a vertical profile, the dome-shaped volcanoes have a complicated funnel shape (Fig. 2b). In the portion near the contact the volcanoes are composed of clastic quartz porphyries, felsite porphyries, and felsite-like quartz porphyries; in the central part, the volcanoes consist of quartz porphyries, younger subvolcanic granosyenite-porphyries, and granite porphyries. These rocks of the volcanoes are frequently intersected by explosion pipes and complicated clastic lavas with which the volcanic activity ended within the caldera under consideration.

The subvolcanic bodies of granosyenite-porphyries and granite porphyries compose small stocks, irregular fissured bodies, and dikes in the form of half-rings and arches. A huge arched dike of granite porphyries fills the caldera edge fault. Small half-ring dikes of granosyenite porphyries are known in the zone to the northeast of the extrusive massif near the caldera.

Dikes of regional composition (microgranophyres, spherolite-porphyries, felsites, and diabasic porphyrites) are superimposed on the volcanic formations of the massif. In particular, a belt of micrgranophyres, which is associated with an arched zone of conical periclinal faults and huge fissures, is found in the Northern part of the massif.

The structural details of this caldera are given by a complicated combination of linear tectonic faults and ring faults. A depth fault and conjugated faults issuing from the depth fault in Northwestern direction are the main tectonic disturbances. A ring fault on the caldera edge is one of the huge ring faults. Dikes and stocks of subvolcanic bodies and several dome-shaped volcanoes are situated on that fault. In the Northern massif zone near the caldera, relatively small half-ring-shaped faults with dikes of granosyenite-porphyries and an arched zone of conical periclinal faults and fissures with dikes of micrograno-phyres are known. An external arched fault is recognized in the Southwestern exocontact of the massif.

In addition to the ring faults, there exist huge rectilinear fissures and faults which grew during the formation and further development of the extrusive massif. Among these faults there are steeply dipping radial rupture fissures and steeply dipping rupture fissures running across the long axis of the cupola. Of particular importance are huge, almost horizontal fissures in the Northern and Northwestern part of the massif close to the caldera zone (see Fig. 2b). These fissures probably developed as rupture fissures during the caldera collapse and piled up along the almost horizontal surfaces of the arched part of the extrusive cupola. More recently, during block transpositions along tectonic faults, huge selvages of tectonic clay developed.

The geological history of the extrusive formations is rather complicated and comprises the following stages: 1) evolution of a ring fault on the edge under the influence of intruding acid magma; 2) formation of a ring horst and a huge extrusive dome covering the horst (development under the influence of the magma); 3) development of younger dome-shaped volcanoes at the intersection points of the ring fault on the edge with linear tectonic faults; 4) graben-shaped collapse of the ring block over the depleted magma focus near the surface and development of the collapse caldera; development of almost horizontal rupture fissures and of zones of conical periclincal faults and fissures along with external and internal arched faults during the sagging of the ring block; 5) intrusion of the younger subvolcanic bodies, formation of local half-ring-shaped faults in the zone near the caldera; resumption of the development of tectonic faults; 6) penetration of clastic lavas into the dome-shaped volcanoes (formation of explosion pipes); 7) occurrence of huge tectonic motions, resumption of the development of tectonic faults, and initial stages, new development, and shifting of some ring faults, and intrusion of regional dikes; and 8) resumption of the development of tectonic and ring faults and fissures, and postmagmatic processes.

The uranium deposits are located in the dome-shaped volcanoes (see Fig. 2a) which gravitate toward the intersections of the caldera edge fault with linear tectonic faults. These sections are deep magma channels which have been active for a long time and over which hydrothermal solutions moved in the post-

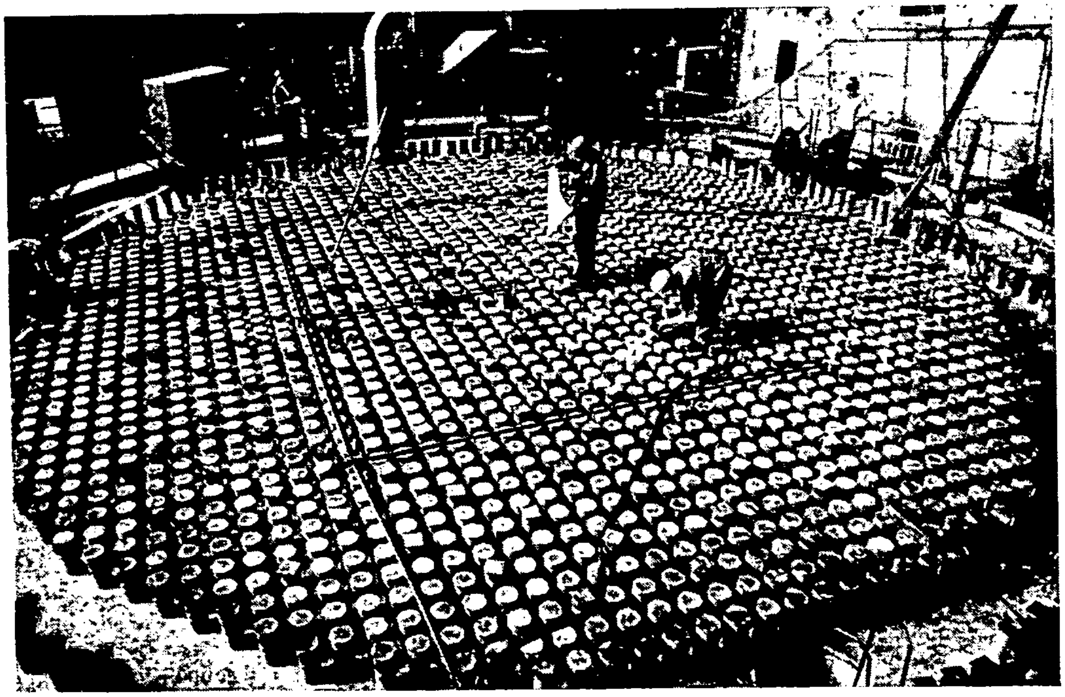

Fig. 3. On the atomic constructions sites of the USSR. Chernobyl'sk Atomic Electric Power Plant. Assembly of the reactor. Photo by I. Pap.

magmatic stage. The distribution of the deposits seems to be substantially affected by the structural details of the base of the caldera and, particularly, by protrusions of granitoids.

The ore bodies of the uranium deposits are represented by steeply dipping ore zones of irregular shape and by lenses, steeply dipping ore columns, and gently sloping ore beds with stockwork structure. The following details of the spatial distribution of the mineralization are recognized: 1) mineralization dominating in the root zones of volcanic and subvolcanic bodies; 2) mineralization confined mainly to the most recent subvolcanic bodies, i.e., granite porphyries and granosyenite-porphyries; 3) many of the ore bodies are situated along gently sloping fissures (Fig. 2b) of the conical periclinical zone and at points at which the zone is intersected by vertical tectonic faults and fissures; 4) mineralization confined to the areas in which tectonic faults intersect and merge; these areas are very important; 5) influence of rocks with favorable physical and mechanical properties upon the distribution of the mineralization. The vertical spread of the mineralization varies between several hundred meters and one kilometer and more at the deposits under consideration.

Deposits in the Exocontact of the Calderas. The structural and geological details of these deposits will be considered in the example of an uranium-bearing ore field situated in the exocontact of a caldera which is probably of the second type and is located in the Paleozoic belt formed in the late orogenic development stage of the folded region [8]. The enclosing volcanogenic-sedimentary stratum is intersected by vent-type facies of volcanites, a massif of granites, and acid and basic dikes of various ages. Pregranitic dikes of quartz microdiorites and granophyres and postgranitic dikes of granite porphyries, quartz porphyries, and porphyrites are recognized among the dikes.

A half-ring-shaped belt of arched faults and huge fissures is recognized in the Northern exocontact of the caldera. Vertical and adjacent cylindrical, and even conical, centroclinal and periclinal faults gently dipping toward the center of the caldera or in the opposite direction are found among the faults and fissures. Dikes and irregularly fissured bodies of granites and all the above-listed pregranitic and postgranitic dikes are confined to this area.

The volcanogenic—sedimentary layer of the ore field and the arched faults are dissected by steeply dipping tectonic faults and huge fissures. Of particular importance are submeridional faults of the base, which have dissected structures in the volcanogenic layer, and faults with a compact structure and northeastern (45-70°) extension.

In the areas in which the arched faults are intersected by submeridional tectonic faults, rather deep hidden vertical volcanic necks of satellite craters are found; at some points the volcanic necks are hidden by extrusive cupolae having rounded or elliptical or, less frequently, irregular shape in a horizontal cross section. The upper portions of some of the volcanic necks are broadened. The filling of the volcanic necks began with the intrusion of eruptive breccias. Younger acid melts intruded and the formation of extrusive felsitic porphyries began. In the region of the endocontact, these porphyries were transformed into fluidal varieties and eruptive breccias. The volcanic processes in the volcanic necks were terminated by the intrusion of clastic lavas which form irregularly shaped volcanic pipes.

Once the volcanic activity had ended, the intrusive processes developed. The intrusive processes are associated with the development of the above-mentioned pregranitic dikes. More recently rather huge intrusions of granites and their veined bodies began; after that, the postgranitic dikes were formed. All these rocks, in the form of dikes and dike-shaped bodies, cut through the volcanic necks.

The ore-bearing sections of the ore field are the individual deposits of an uranium—molybdenum formation and are located, in the majority of cases, in the aforementioned volcanic necks and, occasionally, in developed extrusive domes. The ore bodies have the form of tubular beds, lenses, stocks, isometric stockwork-like bodies, and fissured veins. Tectonic faults and fissures determine the distribution of the ore bodies. The tubular bodies are usually situated in the endocontact of volcanic necks, at points at which the contact is intersected by tectonic faults. Lens-shaped beds are frequently related to slanting zones which are situated in the apical portions of extrusive domes under shielding horizons of fluidic felsites. The fissured veins are related to conjugated ruptures and fanning ruptures of tectonic faults. The ore distribution depends upon the areas in which faults of various directions intersect; these areas comprise intersections of slanting zones with steeply dipping zones. Most of the ore is situated in felsitic porphyries having the texture of the massif. These porphyries are characterized by the strongest fissure formation which is given by their physical and mechanical properties favoring the development of fissures. Rocks of increased porosity also were important for the distribution of the ore. As to its age, the mineralization is younger than the most recent veined rocks, i.e., the dikes of the porphyrites.

The above data on the geology of the deposits make it possible to extend the considerations to the genetic features and, specifically, to the relation with the magmatic foci. Magmatic foci of two types can be recognized in the regions. There exist small pockets near the surface, which were direct sources of volcanites contributing to the structure of the volcanic formations. According to the geophysical data, these pockets are younger than many contemporary volcanoes [9]. The collapse calderas evidently developed because the above foci were depleted in the final period of the development of the volcanism.

On the other hand, there certainly existed much greater magmatic foci which were located at greater depths and which not only supplied the above local pockets but also were direct sources of magma in the ensuing development period of intrusive magmatism (recent granites and, apparently, regional dikes).

In the uranium-bearing ore fields under consideration, intrusive magmatism, which is very clearly visible in the latter example, developed after the volcanism had subsided. Once the volcanism had ended, pregranitic dikes intruded; after that, granites and postgranitic dikes developed and, only thereafter, the postmagmatic process and the ore deposition took place.

Our conclusion is that huger depth foci rather than local near-surface pockets were the basic sources of the ore material of the uranium deposits under consideration. This conclusion is confirmed by the distribution of the mineralization in the deposits, i.e., by the rather pronounced vertical spread of the mineralization, though the depth dependence of the type of mineralization is relatively weak. It seems that the ore-bearing foci were situated at the roots of the zone in which the granite layer of the crust melted.

CONCLUSIONS

1. Uranium deposits are found in two types of calderas. Mainly volcanic covers with volcanites having a composition from basic to acid contribute to the structure of the calderas of the first type which have a relatively long geological development history. Caldera faults developed in the final state of the volcanism and therefore have a relatively short history. The uranium deposits are situated in the internal part of the calderas.

Extrusive domes and subvolcanic formations composed of acid rocks contribute to the structure of the calderas of the second type which are characterized by the shorter geological development history. A ring fault on the edge limits the dome; this fault developed at the very beginning of the volcanic processes and continued to grow until the magmatic activity subsided. Uranium deposits in calderas of this type are situated in the zone of the caldera edge fault.

2. The uranium deposits under consideration have a close spatial correlation to the magma-carrying channels in the depth, which developed for a long time and were the paths of movement of hydrothermal solutions during the postmagmatic stage. These channels are confined to the structural elements of highest mobility in the calderas. In the calderas of the first type, areas in which the huge tectonic fractures intersected in a zone of hidden faults of the base were the structural elements of highest mobility. In the calderas of the second type, these elements were formed by areas in which caldera edge faults were intersected and actived by huge linear tectonic fractures.

3. The distribution of the mineralization in the calderas of the first type depends significantly upon the positions of volcanic rock horizons having appropriate physical and mechanical properties (increased porosity and brittleness). In the calderas of the second type the uranium mineralization was confined to the young volcanic domes and to the subvolcanic intrusions whose internal structure determines both the distribution and the morphology of the ore bodies.

4. The basic source of the ore material of the deposits was formed by deep crustal zones of magma formation rather than by small foci near the surface under the volcanoes. The deep crustal zones of magma formation also supplied the upper magma pockets. Thus, the ore-enclosing volcanic rocks of the volcanic formations described are characterized by a paragenetic relation to the mineralization. The deposits have a structural relation to the volcanic formations; the structural relation confines the deposits to deep magma-carrying channels which existed for a long time and which recently were converted into ore-supplying systems.

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CONTINUOUS UNDERGROUND ORE-MINING OPERATIONS WITH THE AID OF NUCLEAR EXPLOSIVES

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In many Russian and foreign mines working on huge, steeply dipping ore deposits with systems of block caving, excellent technological and economic indicators are obtained. For example, the average monthly output of the workers of the most important professions in the underground winning of ore in one of the combines was as follows in 1973: drill operator in cutting operations – 695 tons; powderman filling charges into boreholes – 1639 kg; and scraper operator – 384.4 tons. The specific removal of preparatory working per 1000 tons of ore reserves in a block is between 2 and 3 m, and the consumption of explosives for the first and second cutting is 0.46 and 0.14 kg/ton, respectively. The cost of production in ore mining is relatively low in the mines of the combine and varies between 2.04 and 2.43 ruble/ton. These values were reached because the best mining systems, highly efficient drilling equipment, and mechanical UZDM-1 and ZMBS-2 charging devices for filling boreholes with the 79/21 granulite were used; the ore was delivered with the aid of vibrating feeder systems; and the ore was hauled with the aid of a 100-150 kW scraper loader. A further increase in the output and the total production volume by improving the existing methods and the equipment encounters certain difficulties, because the possibilities offered by some of these methods and machines have been exhausted. This applies to the large-scale crushing of ore and to the delivery of the cut ore from the stoping blocks and its transfer to intermediate levels.

It is generally accepted that the cost of the mining operations, boreholes, materials, and labor in crushing ore reaches 30% of the total cost of winning. In order to reduce the expenditures for material and labor in this important technological process, one can use the extremely compact, low-price explosives such as relatively powerful nuclear devices.

The underground mining technique with crushing of the ore by nuclear explosions was proposed by several authors and is based on investigations of the mechanical effect of single camouflet explosions. The cavity formed in the explosion caves in the course of time and the so-called cave-in tube, which is filled with rock debris, develops. The dimensions of the debris depend mainly upon the natural jointing of the rocks. The rocks can be extracted from the cave-in tube with known mining techniques. In order to remove the rocks beyond the cave-in tube, an additional crushing of the rocks is required. It is therefore necessary to provide tunnels to the mining points and to drill holes. This minimizes the advantages of the nuclear explosions.

Some authors have therefore concluded that, for obtaining the greatest possible amount of crushed ore, one must try to obtain by the nuclear explosion a cave-in tube of greatest possible size. This approach to the solution of the problem implied that, for increasing the volume of removable ore, one must use explosions with a power of several dozen or even hundreds of kilotons. The diameter of the resulting cavity is so large that the cave-in of the rocks propagates upwards over several hundred meters. In this case, the volume of the cave-in tube and, hence, of the fragmented ore amounts to as much as hundreds of thousands of cubic meters and the utilization of nuclear explosions for underground mining may be profitable (see Fig. 1).

*Deceased.

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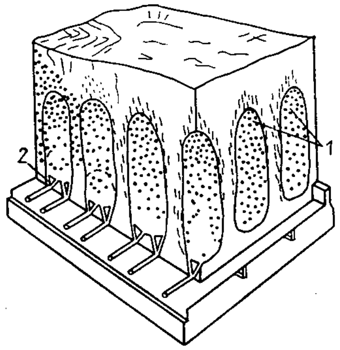

Fig. 1

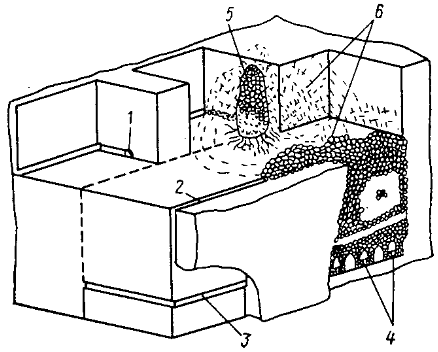

Fig. 2

Fig. 1. Technique of underground mining with rock crushing by nuclear explosions of high power: 1) cave-in tube; 2) workings for transporting the ore.

Fig. 2. Technique of underground mining with ore crushing by nuclear explosions and use of shielding slits: 1) working for the charge; 2) vertical dissecting slit; 3) horizontal undercutting slit; 4) working for the transport of the ore; 5) cave-in tube; 6) crushed ore.

However, an analysis of the real mining conditions in known ore deposits has shown that explosions with a power of at most 1-3 kton of a single charge can be employed in the majority of cases. The morphology of the ore bodies and the seismic effect upon underground sections of a mine and buildings on the surface makes it necessary to limit the power of the explosion.

Owing to the relatively small diameter (15-25 m) of the cavity created by a single camouflet explosion with a power of 1-3 kton in the case of compact ore, the amount of crushed rock within the cave-in tube amounts to several ten thousand cubic meters. This is insufficient for both continuous operation and obtaining advantageous economical conditions from the utilization of nuclear explosions. In mining work, explosions with a power of 1-3 kton may be useful, provided that ways will be found for increasing the amount of the crushed ore and for improving the quality of the crushing effect; furthermore, apparatus must be developed for hauling and transporting large lumps of ore so that the problems of continuous mining are solved with a combination of methods. One of the methods of improving the crushing of ore is based upon the use of reflecting surfaces located on the path over which the blast wave propagates.

The results obtained in nuclear underground explosions set off at a reduced depth of 55-60 $m/kton^{1/3.4}$ from the surface indicate that artificial free surfaces can be used to increase the amount of crushed rock. At the present time no reliable methods of estimating the disintegration of rocks by nuclear underground explosions have been developed for the case of reflecting surfaces. Therefore, both the amount and the quality of the crushed rock are tentatively estimated with experimental data.

Let us consider as an example the results of the American explosion "Sulky" in which, in a first approximation, the earth's surface played the role of the reflecting surface [1]. A 0.085-kton nuclear charge was detonated at a depth of 27.4 m or 58 $m/kton^{1/3.4}$ in basalt. The incomplete camouflet effect (bulging) of the explosion produced a hillock of crushed rock with a radius of 24 m, a height of 6.3 m, and a small funnel with a depth of 2.9 m and a radius of 8.2 m formed by subsidence in the center of the hillock.

The total volume of crushed rock amounted to about 30,000 m^3. The granulometric composition of the rock in the upheaved hillock was suitable for continuous underground mining operations. At a size of the pieces of 0-10, 10-30, 30-50, 50-70, 70-100, 100-120 cm, the concentrations were 12, 26, 25, 12, 10, and 15%, respectively.

Compared with the nuclear camouflet explosions "Hardhat" and "Shole" in granite, in which the yield of crushed rock amounted to 19,800 and 19,200 $m^3/kton$ [2], the yield in the explosion "Sulky" is almost 20 times greater. It must be noted that this considerable difference in the specific yield of crushed rock can possibly be explained with a scale effect of the explosion (0.085 kton in the case of Sulky, 5.4 kton in the case of Hardhat, and 12 kton in the case of Shole). Therefore the results of the "Palangin" explosion are a more adequate typical example of the influence which reflecting surfaces have upon the increase in the specific yield of crushed rock [3].

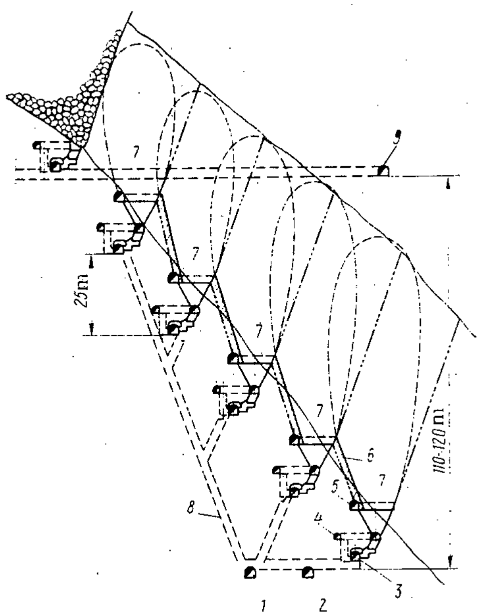

Fig. 3

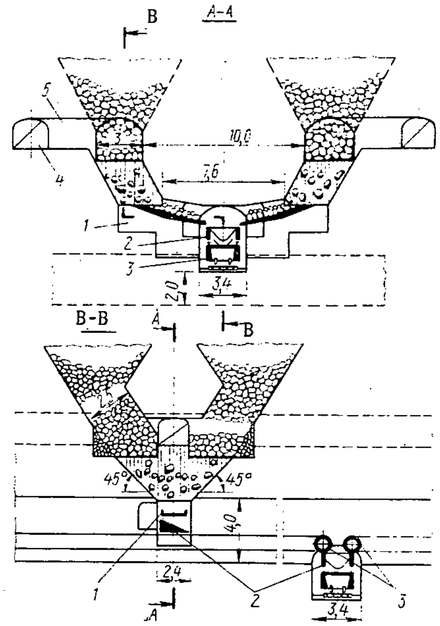

Fig. 4

Fig. 3. Version of the underground mining of an inclined bed with ore-crushing by nuclear explosions: 1-5) transport drift, auxiliary drift, conveyor drift, control drift, and undercutting drift; 6, 7) inclined and horizontal shielding slits; 8) ore chute; 9) chamber with the charge.

Fig. 4. Equipment for the extraction of ore from the workings: 1) vibrating feeder; 2) reloading chute; 3) conveyor; 4) control galley; 5) manway for inspection.

A 4.3-kton charge was detonated at a depth of 85.3 m (55.6 m/kton$^{1/3.4}$) in porphyritic trachyte (dry, compact rock). Crushing of the rock up to the surface and bulging of a dome in analogy to the Sulky explosion were anticipated in that explosion. But owing to a premature escape of gases, a visible funnel with a diameter of 72.6 m, a depth of 24.0 m, a 6.47-m-high crest of piled-up material, and a radius of 44.6 m was formed on the surface. The volume of the visible arch of the funnel was 98,760 m^3 and the funnel volume of 35,570 m^3. The total volume of crushed rock within both the cave-in tube and the visible barrier of the funnel amounted to 1000 $\cdot$ 10^3 m^3 (approximately 230 $\cdot$ 10^3 m^3/kton), i.e., this volume is 11.6 times greater than in the Hardhat explosion. Thus, reflecting surfaces situated at certain distances from the charge can ten times increase the yield of crushed ore per unit power of the charge.

For underground mining, intersecting slits and cuts from below can act as exposed surfaces (Fig. 2). Experience has shown that these surfaces protect the workings behind them from crushing. This is of great importance. Slit-shaped workings make it possible to use the energy of the reflected waves for additional crushing of the ore within certain boundaries and to reduce the effect of the explosion upon the rocks which, after the explosion, are intersected by workings required for recovery of the ore.

The removal of the chipped-off ore from the stoping blocks is an important process in continuous ore-winning operations. The principal method for improving the hauling of ore in huge Russian mining operations comprises the development, testing, and implementation of vibrating feeders of various designs (see Table 1). A vibrating feeder on the basis of the PVR-4.5/1.6 machine is recommended for further development and industrial implementation.

A pilot experiment has shown that the following important technological process, viz., the continuous transfer of the huge lumps of ore (with a size of up to 1200 mm) can be effected with a band conveyor on movable supports (KLT-120 machine) developed in accordance with the scheme of A. O. Spivakovskii, Corresponding Member of the Academy of Sciences of the USSR, in the State Institute for the Planning of Nickel Industry Establishments. The conveyor was tested under industrial conditions in a special section of a stoping block. The combination of the mechanisms for the removal and the hauling of the chipped-off ore comprised PVR-3 vibration feeders and a 150-m-long KLT-120 conveyor on movable supports, which provided ore to an ore chute with a capacity of 19,000 tons. We list below the parameters of the operation of the machinery during the tests:

Volume of the ore delivered (tons)	160,000
Output of the machinery per hour of pure work (tons) ..	1500-1800
Maximum size (mm) of a transported ore lump	2000
Size (mm) of a standard piece	1200
Explosive charge spent for the output (kg/ton)	0.039

The industrial experiment rendered initial data for setting up continuous recovery operations with a hauling and release of ore lumps with a size of up to 1200 mm.

The principles of the new technology based on the use of nuclear explosions are as follows:

1. The ore block to be crushed must be bounded by reflecting surfaces which can be formed by dissecting slits, bottom cuts, or contacts between the massif and the cave-in ore.

2. The charge, or a group of charges, must be located above the level on which the removal and the transport take place, as in the usual ore-recovery work.

3. The main workings which are used for removing ore from stoped space must be arranged behind shielding surfaces and must be passable after a nuclear explosion.

4. The charge must be placed so that the zone of increased radioactivity is beyond the limits of the ore body.

5. In order to prevent the radioactivity which is concentrated near the point of the explosion from spreading over the stoped space, in the transposition of the ore in the block one must:

a) select the removal conditions so that the ore which is situated under the radioactive zone is the last to be removed, whereas the zone proper has subsided to the bottom by the time at which the material is removed from that section; and

b) provide for binding (attachment) of the radioactive zone with the aid of cementing admixtures of bitumen to obtain a single mass which reduces the mobility of the radioactive zone and makes it impossible that radioactive residues are mixed into the ore to be extracted.

In principle, a technique with which the radioactive mass is made to remain in the mine is feasible even without binding by a cement, but additional investigations are required to determine the details of this method.

6. The construction of the level receiving the ore and the means for mechanizing the hauling and the transport of the ore from the block must guarantee high intensity of the ore recovery. This is obtained by vibrating delivery means, conveyor transport with remote or automatic control, ore chutes of great volume, etc.

7. Both the dimensions and the locations of the shielding workings must make it possible that the massif is fractured along fissures before the explosion takes place in the bulk of the material.

Figure 3 shows a new version of a mining system in which nuclear explosions and shielding workings are employed. The ore body is worked on levels with a height of 110-120 m. The main workings of the accumulating horizon for the hauling, on which ore is received from intermediate conveyor drifts, forms the base of a level. The conveyor drifts are cut out on various heights of the level and equipped with rolling band conveyors. The distance between the intermediate galleys depends upon the angle of dip of the ore body and the requirement that the extraction of the ore be optimal; the spacing of the intermediate galleys is 20-25 m (see Fig. 3).

TABLE 1. Parameters of the Operation of the Vibrating Feeders

Type of machine	Volume 10^3 ton of the extracted ore	Technical output (tons/h)	Possible yield during exploitation (tons/cm)	Specific consumption of explosive g/ton for ore recovery
PVR-3.0/1.3	40,6	800—1000	2400	46
PVR-4.5/1.6	42,4	900—1300	3200	39
VDPU-4TM	1146,4	600—800	1500	62
KVS-4.5	58.4	700—800	2150	52

Vibrating feeders are used to transfer the ore onto the conveyors and, thereafter, through an ore chute onto the output collecting level. The spacing of the feeders on the conveyor drifts is given by the dimensions of the output configuration and was assumed to be equal to 20 m. Ore chutes are arranged along the ore body every 200 m in relation to the sensible length of the conveyor.

A level for controlling ventilation, a duplex configuration of releasing funnels, and the design of outlet openings are important details of the new system (see Fig. 4). The workings of the level for control and ventilation extend over the vibrating feeders; the outflow and the quality of the ore are checked from that level with fast-response geophysical methods; pieces which are not to size are crushed before they enter the feeder and the air which was filled with dust and contaminated by explosion products is removed and other auxiliary operations are made on the level for control and ventilation. The level for control and ventilation is arranged at a distance of 4.5-5.0 m above the conveyors; the undercutting level is located 12 m above the level for control and ventilation. A reliable extraction of ore from the stoping space is obtained by increasing the cross section of the workings for ore removal. The height of the moving layer reaches 3 m in the release opening.

The zone which is served by the vibrating machinery in the recovery operations is increased by arranging workings for ore extraction in lateral positions relative to the vibrating feeders. The ore arrives on each feeder from two outlet openings so that two simultaneously moving figures (ellipsoids) of the output material are formed. This reduces the probability of an arch being formed. In this design the feeders are not subjected to the pressure resulting from a thick layer of caved-in ore so that stable operation of the feeders is guaranteed. The galleys for control and ventilation make the workings for the transport free from blasting work required for ore extraction. This, in turn, guarantees a high duty factor of the machinery.

The rock massif is crushed by nuclear charges which are arranged beyond the ore body in a blast chamber. The blasting is directed onto exposed surfaces, among them the boundary with the caved-in rock, the lower horizontal shielding slit, and the inclined shielding slit at the boundary of the intersection with the massif. The slits are obtained by explosions in boreholes which are drilled from an undercutting working disposed underneath the section to be crushed. In order to place the nuclear charges, a receiving working is made. A branch drift, which ends in a blast chamber, is made from this working for each charge. The charges can be detonated either in succession or at the same time.

The preparations and the cutting work are made in two stages. Before the nuclear charge is detonated, workings are made for inserting the charge and for the collecting transport drift. The undercutting and intersecting workings along the boundary between the section of the explosion and the massif are determined. Ore from the undercutting and intersecting workings is hauled to the transport level through ore chutes. In the second stage, i.e., after the explosion, workings are made for the extraction of the ore (see Fig. 3), the assembly of units of rolling band conveyors, vibrating feeders, etc.

In order to preclude radioactive contamination of the ore during its recovery, the near zone of increased radioactivity is compacted with the aid of a cement solution which is injected into boreholes drilled into the remaining part of the upper level working. After solidification of the cement solution, a block with a length of several dozen meters is obtained; as predicted by the mechanics of bulk materials, the mobility of the block is restricted to the extent that the block does not proceed to the outlet opening before all the other ore material has been discharged.

The detonation of nuclear charges in the presence of workings for shielding purposes must substantially increase the size of the zone in which a useful effect is obtained from the explosion and must improve the crushing of the ore and, hence, the technological and economical parameters of this working technique. Calculations which included the mechanization of the recovery and hauling of the chipped-off ore have shown that this working method increases the output of a hauling worker to 750 tons per shift and the output of a miner to 40 tons per shift, whereas the divisional cost of production is 20% lower than in the best methods making use of chemical explosives.

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INFORMATION

ATOMIC ENERGY IN THE USSR IN THE NINTH FIVE-YEAR PLAN

L. M. Voronin and E. Yu. Zharkovskii

An extensive program of introducing atomic electric stations into the national economy of the land is being accomplished in the Soviet Union. Not only have new power intensities been introduced in atomic energy stations during the years of the Ninth Five-Year Plan, but a large seal has been produced for the further precipitate growth of atomic power. At this time, atomic electric stations (AES) of more than $6 \cdot 10^6$ kW total rated power operate in the nation. Large-scale power modules of about $20 \cdot 10^6$ kW total power are being erected. Our atomic power has gone from the first experimental-industrial low-power modules of the Novovoronezh and Beloyarsk electric stations over into the construction of precipitate, serial modules of high unitary power which have been checked out in utilization.

Atomic energy stations with two kinds of thermal reactors have received the main development: WWPR – (watermoderated, water-cooled power reactor) a vessel with water under pressure for $440 \cdot 10^3$ kW and $1 \cdot 10^6$ kW, and a BWR (boiling water reactor) – channeled uranium–graphite with boiling water for $1 \cdot 10^6$ kW.

Industrial mastery of the AES with WWPR started more than ten years ago at the Novovoronezh AES with the commissioning of the first $210 \cdot 10^3$-kW power unit in 1964. The affirmative experience of using the first module permitted commissioning a $365 \cdot 10^3$-kW power unit at this same AES and with practically the same equipment in 1969, which had improved technicoeconomic indices, and then later going over to a new stage in the development of AES with WWPR. A third and fourth module with WWPR reactors with $440 \cdot 10^3$-kW power were commissioned in 1971-1972 for the Novovoronezh AES, and marked the beginning of industrial insertion of serial two-module AES.

By duplicating the principal solutions of the first two modules of the Novovoronezh AES, serial AES with WWPR-440 have substantial differences permitting noticeable improvement in the technicoeconomic indices of AES. Among such perfections is firstly enlargement of the unitary power of the equipment, raising the heat carrier parameters, using boron regulation of the reactivity, raising the reliability of the electrical supply of the main circulation pumps (gasket-free) because of the use of special intrinsic need generators, diminishing the number of separate buildings because of the arrangement of the main construction in the principal housing and the set-up of two reactors in a common room, etc. According to the serial design the Kol'skaya AES, the AES "Nord" in the GDR and the AES "Kozlodyi" in Bulgaria have been erected and commissioned in 1973-1975.

All the main structural work on the objects of the starting complex has been completed on the Armenian AES, erected according to this same design, and start-up-adjustment work is going on full speed ahead. Auxiliary shops and structures have been put in operation: the chemical water purifier, the open distributor unit, etc. In the near furture the first unit of the station will go into industrial operation. Experience with the utilization of WWPR AES indicates their reliability and competitiveness with thermal electric stations (TES) operated with imported organic fuels; this experience also permitted essential perfection of the main and auxiliary equipment, the technological diagrams and the structures. New progressive solutions, aimed at a further rise in the reliability and improvement of the utilization conditions, were introduced into the design of a two-module AES in 1973-1974. The West Ukraine AES was erected according to this design, and expansion of the Kola AES and the preparation for construction of a number of other AES of this type at home and in other countries, members of the CEMA, have been carried out.

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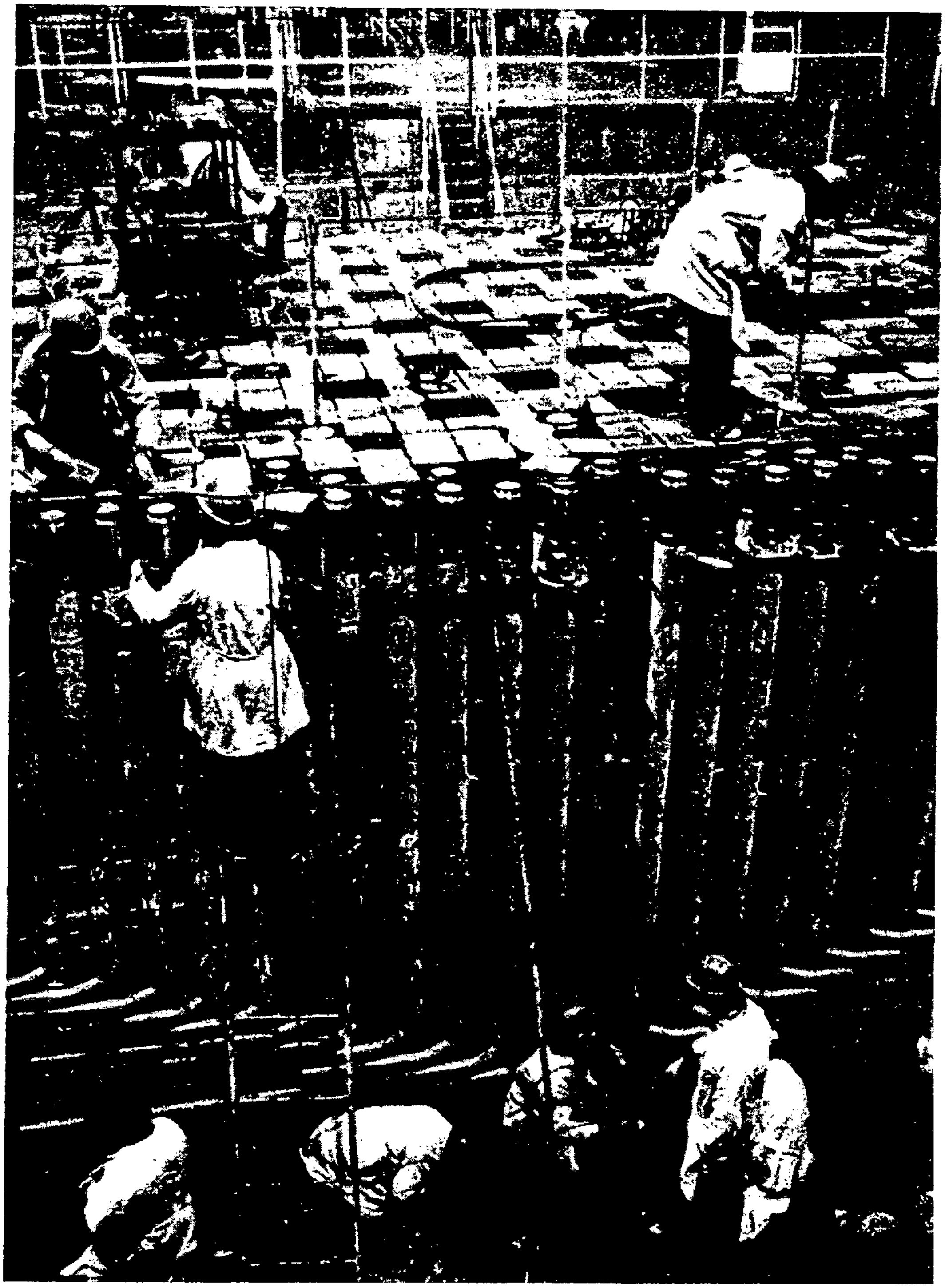

Fig. 1. In atomic buildings of the USSR. Kursk AES. Mounting the equipment in the reactor room. A Pakhomov photo.

The construction of the main buildings has started at the West Ukraine AES. Work on producing the pit under the main and combined auxiliary housing has been performed, concreting the slab to the mark 6.5 of the main housing has terminated. Construction and assembly of the starting-back-up reactor have finished. The first line of cleaning buildings of the economic fecal drainage, the mechanical repair shop, and a number of other objects on a structural base have been set in operation. Concreting the slab of the main housing has also been terminated at the Kola AES.

The design and erection of AES with VVÉR-1000 reactors of $1 \cdot 10^6$ kW unitary power started in the Ninth Five-Year Plan. The design of a two-module AES with such reactors and turbines with $1 \cdot 10^6$ kW power,

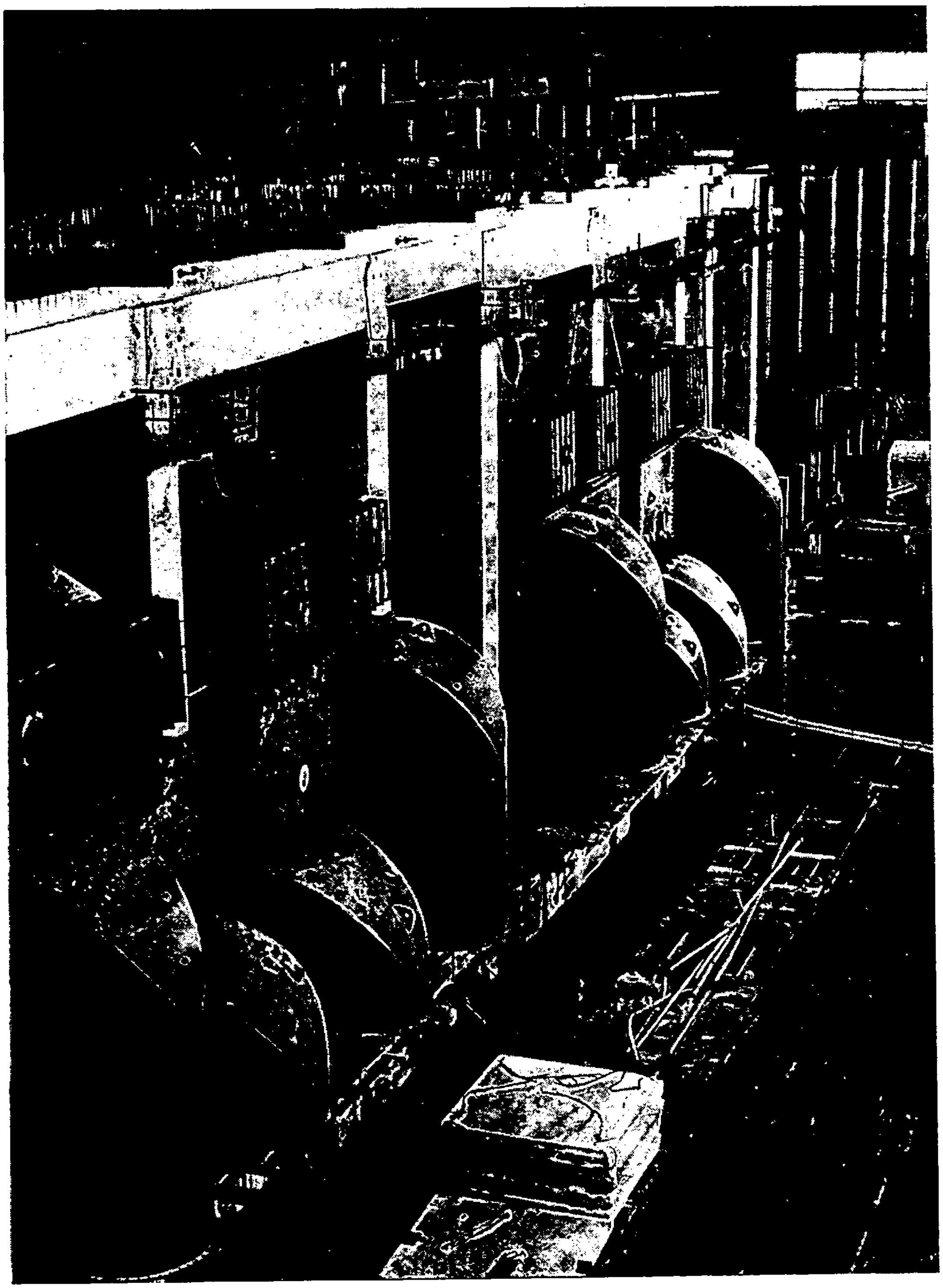

Fig. 2. In atomic buildings of the USSR. Chernobyl'sk AES. Machine room. V. Bratchikov photo.

which underlies the development of atomic energy for the next 10-12 yr, has been developed. Already a number of AES with VVÉR-1000 reactors have been constructed: Kalinin, South Ukraine, the fifth unit of the Novovoronezh and the third unit of the West Ukraine AES. Design documentation for other AES is being prepared.

The industrial mastery of AES with uranium–graphite channel reactors began more than ten years ago, exactly as for AES with VVÉR. The first industrial AES of this kind in the USSR is the Beloyarsk AES. Its reactors are a further perfection of the first AES reactor in the world. The affirmative experience of utilizing the Beloyarsk AES (BAES) permitted making the deduction that channel reactors with

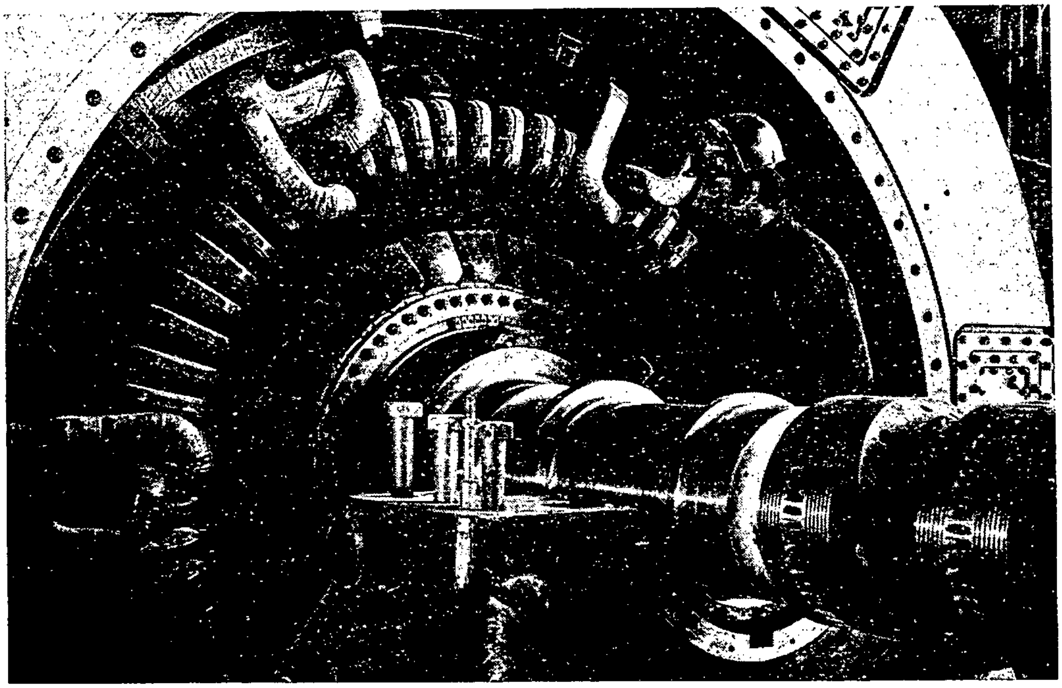

Fig. 3. In atomic buildings of the USSR. Armenian AES. Assembly of the turbo-generator. V. Bratchikov photo.

graphite moderators enclosed by boiling water and steam, can be used in AES of especially high unitary power.

A uranium–graphite channel reactor RBMK-1000 of $1 \cdot 10^6$-kW unitary power has been developed, on whose basis the two-module AES has been designed. Two turbines of $500 \cdot 10^3$-kW power are set up with one reactor in each unit. The important peculiarity and definite advantage of the RBMK-1000 is the possibility of an "on-line" fuel overload without stopping the reactor, which permits raising the utilization factor of the rated power. The V. I. Lenin Leningrad Atomic Electric Station of $2 \cdot 10^6$-kW power, produced according to this design and commissioned in 1973-1975, is the largest AES in the USSR and one of the largest in the world. At this time, mastery of the design power of the second unit goes forward and work on expanding this AES to more than $2 \cdot 10^6$ kW has simultaneously been started.

The Kursk, Chernobyl'sk, and Smolensk AEC with RBMK-1000 reactors have been produced on the basis of the LAES design, and their construction has begun. At this time, the erection of the starting complex of the first unit has practically been completed and the starting-adjusting work and the preparation to power startup proceed. Construction and assembly of the first unit of the Chernobyl'sk AES proceeds at full speed. The experience with the LAES is used extensively in erecting these AES, improvements are introduced which will result in raising the reliability and improving the utilization conditions. The experience with LAES operation showed the possibility of a substantial increase in RBMK power and the production of power units with 1500- and 2000-MW power for such reactors. The research in progress in this area already permits making a deduction about the economic expediency and prospects of RBMK reactors.

In conclusion, it should be recalled that more than $20 \cdot 10^9$ kWh has been developed by AES in the USSR in just 1975. This permitted a saving of about $7 \cdot 10^6$ tons of provisional fuel.

EFFECT OF NEUTRONS REFLECTED FROM THE WALLS OF A ROOM ON PULSE PARAMETERS IN FAST REACTORS

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UDC 621.039.514

The pulse parameters in fast bare reactors depend on the properties of the materials surrounding the core. In particular the pulse characteristics for a given initial reactivity or energy release are different for a reactor located in a building or outside it in free space. In the first case it depends also on the size of the room [1, 2]. This difference in pulse parameters is due to neutrons reflected from the walls of the room.

We present the results of calculating pulses in certain types of fast reactors, taking account of the walls of the room. The reflected neutrons [1] are considered as ten fictitious groups of delayed neutrons.

We use the following equations for the dynamics of a fast pulsed reactor:

$$\frac{dn}{dt}=\frac{k(1-\beta)-1}{\tau}n+\sum_{i=1}^{11}\lambda_i C_i;$$
$$\frac{dC_i}{dt}=\frac{k\beta_i}{\tau}n-\lambda_i C_i;\quad i=1,2,\ldots,11; \tag{1}$$
$$k(t)=k_0-\sum_{j=1}^{3}\alpha_j v_j(t);\quad \beta=\sum_{i=1}^{11}\beta_i;\quad \frac{1}{\omega^2 j}\frac{d^2v_j}{dt^2}+v_j=\int_0^t n\,dt;\quad j=1,2,3,$$

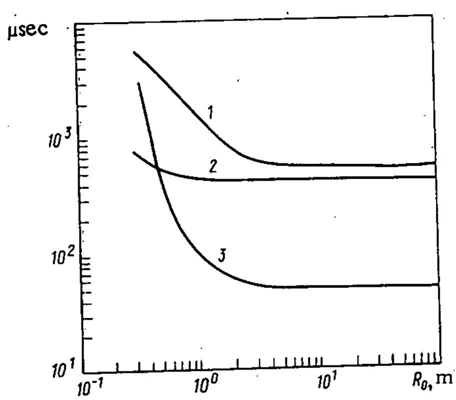

Fig. 1

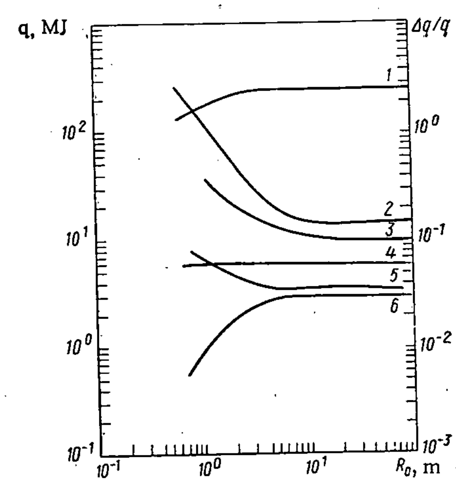

Fig. 2

Fig. 1. Pulse width as a function of radius of room: 1, 2, 3) reactors 1, 2, 3, respectively.

Fig. 2. Dependence of q and Δq/q on R_0: 1, 3) reactor 3; 4, 5) reactor 2; 6, 2) reactor 1.

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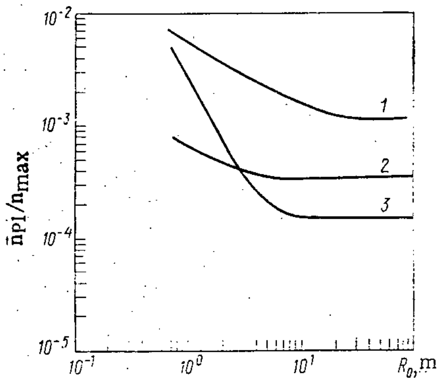

Fig. 3. Dependence of $\bar{n}_{pl}/n_{max}$ on R_0: 1, 2, 3) reactors 1, 2, 3, respectively.

TABLE 1. Reactor Parameters

Characteristic	Reactor		
	1	2	3
τ, sec	$1.1\cdot10^{-8}$	$0.5\cdot10^{-6}$	$2.1\cdot10^{-7}$
ω_1, rad/sec	$0.306\cdot10^{5}$	$3.19\cdot10^{4}$	$1.27\cdot10^{4}$
ω_2, rad/sec	$1.46\cdot10^{5}$	$4.50\cdot10^{4}$	$0.63\cdot10^{4}$
ω_3, rad/sec	$2.84\cdot10^{5}$	$6.00\cdot10^{4}$	$0.44\cdot10^{4}$
α_1,1 MJ	$0.906\cdot10^{-3}$	$0.80\cdot10^{-3}$	$0.117\cdot10^{-5}$
α_2,1 MJ	$0.020\cdot10^{-3}$	$0.50\cdot10^{-3}$	$0.551\cdot10^{-5}$
α_3,1 MJ	$0.004\cdot10^{-3}$	$0.15\cdot10^{-3}$	$0.332\cdot10^{-5}$
r_0, cm	10.9	10.9	40.0

where n(t) is the power; v_j, ω_j, α_j and are, respectively, the relative displacement, the frequency, and the relative reactivity coefficient pertaining to the j-th harmonic of the oscillations; k is the neutron multiplication factor taking account of 11 groups of delayed neutrons. The rest of the notation is standard. Constants with a subscript i = 1-10 refer to neutrons reflected from the walls, and with i = 11 to ordinary delayed neutrons reduced to one effective group.

To determine λ_i and β_i (i = 1, 2, ..., 10) the reflected neutrons, whose spectrum was supposed known, were divided into ten energy groups. It was assumed also that the reactor and the room are spherically symmetric.

$$\frac{1}{\lambda_i} = \tau_1 + \frac{\tau_2^{(i)} + \tau_3^{(i)}}{1-\alpha}; \tag{2}$$

$$\tau_1 \simeq \frac{R_0 - r_0}{v_0} \simeq \frac{R_0}{v_0}, \quad \tau_3^{(i)} \simeq \frac{R_0}{v_i}, \tag{3}$$

where τ_1 is the time of flight of neutrons from the reactor to the walls; $\tau_2^{(i)}$ is the time for neutrons to slow down to the energy of the i-th group in the wall material; $\tau_3^{(i)}$ is the time of flight of neutrons of the i-th group from the walls to the reactor; r_0 and R_0 are, respectively, the radii of the reactor and the room; α is the albedo of the walls; v_0 and v_i are, respectively, the velocities of reactor neutrons and neutrons of the i-th group.

$$\beta_i = \gamma_1\gamma_2\gamma_3^{(i)}\gamma_4^{(i)} \quad \text{for} \quad i = 1, 2, \ldots, 10; \tag{4}$$

$$\beta_{refl} = \sum_{i=1}^{10} \beta_i = \gamma_1\gamma_2 \sum_{i=1}^{10} \gamma_3^{(i)}\gamma_4^{(i)}, \tag{5}$$

where γ_1 is the fraction of fission neutrons leaving the reactor; γ_2 is the fraction of neutrons returning to the reactor as a result of reflection from the walls; $\gamma_3^{(i)}$ is the fraction of the neutrons in the i-th group $(\sum_{i=1}^{10} \gamma_i = 1)$; $\gamma_4^{(i)}$ is the relative importance of the reflected neutrons of the i-th group.

We assume that $r_0 \ll R_0$ and that the neutrons are radiated by the walls with equal probability in all directions within a solid angle of π steradians. In this case the equations take the form

$$\gamma_1 = 1 - \frac{\Sigma_a}{\nu\Sigma_f}, \quad \gamma_2 \approx \frac{\alpha}{2(1-\alpha)}\left(\frac{r_0}{R_0}\right)^2. \tag{6}$$

The factors $\gamma_4^{(i)}$ are determined by going to secondary neutrons appearing in the reactor as a result of the interaction of reflected neutrons with core material, and using perturbation theory:

$$\gamma_4^{(i)} \approx \frac{4\beta_1^{(i)}}{r_0^2} \frac{\int_0^{r_0} r^2\varphi(r)\,dr}{\int_0^{r_0} r^2\varphi^2(r)\,dr} \int_0^{r_0} r^2 B^{(i)}(r)\,\varphi(r)\,dr; \tag{7}$$

$$B^{(i)}(r)=\frac{1}{r}\int_0^r ch\alpha_1^{(i)}x\exp\left(-\alpha_1^{(i)}\sqrt{r_0^2-r^2+x^2}\right)dx,$$

where $\varphi(r)$ is the neutron flux in the critical reactor; $\alpha_1^{(i)}$ and $\beta_1^{(i)}$ are the Peierls core parameters for neutrons of the i-th group. In most cases it is possible to use simpler expressions following from Eqs. (7)

$$B^{(i)}(r)\approx\frac{1}{4\alpha_1^{(i)}}\delta(r-r_0);$$

$$\gamma_4^{(i)}=\frac{\beta_1^{(i)}}{\alpha_1^{(i)}}\frac{\varphi(r_0)\int_0^{r_0}r^2\varphi(r)\,dr}{\int_0^{r_0}r^2\varphi^2(r)\,dr}. \qquad (8)$$

Calculations were performed for three idealized reactors representing basic types of fast pulsed reactors. Reactor 1 is close to the HPRR, FBR [3, 4], and the SPR-II [5] which are characterized by small size and short prompt neutron lifetimes; reactor 2 corresponds to systems of the same composition but with a prolonged prompt neutron lifetime; reactor 3 corresponds to systems of the Super Kukla type [6] distinguished by large size and high dilution of fuel by inert materials (Table 1). The albedo of the walls was taken equal to 0.35.

The results of the calculations are shown in Figs. 1-3 which indicate that as R_0 decreases, the pulses in reactors 1 and 3 for constant initial reactivity become broader, the maximum value of the power and its ratio to the average power in the plateau decrease, and the relative fraction of energy release in the plateau increases. The changes in T and q become important for $R_0 < 2$ m, $\Delta q/q$ for $R_0 < 5$ and 7 m, n_{pl}/n_{max} for $R_0 < 5$ and 13 m for reactors 1 and 3, respectively. Neutrons reflected from the walls have practically no effect on the pulse characteristics in reactor 2.

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SLIPPING CONDITIONS IN THE PROBLEM OF THE MINIMUM CRITICAL MASS

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UDC 621.039.51

The problem of the minimum critical mass of a reactor is formulated in the following way: To find the fuel distribution in a reactor of finite dimensions such as to make the reactor critical while the total amount of fuel is as small as possible; the moderator is distributed uniformly in the rector and its properties do not depend on the fuel concentration; the fuel concentration is limited. Let us consider the one-dimensional model of a thermal reactor in cylindrical geometry. If we assume the validity of the two-group approximation

$$\begin{aligned} &\frac{1}{r}\frac{d}{dr} rD_1(u)\frac{d\Phi_1}{dr} - \Sigma_{ad}(u)\Phi_1 + \nu\Sigma_f(u)\Phi_2 = 0; \\ &\frac{1}{r}\frac{d}{dr} rD_2(u)\frac{d\Phi_2}{dr} - \Sigma_a(u)\Phi_2 + \Sigma^{1-2}(u)\Phi_1 = 0, \end{aligned} \quad (1)$$

with boundary conditions

$$\frac{d\Phi_1(0)}{dr} = \frac{d\Phi_2(0)}{dr} = 0; \quad \Phi_1(R) = \Phi_2(R) = 0,$$

where R is the external extrapolated boundary of the reactor and u(r) is the fuel concentration at the point r, the mathematical formulation of the problem of minimum critical mass will have the following form.

Problem 1. To find a distribution u(r) yielding a minimum of the functional

$$I = \int_0^R f_0(u, r)\,dr = \int_0^R u(r)\,r dr \quad (2)$$

subject to the limitations

$$\left.\begin{aligned} \frac{dx_1}{dr} &= f_1(\mathbf{x}, u, r) = -\Sigma_{ad}(u)x_2 + \nu\Sigma_f(u)x_4 - \frac{1}{r}x_1; \\ \frac{dx_2}{dr} &= f_2(\mathbf{x}, u, r) = -\frac{1}{D_1(u)}x_1; \\ \frac{dx_3}{dr} &= f_3(\mathbf{x}, u, r) = -\Sigma_a(u)x_4 + \Sigma^{1-2}(u)x_2 - \frac{1}{r}x_3; \\ \frac{dx_4}{dr} &= f_4(\mathbf{x}, u, r) = -\frac{1}{D_2(u)}x_3; \end{aligned}\right\} \quad (3)$$

$$x_1(0) = x_3(0) = 0; \quad x_2(R) = x_4(R) = 0; \quad (4)$$

$$u_{\min} \leqslant u \leqslant u_{\max}. \quad (5)$$

According to certain authors [1-3], if the cross sections depend linearly on concentration the optimum arrangement may consist of a finite number of zones of the following types: $u(r) = u_{min}$, $u(r) = u_{max}$, and a special distribution zone. Let us investigate what kind of solution may exist for problem 1 on assuming that the cross sections in Eq. (3) depend on the distribution in a nonlinear manner. One of the necessary conditions for the existence of an optimum distribution in a class of measurable functions is the convexity of the set

$$V(\mathbf{x}, r) = \{f_0(u, r), f_1(\mathbf{x}, u, r), \ldots f_4(\mathbf{x}, u, r) / u \in [u_{\min}, u_{\max}]\},$$

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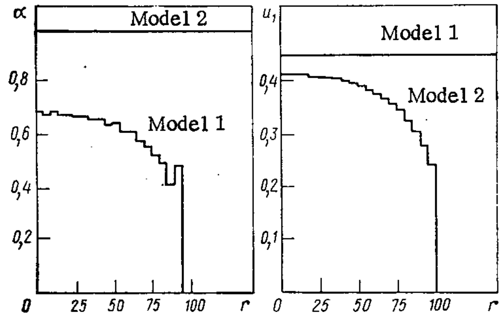

Fig. 1. Optimum values of the weighting function $\alpha(r)$ and the basic distribution $u_1(r)$. The basic distribution $u_2(r) = 0$.

which is formed in an expanded velocity space when the distribution takes all possible values within the range $[u_{min}, u_{max}]$ for fixed x and r [4].

The set $V(\mathbf{x}, r)$ is convex in the case of a linear dependence of the cross sections on u, and in general is nonconvex for a nonlinear relationship. Thus for a nonlinear dependence of the cross sections on the distribution no optimum distribution can exist in the class of measurable functions. However, for the majority of problems, and in particular the problem of the minimum critical mass, there is always a sequence of permissible distributions such that the corresponding sequence of trajectories of Eq. (3) reduces to a certain limiting curve satisfying the boundary conditions, while the value of the functional (2) for which the minimum is being sought tends toward its lower boundary. If the limiting curve is not a solution of Eq. (3), it is said that the minimum of the functional is attained in a "slipping condition," while the limiting curve is called the zero proximity function of the slipping condition [5]. In the slipping condition the concentration at a certain point may be pictured as jumping between the so-called basic concentrations at an infinite frequency.

Let us consider the auxiliary problem [5] in which the expanded set of velocities is the convex closure of the set of velocities of problem 1.

Problem 2. To find distributions $u_k(r)$, $\alpha_k(r)$, $k = 1, 2, \ldots, N$ giving a minimum of the functional

$$I = \int_0^R \sum_{k=1}^N \alpha_k f_0(u_k, r)\, dr \tag{6}$$

subject to the limitations

$$\frac{dy_i}{dr} = \sum_{k=1}^N \alpha_k f_i(\mathbf{y}, u_k, r); \quad i = 1, 2, 3, 4; \tag{7}$$

$$y_1(0) = y_3(0) = 0; \quad y_2(R) = y_4(R) = 0; \tag{8}$$

$$u_{min} \leqslant u_k \leqslant u_{max}; \quad 0 \leqslant \alpha_k \leqslant 1; \quad k = 1, 2, \ldots, N; \tag{9}$$

$$\sum_{k=1}^N \alpha_k = 1; \quad N \leqslant 5. \tag{10}$$

The convexity of the expanded set of velocities in problem 2 implies the existence of optimum distributions $u_k(r)$ and $\alpha_k(r)$, $k = 1, 2, \ldots, N$ in the class of measurable functions. Except for a number of degenerate cases, we may almost always obtain exhaustive information regarding the solution of problem 1 by solving problem 2. Thus the optimum phase trajectory $\mathbf{y}(r)$ of problem 2 coincides with the limiting curve of problem 1, while the minimum value of the functional (6) is the exact lower boundary for the values of the functional (2). If in a certain segment $[r_1, r_2]$ in which $0 \leq r_1 < r_2 \leq R$, $\alpha_j(r) \equiv 1$, the corresponding $u_j(r)$ will be the optimum distribution for problem 1 in the segment $[r_1, r_2]$; otherwise $u_k(r)$, $\alpha_k(r)$ are respectively the basic distributions and weighting functions of the slipping condition in the segment in question (this subject is treated in greater detail in [5]).

Let us apply the foregoing method to the solution of a methodical problem regarding the minimum critical mass of a homogeneous thermal reactor based on natural uranium. We shall assume that the probability of avoiding resonance capture at the point r_1 with a uranium concentration $u(r_1)$ may be approximately calculated from the equations representing the probability of avoiding resonance capture in an infinite medium with a uranium concentration $u(r_1)$. Let $D_1 = 1$; $D_2 = 0.853$; $\Sigma_{ad} = 0.01$; $\Sigma_a(u) = (0.077 + 7.46u) \cdot 10^{-3}$; $\varphi(u) = \exp(-0.245\, u^{0.58})$; $R = 140$. We shall study two models: 1) $\nu\Sigma_f(u) = 0.0099u$; $\Sigma^{1-2}(u) = 0.01\,\varphi(u)$; 2) $\nu\Sigma_f(u) = 0.0099u\,\varphi(u)$; $\Sigma^{1-2} = 0.01$. For the selected dependence of the cross sections on the distribution it is sufficient to have $N = 2$ in order to ensure convexity of the expanded set of velocities in problem 2. Allowing for Eq. (10), we shall consider that the basic distributions $u_1(r)$ and $u_2(r)$ are reflected by the weighting functions $\alpha(r)$ and $1-\alpha(r)$. The solution of problem 2 is then conveniently carried out numerically, using the gradient method.

Figure 1 shows the solution for $u_{min} = 0$, $u_{max} = 0.45$. In this case the slipping condition appears in model 1. If we take the limitation $u_{min} = 0.45$, $u_{max} = 0.8$, the slipping condition appears in model 2 and not in model 1. This example indicates that in reactor optimization problems nonlinear with respect to the concentration distributions we should normally expect slipping conditions to develop.

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EFFECTIVE HALF-LIFE OF ^{252}Cf

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UDC 539.163.1:546.799.8

The effective half-life of ^{252}Cf was measured with a fast fission chamber C containing $4.1 \cdot 10^{-3}$ μg of californium, and a stilbene neutron detector D (Fig. 1). The arrangement differs from that described in [1] and [2] in that in addition to the blocking of starting pulses from recorded fissions in the chamber there is also blocking of neutron pulses correlated with fission pulses (LG) which were recorded but did not trigger the analyzer (the blocking time is equal to the time scale of the analyzer T). In other respects the relation between the electron blocks is the same as in the operation of the block diagram described in detail in [3]. Figure 2 shows the experimental time distribution of the neutrons. At a distance τ from the beginning of the scale, where τ is the delay time of the pulses from the neutron detector, a peak (counts per channel) is observed due to the recording of pulses from neutrons corresponding to those fissions whose pulses triggered the analyzer.

The number of counts in the peak is

$$N_{\text{peak}}=\frac{\varepsilon_f N_n}{(1+N_f T)}\, t_{\text{meas}} \tag{1}$$

where ε_f is the efficiency of counting fragments in the fission chamber; N_f is the rate of counting pulses from recorded fissions ($N_f = \varepsilon_f Q$); T is the time scale of the analyzer TA (256 μsec); N_n is the rate of counting pulses from recorded neutrons, sec^{-1}; t_{meas} is the time of measurement, sec.

The background to the left and right of the peak (N_1) was caused by the recording of fission neutron pulses which were not recorded in the chamber.

$$N_1=\frac{Q\varepsilon_f(1-\varepsilon_f)}{(1+TN_f)}\, N_n \Delta t\, t_{\text{meas}} \tag{2}$$

where Q is the absolute fission rate in the chamber in sec^{-1}, and Δt is the width of the time analyzer channel (1 μsec).

We find from Eqs. (1) and (2)

$$Q=N_f+\frac{N_1}{N_{\text{peak}}\Delta t}. \tag{3}$$

If the efficiency of the fission chamber is close to unity [1, 2] Q can be measured very accurately. It is clear from Eq. (3) that the method described for measuring the half-life eliminates the necessity of calibrating the efficiency of the fission chamber [4] or the efficiency of the neutron detector [5, 6] and eliminates the systematic error related to them. The fast fission chamber with the fast discriminator used in these measurements had an efficiency of 0.95–0.96 for counting fragments, and the accuracy of an individual measurement of Q was 0.03–0.05% Since the chamber was completely sealed and the distance between electrodes and the size of the active spot are very much less than the size of the electrodes there was no possibility of the escape of californium nuclei from the working volume

TABLE 1. Measured Values of Half-Life

$\tau_{1/2}$, yr	δT, %	Reference
2,646±0,004	0,15	[4]
2,631±0,006	0,23	[7]
2,621±0,006	0,23	[8]
2,659±0,010	0,36	[9]
2,638±0,007	0,27	[5]
2,628±0,010	0,38	[6]
2,637±0,005	0,19	Our data

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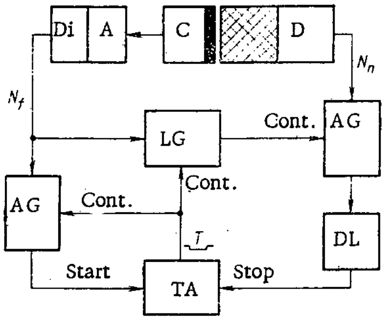

Fig. 1. Block diagram of experiment: A) fast current amplifier; Di) discriminator; LG, AG) linear grating circuits in allowed and forbidden regimes (linear gates and antigates respectively); DL) delay line (passive); TA) time analyzer; Cont., Start, Stop) inputs of control, start, and stop signals.

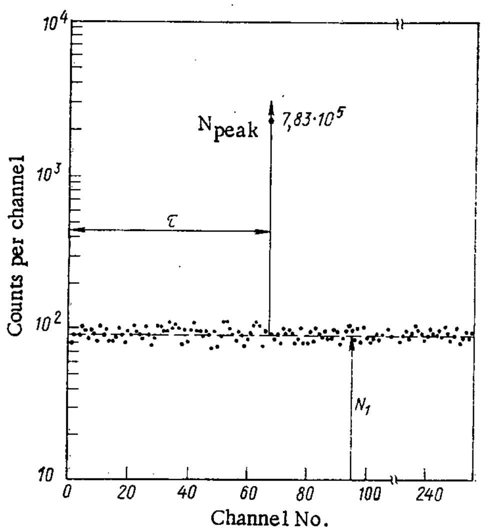

Fig. 2. Time distribution of neutrons.

during the measurements. The high accuracy of measuring the californium fission rate permitted measurements of the effective half-life to be completed in about six months. In determining the absolute number of ^{252}Cf fissions the presence of two other isotopes ^{250}Cf and ^{254}Cf which emit neutrons in spontaneous fission was taken into account. In the five-year period between the chemical purification of the californium and the beginning of the measurements practically all the ^{254}Cf decayed, and the ratio (Δ) of the ^{250}Cf fissions to the ^{252}Cf fissions was 1.28%. It was shown in [5] that the average number of neutrons per fission is the same for ^{250}Cf and ^{252}Cf, and therefore in measuring the absolute number of ^{252}Cf fissions it is necessary to correct only the change in Δ due to the change in isotopic composition. This correction amounted to 0.16% during the whole cycle of measurements. The value obtained for the effective half-life of ^{252}Cf is 2.637 ± 0.005 yr. Table 1 shows the values of $T_{1/2}$ obtained in other experiments.

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MEASUREMENT OF THE EFFECTIVE CROSS SECTION FOR THE FISSION OF ^{252}Cf BY FAST REACTOR NEUTRONS

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UDC 621.039.51

The high radioactivity of isotopes of transplutonium elements creates substantial difficulties in nuclear-physics research. By using dielectric track detectors the fission characteristics of alpha- and beta-active isotopes with very short half-lives can be studied in intense neutron and gamma fields. However, spontaneous fission gives rise to a nonremovable background in work with any kind of detector. In order to reduce this background to a minimum in fission cross section measurements it is expedient to use an intense pulsed neutron source and a system ensuring the recording of fragments only during the neutron pulse.

To measure the cross section for the fission of ^{252}Cf by neutrons, a fast-pulsed reactor apparatus was constructed in which a glass plate detector was displaced by springs relative to the fixed layers of fissionable material at the instant of the neutron pulse. This method leads to a significant increase in effect over background of spontaneous fissions on a certain part of the detector plate. Let D be the diameter of the projection of the layer of fissionable material on the surface of the detector, and v the velocity of the detector at the instant of the pulse. If the width of the neutron pulse $\Delta t \ll D/v$, the ratio of the number of tracks from spontaneous and induced fissions on a section is

$$n_{sp}/n = \frac{0.693D}{\sigma_f T_{sf} \Phi_n v},$$

where Φ_m is the neutron fluence per pulse, σ_f is the effective fission cross section, and T_{sf} is the period of spontaneous fission. In our case $D \approx 0.5$ cm, $v \approx 100$ m/sec, $\overline{\sigma}_f \approx 10^{-24}$ cm^2, $T_{sf} = 58.2$ yr, and $\Phi_m \approx 10^{13}$ neutrons/cm^2. In this case $n_{sp}/n \approx 10\%$, which is quite acceptable for measuring the fission cross section. The sample of californium used in the measurements had the following isotopic composition: 79.8% ^{252}Cf, 3.8% ^{251}Cf, 10.9% ^{250}Cf, and 5.5% ^{249}Cf. The relative error in determining the $^{249-251}$Cf isotopic contents was no more than 3%.

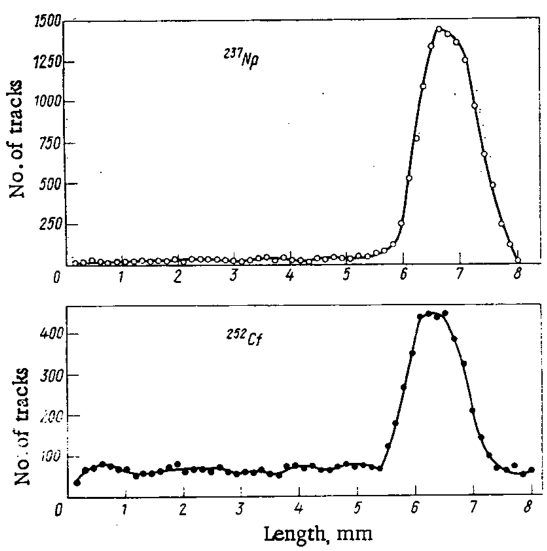

Fig. 1. Distribution of tracks along the length of the detecting plate.

A layer of ^{237}Np serving as a standard was irradiated simultaneously with the layer of californium. The distribution of tracks over the length of the detecting plate obtained in one of a series of measurements is shown in Fig. 1. On the curve for ^{237}Np the tail to the left of the peak is due to delayed reactor neutrons. The pedestal under the peak of induced californium fissions is due to spontaneous fissions and delayed neutrons. The reactor neutron spectrum in our measurements was close to the fission spectrum, somewhat softened as a result of inelastic scattering.

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Data from [1] were used in calculating the effective cross section for the fission of ^{237}Np by neutrons from a pulsed reactor. In processing the results of the measurements it was assumed that the effective fission cross sections of ^{249}Cf and ^{251}Cf are 2.01 ± 0.13 b [2] and that the ^{250}Cf and ^{252}Cf cross sections are equal. The measured value of the effective cross section for the fission of ^{252}Cf by fast reactor neutrons is

$$\bar{\sigma}_f = 1.58 \pm 0.14 \text{ b}.$$

Within the limits of experimental error this value agrees with data in [3] obtained by using neutrons from a nuclear explosion. To compare the results the curve for $\bar{\sigma}_f(E_n)$ from [3] was averaged over the fast reactor neutron spectrum.

In comparison with lighter even–even nuclei studied so far ^{252}Cf has the largest cross section in the first plateau ($\sigma_{f0} \approx 2.5$ b [3]). The ratio of the widths of the compound nucleus ^{253}Cf for the above threshold region ($\langle \Gamma_n/\Gamma_f \rangle \approx 0.3$) differs appreciably from present data on semiempirical systematics [4, 5] based on the properties of ligher nuclei.

The measurements again confirm the considerable influence of the neutron shell (N = 152) on the fission characteristics of heavy nuclei. An appreciable lowering of the fission barriers and a corresponding increase in fissility was noted for the compound nuclei ^{249}Cm (N = 153) and ^{250}Cm (N = 154) [6, 7]. Evidently a similar effect, i.e., a lowering of the fission barriers in passing through the N = 152 shell, occurs for californium isotopes also.

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UNUSUAL MINERAL ASSOCIATIONS IN THE OXIDATION ZONE OF SULFIDE-FREE URANIUM DEPOSITS

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UDC 553.068.41

Anomalous associations of minerals in the oxidation zone often give rise to mistakes in the determination of the type of oxidation zone involved and, consequently, in the composition of the primary ores. The study of such associations is of interest to scientists working on problems involving the laws governing the development of oxidation zones and to members of prospecting parties.

The close association of hydrous oxides and silicates of uranium with hydrous oxides of iron in the oxidation zones of sulfide-free uranium deposits is a rare phenomenon. The extensive development of hydrous oxides of iron is more characteristic of the oxidation zones of sulfide–uranium deposits; they are found most often in association with uranites, and much more seldom with redeposited silicates. The results obtained by processing specimens of pitchblende–carbonate deposits collected in the process of geological mapping of an area form the basis for the present communication [1].

This deposit was formed in a subvolcanic intrusion of felsite-porphyries, which lies within the limits of the foredeep of the last stage of development of a geosyncline zone. The ores of the deposit were formed after the youngest magmatic rocks of the region – dikes of diabasic and dioritic porphyrites – and therefore have no genetic connection with the felsite-porphyries. Apparently the ores are related to them in having a common magmatic focus. The ore bodies are confined to zones of steep meridional discontinuous disturbances, which are the basic structures used for monitoring ores.

Ores in which there is a sharp predominance of carbonates of different ages are distinguished by a relatively simple composition.

Four stages of mineral formation were found in the deposit.

The arsenopyritic stage is relatively less marked; it is formed by the diffuse dissemination of arsenopyrite. Occasionally one may observe streaky accumulations of arsenopyrite with a small amount of quartz and sericite, intersected by carbonate and carbonate–pitchblende streaks.

The ankeritic stage is widely developed. It is formed by veins of ankerite which is yellowish-gray or, less often, pink. The thickness of the veins may reach tens of centimeters, and the length varies from a few meters to 40-50 m. In addition to the ankerite, the veins contain small quantities of quartz, fluorite, and pyrite. The ankerite veins are cut by calcite and calcite–pitchblende streaks.

The calcite–pitchblende stage is also fairly widespread. The predominant mineral is calcite. The calcite and the pitchblende were deposited from solutions, repeatedly alternating and replacing each other. Later, sericite, growing on the pitchblende in the form of festooned fringes and filling the central part of the streaks, detached itself from the pitchblende.

The quartz–carbonate stage within the intrusion of felsite-porphyries is less marked than the preceding two stages. The veins are not very thick and have clear-cut intersections with the calcite–pitchblende streaks. There is replacement of the calcite by dolomite, and less often by quartz.

Thus, close association with calcite and practically complete absence of sulfides are typical of pitchblende. All of this was reflected in the oxidation zone, which is of the hydrated oxide–silicate type [2]. Along the pitchblende there are borders of lead–calcium hydrous oxides of uranium (of the wolsendorfite type), alternating with silicates, chiefly uranophane, in whose composition a notable role is played by

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magnesium. As a result, there are characteristic "humite fringes" of a yellow-orange color. In close association with them we find argillocarbonate accumulations which are the interior zones of metasomatic changes near the ore which are produced by oxidation processes. In addition to such typical changes of the primary ores, we also observe on the site a slight degree of efflux and redeposition of uranium in the form of secondary uranium minerals at distances of up to 10 m. Here we find greenish-yellow incrustations, which are a mixture of alternating thinner layers of uranium silicates of the ursilite type, calcite, an argillaceous mineral of the allophane type, and an almost nonluminescent calcium–magnesium carbonate of uranium, of the rabbitite type.

Composition and Structure of the Incrustations. The interstratification of uranium and nonuranium minerals (calcite, ursilite, allophane, and rabbitite), the formation of "diluted" uranium minerals (not uranophane but ursilite, not schwartzite but rabbitite) indicates the alkaline or weakly alkaline nature of the solutions and the relatively small amounts of extracted uranium, i.e., indicates conditions most characteristic of the formation of oxidation zones of almost sulfide-free uranium deposits.

However, in a number of outcrops we find completely different mineral associations. Thin hydrous oxide–silicate streaks are connected with accumulations or incrustations of loose hydrous oxides of iron, tens of times as thick as the hydrous oxide–silicate streaks. Sometimes the hydrous oxides of iron are arranged along uranium minerals, enveloping them with a thin limonite covering reminiscent of an iron hat.

As can be seen from the brief characterization of the primary composition of the ores, in the deposit there is an especially extensive development of ankeritic veins of the second stage of mineralization, and the iron content of the ankerite is 7%. It is known that iron-bearing carbonates are transformed very intensively in the oxidation zone. In [3] it is shown that carbonate minerals including iron are completely unstable in the presence of free oxygen. The product of their oxidation under the ordinary water–air conditions is limonite. Sideritic, ankeritic, and other ores in the oxidation zone form actual iron hats which are sometimes difficult to distinguish from hats which originated from sulfide material.

From a comparison of the surface outcrops of the above-described deposit with deep-lying samples, it is found that anomalous mineral associations of the oxidation zone appear in those segments in which the pitchblende–calcite streaks intersect the ankeritic veins of the second stage. If the intersection is at an angle close to a right angle, the anomalous intervals are very short. If the intersection is at a small angle, the segment with anomalous mineral associations, with small interruptions, can be traced over a length of several meters.

The correct determination of the type of oxidation zone encountered in such cases may be of considerable help in selecting the most rational method of exploring a deposit. In uranosulfide deposits with a limonite type of oxidation zone, industrial ores are, as a rule, concentrated below the groundwater level. In sulfide-free uranium deposits, even with anomalous limonite outcrops, industrial ore bodies may be found above this level as well.

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EFFECTIVE GAMMA-RAY ATTENUATION COEFFICIENTS FOR RADIOACTIVE ORES

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UDC 550.835

The effective γ-ray mass attenuation coefficient μ/ρ (ρ is the density of ore or rock) is a basic parameter in the calculation of γ fields recorded with radiometric equipment [1]. Values of μ/ρ have been studied sufficiently extensively only for the integral γ spectra from uranium and thorium ores [2]. The development of spectrometric studies of natural materials made it necessary to establish the values of μ/ρ for portions of the differential γ spectrum used for radiometric measurements of U(Ra), Th, and K concentrations in rocks and ores.

We experimentally determined values of μ/ρ (see Table 1) for the standard energy windows of a scintillation γ spectrometer [3] by comparison of the saturation counting rate curves N/N_∞ observed for a 2π geometry with curves calculated from the well-known formula [1]

$$\frac{N}{N_\infty} = 1 - E_2(x),$$

where $E_2(x) = e^{-x} + xE_i(x) = e^{-x} - x \int_x^\infty e^{-t}t^{-1}dt$ is the tabulated exponential integral function (King function) of argument $x = \mu/\rho\,(H\rho)$; N_∞ is the counting rate above an infinite half-space; H is the thickness of a radiating horizontal layer which is saturated along its course.

Measurements were made on models of uranium, thorium, and potassium ores $100 \times 200 \times 200$ g/cm^2 in size with an elongated CsI(Na) scintillator [4] 30 mm in diameter and 70 mm long in an iron shield 2.5 mm thick. The relative error in measurement of counting rate was no more than 1%.

Experimental saturation curves are shown in Fig. 1 for the energy intervals 1.05-1.35 and 2.05-2.65 MeV. The curves for the other intervals given in Table 1 are located close to them. For radiation from U(Ra) and Th, saturation practically begins at a layer thickness of 75-80 g/cm^2.

TABLE 1. Effective γ-Ray Mass Attenuation Coefficients in Rocks and Radioactive Ores (Z = 13-15) for Various Emitters

γ-spectrum energy interval, MeV	μ/ρ, cm^2/g		
	U (Ra)	Th	K
1.05—1.35	0.034	0.032	—
1.00—1.60	0.034	0.032	—
1.35—1.55	0.034	0.032	0.050
1.65—1.85	0.036	0.033	—
2.05—2.65	0.035	0.036	—
2.40—2.80	—	0.037	—

Note the steeper saturation curve and the higher value of μ/ρ in the energy interval 2.05-2.65 MeV in comparison with interval 1.05-1.35 MeV. The reason for this is that the effective coefficients reflects the attenuation of the direct and scattered radiation in the emitting medium and in the scintillator [5]. In the region of γ lines with maximum energy, the effective coefficient μ/ρ is close to the total attenuation coefficient μ_0/ρ for the primary radiation. For the middle regions of the spectrum, the value of μ/ρ is reduced because of the effect of the scattered radiation associated with γ rays of higher energies.

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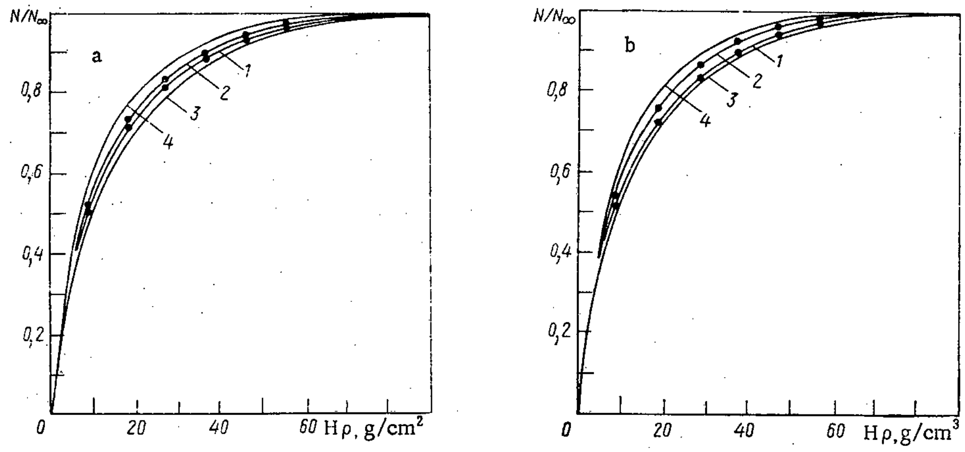

Fig. 1. γ-ray saturation curves for uranium (a) and thorium (b) ores: 1, 2) experimental curves for E = 1.05-1.35 and 2.05-2.65 MeV; 3, 4) theoretical curves for μ/ρ = 0.030 and 0.040 cm³/g, respectively.

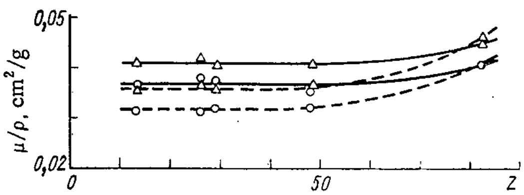

Fig. 2. Dependence of μ/ρ on the Z of the absorbing medium obtained experimentally in a broad-beam geometry for γ radiation from Ra (——) and Th (- - - -) in the spectral intervals 1.00-1.60 (○) and 2.05-2.65 (△) MeV.

The dependence of the effective coefficients μ/ρ on the effective atomic number Z_{eff} of the medium was studied in broad-beam geometry with point sources of Ra and Th. Sheets of Al, Fe, Cu, Cd, and Pb from 10 to 40 g/cm² thick were used as absorption filters. Counting-rate measurements were made in the γ-spectrum regions 1.00-1.60 and 2.05-2.65 MeV with a relative statistical error of no more than 1%. The effective coefficients μ/ρ were calculated from the exponential law for radiation attenuation. The values of μ/ρ (Fig. 2) remain practically constant over the range of Z from 13 to 50. In the range of Z from 51 to 82, the coefficient μ/ρ increases by 10 and 25% respectively for Ra and Th radiation. As in a radiating layer, the coefficient μ/ρ for the energy interval 2.05-2.65 MeV is approximately 10% higher than that for the interval 1.00-1.60 MeV.

A comparison of the experimental values of μ/ρ for emitting (see Table 1) and absorbing (see Fig. 2) media show satisfactory agreement of the results for the Th γ spectrum and somewhat poorer agreement for Ra, particularly in the energy interval 2.05-2.60 MeV where the counting rate is low and of the order of background.

The concentrations of U and Th in radioactive ores rarely exceed 1.0-1.5%. Calculations show that an increase in the concentration of U or Th to 1.5% in an ore base with a Z_{eff} from 13 to 27 leads to an increase of only 3% in μ/ρ for the energy intervals in the γ spectra studied. Therefore one can consider that the value of μ/ρ remains constant over the entire actual range of Z_{eff} for rocks and radioactive ores with an accuracy sufficient for spectrometric determinations of U(Ra), Th, and K. For rough calculations of γ fields at energies of 1-3 MeV, one can use the average value $\bar{\mu}/\rho$, which is close to μ/ρ for the integral γ spectrum from U(Ra) or Th when the radiation is detected with gas-filled counters or scintillation detectors with an energy threshold in the neighborhood of 0.2 MeV [2, 6].

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Yu. A. Gulin

THE GAMMA – GAMMA METHOD OF INVESTIGATING OIL WELLS*

Reviewed by E. M. Filippov

Density gamma–gamma logging, along with the neutron methods neutron–neutron and neutron–gamma logging, is widely used in oil-field geophysics. The book being reviewed is the first book on oil-field geophysics.

It consists of a foreword, nine chapters, conclusions, and a list of references.

The book contains descriptions of different methods of calibrating instruments, determining thicknesses of beds, and finding densities of the rocks in the beds. Many actual calculations and experimental results are cited. These refer to the relation between recorded counting rates and different parameters of the instruments and conditions of measurement in the wells.

A number of observations may be made concerning the book. The author uses a word difficult to pronounce, "plotnostnomer" (density meter), in place of plotnomer (densimeter) for an instrument that measures rock density. For the modification of the gamma–gamma method introduced by Yu. A. Gulin into the literature of nuclear geophysics, the designation gamma–gamma logging was introduced without regard to the energy of the source employed. The method, based on the recording of radiation passing through the investigated medium, has been named the absorption method or the method based on absorption and it is designated the gamma method (gamma method based on primary radiation). This method is not a variety of the gamma–gamma method for the conditions of measurement investigated by the author.

Editorial errors also appear in the book. Thus, Fig. 5 shows a threshold spectrum which the author calls integral. For an integral spectrum the curves should tend not toward zero with increase in energy but toward the integral asymptote. The author does not explain how to distinguish the coefficients of radiation attenuation for nonmineralized water in columns 3 and 5 of Table 23. Chapters 3 and 4 should be combined into a single chapter, since the material in them is all closely interrelated. This would permit avoidance of some repetition. In Chap. 5 the results from interpretation of the gamma–gamma data are not discussed. It would therefore be more suitable to term the chapter "Method of determining rock density."

In Chap. 6, devoted to geologic interpretation of the results, some practical examples should be given of interpreting logs with the aid of the given nomograms, i.e., examples like those in Chap. 7.

The author gives little attention to description of the two-sonde method with nonazimuthal collimation of radiation (instruments without spring clamps). Instruments manufactured on this principle have an advantage over instruments with spring-clamp arrangement when wells of small diameter (no greater than 190–220 mm) are being investigated. For wells of different diameters it would be possible to manufacture an instrument with centering devices and with adapters of various diameters in like manner to the instrument designed by Yu. A. Gulin and others for exploration of boreholes for coal.

The book would gain considerably if the author would consider (such as from foreign results) the possibility of using computers, of providing automatic data processing and logs of the directly studied parameter (density or porosity). The remarks presented are not fundamental.

*Nedra, Moscow (1975).

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On the whole the book leaves a good impression. It furnishes varied and interesting material, obtained through the author's efforts with mathematical modeling and experiments. The book will be useful for practical workers and investigators in the appropriate specialties.

19th CONFERENCE OF THE CMEA PERMANENT COMMITTEE ON ATOMIC ENERGY USE

Yu. I. Chikul

The 19th Conference of the Permanent Commission of the Council on Mutual Economic Assistance convened in Moscow during November 1975. Delegations from Bulgaria, Hungary, the GDR, the Republic of Cuba, Poland, Romania, the USSR, and Czechoslovakia took part in the work of the Commission. Representatives of the international economic combines "Interatominstrument" and "Interatomenergo" and the Joint Institute for Nuclear Research (JINR) were present at the conference.

The Commission discussed measures in the area of atomic power following from the resolutions of the CMEA Session taken at the 19th Conference, and of the Organizing Committee of the Council; reports on the results of developing thermal and fast high-power energy reactors and proposals for further cooperation and acceleration of the introduction of these reactors were examined; a report of the Temporary International Scientific-Research Collective for carrying out reactor physics investigations was heard, the importance of its research in the area of VVÉR-type power reactors was noted, and it was acknowledged expedient to extend the lifetime of the agreement about the creation of this collective to 1980.

The Commission analyzed the state of research on the development of multilateral international specialization and cooperation in the production of isotopes, the application of radio-isotope methods and radiation technique in various branches of the national economy of the member nations of CMEA. Questions associated with raising the efficiency of operation of "Interatominstrument" and a further development of cooperation with the International Atomic Energy Agency (IAEA) were discussed.

The research plan of the Commission for 1976-1977 and the research plan in the area of standardization in 1976 announced. A number of other questions were also discussed. Appropriate recommendations and resolutions were taken on all the questions considered.

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WORK OF THE COORDINATING SCIENTIFIC – TECHNICAL COUNCIL ON REPROCESSING IRRADIATED FUEL OF AES

V. I. Zemlyanukhin

The coordinating scientific–technical council (CSTC) on reprocessing AES irradiated fuel was created in 1971 as a working organ of the CMEA Permanent Committee on Atomic Energy Use (PCAEU). The problems of this CSTC are the coordination of efforts of interested CMEA member nations in conducting research in the area of reprocessing spent fuel elements of AES with thermal (VVÉR type) and fast reactors and the creation of scientific–technical cooperation on the problem "Investigations on Reprocessing the Fuel Elements of Atomic Electric Stations." Many CMEA member nations are interested in solving equations of reprocessing spent AES fuel elements. Specialists from Bulgaria, Hungary, the GDR, Poland, Romania, the Soviet Union, and Czechoslovakia took an active part in the work of the Council. The representative of the CSTC is the vice-director of the Institute of Nuclear Research of Czechoslovakia, M. Podesht.

In 1971-1975 the program of cooperation between the CMEA member nations on the problem under consideration provided for conducting research on five topics: preparation of the fuel elements for reprocessing, hydrous methods of reprocessing irradiated fuel elements of VVÉR-type reactors, and the possibility of using them to reprocess fast-reactor fuel elements; fluoride methods of reprocessing spent fuel elements; development of methods of checking and controlling technological processes; technicoeconomic questions of the reprocessing of spent fuel elements; safety of the surrounding medium. Questions of the transportation of spent nuclear fuel occupied a significant place in the work of the Council. The experience with transporting spent AES fuel elements in Rhinesburg (GDR) was studied and proposals to simplify organization of the transport were prepared. Questions of the construction and unification of railroad transport facilities for the transfer of fuel elements of the VVÉR-440 reactors were coordinated. In connection with the fact that data about the fissionable material in the fuel elements are needed in preparing the spent fuel element assemblies for transportation to the radiochemical reprocessing factories, the Council worked out a proposal on the development of investigations to determine the content of such fissionable materials. In addition to using the design methods, it was recommended that nondestructive methods be developed, such as the determination of plutonium by a fluorescent method, the determination of ^{235}U and ^{239}Pu on the basis of recording instantaneous and lagging neutrons which are formed after additional irradiation by fast neutrons or gamma quanta; the determination of the degree of burn-up on the basis of gamma-spectrometry.

The Council prepared "Technical Conditions. Assembly of Fuel Elements (Spent) of Atomic Electric Stations with VVÉR-440 Reactors" for consideration by the CMEA PCAEU. These technical conditions have been recommended for use by the CMEA member nations in delivering the spent nuclear fuel to the factory for reprocessing. Worked out jointly with the Secretariat Branch of the CMEA were "Rules for the Safe Transportation of Spent Nuclear Fuel from the AES of CMEA Member Nations by Railroad." At present, the "Rules" are under consideration by the CMEA PCAEU. The development of an analogous document regulating the transportation of spent fuel by waterway and hybrid waterway–railroad transport was noted.

Investigations in the area of hydrous and fluoride methods of reprocessing irradiated fuel elements were conducted with a view to perfecting individual technological components and equipment. The results of these developments are used in technico-economic computations and in design work. In particular, work on perfecting extraction schemes for VVÉR-440 fuel elements by using a heavy diluent, an investigation on the influence of different kinds of diluents on the radiation chemical stability of tributyl phosphate, the

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selection of flocculents for clarification of solutions of VVÉR fuel elements, the creation of a model of an endless screw dissolver, the development of pulsating, turbine, and centrifugal extractors and miniature dosers for radioactive solutions should be noted. The fluoride gas technology of reprocessing fuel elements of fast reactors has been verified in a large scale laboratory set-up (in an example of fuel elements of a BOR-60 reactor), cut-off valves and flowmeters intended to operate in fluoride gas media as well as the technology for fabricating sorbents used to cleanse the fluoride have been developed.

The "Scheme to Check the Extraction Reprocessing of Spent VVÉR-Type Fuel Elements" for refinement of the problems on working out methods to check and control the technological nuclear fuel regeneration processes has been prepared by specialists and has been agreed upon.

The Council agreed to list standard specimens of substances intended to verify the methods and calibration of analytical checking instruments. The opinion has been expressed that it is expedient to direct the main efforts in the area of perfecting methods of analysis in the next five-year plan (1976-1980) towards the creation of automated complexes for the analysis of highly active products. To do this it is necessary to develop instrumental analysis methods (absorption, x-ray fluorescent, spectrophotometric, thermochemical, acoustic, and neutron). The problem of assuring the possibility of their use as sensors for atomic power plant technical regulations is posed in developing the instruments. Work to produce effective atomic power plant technical regulations has started. It is directed towards the creation and refinement of mathematical models of the processes used in technological fuel element reprocessing schemes.

The data obtained about the economics of fuel reprocessing were used to predict the development of atomic power in the CMEA member nations.

The Council has organized information about the history and results of investigations included in the 1971-1975 program of cooperation. Progress reports, examined in the Council, appeared annually at all stages of the research. Expanded reports appeared about the terminated stages, published in the journal "Nuclear Energy" (Czechoslovak SSR). Moreover, at the CSTC conferences (held twice a year) and at the meetings of specialists, the state of the research on the individual questions was discussed. The Symposium "Investigations in the Area of Irradiated Fuel Reprocessing," held in 1974 in Czechoslovakia, was of great value. Participating in it were 136 representatives from Poland, Hungary, the GDR, Romania, the Soviet Union, Czechoslovakia, Yugoslavia, colleagues from the CMEA Secretariat, and also a representative of the International Atomic Energy Agency (IAEA). There were 93 reports on hydrous and anhydrous fuel-element reprocessing processes and on analytical and economic questions. The tendencies towards the development of investigations on spent fuel reprocessing all over the world, the state of individual developments in the CMEA member nations were analyzed, and paths of further research on all aspects of the problem under consideration were determined.

The Council prepared proposals on cooperation in the area of scientific and technical investigations conducted by the CMEA member nations on the problem of "Investigations on Reprocessing Fuel Elements of Atomic Electric Stations" in 1976-1980. Five topics were included in the cooperative plan for this period: transportation of spent fuel elements and nondestructive methods of determining their content of fission material; discovery and preparation of spent fuel elements for reprocessing; perfection of the technology and creation of the equipment for regeneration of spent fuel elements of VVÉR type; development of methods and instruments to check and control the technological spent fuel element regeneration processes; development of the technology and creation of the test equipment for spent fuel element regeneration of spent fuel elements of VVÉR type; development of methods and instruments to check and control the technological spent fuel element regeneration processes; development of the technology and creation of the test equipment for spent fuel element regeneration for fast reactors. Working plans on all the themes of the problem for 1976-1980 were agreed upon at the 8th Congress of the CSTC (September 1975).

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RESULTS OF THE WORK OF THE COORDINATED SCIENTIFIC–TECHNICAL COUNCIL ON RADIATION TECHNIQUES AND TECHNOLOGY (KNTS-RT)

A. K. Zille

In the member countries of the Council for Mutual Economic Aid (CMEA), a broad front of work is being undertaken in the field of radiation techniques and technology, which is at quite a high scientific-technical level. Representatives from eight countries participate in the work of KNTS-RT: Bulgaria, Hungary, German Democratic Republic (GDR), Republic of Cuba (since October 1975), Poland, Romania, USSR, and Czechoslovakia. In 1971-1975, the Council conducted a large operation for creating the prerequisites for an increase in the scientific–technical level of exploitation and the most rapid introduction of radiation processes and equipment in the national economy of the interested countries. During this period 10 Conferences, seven meetings of groups of experts, and a scientific-technical conference have been held. At the 1st conference (April 1971), the Council worked out a program of cooperation in the field of radiation techniques and technology up to 1975.

The work of the KNTS-RT in achieving the individual sections of the program will be considered below.

1. In the field of organization of the introduction into the national economy of the member countries of CMEA in the future prospects for radiation processes, the Council discussed measures for the commercial achievement of processes for the radiation cross-linking of polymers and the radiation vulcanization of rubbers. Having discussed the report on "The Organization on Production Scales of the Radiation Processing of Food and Agricultural Products" (3rd conference, April 1972, CzSSR), the Council acknowledged this method of overcoming losses of foodstuff raw materials, which was promising and had a number of advantages over the well-known methods. In order to develop measures for the introduction of radiation processing of food and agricultural products in the national economy of the interested countries within the scope of the KNTS-RT, a special group of experts was formed. Work in this direction was carried out jointly with specialists of the Permanent Commission of CMEA on the food industry. It was noted, according to the report on "The Commercial Achievement of Processes for the Radiation Production of Woodpulp–Plastic Materials" (4th conference, October, 1972, Budapest), that the use of wood-pulp–plastic materials obtained by the radiation route is promising in the shipbuilding and chemical industries, civil construction, etc. Having heard and discussed the report on "The Commercial Achievement of Processes for the Sterilization of Medical Goods" (5th conference, April 1973, GDR), the Council noted that radiation sterilization at the present time is one of the most developed and widely used radiation-technological processes. Measures were considered and approved for the introduction of radiation sterilization processes in industry for the interested member countries of the CMEA.

2. The Council coordinated the principal trends and the structure of the scientific forecasting of the main trends of development of radiation techniques and technology up to 1990 and has recognized it as necessary, on the basis for its exploitation, to set up an inventory of completed and carried-out applied scientific-research and experimental-structural projects in the member countries of the CMEA.

3. A unified terminology in the field of radiation techniques and technology (basic terms) was worked out within the scope of the Council (3rd conference, April 1972, CzSSR).

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4. The Council considered and endorsed the report on "The Technical-Economical Principles of the Industrial Application of High-Powered Sources of Ionizing Radiations" (2nd conference, November 1971, Moscow), on the basis of which was developed "A Procedure for Determining the Economic Efficiency of Radiation Techniques" (5th Conference, April 1973, GDR), and endorsed at the 24th Session of the Commission (July 1975, CzSSR). The procedure was recommended for use in the member countries of the CMEA.

5. "Tentative Technical Requirements for Research Multipurpose and Specialized Facilities for Radiation Processes" (3rd conference, April 1972, CzSSR) and also "Unified Health Regulations for the Installation and Operation of High-powered Radioisotope Gamma-Facilities" (9th conference, April 1975, CzSSR) were worked out and approved, and were recommended by the Commission for use in all the member countries of the CMEA taking into account the national legislation of the countries.

6. In the field of the joint conduct and coordination of applied projects, a plan of cooperation was considered for the establishment of a Tentative International Association to investigate the radiation sterilization of medical products (VMK-RS), attached to a branch of the Scientific-Research Textile Institute in CzSSR (7th conference, April 1974, Lodz, Polish Peoples' Republic), and a plan for a program of work of this body (10th conference, October 1975, Minsk, USSR). Work on the establishment of the VMK-RS will be undertaken in 1976.

7. In effecting measures for the Standardization and Unification of Radiation Facilities, the Council considered the question of the development of unified units of radiation facilities (3rd conference, April 1972, CzSSR). Further work is planned in 1976-1980 on the unification of units of radiation facilites and the development of plans for standard radiation facilities.

8. In working-out proposals for the specialization and cooperation of production, the Council discussed provisional reasons for organizing the output of radiation-modified polymer and rubber-technical production (8th conference, September 1974, Tashkent) by means of specialization of production of this type of product in the individual member-countries of the CMEA. A technical proposal on this question and also "A Technical Proposal for the Organization, for Member Countries of the CMEA, of the Commercial Production of Radiation- Modified Wood-Pulp Goods," prepared by the Secretariat Division of the CMEA (9th conference, April 1975, CzSSR) were discussed. The urgency of the proposal was stressed, concerning the creation of an international undertaking for the commercial manufacture of such goods. The Council considered and discussed a "Technical Proposal for the Organization of Production of Radiation-Sterilized Materials and Goods for Medical Purposes" and acknowledged it to be expedient to organize all-round collaboration of the member countries of the CMEA in this direction by preparing the appropriate Agreement between the interested countries.

9. The Conferences and Symposia were molds for experience and scientific-technical information. A significant role in this was played by a conference, organized on the initiative of the Council, on the problems of the introduction of high-powered radiation facilities and radiation technology (October 2-4, 1972, Budapest). In accordance with the plan, the project comprised a symposium on the radiation preparation of food and agricultural products (October 15-17, 1973, Sofia) which created great interest for specialists–dieticians, physicians–hygienists, and exploiters of radiation facilities, and also a Symposium on the problems of radiation graft-polymerization (September 2-4, 1974, Tashkent, USSR).

Radiation sterilization occupies an important place in the medical industry of many countries. This method is effective, and sometimes also the only possible method, for ensuring the sterility of a number of medicinal preparations and transplants. This was demonstrated well at the symposium on the scientific problems of radiation sterilization of medicinal substances (April 7-9, 1975, CzSSR), which is an important stage on the path for the future development of work in this field.

10. Taking into account the necessity for improving the training of specialists, the Council approved the proposal of the Romanian delegation, concerning the organization at the Center in the training and specialization of Romanian personnel,, for a course of lectures and practical work.

Thus, the program contemplated in 1971-1975 is essentially acoomplished.

The measures underlying the achievement within the scope of the KNTS-RT in the next few years consist in the draft of a program and a plan of work of the Council in 1976-1980. The principal problems which it is proposed to solve in the forthcoming period are:

1. To work out a forecast for the development of radiation technique and technology.
2. To develop unified standard-procedural documents in the field of safety of operation of radiation facilities, technological dosimetry, radiation sterilization, and the treatment of food products.
3. To work out proposals for the industrial introduction of the most promising radiation processes and methods (in accordance with provisional data of the Soviet delegation on forecasting the development of radiation techniques and technology).

In the program of cooperation between the member countries of the CMEA in the field of radiation techniques and technology, the sequences and periods of achievement are defined for all the work which must be carried out in the next five years.

JOURNAL OF COLLABORATION

The 8th Conference of the Coordinated Scientific-Technical Council (KNTS) on the Reprocessing of Nuclear Power Station Fuel and the meeting of specialists on the Regeneration of Spent Fuel Elements from Fast Reactors took place on September 23-26, 1975 in Dresden (GDR). At the meeting, the reports of specialists from GDR, Poland, the Soviet Union, and Czechoslovakia were considered, concerning the results of investigations from the development of a technology and the construction of an experimental plant for the regeneration of spent fuel elements from fast reactors. These investigations provide for a program of collaboration between member countries of CMEA concerning the problem considered in 1971-1975. The work carried out was related with both aqueous and fluoride technology for reprocessing fast reactor fuel elements. Specialists of the USSR presented a summary report, in which the directions are defined for carrying out investigations in the next period, and which take into account the essential special features of spent fuel elements from fast reactors – deep burnup, high buildup of fission products and the relatively short time of cooling. The meeting of specialists and the KNTS endorsed a list of the most important routes for investigating the development of a technology for the regeneration of fuel elements, a work plan for 1976-1980 on the theme, "Development of a Technology and the Construction of an Experimental Plant for the Regeneration of Spent Fast Neutron Reactor Fuel Elements" and technical assignments on the stages of the work, which will be carried out in the countries in accordance with the work plan.

The Council ratified the work plans for 1976-1980 on all themes of the problem "Investigations on the Reprocessing of Nuclear Power Station Fuel Elements." During the discussion of the results of the Conference, the Council acknowledged it to be advantageous to organize, within the scope of the KNTS, a comparison of the results of the determination of the components in solutions of spent nuclear fuel from water-cooled/water-moderated power reactors (VVÉR). Data on the assignments completed in 1971-1974 on the problem and on the introduction into practice of their results are worthy of merit.

The 21st Conference of the Working Group on Reactor Science and Technology and Nuclear Power Generation was held in Burgas (Bulgaria) on September 23-26, 1975. The draft of the work plan of the Permanent Commission in 1976-1977 in the field of reactor science and technology and nuclear power generation was discussed and ratified. A report was heard and discussed, on the results of work on the theme "The Investigation, Development, and Improvement of Water-cooled/Water-moderated Reactors, in particular, the construction of a Reactor Facility of the VVÉR type with an Electrical Output of 1000 MW" and proposals for the future collaboration and acceleration of the introduction of thermal power reactors of high capacity in the period after 1980. A report was discussed and agreed on "The Results of Work undertaken in the member countries of CMEA on the theme Control and Management with Nuclear Reactors and Plant of Nuclear Power Stations" in the period 1971-1975. The participants in the conference ratified the agenda for the carrying out and notification for the second conference of specialists of the member countries of CMEA on the problems of shielding of nuclear power station reactors. The proposals of the delegations on the theme "Determination of the Optimum Strategies of Nuclear Power Generation within the Scope of CMEA" were discussed. A number of problems, following from the resolutions of the 29th Session of CMEA and of the working units of CMEA were also considered, and coordinated solutions were accepted.

The 8th Conference of the KNTS on Fast Reactors took place on September 29 to October 3, 1975 in Dresden (GDR). The results of work were given on the achievement of the Program of scientific-technical collaboration in the field of fast reactors in 1974-1975. Reports were heard and discussed concerning the work accomplished in sequence with that of the work plan for accomplishing the program of collaboration on the problem "Investigations in the Field of Fast Neutron Power Reactors." A plan of work for the Council in 1976-1977 was discussed and approved. Proposals for a conference of specialsts on the various

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aspects of safety of "sodium–water" steam generators were considered. The Soviet delegation gave information on the operation of the reactor power plants BOR-60 and BN-350. A report on the results of work in achieving the program of collaboration on the problem being considered, and proposals for future collaboration and a speed-up of the introduction of fast power reactors of high capacity in the period after 1980, was discussed and approved. Proposals were considered for future cooperation in the manufacture of plants for nuclear power stations with high capacity fast reactors. Data on the introduction into practice of the results of completed investigations, carried out in 1971-1974, were discussed. The Council considered a number of problems resulting from the resolutions of the 29th Session of the CMEA and the working units of the CMEA, and accepted coordinated solutions.

A CONFERENCE ON THE PROBLEMS OF THE DESIGN, ASSEMBLY, STARTING, AND OPERATION OF ATOMIC ELECTRIC POWER PLANTS

Yu. I. Mityaev

The conference was held in Kiev from Oct. 2 through Oct. 4, 1975. It was organized by the Ministry of Power Engineering and Electrification of the Ukrainian SSR, the Ukrainian Republic Board of the Scientific Technical Society of Power Engineering and Electrotechnical Industry, and the Academy of Sciences of the Ukrainian SSR. About 330 specialists from 63 organizations and enterprises of the country took part in the work of the conference. The aim of the conference was to better acquaint its participants with domestic nuclear-power-engineering practice, to establish and strengthen contacts among leading scientific-research, construction, and design institutes, with assembly and maintenance organizations, and with atomic electric power stations APS. In opening the conference, the Minister of Power Engineering and Electrification of the Ukrainian SSR, A. N. Makurin, briefly characterized the progress of power engineering in the Ukraine under the Ninth Five-Year Plan, described the Chernobylak, West Ukrainian, and South Ukrainian APS which are under construction in the Ukraine and which are projected under the 10th Five-Year Plan to have an installed power of about $6 \cdot 10^6$ kW, and formulated the most important scientific and technical problems of immediate concern in the development of nuclear power engineering. The participants in the conference were greeted by Academician of the Academy of Sciences of the Ukrainian SSR V. I. Tolubinskii and by the president of the Scientific Technical Society of Power Engineering and Electrotechnical Industry of the Ukrainian SSR, G. A. Klimenko.

The lectures which were heard in the first two days touched on the most varied problems of nuclear power engineering. In a lecture entitled "Perspectives for the Development of Nuclear Power Engineering in the USSR," E. P. Karelin reported on the main technical and economic characteristics of domestic APS using water-moderated/water-cooled power reactors (WWR) and channel-type power reactors (CPR), which, as is well-known, constitute the basis of nuclear power engineering in the USSR in the next few decades. The state and main directions for further development and improvement of reactor energy channels (of the CPR type) and casings (WWR type) were discussed in lectures by Yu. M. Bulkin and V. A. Sidorenko. At the beginning of his address, Yu. M. Bulkin announced that on the eve of the opening of the conference the State Commission had accepted into commercial operation a second CPR at the V. I. Lenin Leningrad APS. This power station thus became the most powerful APS in the USSR and Europe. Both speakers indicated, in particular, the means of increasing the unit output of the reactors. Yu. M. Bulkin described a section-block reactor with an electrical power output of 2.0 GW using nuclear superheated steam (CPRS), and V. A. Sidorenko spoke about certain improvements made in the WWR-1000 reactor.

V. E. Doroshchuk devoted his remarks to the problems involved with scientific investigations in the field of nuclear power engineering. He noted the need for a more extensive use of the experiences in traditional thermal-power engineering, and discussed some problems which have become more pressing with the introduction of APS, in particular, plant input and running control, APS instrumentation, organizing and providing for maintenance operations, providing for safety and security at the APS in emergency situations, etc. The ever increasing need for mobie APS was noted.

There were also lectures given which reviewed the work going on in several areas, viz.: APS design (by V. P. Goncharenko), construction details of the Kharkov turbine plant and the problems involved in

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increasing their unit power (by Yu. F. Kosyak), the experience of manufacturing heat exchange and water-preparation equipment for APS (by V. A. Minos'yan), and safety problems at APS (by Yu. V. Shvyryaev). Questions of safety were also dealt with in the lectures by I. K. Nikitin entitled "The Interaction of APS with the Surrounding Medium." Reactor safety problems were also discussed by V. P. Vasilevskii, who studied several breakdowns of the CPR reactor with flows and fractures of piping 50-300 mm in diameter, and described facilities and systems to ensure the safety of APS in the event of such failures.

B. Ya. Prushinskii reported some interesting factual data concerning engineering and economic indices for the period 1974-1975 of APS from the USSR Ministry of Power Systems. He presented figures for the yearly average power, production and distribution of electrical energy, the consumption of electrical energy on characteristic necessities, efficiency, coefficient of utilization of installed capacity, net cost of electrical energy, as well as an analysis of the planned and forced shutdowns for the Novovoronezh, Beloyarsk, Kol'sk, and Bilibinsk APS. The relatively high annual average coefficients of utilization of installed capacity of APS, which in 1974 amounted to 0.71, should be noted. By way of comparison, it should be pointed out that the average coefficient for all of the APS in the USA in the period 1973-1974 was 0.56-0.57 [see Atomwirtschaft, Nos. 7/8, 316 (1975)]. More detailed results of the operation of APS were given in separate talks by L. I. Vitkovskii (on the Novovoronezh APS), by R. Kh. Gabitov (on the Beloyarsk APS), and by B. A. Trifimov (on the Kol'sk APS). K. K. Polushkin reported to the conference on the results of starting up, effecting the design parameters, and the first results from operating the leading CPR reactor, which was put into operation in December, 1973 at the V. I. Lenin Leningrad APS. V. P. Akinfiev gave a progress report on the construction, assembly, and the starting preparations of the Chernobylsk APS. This APS will probably be started up in 1976.

Several reports were devoted to questions of control during the assembly and operation of APS (V. F. Zlepko, Yu. A. Yakobson, and G. L. Levin), and also to questions of manufacturing quality and equipment maintenance (V. A. Moiseitsev, Yu. K. Byvshev, and V. A. Maslenok). É. P. Kazakova and V. A. Beletskii summarized and analyzed factual data on the hydrochemical regime APS of the USSR Ministry of Power Systems during the period 1974-1975, and discussed the experience and problems of starting and maintaining APS.

Problems involving the physics and shielding of power reactors were discussed in lectures by A. S. Dukhovenskii ("Neutron-physical Characteristics of the WWR-1000 Reactor and Problems of Nuclear Safety") and V. N. Mironov ("Initial Tests of the Efficiency of Biological Shielding of APS using WWR"). Interesting reports were given by V. A. Chevychelov on the distribution of APS in the Ukraine and their connection to the power system, and by G. A. Kopchinskii on the main directions of development of foreign nuclear power.

All in all, there were about 35 lectures and reports delivered at the conference; the theses of the majority of these were printed by the beginning of the conference. In particular, one of the decisions notes the advisability of organizing in Kiev in 1976 a Republic interdepartmental conference on the problems of optimizing systems of engineering water supply for APS, reducing thermal pollution, and the utilization of low-grade heat.

It should be noted that the conference was able to function successfully thanks to its having been well-organized, even though there was no regular plan followed in the order of the lectures presented.

CONFERENCE ON THE TECHNICAL APPLICATIONS OF SUPERCONDUCTIVITY

A. G. Plesch

A conference on the technical applications of superconductivity was organized and carried out by the USSR State Committee for the Use of Atomic Energy and the Ministry of Electrotechnical Industry from Sept. 16 through Sept.19, 1975 in Alushta. It excited the lively interest of Soviet and foreign specialists. It was participated in by the representatives of more than 60 ministries, departments, scientific-research organizations, industrial enterprises of the USSR, and also by the representatives of 11 foreign organizations.

The subject matter of the conference included an extremely wide variety of questions involving the so-called high-current superconductivity. The problems discussed included, in particular, problems involved in creating specific superconducting devices in the manufacture of electrical machinery, in railway transport, in accelerator techniques and experimental techniques of high energy physics, the magnetic systems of equipment used in thermonuclear studies, superconducting electric power transmission lines, general engineering and physical problems connected with developing superconducting magnetic systems, the problems involved in improving the superconducting materials put on the market by the industries of several countries (England, the USSR, the USA, France, Japan, and others), the perspectives for creating superconductors with industrial value, and problems involved with cryogenic maintenance of superconductors.

Superconducting Devices in Accelerator Technology and in High-Energy Physics. In this area there has been accumulated a considerable body of experience in operating superconducting magnetic systems operating at constant current and ranking with other experimental facilities, e.g., in using the bubble chambers at the Batavia (USA) and CERN accelerators, the magnetic particle analyzer at the "DESY" (FRG) accelerator, etc. The construction and parameters of these devices are widely known, but a sufficiently complete estimate of the results of their operation was made perhaps for the first time at the conference. Attention was called to the quite successful experience in operating the Fermilab (Batavia, USA) bubble chamber, which was put into operation in August, 1972, and which has since that time operated for over 10,000 h. The parameters of its superconducting magnet are as follows:

Operating current, kA	5
Current density in the winding, kA/cm^2	3.7
Internal diameter of the winding, m	4.27
Stored energy, MJ	396
Field at the center, T	3.01

The reliability of the superconducting magnet system turned out to be better than that of several other chamber systems. No emergency situations occurred during the whole period of operation. The experience in operating other large superconducting systems of this type can also be considered to be successful.

Another trend in the construction of permanent magnets is exemplified by the large permanent dipole and quadrupole magnets. The experience in building such devices in the USSR, the USA, and the FRG has been favorable. It is significant that, e.g., four such magnets used in experiments at Fermilab use 99% less electrical energy than magnets which perform in the usual way. The coolant (nitrogen and helium) supplies in their cryostats are such that continuous operation for a week without recharging is possible. Some of their magnet parameters are:

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Total mass, tons	65-165
Stored energy, kJ	300-20,000
Total field, T	1.8-2.0
Nitrogen loss, liters/day	18-30
Helium loss, liters/day	36-50

Recently, in the USA. USSR. and in the FRG considerable success has been attained in constructing pulsed dipole magnets for accelerator circuits. In particular, a dipole with field intensity 4.5 T constructed at the Karlsruhe Institute of Experimental Nuclear Physics has operated for more than 250 thousand cycles. No changes in the properties and characteristics were observed during this time, and the reproducibility of the characteristics was well within the tolerances of high energy physics. The reliability and capital investment of pulsed magnets are estimated to compare quite favorably with similar magnets of the usual sort of performance, and their operating cost is considerably less because of the reduction in electrical energy requirement by a factor of 10. As yet, unfortunately, there is not much experience in the practical operation of such magnets.

Considerable progress has been made in the development of superconducting high-frequency devices for use in ring accelerators, linear accelerators, and particle separators, and reports on these were given by Soviet and West German specialists.

We note from the projects discussed some proposals for modernizing the largest accelerators at Batavia and Serpukhov. In view of their scale, these proposals are remarkable; they call for increasing the energy of the accelerated protons to several thousand gigaelectronvolts – a necessary condition, evidently, for further progress in high energy nuclear physics. In practice, this will be possible to achieve only by using superconducting magnets employing pulsed and continuous current in accelerating circuits and beam extraction systems. The electrical energy requirement for the alternative approach is unreasonably large, and the geometrical dimensions of the accelerator ring are several times as large. Both Soviet and foreign scientists and development engineers consider it feasible to reduce the electrical energy required by these accelerators by several times by replacing their magnetic systems by superconducting systems, using already developed and tested technology.

Superconducting Magnetic Systems for Thermonuclear Investigations. A thought frequently expressed in the lectures of Soviet, American, and West German specialists was that the solution of the problem of controlled thermonuclear synthesis and the creation of a demonstration reactor with a magnetically contained plasma (of the Tokamak type) is only possible with superconducting-type magnetic systems. According to current estimates, the energy requirement for the magnetic systems of a reactor with the usual performance amount to tens and hundreds of gigaelectronvolts, and therefore a positive energy output of the reactor is hardly realizable. The power supply requirement for superconducting reactor magnetic systems is only hundreds or even tens of MeV. As a result, such studies are developing within the limits of the thermonuclear program at a rapid pace. In the FRG, for example, they are being carried out in accordance with a complex program, which includes not only the development of the superconducting systems themselves, but also the development of blanket materials and low-temperature construction (including superconducting and insulating) materials. Extensive cooperation between government laboratories and private firms both within the FRG as well as on a European scale is typical of these investigations. The Karlsruhe Institute of Experimental Nuclear Physics has emerged as the coordinator of the superconductor development.

As was indicated at the conference, the most advanced work in developing superconducting devices for thermonuclear facilities is evidently being done in our country. The relatively small LIN-5 apparatus for studying the storage and stability of a plasma in open systems has already been operating successfully for three years at the I. V. Kurchatov Institute of Atomic Energy. This has simplified the conditions for carrying out experiments, and made it possible to dispense with large sources of power for establishing the magnetic field. The basic figures for this facility are: magnetic induction at the center, 2.4 T; magnetic induction at the mirrors, 5 T; stored energy, 0.5 MJ. The next step in this direction was the creation of the larger TsMS-0.25 apparatus for studying ultrahigh-frequency heating of a plasma. The Institute of Atomic Energy gave this apparatus to the Institute of Physical Problems of the Academy of Sciences of the USSR, where it is also being used. This apparatus, in contrast to the above-cited devices of the immersion type, is cooled by liquid helium which is pumped through special channels in the magnet winding, and therefore is called a circulation magnetic system (CMS). In the opinion of the Soviet and a number of foreign scientists, circulation systems are technically more ideal, although the technology involved in their production is more difficult.

Particular interest was displayed in a unique superconducting circulation magnetic system for the T-7 type tokamak. The assembled parts of the system, which comprise 1/8 of a torus, were brought to the testing stage during the conference. Several of the main parameters of the T-7 are:

Large radius, cm	122
Inside radius of chamber, cm	35
Field at center, T	3
Plasma current, kA	500 (at q = 2.5)
Stored energy, MJ	20
Constructive current density, A/cm^2	4000
Mass, tons	12

Several foreign scientists expressed the opinion that it will be possible (after testing the T-7 and interpreting the results) to proceed to the planned superconducting magnetic systems for thermonuclear reactors, of which sketchy plans exist in a number of countries.

By the time of writing this article, the first cooling and current input had been carried out on the first "eighth" of the T-7, with satisfactory results.

Superconducting Materials. Industry in several countries, including the Soviet Union, has for a number of years manufactured superconducting materials in the main using backings of deformable niobium-titanium alloys. The conference showed that the variety, quality, and amount of production of the materials are to a sufficient extent satisfactory for the current requirements of the developers of large superconducting devices for high energy physics, thermonuclear investigations, and other areas of practical application. The existing materials make it possible to obtain magnetic fields of several tesla in large volumes (up to several cubic meters), and in laboratory devices fields of 8-9 T are attainable. The operating temperatures of these materials are 4-5°K. Nevertheless, much consideration was given at the conference to the problems of producing superconducting materials. This was done firstly because the area of practical application of superconductors is continuously expanding, resulting in the appearance of new requirements for the materials, and secondly because there exists a wide variety of superconducting compounds, mainly with lattices of the A-15 type, which have high values for their critical parameters (I_c, T_c, and H_c). A number of problems have to be solved in introducing such compounds. Finally, there are many possibilities for perfecting the technology of the superconducting materials (perhaps a better name for some of them would be products, since their production frequently demands minute accuracy) already in industrial production. In addition, there are theoretical and experimental grounds for expecting the appearance of commercially useful superconductors with critical temperatures appreciably higher than 20°K and critical fields of the order of several tens of teslas. In this connection, special attention was given by the participants to the report that Westinghouse has developed a superconducting film on a backing of Nb_3Ge by reducing hydrogen chlorides in a gaseous medium at 15°K and in a magnetic field of 7 T. The critical temperature of the film was 22.5°K. The specialists at this firm intend to develop the method during the next several years to the level of an industrial technology.

On the basis of the lectures given and the opinions of the participants, the immediate perspectives of the superconducting materials industry can on the whole be formulated as follows:

1. The outlook for the immediate future, possibly in 1976, is that industrial multiple-strand superconducting materials on Nb_3Sn or V_3Ga backings will make their appearance in a number of countries. This will make it possible to utilize magnetic fields in the region around 12 T, and the rather higher critical temperature of these materials will make it possible to increase the reliability of superconducting devices and to decrease the demands made on cooling systems.

2. The evolution of superconductor technology will lead to the development of new materials for higher fields. At the present time, the most favorable superconductors from this point of view are apparently NbGe, NbAl, NbAlGe, NbSi, and several others. Unfortunately, none of these are very attractive from the engineering standpoint.

3. The immediate outlook which is evident for superconductor technology is for an emphasis on materials with backings of niobium–titanium alloys.

Considerable discussion took place around the problems of applying superconductivity to the manufacture of electrical machinery and to electrical propulsion systems. Promising results have been obtained with operating superconducting (including full-scale models) in all of these areas, and plans have been developed for specific devices. Several of these are being achieved. For example, a joint Soviet–American project is under way involving the construction of a superconducting transmission line near Moscow with a length of the order of kilometers. Many who addressed the conference noted the necessity for greater international cooperation in introducing superconducting devices into contemporary technology. Concrete proposals were made, in particular in relation to the development of superconducting magnetic systems for thermonuclear devices.

The accumulated experience in constructing and operating superconducting devices makes it possible to immediately proceed to make effective use of them in industry, while the progress, for example, in accelerator technology and in the technology of thermonuclear experiments is most directly related to the introduction of superconducting devices. Unsolved physical, engineering, and technological problems which require the tireless endeavors of investigators must not block this introduction.

4TH INTERNATIONAL CONFERENCE ON THERMAL EMISSION ENERGY CONVERSION

A. I. Kulichenkov

The conference took place on September 1-3, 1975 at Eindhoven (The Netherlands). Fifty reports encompassing different scientific and engineering aspects of thermal emission conversion (TEC) were heard.

After the decrease in activity in the area of TEC caused by the abrupt cutback in research on nuclear thermal emission programs in the USA, France, and the Federal Republic of Germany, a noticeable revival has again recently been observed in this area, especially in the USA. Both the scale of the researches and their single-mindedness have changed significantly.

An essential transition from the area of space utilization to the area of more extensive terrestrial application occurred in estimating the sphere of possible TEC applications. This process is accompanied by either a total rejection of the program of space TEC application (West Germany) or by a revision of the time to accomplish it by a 10-15-yr shift (USA). In this connection, new reactorless schemes of TEC application have been investigated. Among them are low-power plants as well as large-scale industrial electrical stations using chemical fuel or solar energy. It is hence proposed to use the thermal emission converters as either the main electricity generating modules or as a high-temperature superstructure for the classical steam turbine cycle. The practical realizability of these schemes is specified by the possibility of a radical improvement in the TEC output characteristics for a substantial reduction in the working temperatures. Preparation of the technological base for the development of perfected TEC, responding to the requirements of the new spheres of application, is the main purpose of the researches conducted in the USA, France, and West Germany within the framework of the national thermal emission programs.

The USA presented a long-range dual-purpose program for the development of research on thermal emission conversion financed by NASA and ERDA. The long-range purposes of this program are the creation of a nuclear power plant of forced parameters for a space system with electric rocket thrust and up to a 50% and more rise in the efficiency of modern thermal stations because of the use of high-temperature thermal emission superstructures. The main tasks of the first stage of the program are demonstration of the possibility of achieving an up to 30% conversion efficiency at a 1400°K emitter temperature and the discovery of technological facilities requiring development in the later stages. Preliminary development of the system as a whole has been carried out in order to give general estimates of the parameters of future plants as well as to maintain the technological level achieved. These developments are based primarily on predictable parameters of TEC, however the possibilities of using already designed modern TEC are not excluded. The realization of more economical TEC discloses the possibility of using new structural schemes, for instance, a reactor with a converter remote from the active zone. In this case the reduction in the radiator temperature is compensated for by higher conversion efficiency, and the possible increase in radiator mass by a reduction in the mass of the shield because of the more compact active zone of the reactor. Such a configuration does not yield to the version with built-in TEC in its specific mass characteristics, however, it assures a considerably greater working lifetime and a rise in system reliability.

Researches on TEC in France and West Germany are conducted within the framework of short-range programs, whose principal purpose is to prove the possibility of raising the economy of TEC and the search for new applications. The main researchers are basic and applied investigation in the area of plasma and surface phenomena in TEC and the technology of electrode materials.

Thermal Emission Power Plants. The report of V. A. Kuznetsov (FÉI) about the results of testing the thermal emission reactor Topaz was received with great interest. The operating experience of the three

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reactor zones showed that such plants can be a sufficiently reliable and convenient energy source for 5-7 kW power with an operational life of at least several thousand hours. Special measures undertaken to improve the degassing and preparation of the emitter surfaces during tests of the Topaz-3 reactor permitted obtaining more than 6-kW power for more than 1200 h in the main section. The total reactor operating time in the electrical power generation mode was 3000 h. In addition to the electrical characteristics, the other reactor characteristics investigated were: temperature coefficients of reactivity, contribution of electronic cooling, leakage currents through the insulation, etc. An interesting peculiarity of the Topaz-3 reactor is the weak self-oscillations of the system at a 0.08-Hz frequency and with 0.1% of the nominal amplitude.

A number of reports were devoted to the question of using TEC in industrial power. Schemes to apply TEC as a high-temperature superstructure to the water steam-turbine cycle of industrial parameters with different heat sources such as a nuclear reactor, chemical fuel, solar energy, were examined. A thermodynamic analysis of a cycle with thermal emission superstructure for different reactor schemes is given in the report of L. E. Apatovskii et al. (Leningrad). Estimates show that a TEC as a high-temperature superstructure possesses advantages in its structural peculiarities and temperature parameters over other combination schemes, and is comparable in economy with a gas-turbine superstructure at the same temperature level. Application of a thermal emission superstructure to a thermal station permits obtaining a total efficiency of around 50% at 1400-1500°K emitter temperatures, which will yield a significant fuel economy and diminish heat waste (E. Britt et.al., USA). A structural concept of a large-scale thermal emission module is proposed, in which high output currents will permit getting rid of metal-ceramic sealing of the gap because of using a metal diaphragm. Inductive coupling to the load by means of a transformer built into the module is provided.

Collectors with Low Work Function. A reduction in the collector work function is perhaps the most real means for raising the economy of thermal emission converters. Primary attention is spent on this problem in practically all nations. The general direction in investigations on electrode materials with low work function is towards activation of both the materials which are classic in TEC practice and of the relatively new materials which are distinctive by the electrically negative admixtures (principally oxygen) in the presence of cesium vapors. Among them are the pure refractory (Nb, Mo, W, Re) and the noble (Ag, Au, Pt) metals, impregnated systems, hexaborides of the rare-earth metals (LaB_6, CeB_6, PrB_6), cermets (UO_2-Mo), semiconductors (Si, GaAs, GaP). The broadest investigations in this area are conducted in the USA. Of the 24 materials studied, a work function less than 1.2 eV has been achieved successfully for six. These are systems based on pure metals (W and Ag) at a 660°K temperature, impregnated systems, and hexaborides at 450°K. It is shown in the paper of J. Deplat (France) that the minimal work function for the W(100)—O—Cs system is 1.1 eV for both a monolayer coating (co-adsorption of Cs and O_2) and for thick cesium oxide layers on the order of 100 Å. Other admixtures besides oxygen were also studied. Results of investigating the emission and structural properties of systems under the combined adsorption of cesium and different chalcogenides are presented in the report by specialists from FÉI and Kiev State University. The influence of oxygen has been studied for two different fillers, barium and cesium, in the paper of M. Bradke et al., (West Germany). It is shown that activation of the substrate by oxygen results in a reduction of the work function to 1.8 and 1.1 eV, respectively (as compared with 2.15 and 1.6 eV without oxygen) in barium and cesium vapors. It is proposed to use the cermet UO_2—Mo as emitter material. Its advantage is not only the high emissivities at low cesium vapor pressure, but also the capacity to extract oxygen in the intraelectrode space in sufficient quantity to reduce the collector work function. A majority of the optimistic results with oxide collectors has been obtained under nonequilibrium oxygen adsorption conditions. The paper of F. Rufe et al. (USA) is devoted to the insertion of oxygen in real TEC. The use of a tungsten collector covered by tungsten oxides by the method of recondensation would permit reduction of the effective losses in a TEC to 1.87 eV as compared to 2.1 eV for a TEC with a pure tungsten collector. It is proposed that the lifetime of such a collector would be several years.

The growth of the volume and level of surface phenomena investigations, the extensive introduction of modern methods of analyzing the composition and structure of the adsorption systems (Auget spectroscopy and slow electron diffraction methods) must be noted. Perfected methods of studying the work function (thermal and field emission in a retarding field) under conditions approaching most closely to the electrode operating conditions in a TEC have been developed and mastered.

Perfected TEC Modes. A number of reports (USSR, USA, France) were devoted to the problem of reducing the transport losses in a TEC. Among the schemes considered are various triode systems, pulse diodes, and diodes of mixed type. The results presented by both the Soviet and American papers confirm the efficiency of an auxiliary discharge and pulsed excitation in inert gases. The efficiency of the auxiliary discharge turned out to be unexpectedly low in cesium vapors: a gain coefficient of 10-20 and effective losses of about 0.4 eV. It is assumed that this is explained by the unfavorable relationship between the scattering and ionization cross sections during electron interaction with the cesium atoms.

The possibility of perfecting a pulsed TEC, elucidated in the report of S. Manikopoulos et al. (USA) is interesting. It is proposed to use a mixture of molecular nitrogen with cesium vapors as the filler in pulsed diodes. It is assumed that upon excitation of such a mixture, a part of the electron energy will be transmitted into excitation of the vibrational levels of the nitrogen molecules, and then ionization of the cesium atoms during plasma dissociation upon collision with the excited nitrogen molecules. Such an ionization mechanism practically eliminates heating of the electron gas and significantly increases the plasma dissociation time. The phenomenon of low-frequency self-oscillation in a triode with an auxiliary discharge is disclosed in the paper from the I. V. Kurchatov Institute of Atomic Energy (V. Z. Kaibyshev et al.) because of feedback between the main and auxiliary loops by way of the common plasma between the electrodes. The possibility of regulating the oscillation frequency and mode, as well as a number of other favorable factors discloses the prospect of using such triodes in high-temperature and radiation-safe electronic circuits. Of interest are the results of investigating the reflexivity of molybdenum and tungsten during the adsorption of cesium and oxygen for low energy electrons (S. Balestra et al., USA). The high value of the coefficient of reflection of hot electrons (around 0.5) for electrodes with a pure cesium coating can be the reason for limiting the output characteristics of modern TEC.

* * *

Upon termination of the conference on September 4, 1975, the next meeting of the International Communications Group (ICG) on TEC took place, where summaries of past conferences were supplied. A resolution was taken to prepare and publish a combined report on the state of research with and the prospects of using TEC in the participating nations of the ICG jointly with the transactions of the conference. The session considered it expedient to conduct further IGC work on the previous level. The proposal of the Soviet representatives to organize the next international conference of TEC specialists in the USSR in 1977 was accepted. Professor V. A. Kuznetsov (FÉI, USSR) was elected a new representative of the International Communications Working Group on TEC.

7TH EUROPEAN CONFERENCE ON CONTROLLED THERMONUCLEAR FUSION AND PLASMA PHYSICS

Z. I. Kuznetsov

The conference took place on September 1-5, 1975 in Lausanne, Switzerland. More than 400 representatives from Europe, USA, Japan, Australia, Argentina, India, and Egypt participated. About 200 original and 24 review papers were presented which dealt with different systems and directions of research on controlled thermonuclear fusion and high temperature plasma physics, including tokamaks and stellarators, open magnetic traps, pinch and plasma focus devices, laser systems, electron and ion beams, turbulent and RF plasma heating, impurity and wall-plasma interaction problems, interaction of particle beams with plasmas, anomalous transport phenomena in plasmas, and problems in the theory of plasma physics.

A central place was occupied by the results of tokamak research, to which more than half of the talks were devoted. The topics discussed included the determination of the dependence of the energy lifetime of the plasma on the plasma parameters, clarification of the mechanisms which limit the particle lifetime and energy containment, the determination of the allowable values of the (stability) safety factor of the plasma column and of the ratio of the plasma pressure to the magnetic pressure, studies of impurity behavior, and the development of supplementary plasma heating and stabilizing techniques.

The work done on the large French tokamak, TFR, was of great interest. With additional plasma heating by injection of atomic hydrogen, an ion temperature of 1.4 keV at a density of $5 \cdot 10^{13}$ cm^{-3} with a confinement of time greater than 20 msec has been achieved. Ion energy loss obeys a neoclassical law; however, anomalously rapid electron energy loss is observed. These losses, in the authors' opinion, may be due either to inner turbulence or to a dissipative plasma instability.

On the Soviet Tokamak-6 machine a series of experiments has been carried out on the disruptive instability (peak instability), which is observed when the plasma density is increased. The authors believe that the process begins with pumping of some surface mode of an MHD instability, which then drives higher modes. Then the original mode is damped and the energy stored in the other, strongly developed modes causes a number of effects which lead to the disruptive instability. On the small American high magnetic field tokamak "Alcator," a high density (of order 10^{14} cm^{-3}) plasma has been obtained with pulsed injection of gas. On the American ATC machine supplementary heating is by injection of atomic hydrogen, RF heating at the lower hybrid resonance, and plasma compression by the magnetic field. As a result, it has been possible to raise the ion temperature to 1.2 keV at a density of about 10^{13} cm^{-3}. On the small Soviet tokamak TM-3, the possibility has been demonstrated of heating the electron component of the plasma by means of RF heating at the electron cyclotron resonance at magnetic fields of up to 25 kOe. The lifetime of the plasma column increased in this case with the increasing electron temperature and the heating mechanism was classical.

The results of experiments on adiabatic heating of the plasma in the Soviet Tuman-2 machine demonstrate significant improvement in energy confinement during the time the plasma is compressed. The ion component was heated according to an adiabatic law while the electrons were more strongly heated.

In a number of Soviet and American devices, experiments are being conducted to verify the theory that a noncircular cross section increases the (stability) safety factor of the plasma column. The development of MHD instability modes of the plasma column is in good agreement with theory. On the Soviet RT-4 machine the possibility has been shown of stabilizing the disruptive instability in tokamaks with the aid of a variable longitudinal current of magnitude on the order of 0.2 times the main discharge current.

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A large group of experiments on the behavior of impurities in various tokamak plasmas has shown that the main source of impurities is the discharge chamber walls and the impurity flux is proportional to the electron density. The impurities may enter the plasma due to plasma radiation falling on the wall or to wall bombardment by neutral particles produced by charge exchange in the plasma or by charged particles. In the typical machine operating regimes, the principal impurities are carbon and oxygen, while in low electron density regimes heavy impurity ions from the diaphragm and chamber walls are present.

A small number of papers were devoted to stellarator research, but a number of the results were of indubitable interest. It has been shown on the Soviet stellarator Uragan-2 that ions can be heated to several hundred electron volts by ion cyclotron heating. In West Germany, plasma containment with a high ratio of the plasma pressure to the magnetic pressure is being studied on the Pulsator and Isar-T1-B machines. High MHD stability of the plasma has been obtained on both machines. It has been shown that in the current discharge regime of the W-IIB machine the anomalous electron thermal conductivity observed in stellarators cannot be explained as an impurity effect. Anomalous electron thermal conductivity has also been observed in the Soviet stellarator Sirius and linked by the authors to drift oscillations.

Studies on the application of feedback systems to the American Scyllac machine (a toroidal theta pinch) have shown that although it is possible, in principle, to use such a system for stabilization and support of the plasma column equilibrium, its realization in the case of a plasma with thermonuclear parameters will encounter serious difficulties. In this regard, it is planned to test the possibility of stabilizing the plasma with a conducting wall in combination with shock heating with low plasma compression. Recently, plasma focus type systems with supplementary laser heating of the plasma have appeared (USSR and Poland). The neutron yield increases in such a system by several times when the processes are well synchronized.

In the talks on open magnetic traps the main topic was the problem of eliminating losses of particles and energy through the open ends of the traps. In the American 2X-IIB machine the drift cone instability has been suppressed by injecting a cold plasma along the system axis. When 370 equivalent amperes of atomic deuterium was injected, a plasma with a density of about $4 \cdot 10^{13}$ cm^{-3}, an ion energy of 13 keV, and a confinement time of several milliseconds was obtained. Interesting studies have been made on the Jupiter electromagnetic traps (USSR) in which a plasma with a density of $5 \cdot 10^{12}$ cm^{-3}, and ion temperature of 1 keV, and a lifetime of 5 msec has been obtained.

In the talks devoted to laser-produced plasmas, the most attention was given to the interaction of radiation with a solid target and a plasma, in particular to the study of nonlinear radiation absorption processes, the establishment of thresholds for the appearance of nonlinearities, production of harmonics in laser-produced plasmas, and the measurement of the angular intensity distribution of reflected light and harmonics. The majority of the experimental work has been done with nanosecond pulsed CO_2 lasers. It was shown that light is reflected mainly backward and the overall reflection coefficient of the plasma does not, as a rule, exceed 15% up to fluxes of $5 \cdot 10^{12}$ W/cm^2. Preliminary experiments have demonstrated the existence in a laser plasma of intrinsic magnetic field strengths of up to several megagauss.

The results of computations and experiments on the irradiation of targets by powerful electron beams were presented in a number of talks. Focussed relativistic electron beams, produced in special diode devices, make it possible to achieve target compression at powers of up to 10^{12} W. About half of the energy stored in the beam is dissipated in the irradiated shell of heavy elements surrounding the D–T target.

Various systems for creating long pulses and proton layers for capturing and containing hot plasmas are being developed and built. Pulsed proton beams with currents of several kiloamperes and 10^{14} total protons in the pulse have been obtained. It is proposed to compress these beams in a magnetic field to increase the parameters of the trapped plasma to thermonuclear values.

The results of work during the two years since the last conference indicate a considerable expansion in research on controlled thermonuclear fusion with the use of modern diagnostic techniques and of computers for data processing and numerical calculations. The new data obtained from experimental thermonuclear machines have made it possible to proceed in developing the next generation of powerful machines in the European countries, the USA, and Japan. These new devices include, first of all, tokamaks intended for production of plasmas with thermonuclear parameters. Construction is planned to begin on these machines in the end of the 1970's. The successful demonstration in them of a thermonuclear reaction with an energy yield equal to the energy delivered to the plasma will make it possible to proceed to the construction of thermonuclear power reactors.

SOVIET–WEST GERMAN SYMPOSIUM "ARMATURES AND PUMPS FOR POWER STATIONS"

R. R. Ionaitis

The Symposium was held on October 15-16, 1975 in Moscow. It was organized by the first UGT (Union for Gas Techniques GmBH). Leading firms of the Federal German Republic in the field of armatures and pump construction for nuclear power stations were represented.

On the first day of the sessions, detailed papers by West German specialists were read, devoted to shut-off and control valves, welded slide-valves and gates, check and safety valves, and on the second day papers were presented and discussed on the level of development of feed and circulating pumps for nuclear power stations, and the conditions for their technical servicing and testing.

Reactor technology in the Federal German Republic is relatively young. In 1961, a nuclear power station was started up in Kehl with a capacity of 15 MW; in 1966 there followed the start-up of the nuclear power at Gundremmingen with a capacity of 250 MW and in 1968 the reactor at Obrigheim started up (345 MW). Now, a nuclear power station with a capacity of 1000-1300 MW is being brought on stream. The earlier phase of reactor technology in the Federal German Republic was characterized by the large number of individual special designs (for a specific reactor). At the present time, the construction of reactors has emerged from the experimental stage. A defined standardization of reactor structural units, including armatures is being planned. The successful experience in the operation of nuclear power stations has removed the doubts and variations, which earlier had appeared in small structural armatures. Consequently, the general line of development of a reactor armature, as shown in the report by M. Moersdorf – the Use of an Armature for Thermal Stations – has been improved by taking account of reactor requirements. The principal one is the tendency to prevent damage to components and units which are under pressure, in order to avoid the escape outside of the radioactive working liquid. For this, first of all the volume of testing of the reactor armature is increased significantly in comparison with the usual testing. Seals of increased reliability are used for preventing leakage of radioactivity and contamination of the surrounding space. The reactor armature, as a rule, is welded into the manifold (and is not mounted on flanges). At first, the cover also was welded to the housing, but this has not been done for the past 10 years, as the gaskets used (spiral, asbestos, self-sealing double gaskets with suction) are sufficiently reliable. In the shaft duct, leakage is prevented in the most reliable way by glandless seals, operating from the pressure of the working medium. However, the sylphon seal has received the most widespread use, although it was considered to be exotic 20 years ago. According to the report of the speaker, of 100,000 sylphons in operation only a few are out of service (due to corrosion). Today, sylphons are being used up to working pressures of 320 bar and duct conditions of up to 450 mm. Membrane valves (SISTO brand) with elastomer seal are used up to a pressure of 12 bar and a temperature of 130°C and, finally, for large diameters and strokes, packing glands with suction and a spring clamp are used. The West German specialists are convinced that certain slide-valves made up to 4000 lifts before any leakage through the gland appeared.

A report in detail was made about the materials used for a reactor armature, the technology of manufacture, actuators, and also about the cleanliness which accompanies manufacture, the assembly, and shipment of the armature. However, it was mentioned that the efforts maintaining cleanliness come to nothing with the builders of nuclear power stations, which creates specific difficulties during startup of the reactor, especially in ensuring sealing.

In the report by H. Wieland, it was explained in detail why, for nuclear power stations, steel forge-welded slide-valves are manufactured and how these slide-valves and check valves are installed, which have been supplied to 12 nuclear power stations in the Federal German Republic and to four stations abroad.

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P. Heiler reported on slides with an arbitrary diameter of up to 3000 mm and a flow rate of the medium of tens of thousands of cubic meters per hour, and methods and means of controlling such large shut-off devices. The report by H. Richter on safety valves for nuclear power stations created great interest, especially such large ones which, at a pressure of 210 bar and a temperature of 540°C can pass up to 1700 ton/h of hot steam. Four such devices serve for the protection of a nuclear power station with a capacity of 1200 MW.

Reports on pumps for nuclear power stations created no less interest. The report by H. Diderichs and D. Born discussed single-stage pumps operating at a temperature of up to 180°C and feeding up to 4000 ton/h of water at a pressure of up to 80 bar. It related in detail the most important sealed unit of pumps and the construction of glandless circulation pumps with a capacity of up to 8500 m^3/h (pressure head 32 bar, pressure 320 bar, temperature 375°C, and arbitrary diameter up to 600 mm). R. Dernedde et al., carefully analyzed what effect the pump design has, especially the point of location and the type of seals and bearings (hydrostatic or hydrodynamic), on the duration of maintenance and the number of servicing personnel of the nuclear power station. This question, in connection with extension of nuclear power station construction, is of extremely pressing importance, as the data given in the report concerning the radiation doses at the Gundremmingen, Obrigheim, and Stade nuclear power stations show that personnel are subjected to the maximum irradiation during disassembly and decontamination of the contaminated parts of the pump seals. In another report, R. Dernedde et al., noted that the requirements of nuclear power stations for high-powered pumps considerably outstrip the capabilities of the firms manufacturing them. Pump constructors are not able properly to strip down in operation a pump of one capacity, designed by them, before proceeding with an order for a pump of much greater capacity. Actually, as it follows from the table given in the report, over 10 years the capacity of the nuclear power station has increased by a factor of five (from 250 to 1300 MW), the rating of a single pump by a factor of four (from 5.7 to $24 \cdot 10^3$ m^3/h), the pressure by a factor of four (from 4.4 to 17.3 bar) and the pump capacity by a factor of nine (from 0.8 to 7.4 MW). In the Federal German Republic, a test bench has been designed, installed in a mounting-experimental housing with a length of 83 m, a width of 28 m, and a height of 22 m for full-scale testing under full load. The circuit is calculated on a pressure of up to 180 bar, a temperature of up to 350°C, a flow rate of up to 6000 m^3/h, and a motor power of 12 MW..

The Soviet specialists speaking in the debates noted that the general trends of development of the reactor armature and pumps in the Soviet Union and in the Federal German Republic in many respects coincide, however the detailed interpretation of the details of this development by the West German specialists was extremely useful.

In conclusion, the excellent organization of the symposium should be noted. Reports with Russian text and detailed lists of the firms participating in the symposium were distributed to every participant; reports were read out in Russian and the answers to questions were translated rapidly and skillfully.

TECHNICAL CONFERENCE NUCLEX-75

L. N. Podlazov

The 4th International Fair and Technical Conference on the nuclear industry Nuclex-75 took place in Basle (Switzerland) on September 6-11, 1975. The Fair is held every three yars, and is intended to demonstrate world achievements in nuclear industry and technology. The increasing authority of the Fair is indicated by the fact that in 1972 it was attended by 332 companies from 22 countries and in 1975 by 433 companies from 25 countries. Among the participating organizations those specializing in nuclear power predominated, thus indicating the increasing role of nuclear stations in power production.

The papers read at the conference were presented and discussed in eight sections. These embraced a wide circle of questions associated with the development of new and the perfection of old reactor installations of various types, and also with their safety and reliability in service. There were also eight specialized colloquia in which the participants of the exhibition were able to present short descriptions of their companies' products. In all the sections and colloquia over 200 papers and communications were read.

The Soviet Union presented six contributions (three on fast reactors and three on uranium–graphite reactors of the channel type with a boiling coolant). Those on fast reactors concerned the BN-600 reactor, sodium pumps for the BOR-60, BN-350, and BN-600, and also the combined use of ^{235}U and plutonium in fast reactors. The papers on the channel reactors were devoted to the characteristics of the RMBK-1000 reactor obtained during its initial service and to reactor safety measures, and also to a module channel reactor with nuclear steam heating. The Soviet contributions were of great interest to the delegates. At the opening of the Conference papers were read in connection with the prospects and structure of nuclear power development and also its role in the general power supply.

The first section considered fast breeder reactors. The service characteristics of existing reactors were discussed, together with special features encountered in the operation of individual reactor equipment, the prospects of further study, and the perfection of these reactor systems for producing industrial energy and fuel breeding. The second section was devoted to high-temperature gas-cooled reactors, the basic principles of their development, and problems relating to the development of equipment for the first circuit, including a helium turbine for incorporation in the direct cycle. Much time was devoted to the development and study of fuel composites, fuel elements, and the materials of the active zone. Questions of safety were considered from the point of view of ecological problems. Papers presented to the third, fourth, and sixth sections mainly concerned the technology of proven reactor installations and also improvements to their service properties, increases in safety and reliability, and the prospects of developing reactor installations with large unit powers. Considerable place was also given to various aspects of the study of fuel-element working characteristics, as well as fuel composites and the perfection of these.

The 5th section discussed the safety of nuclear power stations in relation to the population and environment and methods of estimating risks. Measures to ensure safety in a major emergency involving the dehermetization of the circuit and the loss of coolant were considered, as well as the possible consequences of such an emergency and the influence of seismic factors on the safety of nuclear power stations. The principal method of estimating the safety of reactor installations employed in the contributions was the statistical method based on the ideas of Farmer. Attention was drawn to the necessity of increasing present efforts to assemble and analyze information regarding the statistics of operational faults in individual items and the power station as a whole. The seventh section considered methods of increasing the heat utilization and working efficiency of reactor installations and ways of improving the final coolant systems; the eighth

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considered instruments and automatic devices for controlling nuclear power stations. The objective necessity of improving the monitoring and control systems in view of the increasing power of nuclear power-station units and the complication of control problems was emphasized.

The specialized colloquia considered construction materials for nuclear technology, new uses of radioisotopes, instruments and devices for monitoring and controlling nuclear installations and measuring radioactive radiations, new methods of reprocessing and removing radioactive waste from nuclear installations, etc.

The papers read at the technical conference were published as soon as it opened in the form of 17 collections of articles, which indicated the present expansion of the sphere of application of nuclear installations, the tendency for atomic power-station units to increase in size, and the increasing role of nuclear power in the life of modern society.

CONFERENCE OF SPECIALISTS ON DATA PROCESSING FOR REACTIONS WITH CHARGED PARTICLES

L. L. Sokolovskii

In September 1975 the Section for Nuclear Data of the International Atomic Energy Agency held a conference on technical problems related to the gathering and exchanging of nuclear data for reactions involving charged particles. The participants of the Conference came from several centers and groups; among them there was a representative of the Center for Data on the Structure of the Atomic Nucleus and Nuclear Reactions of the State Committee for the Use of Atomic Energy, founded in 1972. The activities, the organization of international cooperation for creating file-evaluated data, the exchange of bibliographies, of numerical, and evaluated data, and the formats of bibliographic records and of numerical and evaluated data were discussed. In the first stage the Conference limited its work to "integral" data for reactions involving charged particles; the term relates to excitation and yield functions for thick targets of nuclei with $Z > 4$ and to outgoing p, α, d, ^{3}He, and T particles at energies of 50-60 MeV.

Though integral data on nuclear reactions involving charged particles at the present time make up about 5% of the total of nuclear data for reactions involving charged particles, efforts in the compilation and evaluation of the data are indicated, because the number of the users of the data in applied science and technology increases and because the data are needed for organizing international collaboration. Ninety percent of all nuclear data for reactions involving charged particles are differential data for which only a limited number of users exist, who are mainly working on basic research problems. The Conference approved work on the compilation and decided over the continuation of this work and its possible expansion in relation to nuclear data for reactions involving charged particles. Both the compilation and the exchange of integral nuclear data on reactions involving charged particles are to be given priority. The efforts of the group of Professor H. Munzel (Karlsruhe, Federal Republic of Germany) on the systematic compilation of integral data on nuclear reactions involving charged particles in a format resembling the neutron format of EKSFOR were considered very valuable. The recommendations of the Conference were based on the assumption that the group in Karlsruhe will maintain the main file of evaluated integral data on nuclear reactions involving charged particles. In this connection the Conference recommended: the Karlsruhe group shall continue the compilation of integral nuclear data on reactions involving charged particles; the Center for Data on the Structure of the Atomic Nucleus and Nuclear Reactions of the State Committee for the Use of Atomic Energy shall compile integral nuclear data on reactions involving charged particles for the USSR; the National Center of Neutron Cross Sections (Brookhaven, USA) shall compile similar data for the USA and for Canada; and a modified EKSFOR format must be used for the exchange of integral data on reactions involving charged particles.

In regard to cooperation and free exchange with other centers, the Conference assigned to the USA the main responsibility for the worldwide bibliographic system of nuclear data on reactions involving charged particles. In the near future, the National Center for Neutron Cross Sections must agree with the Center for Data on the Structure of the Atomic Nucleus and Nuclear Reactions of the State Committee for the Use of Atomic Energy and with the Karlsruhe group on the format and the arrangement of the future bibliographic system. The participants of the Conference agreed that the creation of a main file of evaluated integral nuclear data on reactions involving charged particles is possible only with close international collaboration and free exchange of bibliographic, numerical, and evaluated data.

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In April-May 1976 a conference of the International Atomic Energy Agency on problems of the compilation and exchange of data on the structure of the nucleus shall be convened. Then some problems related to the execution of the recommendations of the Conference on the compilation of nuclear data on reactions involving charged particles will be summarized.

24TH SESSION OF THE SCIENTIFIC COMMITTEE OF THE UNITED NATIONS

R. M. Aleksakhin and A. A. Moiseev

The Session took place on September 15-19, 1975 in New York. Representatives from 19 countries and five international organizations (IAEA, ICRP, ICRU, WHO, and UNEP – Programs of the United Nations on the preservation of the surrounding medium) participated in its work. The work proceeded in two subgroups – physical and biological (a genetic commission was allotted to the composition of the latter). A number of documents were discussed by the physical subgroup (Chairman, A. Jammet, France) which dealt with the level of present-day contamination of various items of the external medium, due to the global fallout of artificial radionuclides; natural sources of ionizing radiations were analyzed in detail. The important role of man's economic activities in changing the natural radiation background was emphasized (the use of structural materials with an enhanced content of natural radioactive substances, the use in agriculture of fertilizers with an increased concentration of uranium, radium, thorium, etc.). A more rigorous estimate of the background irradiation of man has allowed the significance of the additional irradiation from artificial radiation sources to be assessed more correctly. Particular attention was paid to the estimation of the possible contamination of the surrounding medium during the peaceful utilization of atomic energy (mainly nuclear).

The collective doses of irradiation of the population have been calculated at all stages of the nuclear-power generation cycle (ore extraction, its processing, radiochemical production, reactor operation, transportation of radioactive substances, and the storage of wastes). Calculations of this kind in full volume were carried out for the first time and this scientific information is of first-degree importance. They show convincingly, very low levels of irradiation of man in the future (up to 1990-2000) at all stages of production of nuclear power.

Among the problems discussed in the biological subgroup (Chairman, E. Pochin, Great Britain), new data were considered concerning radiation carcinogenesis with laboratory animals, the basic principles of the biological action of radiation, and estimates of the radiation doses to man.

The section dealing with the mechanisms of radiation carcinogenesis, submitted a complete revision with the assistance of genetics. In light of modern concepts, the general biological mechanisms of mutagenesis and carcinogenesis, due to the activation of an oncovirus, were compared with principle. This is very important for considering the effect of low doses of ionizing radiations and nonspecific carcinogenesis during irradiation, occurring on a background of diverse immune and hormonal reactions, due to a complex set of postradiation changes in the organism.

Lively discussions generated the question concerning the relation between carcinogenic effects and the space and time distribution of the absorbed energy of ionizing radiations. The degree of irradiation hazard to individual tissues was assessed, taking into account the possibility of cancerous transformation of a specific tissue, the contribution of a specific type of neoplasm in the population morbidity and lethality. In this respect, in the first place is the thyroid gland, followed by the bone and bone marrow and then, considerably lower in degree of hazard, the lungs, skin, and liver. The estimate of the genetic danger is given in two aspects: by the direct method and by determining the so-called doubling dose. It is supposed that with animals, in the first generation as a result of irradiation with a dose of 1 rad, there are 1 to 3 mutations per 1 million males. The great care in the forecast estimates for man, in view of the absence of authentic information, attracted attention. A dose of 10 rad is assumed as the doubling dose only in respect of balanced reciprocal translocations.

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By way of the individual documents, the biological subgroup discussed the results of the radiation dose estimates during the medical use of ionizing radiations and radioactive substances for cases of occupational irradiation.

After the introduction of supplements, taking account of the results of the discussions, the documents considered will be accepted in final form for the General Assembly of the United Nations at the 25th Session of the Committee, which is planned to take place on September 6-17, 1976 in Vienna.

SOVIET EXHIBITIONS AT THE 4TH INTERNATIONAL EXHIBITION OF THE NUCLEAR INDUSTRY NUCLEX-75

V. A. Dolinin

The exhibition took place in Basle (Switzerland) in October 1975. Exhibits from the nuclear industry of a large number of countries were presented. For this first time Nuclex-75 offered a specialized exhibition of nuclear technology presented by the International power organization "Interatominstrument." The Soviet Union sent representatives of two member organizations of "Interatominstrument" ("Tekhsnabeksport" and "Izotop"), who demonstrated 15 exhibits.

Special interest was aroused in a system for detecting packs containing dehermetized fuel elements. This system is intended for use in nuclear reactors cooled by means of liquid alkali metals and alloys; it allows successive testing of the packs in the active zone of the shut-down reactor, without having to take them all out of the active zone. On detecting packs with dehermetized fuel elements these may be unloaded in the usual way.

A number of our instruments evoked special interest.

PIR-1 (Radio-Isotope Measuring Converter). Together with the portable pulse counter SIP-1M, this is intended for measuring the volumetric mass (density) in the surface layer of the object being measured and is used for the mass express quality testing of objects and structures made from light and cellular concretes and soils. The instrument is intended for various reinforced-concrete-assembly undertakings, building organizations and laboratories, technical monitoring services, and so on. Special features include the use of a low-activity source of γ radiation, a device for the automatic stabilization of readings, and devices for preventing the accidental movement of the source into the working position and automatically placing it in the transportation position when the instrument is lifted up.

RPI-1 (Radio-Isotope Measuring Converter). Together with the SIP-1M this is intended for the technical and technological nondestructive express quality testing of surface compaction in reinforced and nonreinforced soils, and in asphalt and cement concrete placements up to 30 cm deep. Together with other instruments this constitutes a means of testing the bearing capacity of soils and the strength and resistance of concretes. The instrument may be used under field conditions and is resistant to dust and moisture. Special arrangements are incorporated to eliminate radiation hazard. The RPI-1 is intended for builiding organizations and laboratories, technical monitoring services, etc.

The SIP-1M is a universal, economical, and reliable portable counter intended for recording electrical pulses arising from radioisotope or other measuring converters; it contains a high-stability electronic timer. Information is extracted in decimal form. The counter is provided with stabilized 400 and 12 V voltages by measuring converters operating in conjunction with it.

The IZV-3 instrument is intended for the express determination of inactive dust and short-lived radon decay products in the air; it operates directly in the working regions of industrial sites: mining, metallurgy, quarrying, glass, cement, and automobile manufacture.

As usual there was great interest in radioisotope technological monitoring instruments. The radioisotope-based servo-type level gage UR-8M is intended for the continuous automatic remote measuring and recording of the level of liquid media in open or closed reservoirs and tanks, and also for transmitting

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Fig. 1. "Interatominstrument" exhibits at the Nuclex-75 exhibition.

a pneumatic signal via the secondary instrument KSP-3 to an automatic control system. The UR-8M helps in automating technological processes in the chemical, oil-refining, paper-and-pulp, and other branches of industry.

The general-purpose gamma relays GRP1-1 (single-channel) and GRP2-1 (two-channel) are intended for solving problems relating to the complex automation of production processes in various branches of industry. In contrast to existing Soviet and foreign gamma relays, the statistical information in the GRP1-1 and GRP2-1 is analyzed in digital (discrete) form, using routine-production integrated microcircuits. This improves the technical characteristics (speed of action, reliability, sensitivity, and stability of the parameters).

The Press gave considerable attention to the exhibits of the Soviet section and the "Interatominstrument" exhibits as a whole. A number of contracts were concluded for the sale of items shown at the exhibition.

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CONTROL AND SAFETY OF NUCLEAR POWER REACTORS*

Reviewed by V. I. Plyutinskii

The widespread construction of nuclear-power stations, especially in regions with a high population density, makes extremely important and urgent the problem of the safety of nuclear-power stations. There are no books on the present-day status of investigations in this field in the Soviet literature. The existing monographs deal only with individual aspects of the problem, or are of the nature of manuals. The numerous foreign works, as a rule, ignore the results obtained in the USSR which have made a significant contribution to the problem of safety of nuclear power stations and which are used extensively in the development and operation of Soviet nuclear-power stations.

The appearance of the book being reviewed, in which for the first time in Soviet literature the problem of control of nuclear reactors is considered in toto, is of considerable interest. The authors are well-known specialists in the control and dynamics of nuclear-power stations and have been working in this field from the first days of nuclear-power generation.

Problems of safety and methods of assessing the risk of possible hazards at nuclear-power stations are considered in the book. It is shown that the development of these methods permits the technicoeconomical and sociopsychological requirements to be established for systems which will ensure the safety also of the emergency precautions which, at the present time frequently are achieved on the basis of volitional solutions.

Various aspects of the dynamics and control of nuclear power stations are considered from the point of view of ensuring the required safety and which adds orderliness and logical continuity to the book and permits important theoretical and practical results to be provided. Extensive data from Soviet and foreign investigations are used (the bibliography contains more than 150 references).

In the first section, the theoretical prerequisites are studied for the choice and establishment of the mathematical models describing the dynamics of the reactor. Here, the equations of reactor kinetics and the dynamics of thermophysical processes used at the present time are considered. The account of the methods of solving the equations of reactor space kinetics, their comparative analysis, and comparison of the numerical and experimental results are of special interest, and recommendations are made on their basis for the applicability of the various methods.

Unfortunately, when recounting the equations of dynamics the authors have not touched upon a number of questions which are important for studying high-speed emergency processes, such as the effect of the distribution of the coolant parameters over the radius of the fuel channel, the experimental basis for the accuracy of the results obtained for a single-phase coolant and the numerical methods for solving nonlinear dynamic equations.

In the second section, the effect is considered of special structural features and the dynamic characteristics of the basic reactor types – water-cooled/water-moderated, boiling pressurized, and channel reactors – on the methods of ensuring safety. Based on theoretical and experimental data, numerous factors are studied (the dynamics of feedback, heat exchange crises, fast and slow self-fluctuations, hydraulic instability, spatial fluctuations of the heat release field, and disruption of operating cycles),

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affecting the course of normal and emergency processes. In particular, the possibility is shown of detecting the start of boiling of the coolant in reactors with water under pressure by the analysis of the current variations of ionization chambers. Some practical measures for increasing the safety of the course of transient processes are considered and substantiated. Some objections are raised by the extension of the tightly coupled model of the active zone to all reactors of water-cooled/water-moderated power reactors. The applicability of this model is indisputable for the VVÉR-210 and VVÉR-440 reactors but may prove to be invalid for the more powerful reactors of the VVÉR-1000. There is great interest at present in gaseous and liquid-metal coolants, but in this section problems of safety of these reactors are not considered.

The third section is devoted to problems of design of control and safety systems of nuclear reactors. Here, the considerable practically important data on the analysis of control and safety circuits are generalized and procedures are given for choosing their parameters, based on safety estimates. The control of the average (integral) and local power of a reactor is discussed, an interesting review is given of methods of control using gas, liquid and combined methods for changing the properties of the active zone, and the results are given of the functioning of control and safety systems of different types. However, these data should have been supplemented with data from certain Soviet stations (first and second units of the Beloyarsk nuclear power station, the third unit of the Novovoronezh nuclear power station, the Biblibinsk nuclear power station, and the Shevchenko nuclear power station) and also from foreign stations. The authors give consideration to one of the rapidly developing trends – the application of computers to nuclear power station control systems and the necessity is formulated for the use of controlling computers. One cannot but agree with the conclusion that conversion to controlling computers of all functions would be premature (e.g., protection by neutron flux, the period and the pressure in the circuit). Meanwhile, the status of a work is debatable which, in analyzing the capabilities of controlling computers, must start from their operation to failure, amounting to several hundreds of hours. This quantity, which is true for a single computer, increases significantly (up to several tens of thousands of hours) in standby multimachine complexes. It is precisely the latter value which must be taken into account when estimating the efficiency of a controlling computer.

The book is completed with a section in which problems of the operational control of nuclear reactors are considered. Recommendations are given for the choice of methods of controlling the energy release spatial fields in a reactor, the diagnostics of intrareactor anomalies, organization of reactor startups, and other problems which are important in the development and operation of a reactor.

It should be mentioned that the authors have studied a large circle of problems associated with the safety of nuclear power stations, many of which have been solved, and for others the effective paths for finding solutions have been outlined. Despite the complexity of the problems broached, the book is written in intelligible language, which makes it possible for a wide circle of readers to familiarize themselves with it. The authors have proceeded correctly by omitting many cumbersome mathematical calculations and have confined themselves only to the qualitative analysis of the results obtained.

It may be expected that the book we have reviewed will become a reference book for engineers and scientific workers occupied with problems of the development, construction, and operation of nuclear power stations. It will play an important role in the work of training highly qualified personnel for the rapidly developing nuclear power generation in the USSR and countries of CEMA.

Yu. V. Seredin and V. V. Nikol'skii

PRINCIPLES OF RADIATION SAFETY IN PROSPECTING AND EXPLORATION FOR MINERALS*

Reviewed by E. D. Chistov

The book being reviewed is a first attempt at the generalization of methods and procedures for ensuring radiation safety during prospecting and exploration of minerals. The book presents the basic concepts and special terminology assumed in the literature on problems of ensuring radiation safety and in health-legislative documents, regulating the use of radioactive substances and other sources of ionizing radiations.

Data on nuclear methods used at the present time for the prospecting and exploration of minerals preface the account of the basic information. The principal aspects of the physical and biological principles of protection from ionizing radiations are recounted in a form suitable for practical workers (who are not specialists in the field of nuclear physics, dosimetry, and shielding).

In the main part of the book, practical recommendations are made for the organization and execution of radiation monitoring, the provision of safe working conditions for the storage, transportation, and use of sources of ionizing radiations under laboratory and field conditions, and taking account of the specific characteristics of conducting mineral prospecting and exploration.

A large amount of reference data is given in the book, concerning neutron and gamma sources used in geophysical investigations and also concerning the various remote-control devices and shielding techniques. The appendixes contain models of the documents which it is necessary to keep for the provision of and working with sources, their movement, and burial.

Unfortunately, the authors have not given consideration to problems of ensuring radiation safety during prospecting operations in mine workings, excavations and ditches, although these types of work occupy a notable place in the general complex of prospecting and exploration of minerals. Certain general questions of the interaction of ionizing radiations with different media, and also questions of dosimetry and shielding are explained in unnecessary detail. It must be hoped that these comments will be taken into consideration by the authors in preparing the next edition of the book being reviewed.

In conclusion, it should be noted that this is a useful book, which undoubtedly will help engineering-technical personnel of the geological service in solving practical problems on the rational organization and exceution of work under conditions of professional contact with ionizing radiation sources, and workers carrying out monitoring for the observance of measures for radiation safety, and for the provision of the qualified execution of such monitoring.

*Nedra, Moscow (1975).

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