Russian Original, Vol. 43, No. 5, November, 1977

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SATEAZ 43(5) 971-1074 (1977)

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SOVIET ATOMIC ENERGY

АТОМНАЯ ЭНЕРГИЯ
(ATOMNAYA ÉNERGIYA)

TRANSLATED FROM RUSSIAN

CONSULTANTS BUREAU, NEW YORK

SOVIET ATOMIC ENERGY

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Single Issue: $50
Single Article: $7.50

Prices somewhat higher outside the United States.

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CONSULTANTS BUREAU, NEW YORK AND LONDON

227 West 17th Street
New York, New York 10011

Published monthly. Second-class postage paid at Jamaica, New York 11431.

SOVIET ATOMIC ENERGY

A translation of *Atomnaya Énergiya*

May, 1978

Volume 43, Number 5 November, 1977

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The Russian press date (podpisano k pechati) of this issue was 10/25/1977. Publication therefore did not occur prior to this date, but must be assumed to have taken place reasonably soon thereafter.

SOVIET ATOMIC SCIENCE AND ENGINEERING TODAY

The creators of Soviet atomic science and engineering, i.e., scientists, engineers, and workers, are proud of their successes as they celebrate the 60th anniversary of the Great October Socialist Revolution together with the entire Soviet nation. Having designed and constructed the world's first atomic power plant in a record-breaking time, the Soviet people opened up an era for the growth of nuclear power engineering. In this achievement by the Soviet Union the international community saw an expression of its hopes for the use of atomic energy for the good of mankind.

Large atomic power plants have been built and are in successful operation in our country, making it possible to satisfy the growing energy demands of a number of important industrial regions of the country. At the same time, the most promising types of reactors are being studied and chosen for atomic power plants. The Novovoronezh, Kolsk, and Armyansk atomic power plants have been provided with water-cooled—water-moderated (VVÉR) reactors, the Leningrad, Kursk, and Chernobylsk plants, with RBMK uranium—graphite channel-type reactors; and the Shevchenko atomic power plant, with a BN-350 sodium-cooled fast reactor which, in addition to electricity, also generates industrial heat for distilling seawater. The Siberian, Beloyarsk, and Ulyanovsk atomic power plants are operating successfully. The Bilibinsk atomic heat and electric power plant with a uranium—graphite channel-type reactor built especially for the conditions of the Far North is in operation in Chukotka. Atomic power plants have been built in Czechoslovakia, the German Democratic Republic, Bulgaria, and Finland with the assistance of the Soviet Union.

The following large atomic power plants with reactor blocks of 1 million kW or more are now under construction in the Soviet Union: Smolensk, Kalinin, Rovno, South Ukrainian, Ignalin, and others.

Two years after the first atomic power plant was put into service, construction was started on the atomic icebreaker Lenin, the flagship of the Soviet arctic fleet. The second nuclear-powered ship, Arktika, is now in service and the physical start-up of the reactors of the Sibir is now under way. In the year of the 60th anniversary of the October Revolution, the Arktika completed a heroic voyage, viz., overcoming the ice of the ocean it reached the geographical North Pole.

Soviet scientists and engineers are confidently advancing along the road to harnessed thermonuclear energy. A developed thermonuclear reaction has been achieved in the Tokamak-10. Achievement of a controlled thermonuclear reaction on a commercial scale holds out great promise for providing energy for future society.

Practical results have been obtained in the direct conversion of heat generated in a nuclear reactor. The experimental thermoelectric and thermionic converters, Romashka and Topaz, have been demonstrated in successful operation. Radioisotopic energy sources have been developed for use in meteorology and navigation.

The application of atomic technology in the national economy has not been confined to power generation. In thousands of plants it is employed in the control of technological processes and in quality control of products; in geology, it is used in prospecting for commercial mineral deposits; in medicine, it is used extensively for diagnosis and treatment; and in biology and agriculture, it is employed to accelerate the selection of microorganisms and plants.

The level of development attained by the atomic industry has made it possible to assign it the major national economic tasks formulated and adopted by the Twenty-Fifth Congress of the Communist Party of the Soviet Union (CPSU). The Main Directives for the Development of the National Economy in 1976-1980 envisage the commissioning of 13-15 million kW of capacity in atomic power plants; to continue the construction of atomic power plants with reactors each of 1-1.5 million kW; to provide for the leading development of the nuclear power industry in the European part of the USSR; to speed up the construction and utilization of fast reactors; and to proceed with preparatory work on the use of atomic energy for central-heating purposes.

Translated from Atomnaya Énergiya, Vol. 43, No. 5, pp. 323-324, November, 1977.

The atomic industry is a characteristic example of the accelerated assimilation of scientific developments: from scientific search through basic research and technical development to the practical application of the results. Many institutes of the Academy of Sciences of the USSR were drawn into the work on the creation of the atomic industry, the most prominent industrial organizers were put in charge, special branches of scientific—industrial organizations were set up, factories with their own personnel were brought in, and a considerable part of the work, from search to introduction into practice, was conducted simultaneously. The experience gained from the development of the atomic industry demonstrated the positive aspects of such organization and this found expression in the simplification and acceleration of the process by which scientific developments are introduced into practice as well as timely, high-grade training for personnel for a new branch of the national economy. With the constant control and assistance of the Central Committee of the CPSU, the atomic industry in a very short time became capable of solving problems of immense complexity. This proved possible because throughout the existence of the Soviet state the party and the government have paid constant attention to the organization and development of scientific establishments and leading branches of industry.

As far back as during the first five-year plans an experimental base that was advanced for that time was established for nuclear-physical research. Subsequently, this base was expanded through the construction of larger and larger facilities, including the unique accelerators at Dubna and Serpukhov.

Soviet scientists have enriched atomic science with major achievements such as: the discovery of spontaneous fission of uranium; hypotheses about the neutron—proton structure of the nucleus and the exchange character of nuclear forces; the discovery of nuclear isomerism in artificial isotopes; the discovery of Cherenkov—Vavilov radiation; research on artificial radioactivity; the development of a theory and methods of calculation for the chain reaction of ^{235}U; the discovery of new transuranium elements, including kurchatovium; the discovery of new nuclear particles; measurement and refinement of the fission cross sections of nuclei for thermal and fast neutrons; and many others. Nuclear physics, which even in the 1930s seemed to be abstruse, abstract, and far removed from practical application, has grown into an important independent branch of science, has had a great impact on other branches of science and engineering, and has laid the foundations of nuclear power engineering.

Soviet scientists have played a leading role in the development of the concept of fast reactors possessing the property of breeding nuclear fuel and thus increasing the efficiency of uranium utilization tens of times. Such reactors are now in the stage of experimental-commercial development. The experience gained from the operation of the world's largest fast reactor in the USSR convincingly confirms the promise of this line of development.

The accelerated development of nuclear power generation today is one of the principal lines of work in the rational balancing of fuel and energy resources, especially in the European part of the USSR. It can be said with full justification that the advent of nuclear power has been extremely timely: mankind is facing the problem of a shortage of fuel and energy resources. Nuclear power postpones the problem and, in the long range, after the introduction of fast breeder reactors and mastery of thermonuclear power, will practically completely eliminate this problem. Along with the optimal solution of the fuel problem, nuclear power also facilitates protection of the environment.

Electricity can already be obtained from atomic power plants at lower cost than from power plants operating on organic fuel. In the future, the economy of atomic power plants will increase further, on the one hand, because of the continuous improvements in their design and, on the other hand, as the result of the rising cost of organic fuel as more and more inaccessible and remote deposits are developed.

The application of atomic energy for central heating is an important immediate task. Later on methods will have to be developed for obtaining high-temperature heat in nuclear reactors for industrial purposes, for metallurgical, chemical, and other production.

Science and the results of its development are inseparable from the policy of the party and the government: science has become a direct productive force. Fruitful basic and applied research in the realm of nuclear science and technology is being pursued in many scientific establishments provided with the latest word in equipment. Successes are facilitated by the consolidation and coordination of work done in this field in the socialist countries. "We consider it necessary," Comrade L. I. Brezhnev said at a meeting with the leading officials of the Academies of Sciences of the socialist countries, "to encourage the development of basic science in every way, to see to its being organically united with applied research, and to speed up the introduction of scientific discoveries into the national economy" [Kommunist, No. 4, 9 (1977)].

The continuous care of the party and the government for Soviet nuclear-physical science and atomic industry and the selfless work of scientists, engineers, and other specialists have ensured decisive achievements in this field and laid the foundation for future major successes in the utilization of atomic energy for the development of the material and technical base of Communism.

WATER-COOLED—WATER-MODERATED REACTORS IN NUCLEAR POWER GENERATION OF THE SOVIET UNION

V. A. Sidorenko

The prospects of a new trend in nuclear power generation must be based first and foremost on its economy. Other important requirements for a nuclear power plant as a source of energy in electric power stations are safety and reliability. But it is obvious that the achievement of the required level of safety and reliability appears also in the final count in the efficiency indices of nuclear power stations. The first Soviet water-cooled—water-moderated (VVÉR) reactors demonstrated the validity of the scientific-technological principles invested in them, and the structural design principles for achieving them. The high efficiency of water as a moderator and the small difference in the moderation length in the fuel lattice and in pure water, because of the significant contribution of inelastic moderation in the uranium, have permitted the use exclusively of a compact fuel lattice design, and the production of a high power from a reactor with a small-sized core.

Close-packed fuel lattices ensure a significant cross effect of fast neutron multiplication. A compact core design with the minimum quantity of structural materials allows a comparatively good thermal neutron utilization to be achieved. These qualities enable the required neutron multiplication factors in the fuel lattice to be obtained, with a quite high resonance capture in ^{238}U. The neutron spectrum in the close-packed lattices used is found to be relatively hard, and fission and absorption processes in the epithermal energy region acquire considerable importance. Thus, a high rate of plutonium buildup and its significant contribution to the production of energy for a deep fuel burnup are ensured. As a result, the conversion factor of fissile isotopes in a water-cooled—water-moderated reactor is found to be markedly better than, e.g., in a similar graphite—water reactor.

The choice of sintered uranium dioxide as the fuel has proved to be promising, since it is stable to erosion by water (coolant) and, for this reason, ensures the least contamination of the primary circuit in the event of fuel element failure, and fuel elements of uranium dioxide maintain their efficiency with quite deep burnups, which promotes further improvement of the fuel cycle of these reactors. However, the use of uranium dioxide as the fissile material is not optimal from the point of view of the physics of the fuel cycle, but is a compromise, partially satisfying the physical and technical requirements. Conversion in the future, with the improvement of nuclear fuel technology, to denser compositions based on uranium metal, shows promise for the additional improvement of the economic characteristics of the fuel cycle.

A special zirconium alloy is used as the principal structural material of the core, which allows quite high coolant parameters to be ensured and a satisfactory thermodynamic efficiency of the cooling cycle to be achieved. At the same time, a quite favorable neutron balance in the core has been successfully maintained.

The use of water-cooled—water-moderated reactors in nuclear power stations has led to the introduction into power generation of turbogenerators on saturated steam. This decision has proved to be important in principle, since it has permitted a satisfactory efficiency of the cycle (27-34%) to be achieved in nuclear power stations with a limited coolant temperature in the reactor (300-350°C).

The deciding factors that stipulated the power of the first commercial water-cooled—water-moderated power reactor in the first unit of the Novovoronezh nuclear power station (NVNPS) (Fig. 1) as 210 MW (electrical) were the dimensions of the reactor vessel (external diameter, 3.8 m; length, ~12 m), which are almost limiting with respect to the conditions of manufacture of the vessel at the factory and its transportation by railway. The requirement for the railway transportation of the reactor and other plants of VVÉR reactor installations is to be decided by the future development and improvement of technical characteristics.

Translated from Atomnaya Énergiya, Vol. 43, No. 5, pp. 325-336, November, 1977.

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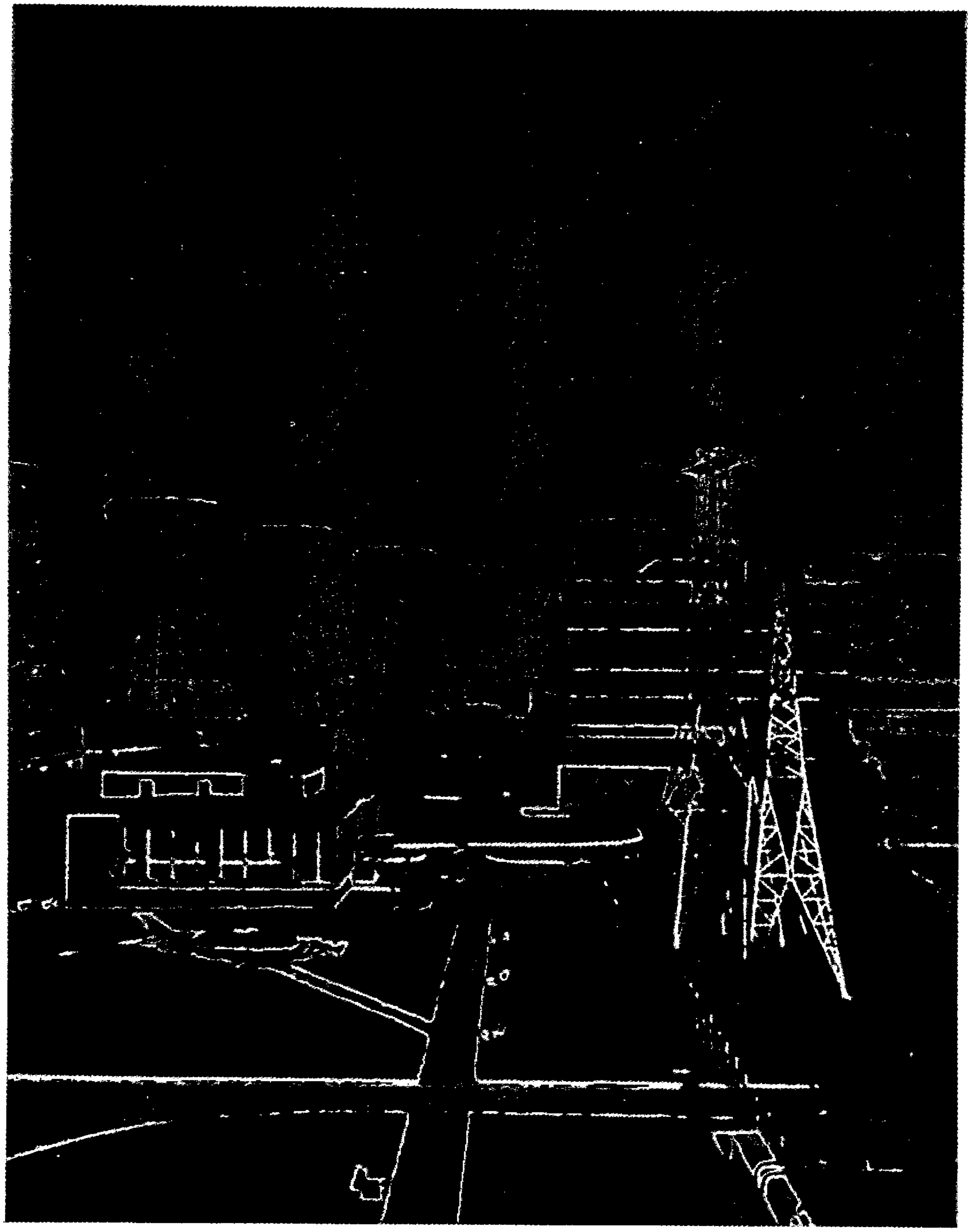


Fig. 1. Fiftieth-Anniversary-of-the-USSR Novovoronezh nuclear power station. The first Soviet nuclear power station with power reactors of the containment-vessel type. Four units are currently operating with a total nominal capacity of 1455 MW.

A high-strength low-alloy steel is used for the reactor vessels which ensures its minimum dimensions and mass.

The first unit of the Novovoronezh nuclear power station came into operation in Sept. 1964 (in the startup year this unit was the most powerful nuclear power station in the world).

The next qualitative stage (second generation) of development of the VVÉR is the VVÉR-440 facility. It forms the basis of the first large-scale series of nuclear power stations, as their satisfactory economic indices have made these stations completely competitive with conventional-fueled stations in almost all the regions of the European part of the Soviet Union. These reactors are extensively used also in certain foreign

TABLE 1. Principal Characteristics of Nuclear Power Stations with Water-Cooled—Water-Moderated Reactors

Characteristic	VVÉR-210	VVÉR-365	VVÉR-440	VVÉR-1000
Year of startup	1964	1969	1971	—
Capacity, MW:				
electrical	3 × 70	5 × 73	2 × 220	2 × 500
thermal	760	1320	1375	3000
Efficiency (net), %	27,6	27,6	32	33
Steam pressure before turbine, kgf/cm²	29	29	44	60
Pressure in primary circuit, kgf/cm²	100	105	125	160
No. of loops	6	8	6	4
Water flow rate through reactor, m³/h	36500	49500	39000	76000
Internal diameter of reactor vessel, mm	3560	3560	3560	4070
Steam production from one steam generator, tons/h	230	325	425	1469
Uranium charge, tons	38	40	42	66
Average fuel burnup in stationary cycle, MW·day/kg U	13	27	28,6	26—40
Av. specific power intensity of core, kW/liter	46	80	83	111
Av. power intensity of fuel, kW/kg U	10,5	33	33	45,5
Specific flow rate of coolant, tons/h·MW	38	30	25	19
Water temp. at reactor inlet, °C	250	250	269	289
Av. heating in reactor, °C	19	25	31	35
Specific capital costs,* rubles/kW (electrical)	406	273	200	—
Cost of electric power†, kopecks/kW·h	0,95 (0,788)	0,743 (0,569)	0,643 (0,584)	0,573

*Data according to the Novovoronezh nuclear power station.

† Design data according to the units of the Novovoronezh nuclear power station. Actual data during 1976 are shown in parentheses.

TABLE 2. Technicoeconomic Characteristics of the Novovoronezh Nuclear Power Station

Year	Electric power output kW·h·10⁶	Utilization factor of installed capacity	Cost of electric power sent out, kopecks/kW·h
1972	5413,4	0,607	0,81
1973	8647,7	0,68	0,752
1974	9664,1	0,76	0,644
1975	9138,1	0,717	0,641
1976	9750,8	0,763	0,632

countries; they are operating and continue to be constructed in the German Democratic Republic, in Bulgaria, Finland, and are being constructed in Czechoslovakia, Hungary, and other countries.

The third generation is the VVÉR-1000, which is being constructed at the Novovoronezh nuclear power station as the fifth unit and is the pilot reactor in a new series.

The reactor facility of the second unit of the Novovoronezh nuclear power station, the VVÉR-365 (Table 1), occupies an intermediate position between the first and second generations. All the principal improvements of the core, developed for a commercial reactor of average capacity and then used in the VVÉR-440, have been incorporated in it. In order not to delay the practical confirmation of these decisions, the nuclear power station and the reactor facility were completed mainly with the plant developed for the first unit and calculated on almost the same parameters. The operating characteristics of the Novovoronezh nuclear power station are given in Table 2.

In order to solve the problem associated with the special features of the physical processes of VVÉR, an experimental base was constructed, extensive experimental data were accumulated, and on this basis improved computational programs were worked out.

The small neutron diffusion lengths predetermine the sharp nonuniformity of the power distribution near inhomogeneities of the fuel lattices. The large temperature and power effects of reactivity, the large excess of reactivity on burnup, require high efficiencies for the control systems. The possibilities of significant

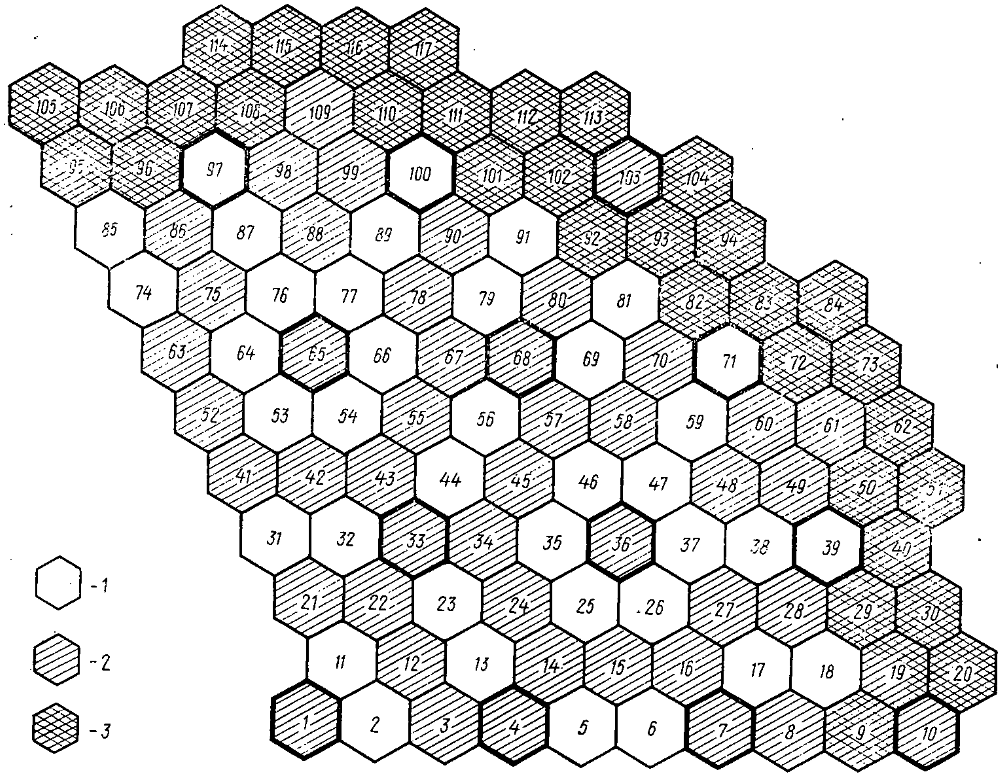


Fig. 2. Recorder chart of the first fuel charges of the reactor of the first unit at the "Kozlodu" (Bulgarian Peoples' Republic) nuclear power station, and the second unit of the "Nord" (German Democratic Republic) nuclear power station. 1) Enrichment 1.6%; 102 working cassettes and 12 control and safety cassettes; 2) enrichment 2.4%; 108 working cassettes and 25 control and safety cassettes; 3) enrichment 3.6%; 102 working cassettes and no control and safety cassettes.

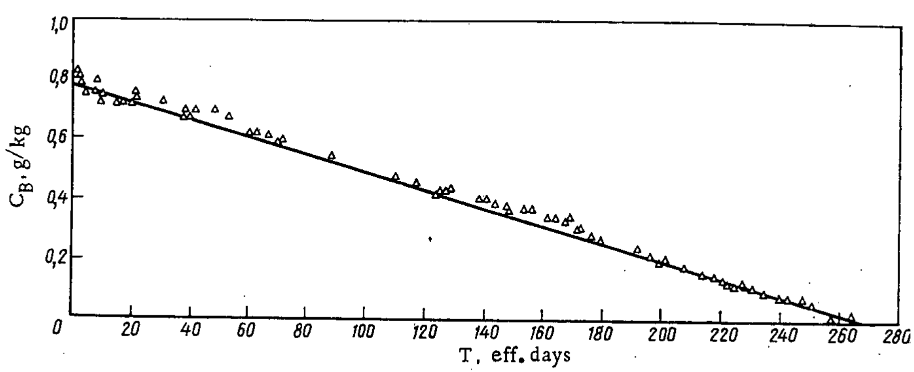


Fig. 3. Change of critical concentration of boron in the coolant during burnup of the first fuel charge of the Novovoronezh nuclear power station fourth unit.

deformations of the power distribution during burnup create the need for the development and use of a complex scheme of interchanging the cassettes in the core during rechargings, and the use of fuel charges with nonuniform composition. All this, in conjunction with the high power intensity of the core, requires great detail and a high accuracy of the neutron-physics calculations. The grouping of part of the core of the VVÉR-440 is shown in Fig. 2. In the I. V. Kurchatov Institute of Atomic Energy, a large system of mathematical programs was set up for investigating VVÉR reactors. The system consists of six interrelated complexes. The first and second complexes represent the fundamental libraries of computed nuclear data and program systems for the preparation of multigroup libraries of cross sections, worked out both in the Institute of Atomic Energy and in the framework of the Temporary International Scientific-Research Staff of Scientists of the Member-Countries of the CMEA. The libraries of nuclear data of the Soviet Nuclear Data Center and other countries have been used.

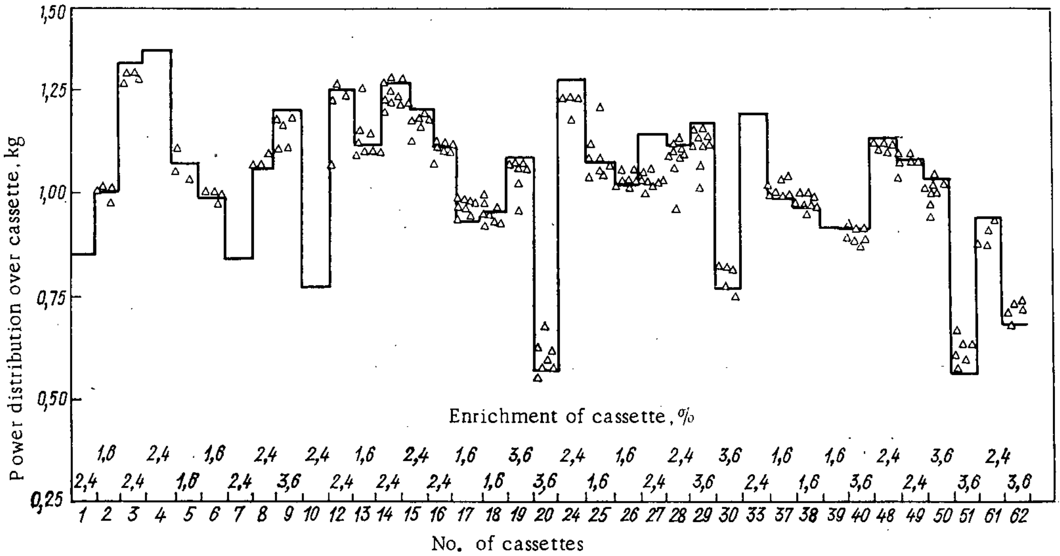


Fig. 4. Power distribution over the cassettes of the first fuel charge of the reactor of the first unit of the Kolsk nuclear power station: ———) calculation by the BIPR-4 program; △) thermocouple readings; $N_p = 100\%$; T = 100 eff. days; $H_{ARK-VI} = 150$ cm; $C_{H_2BO_3} = 3$ g/liter; six operating primary circulation pumps.

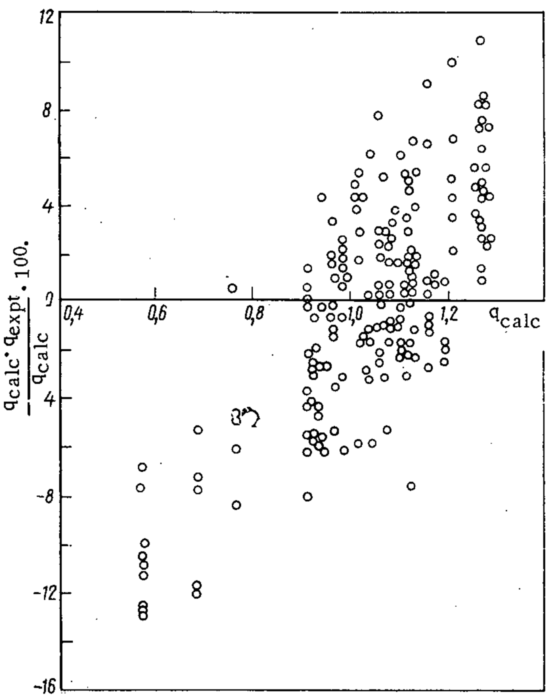


Fig. 5. Deviation of calculated distribution of power over the cassettes of the core, from the measured distribution (first charge of the reactor of the second unit of the "Kozludu" nuclear power station with 100% capacity and charge duration of 80 eff. days): ○) experiment.

The third and fourth programs are complexes for the preparation of libraries of small-group cross sections and precision multigroup programs (one-dimensional and two-dimensional in P_N-approximation) and Monte Carlo programs (one-, two-, and three-dimensional) for detailed investigations of procedural problems.

The fifth complex comprises programs for design calculations on the physics of VVÉR. Here, programs on the preparation of effective cross sections of the lattices in different states of burnup, with a different isotopic composition, etc. (100 points with respect to energy in the thermal group, 3 in the epithermal group, and 120 chains of fission products and heavy isotopes) are coordinated: programs for calculating boundary conditions, spectrum details in the epithermal energy region, and for calculating the power distribution inside the cassettes; one- and two-group three-dimensional physics model of the VVÉR with the reproduction

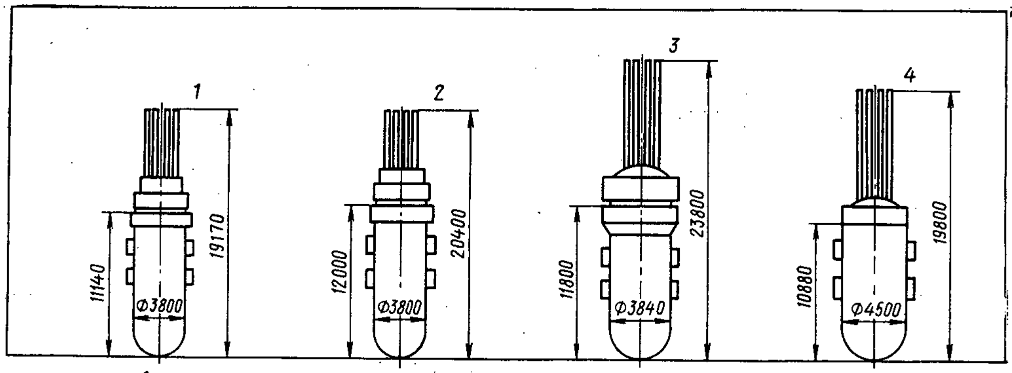


Fig. 6. VVÉR development: 1) VVÉR-210 (mass of vessel and reactor: 223 and 470 tons); 2) VVÉR-365 (241, 523 tons); 3) VVÉR-440 (200, 573 tons); 4) VVÉR-1000 (304, 730 tons).

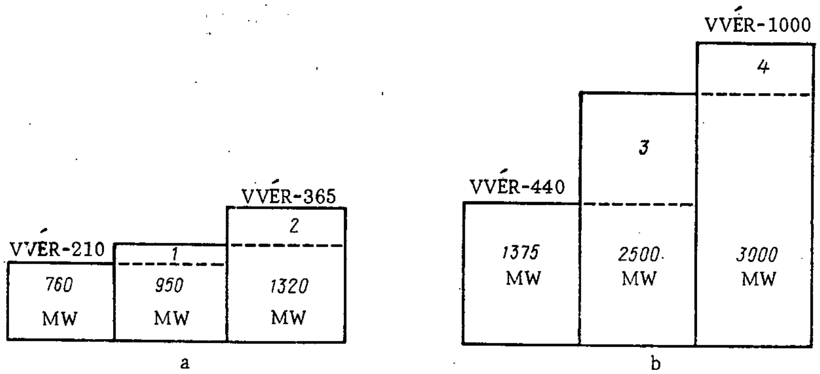


Fig. 7. Role of the different factors in increasing the thermal capacity of the VVÉR in the second (a) and third (b) generations: 1, 3) Coolant flow rate, length of fuel elements (for 3, margins up to limiting values); 2) nonuniformity of heat release; 4) design of pumps (inertialess).

of all burnup control cycles of the fuel charges, taking into account the effects of feedback, etc.; a program for calculating the reactivity coefficients and other parameters of the point kinetics of the reactor; two-dimensional four-group 7000 program, making it possible to analyze the power distribution over all the fuel elements of the reactor. The sixth program is concerned with programs for thermophysical calculations, calculations of safety and reliability, optimization of fuel cycles and the processing of operating data, etc.

The constant improvement of the system of programs developed ensures the following computational accuracy for the main operating characteristics of VVÉR, %:

Coeff. of power nonuniformity of the cassettes	5
Coeff. of power nonuniformity of the fuel elements inside the cassettes	10
Duration of operating period of charge	5
Eff. of control rods	10
Eff. of liquid absorber	5
Temp. coeff. of reactivity	$0.5 \cdot 10^{-4}/°C$
Power coeff. of reactivity	10

These values permit the reliable functioning of all operating reactors and, with sufficient certainty, predict the physical characteristics of reactors in the design stages.

With the bringing into operation of the variants of the program being developed at present, the errors of the calculations are reduced to errors of measurement with regular measuring systems. Figures 3-5 show the calculated and experimental data defining the accuracy of the computational programs.

The most important factor in the development of VVÉR is the increase of the unit capacity of a reactor assembly. Figure 6 schematically shows the power series of VVÉR: 210, 365, 440, and 1000 MW (electrical).

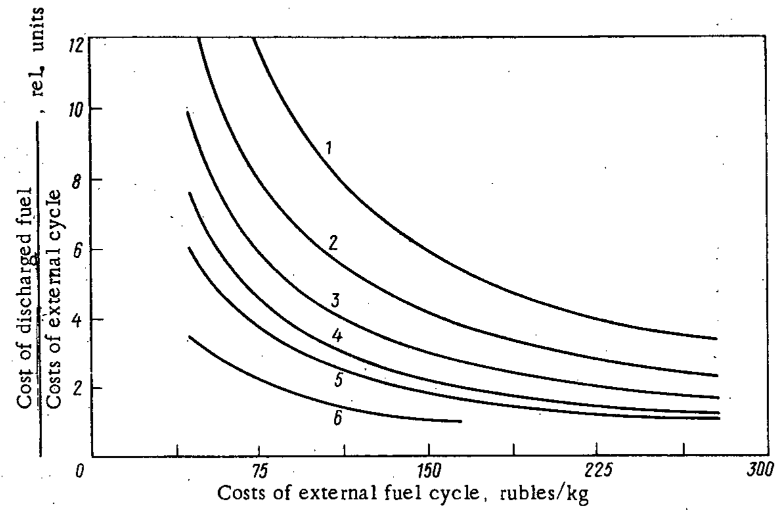


Fig. 8. Earning capacity of the chemical reprocessing of the VVÉR spent fuel: 1, 3) Th_M; 2, 3) UO_2; 4, 6) U_M; for alternatives 3, 5, and 6 the cost of UF_6 and separative work units is 36 and 41 rubles/kg; for alternatives 1, 2, and 4 it is 100 and 54 rubles/kg, respectively.

TABLE 3. Principal Technicoeconomic Indices of Possible Fuel Cycles for the VVÉR-100 (for φ = 0.8)

Index	Dioxide	Dioxide	Metal	Metallic thorium
	open cycle	closed cycles		
Fuel comp. (rel.)	1	0,79	0,66	0,53
Natural uranium consump., kg/MW (elect.) · yr	216	120	77	59
Separative work, kg separative work units/MW(elect.) · yr	132	79	38	52

The development of the VVÉR with increased unit capacity has taken place over two successive stages of increase of the parameters, and has ensured a corresponding increase of the thermodynamic efficiency of the steam power cycle.

The direct means for increasing the thermal capacity of the VVÉR are reduction of the nonuniformity of heat release in the core, increase of the coolant flow rate through the core and of the total length and surface area of the fuel elements; reduction of the margin between the operating and maximum permissible values of the parameters. The increase of the reactor capacity due to the reduction of the nonuniformity of the heat release was accomplished by conversion from the core of the first unit to the core of the second unit of the Novovoronezh nuclear power station.

The VVÉR reactors are oriented mainly on a cycle of three partial rechargings per running period. The design duration of the working period between rechargings, in the second and third generations of reactors, amounts to 6500-7000 eff. hours, which ensures a good utilization factor of the installed capacity of the station and permits recharging to be carried out once per year, in the spring—summer period which is suitable for power generation systems. In all the reactors, with the exception of the VVÉR-210, this recharging cycle is used, in which the fresh fuel is always charged into the periphery of the core, with its subsequent relocation in the central region of the core (remaining there during two operating periods), resulting in its discharge. This ensures the necessary compensation of the heat release in the core; moreover, the difference between the average and maximum burnup of the discharged fuel is reduced. An additional reduction of the nonuniformity of the heat release was provided by the introduction on the second unit of the Novovoronezh nuclear power station of a control of the burnup by the power, by means of a solution of boric acid in the primary coolant.

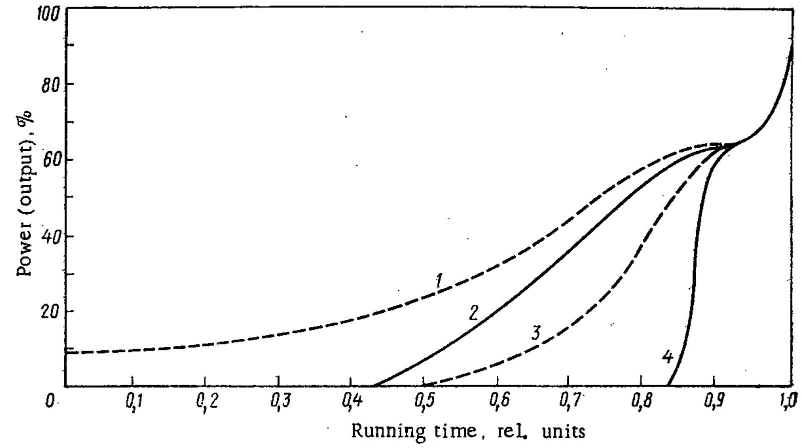


Fig. 9. Adjustability of a unit with VVÉR-440, taking account of the presence of a controlling group of cassettes in the core (the change of water temperature in the case of a power tripout is taken into account): -----) holding at power N and at any instant of rise to nominal is ensured; ——) holding at power N and output at nominal power ensured, if the halt is not more than 1 h; 1, 2) $(1/C_B)(dC_B/dt) = 0.05\ h^{-1}$ in the case of an unlimited and 1-h halt, respectively; 3, 4) $(1/C_B)(dC_B/dt) = 0.20\ h^{-1}$ in the case of an unlimited and 1-h halt, respectively.

An increase of the coolant flow rate requires the development of a new, more powerful pumping plant (or requires the number of loops in the reactor installation to be increased, which is economically disadvantageous). The flow rate of the water in the core and in the reactor vessel restricts the increase of the coolant flow rate in the VVÉR.

The overall length of the fuel elements can be increased by increasing the total charge of uranium in the core or by a reduction of the diameter of the fuel elements. The principle of the factory manufacture of the reactor vessel and its railroad transportation limits the increase of the total uranium charge, and therefore from the very start the necessity arose for developing fuel elements with a somewhat smaller diameter than was accepted in foreign practice. The use of fuel elements of relatively small diameter ensured, in the first stages of development of the VVÉR, a reserve in the linear thermal loading, which makes possible a considerable increase of the power intensity of the fuel in the core volume.

Three factors should be distinguished in the reduction of the margin between the operating and limiting values of the parameters. The first is related to the depth of our knowledge about the processes taking place in the reactor. An example of the improvement of investigations and of the increase of the reliability of the results is a study of a heat exchange crisis in the core. The decisive factor here is the maximum approximation of the study conditions to the natural operating conditions.

In the I. V. Kurchatov Institute of Atomic Energy, the conditions of origination of a heat exchange crisis in water cooled reactors have been studied for a long time, and the results of other organizations as applicable to the cores which have been developed have been generalized. Based on experiments conducted under conditions as close as possible to the operating conditions of the VVÉR, a numerical formula was found for predicting a heat exchange crisis in the fuel element bundles of the VVÉR, with a mean-square error of 5.5% (for uniform heat release). This formula satisfactorily describes the experimental data with respect to bundles with nonuniform heat release. Further systematization of the experiments on the conditions of heat exchange in the fuel element bundles creates a good basis for overloading of the thermal cycle of cores by disposing of the margins in ignorance.

The second factor for reducing the margins is related to the reliability of the knowledge of the parameters achieved in the reactor, and conditioned by improvement of the measurement systems, in the first phase of intrareactor measurements.

The third factor is the increase of reliability of heat removal systems, which will enable the identical effect to be produced in an increase of thermal capacity with a small increase of cost or a large increase of capacity with a fixed increase of cost. This factor made the greatest contribution to the increase of capacity

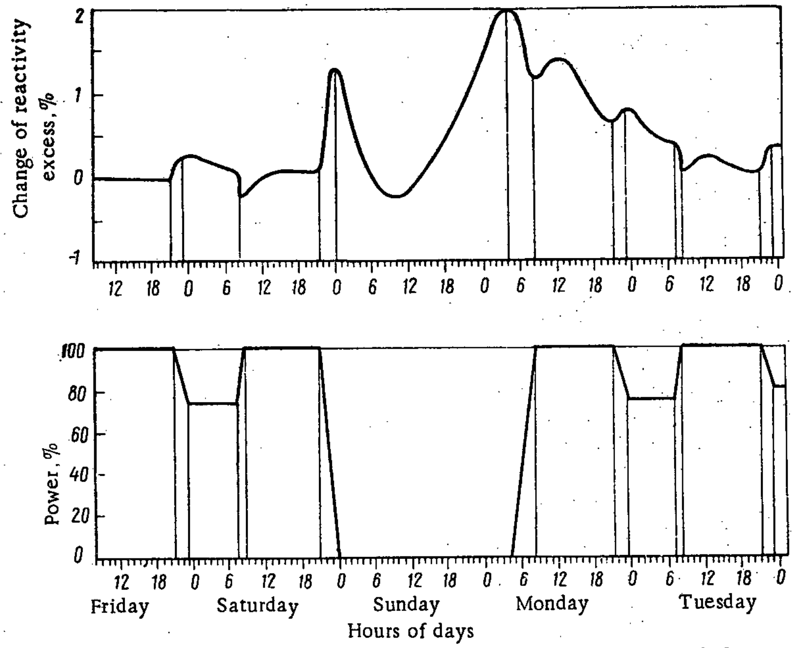


Fig. 10. Change of reactivity excess during operation of the reactor in accordance with a characteristic weekly load chart.

during conversion from the VVÉR-440 to the VVÉR-1000, due to the replacement of the glandless primary circulation pumps, which have a low inertia, by pumps with external electric motors and with forced small flows, equipped with special flywheels. Figure 7 shows the role of the various factors in increasing the power output.

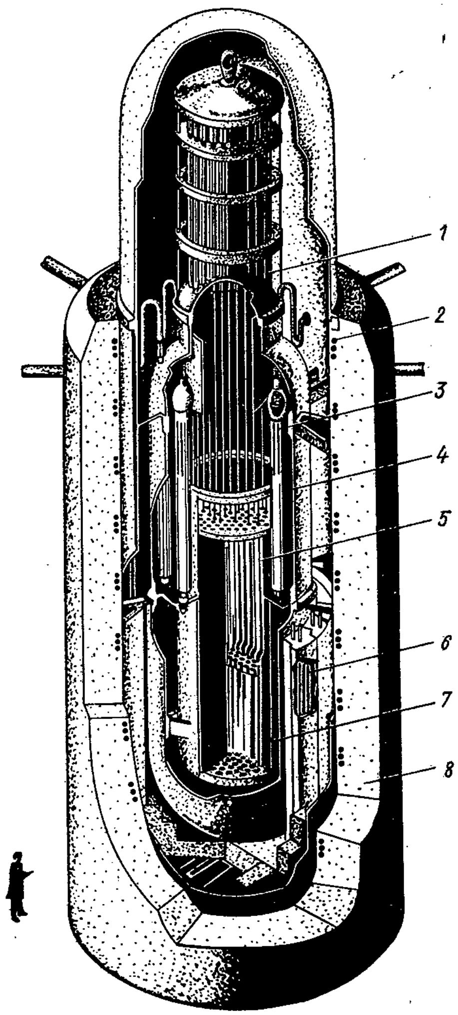
Important factors, which ensure an increase of power, are increase of the fuel burnup and its charge in the reactor core. It should be emphasized that railroad transportation of the reactor vessel facilitates the construction of nuclear power stations in many regions of the Soviet union and other countries, but in practice it limits the unit capacity of a VVÉR reactor to 1000 MW (electrical), as it is difficult to distribute a large load of uranium. In practice, from the point of view of railroad transportation, the maximum dimensions of the vessel have been reached already for the VVÉR-210.

In order to ensure the optimum grouping and structural solutions, an increase of the capacity of the unit must be accompanied by an increase of the capacity and output of the main plant. The tendency to reduce the specific capital costs in commercial installations and to utilize rationally the production capacities of factory-manufacturers reduces to the fact that the tendency to enlargement of the plant acquires an independent nature outside of the relation with increased capacity of the unit.

The plant enlargement, unconditionally, requires a significant increase of its reliability and this, in its turn, leads to a reconsideration of certain principles established in the basic layout of the facility, the grouping of the plant and maintenance methods. The experience built up on all VVER installations takes into account this increase of reliability and confirms the advisability of simplifying the layout decisions and the basic configuration.

In the VVÉR-210, each circulation loop of the primary circuit is installed in an insulated compartment, which allows maintenance of the loop facilities to be carried out during operation of the reactor with the other loops. In the second unit of the Novovoronezh nuclear power station, two loops were installed in each compartment, and in the VVÉR-440 the plant of all six loops was installed in a single compartment. A similar arrangement of the four loops is provided in the VVÉR-1000.

In considering a capacity of ≈ 2000 MW as a further step in construction of VVÉR reactors, several routes can be mentioned for achieving this capacity. The thermal capacity of 6000-6300 MW necessary for this unit can be obtained from the core installed in the vessel of the VVÉR-1000. For this, it will be necessary to use fuel elements with a diameter of 6 mm, to carry out fuel recharging twice per year and, for the purpose of maintaining the total feed of cooling water in the reactor, as accepted in the VVÉR-1000 design, it will be necessary to generate saturated steam at a pressure of 47 kgf/cm^2. An increase of the thermal load of the fuel elements can be provided by heat exchange intensification.



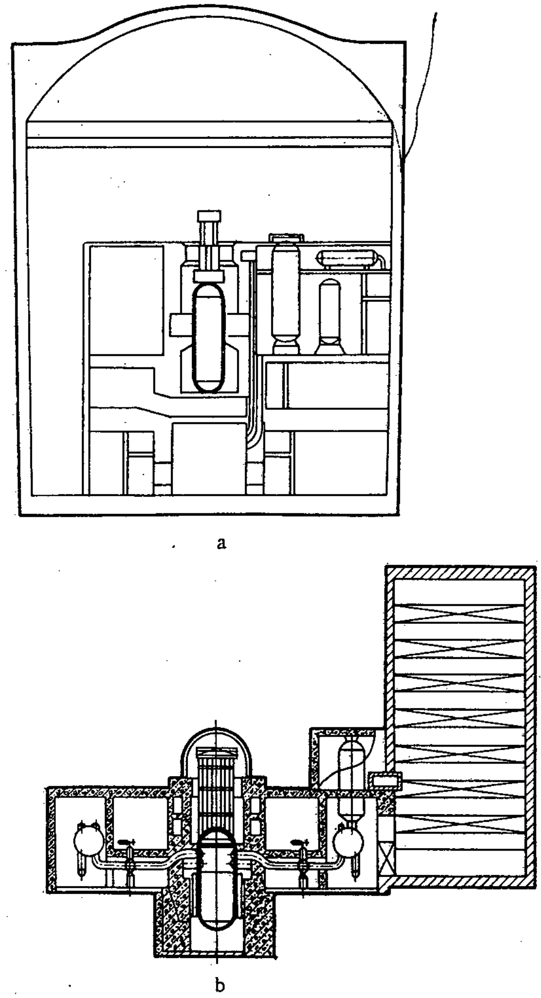


Fig. 11 Fig. 12

Fig. 11. Grouping of a reactor for an automatic thermal power station: 1) absorber servo; 2) cooling for concrete; 3) intermediate heat exchanger; 4) reactor vessel; 5) tractive section; 6) cooling for iron water shield; 7) core; 8) concrete shell.

Fig. 12. Emergency localization systems for $P_{max} = 4$ (a) and $P_{max} = 1$ kgf/cm^2 (b).

More preferable at the present time is the route which retains all the principle solutions with respect to the core and the limiting parameters of the VVÉR-1000. For this unit it will be necessary to construct a reactor vessel with an inside diameter of 5.7 m, in which a fuel charge of about 150 tons can be disposed (scaled to uranium metal); new and more powerful steam generators must be developed. By retaining the VVÉR-1000 pumps, a layout is possible with two pumps per steam generator, and by retaining the layout with 4 main circulating pumps, pumps with an output of $\sim 40{,}000$ m^3/h will have to be built.

The discarding of railroad transportation of the reactor vessel leads to the necessity for using either other means of transportation or undertaking assembly of the steel vessel in situ at the nuclear power station, for which multilayer vessels are more suited, as they do not require heat treatment after assembly. The in situ construction of reactor vessels is possible not only of steel, but also of prestressed reinforced concrete, and its use will remove completely the problem of power limitation of the VVÉR. The impossibility of large cracks appearing in consequence of embrittlement failure makes the concrete reactor vessel safer than a metal reactor vessel. However, in a complex with a concrete vessel, it is advisable to use a VVÉR of the boiling type, as the pressure in it is twice as low as in the nonboiling type. In order to construct high-capacity boiling VVÉR, the large positive experience in the operation of the experimental VK-50 boiling reactor in the Nuclear Reactor Scientific-Research Institute (NIIAR) has been useful. The large volume of investigations on this reactor has enabled a higher specific power density to be achieved with natural circulation in the core, than in the reactor of the first unit of the Novovoronezh nuclear power station. For a boiling reactor with a capacity of 2000 MW, a reactor vessel is required with a diameter of 12-14 m and a height of 21-23 m, and with a concrete wall thickness of ~ 5 m. Steam generators and other complex plants are not required in this case.

Together with the solution of the problem of further increasing the unit capacity of the nuclear power station reactors, the problem of improving the efficiency indices of the fuel and reducing the specific consumption of natural uranium will be urgent for a long time. Up to the present time, the principal route for improving the efficiency of the fuel cycle of VVÉR reactors is to increase the fuel burnup. In the reactors of the second unit of the Novovoronezh nuclear power station, and in the VVÉR-440, the design burnup of ~28,000 MW·days/ton already has been achieved on the average for the discharged fuel, with a maximum burnup of >40,000 MW·days/ton for the fuel elements.

The core of the VVÉR-1000 is oriented on an average burnup of 40,000 MW·days/ton and with a maximum burnup on the average of 44,000 MW·days/ton for the fuel element; for this, it is necessary to supply fuel enriched to 4.4% on recharging. Until mass experience in the operation of fuel at this burnup has been acquired, the core can operate with makeup fuel enriched to 3.3%, with a burnup of 27,000 MW·days/ton.

The currently used cycle of three partial rechargings with an annual cycle cannot be assumed to be the best for the future development of VVÉR reactors. In the first place, an increase in the number of partial rechargings per running period of the fuel from three to six will allow the fuel component cost of electric power to be reduced further by 6-8%; secondly, in the development of large-scale nuclear power generation, it is not obligatory and even inconvenient to "be bound" to recharging to a summer minimum of power requirement. The most important requirement remains the provision of the maximum load factor of the station, in any case, to reduce to a minimum the time lost. On the other hand, the duration of the recharging operations, even for those reactor designs which are calculated on once-per-year recharging, including cooling and heating up, unsealing and resealing the reactor, all assembly and disassembly operations and replacement of spent cassettes with the recharging machine, amounts to about 7-8 days.

This experience allows it to be supposed that the most promising route for increasing the recharging frequency is the improvement and simplification of the existing methods with a reduction, as the immediate problem, of the duration of shutdown of the unit for the next fuel recharging to 1 week. Under these conditions, the changeover of all VVÉR reactors to a regime with six partial rechargings per running period with a 6-month cycle can be planned. In relation to the specific consumption of natural uranium in the stationary regime, this will be equivalent in practice to a limiting regime of continuous fuel recharging. There are other ways also for improving the fuel cycle factors of the VVÉR.

As the irradiated VVÉR fuel contains a large number of fissile isotopes, chemical reprocessing of the fuel and the return of the reprocessed fuel to the cycle permit the consumption of natural uranium to be reduced by 40-50%, and the necessary capacities of the separating factories by 40%.

The development and utilization in the cores of VVÉR of a denser fuel than uranium dioxide (e.g., corrosion-resistant compositions based on metallic uranium), would permit additionally a reduction of the consumption of natural uranium by 35%, the capacity of the separation plant by 50%, and the fuel component by 15%. The data on the use of thorium fuel cycles in VVÉR (Table 3) are close to these results.

Figure 8 shows the variation range of the cost of production of the fuel cycle, for which the expediency of chemical reprocessing of VVÉR fuel is retained, for a different cost of natural uranium and its enrichment process.

The introduction into the power generation system of a country of a large number of nuclear power stations requires a reconsideration of their operating cycles. If, until very recently, work on base loads could be provided for nuclear power stations, then in subsequent years they would have to operate as well in the variable part of the load chart. Experience of operating the VVÉR confirms the simplicity of control of reactors and the feasibility of following a variable load. The VVÉR reactors have a very important quality — negative temperature and power coefficients of reactivity and, associated with this, the capability of automatic regulation and automatic limitation of power. Special attention has always been paid to the conservation of this property of VVÉR cores. In particular, after the introduction into the designs of reactor installations of liquid boron control, the choice of the maximum reactivity excess, compensated by a dissolved absorber, has determined the conservation in all operating cycles of a negative reactivity coefficient with respect to the temperature of the coolant.

The dimensions of the VVÉR reactors (by comparison with the neutron migration length) are very large. Under conditions of a strongly flattened neutron field, variations of the reactor power accompanying the movement of the automatic control rods and the change of the spatial power distribution are dangerous due to the onset of xenon instability. However, in the VVÉR the negative power coefficient of reactivity effectively stabilizes the system, significantly displaces the boundary of instability, and also reduces possible fluctuations of the neutron field in transient processes. The VVÉR-440 reactor is completely free from xenon power "wobbles." The radial dimensions of the VVÉR-1000 also guarantee stability of the neutron field in the horizontal cross section of the core. The increase of the height of the core to 3.5 m in the VVÉR-1000 required a special system of control with the high-level field, which must preserve the axial nonuniformity of the heat release in transient processes within the scope of the permissible values.

In solving the problem of building an adjustable power generating facility, in addition to the basic stability of the reactor, other limiting factors are important, relative to which the units of nuclear power stations with the VVÉR reactors are undoubtedly promising. Here should be mentioned the problems of ensuring the necessary reactivity excess for overcoming the nonstationary poisoning by ^{135}Xe, the deviations of the parameters of the facility from the nominal values, and problems associated with the duration and complexity of the technological operations during changes of operating conditions.

The solution of the first problem in the existing VVÉR reactor installations, in principle, is simplified by the large reactivity excess by burnup, which can be used in transition processes during the greater part of the running period of the reactor, without specific deterioration of the planned fuel cycle of the present day installations. As a rule, this is related with the creation of an adjustable reactivity excess (Fig. 9). Good grounds for solving the second problem consist in the small range of change of operating temperature of the primary coolant with a wide variation of loads (not more than 30°C for a 100% power variation), which is characteristic for the VVÉR installations. Cyclic loads in the plant of the primary circuit are not found to be excessive, and the automatic adjustability of the reactor contributes to maintaining the parameters of the facility within safety limits.

The good controllability of the VVÉR reactors is confirmed in practice; further improvement of the control system and of the technological circuits in new designs is directed at increasing the control operability of the facility in the case of frequent changes of the operating conditions. Already in certain designs of the units with commercial VVÉR-440 reactors, complex power change charts with weekly shutdowns and daily partial load sheddings are being adopted (Fig. 10). Based on the VVÉR-1000 facility, a unit with a VVÉR-500 is being constructed, on which increased demands for adjustability have been imposed. The specific problem requiring special attention remains the assurance of the necessary durability of the fuel elements, constantly operating under variable power conditions.

A new important field in which the experience in constructing VVÉR reactors can be used in future years is district heating and heat supply. The first route to the solution of this problem is the construction of automatic thermal power stations (ATÉTs) based on the developed reactor facilities used for nuclear power stations. Changes may involve only the steam-turbine section of the station and the extent to which solutions must be considered, directed at increasing the radiation safety of facilities, in consequence of their proximity to densely populated regions. Here, in time, a predominance of boiling reactors can be provided in reinforced concrete vessels, which was mentioned earlier. In order to use the VVÉR reactors in heat supplies, it will be necessary to build special reactors for nuclear boilers. Specialization in the production of relatively low-temperature heat (150-170°C) will enable the basic plant design to be strongly simplified and reduced in cost, and a facility of increased reliability and safety to be built. It will be most advantageous for such simplification of the VVÉR facilities to use natural circulation and integral plant grouping, to make maximum use of the principles of automatic control of the facilities, etc. Experience in the development, operation, and study of certain generations of the VVÉR and of the cased boiling VK-50 reactor will enable a facility to be constructed which is optimum in its parameters and characteristics. In a three-circuit facility, the pressure of the coolant in the primary circuit cannot exceed 12-16 kgf/cm^2. The manufacture of the plant is simplified considerably. The necessary level of economic efficiency of automatic thermal power stations can be achieved with a single thermal capacity of the units of $\sim$ 500 MW. A general view of a possible reactor facility for an automatic thermal power station is shown in Fig. 11. In any case, the use of reactor facilities for heat supplies demands special attention to further increase their safety.

In ensuring the safety of the VVÉR reactors, the main attention is paid to the quality and control of the plant, and to the establishment of reliability in the course of operation.

The shielding and insurance measures provided for in the plans of the first stations conformed to the restricted scale of the maximum planned emergencies and increased role accepted for these facilities, and removed the factor of remoteness of nuclear power stations from populated locations. The mass spread of nuclear power stations leads to an intensification of the technical measures for the neutralization of potential danger. These demands have been set before the plans for the latest generation of nuclear power stations with VVÉR reactors, including the new group of stations with the VVÉR-440.

Declassified and Approved For Release 2013/03/22 : CIA-RDP10-02196R000700100001-6

An analysis of the nature of the occurrence of dangerous procedures shows that the dangerous consequences of accidents for the nuclear power station itself and for the surrounding population can be prevented by the creation of reliable and strong means of cooling the core. If emergency cooling prevents melting of the fuel, then the rôle of external insurance barriers of the type of leaktight compartments and health-protective zones becomes less marked.

The loss of coolant by rupture of the largest circulating pipeline of the primary circuit is accepted as the critical design emergency in new designs. At present, a system of emergency cooling of the core is being developed, which will permit the expectation that in the event of a dangerous rupture of the largest pipeline (with a diameter of 500 and 850 mm in the VVÉR-440 and VVÉR-1000 respectively) no melting of the fuel will occur and the fuel elements may only partially lose their hermetic sealing.

In addition to this, certain alternative systems for localizing activity to compartments of nuclear power stations have been developed, which are calculated on the possibility of a dangerous rupture of the largest pipeline (Fig. 12). As applicable to the VVÉR-1000, (a) the installation of a leaktight ferroconcrete shell has been specified and (b), for the VVÉR-440 various alternatives have been devised for localizing compartments, operating on the typical grouping of commercial nuclear power stations with a leaktight steam-generator compartment.

Experience in the operation of working nuclear power stations with the VVÉR reactors shows that these stations are safe sources of power, and have no harmful effect on the surroundings and the population. Nevertheless, the extension of the sphere of application of nuclear power plants, their proximity to population bodies, and in particular, the installation of nuclear boilers, inevitably will lead to a further intensification of safety requirements. Experience in the development of the VVÉR reactors shows that the increased demands on safety can be achieved while maintaining the satisfactory economical indexes of these power plants.

DEVELOPMENT OF URANIUM–GRAPHITE CHANNEL REACTORS IN THE USSR*

A. P. Aleksandrov and N. A. Dollezhal'

The steady increase in the demand for electric energy and the problems in the use of organic fuel require an intensive development of nuclear power. One of the most important ways in which the Soviet Union is meeting the nuclear power problem is by the use and development of channel-type power reactors. These are graphite-moderated reactors cooled by boiling water or steam under pressure in vertical fuel channels. Channel reactors were chosen because of their characteristics and Soviet experience in reactor engineering [1-3].

Channel reactors can be refueled without shutting down the reactor. This ensures a high reliability and technical readiness of the reactor and permits the use of various fuel and structural materials and fuel loading systems. Standard structures and components can be widely used in constructing the reactors and increasing their power. Although the branching of the circulation loop is a certain disadvantage of channel reactors, it permits the construction of coolant loops of relatively small volume, and this simplifies the solution of safety problems of such reactors, particularly as their unit power is increased [4].

Experience in the Construction of Channel Reactors. The development of channel reactors began 50 years ago when research and experiments on the construction of the world's first nuclear power plant (NPP) at Obninsk were greatly expanded. The startup of this NPP in 1954 and its successful operation not only demonstrated the possibility of the peaceful use of atomic energy, but also permitted the testing and verification

* The present article is based on papers by V. G. Aden, Yu. M. Bulkin, A. P. Veselkin, V. K. Vikulov, V. P. Volkov, O. M. Glazkov, V. V. Goncharov, I. I. Grozdov, M. B. Egiazarov, I. Ya. Emel'yanov, A. D. Zhirnov, L. R. Kevorkov, A. Ya. Kramerov, S. P. Kuznetsov, E. P. Kunegin, L. I. Luninaya, Yu. M. Mityaev, V. I. Mikhan, K. K. Polushkin, V. V. Postnikov, V. S. Romanenko, A. P. Sirotkin, V. M. Fedulenko, S. M. Feinberg, V. N. Filippov, and R. T. Shapovalov. The authors are sincerely grateful to these colleagues.

Translated from Atomnaya Énergiya, Vol. 43, No. 5, pp. 337-343, November, 1977.

0038-531X/77/4305-0985$07.50 © 1978 Plenum Publishing Corporation

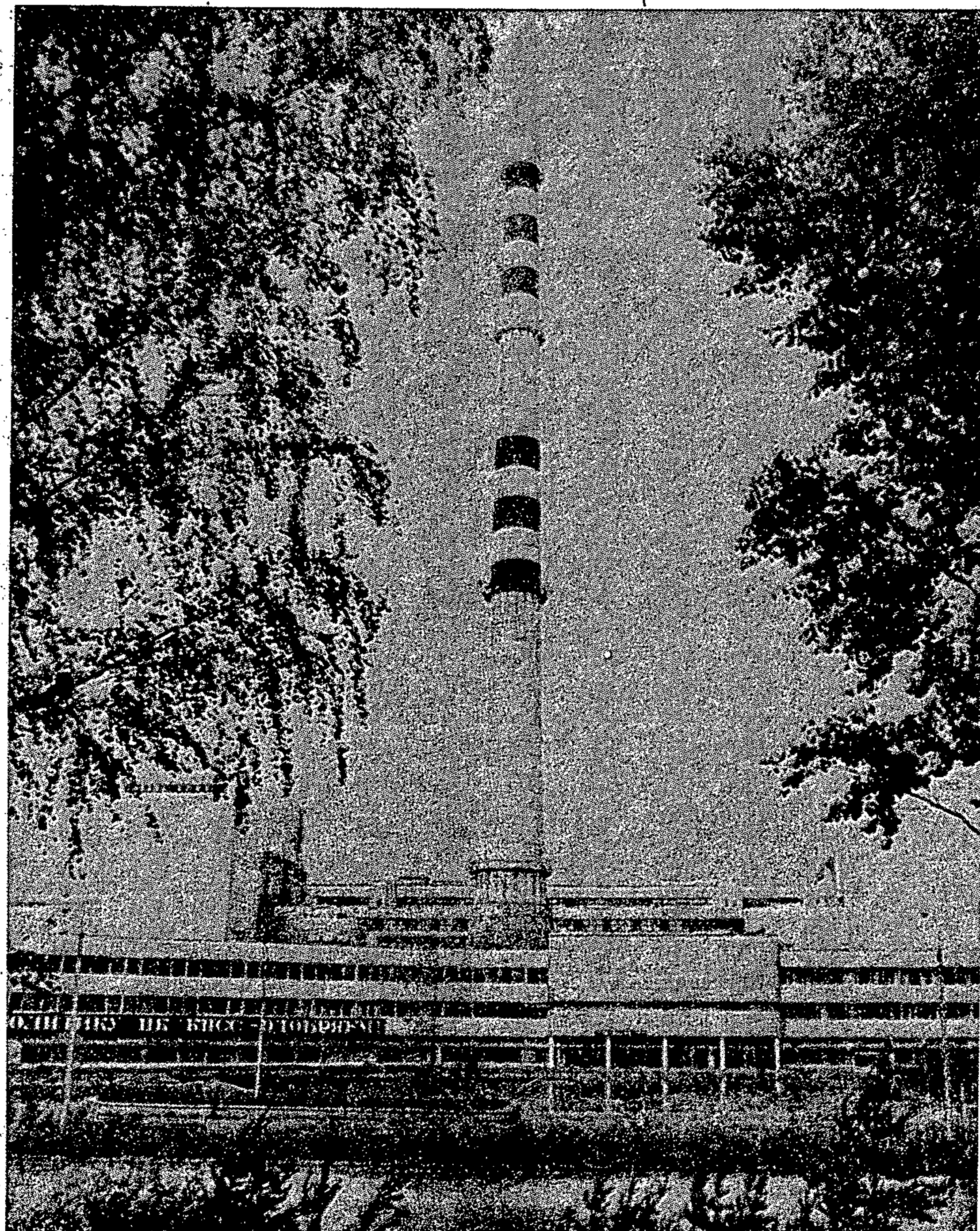


Fig. 1. V. I. Lenin Leningrad nuclear power plant (LNPP). At the present time two units with channel boiling reactors, each rated at 1000 MW, are in operation. Two more units with the same kind of reactors are under construction. The total power of the LNPP will be 4000 MW (electrical).

of a number of technical procedures, many of which were successfully utilized in subsequent reactors. Thus, partial refueling, cooling of parallel fuel channels by boiling water, the nuclear superheating of steam, the proof that graphite stacking can withstand a high temperature, etc. were achieved for the first time at the first NPP. The experience gained in the construction and operation of the first NPP was utilized in the construction of the Siberian NPP which went on line in 1958, and in two reactors of the Beloyarsk NPP (BNPP) which operated at 100 MW from 1963, and at 200 MW from 1967.

The nuclear superheating of steam on an industrial scale was first accomplished at the BNPP by using reactors with stainless steel-clad tubular fuel elements with the direct delivery of the superheated steam to the turbine. The operation of the two BNPP units for more than ten years showed the reliability of the fuel

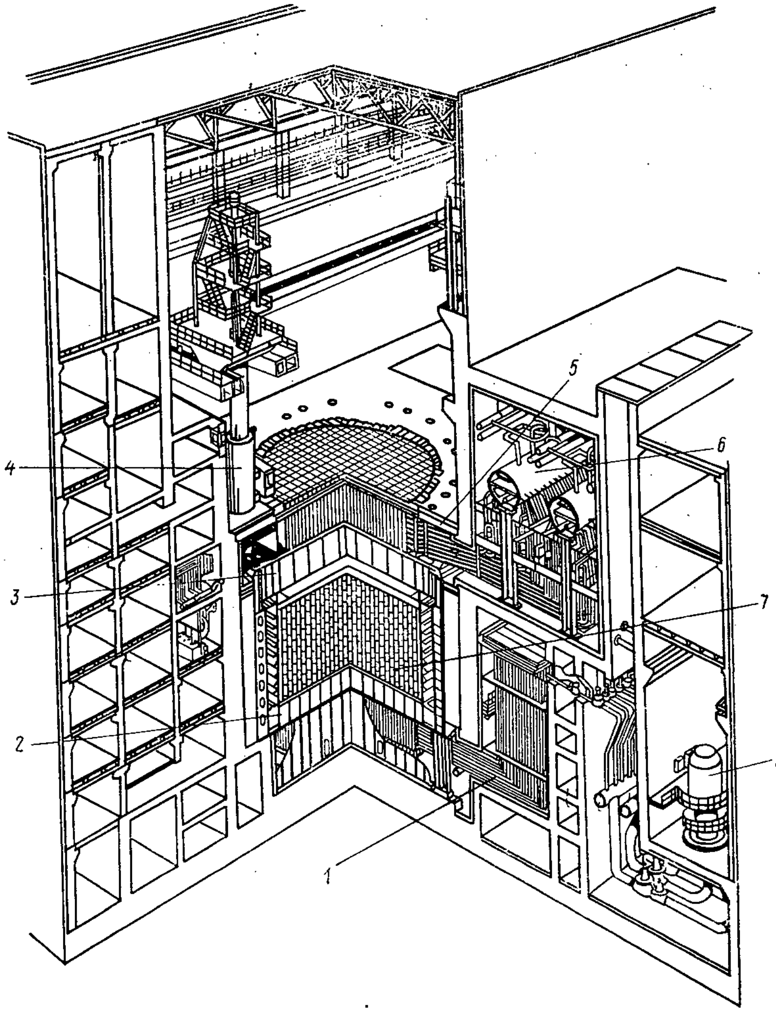


Fig. 2. Arrangement of RBMK-1500 equipment. 1) Water pipelines; 2) lower supporting metal structure; 3) upper biological shield; 4) refueling machine; 5) steam-water pipelines; 6) drum-separator; 7) core; 8) main circulating pump.

channels, including those for the superheating of steam to a very high temperature (550-560°C), with an economically acceptable burnup of uranium (more than 30,000 MW·days/ton). The safe operation of a NPP employing a single-loop thermal circuit with turbines operating with slightly radioactive steam was also confirmed [5, 6].

Channel reactors can also be installed in a NPP of low or intermediate power. For a number of years a plant has been operating successfully at Bilibino in the far northern part of the USSR with four reactors using natural circulation of the coolant. This plant produces both thermal and electric energy in accord with consumer demand, with a capacity factor of 66-68%.

Drawing on the experience in the construction and operation of such reactors and NPP and the successes of the reactor construction industry and technology, a high-power commercial reactor, the RBMK-1000 with a power of 1000 MW (electrical) [2] was designed and developed to make more complete use of the advantages of channel reactors.

The reactor is set in a thin-walled pressure vessel with graphite stacking penetrated by vertical fuel channels in a zirconium alloy. Two fuel assemblies, each containing 18 fuel rods, are placed in tandem in each channel. Heat is removed by boiling water. Separated dry steam is fed to two 500 MW turbines. The RBMK-1000 has practically the same single-loop thermal circuit as the second unit of the BNPP, but the steel fuel rod cladding is replaced by zirconium alloy. This decreases the fuel component cost of electric power, but it requires a certain relative decrease of the coolant parameters.

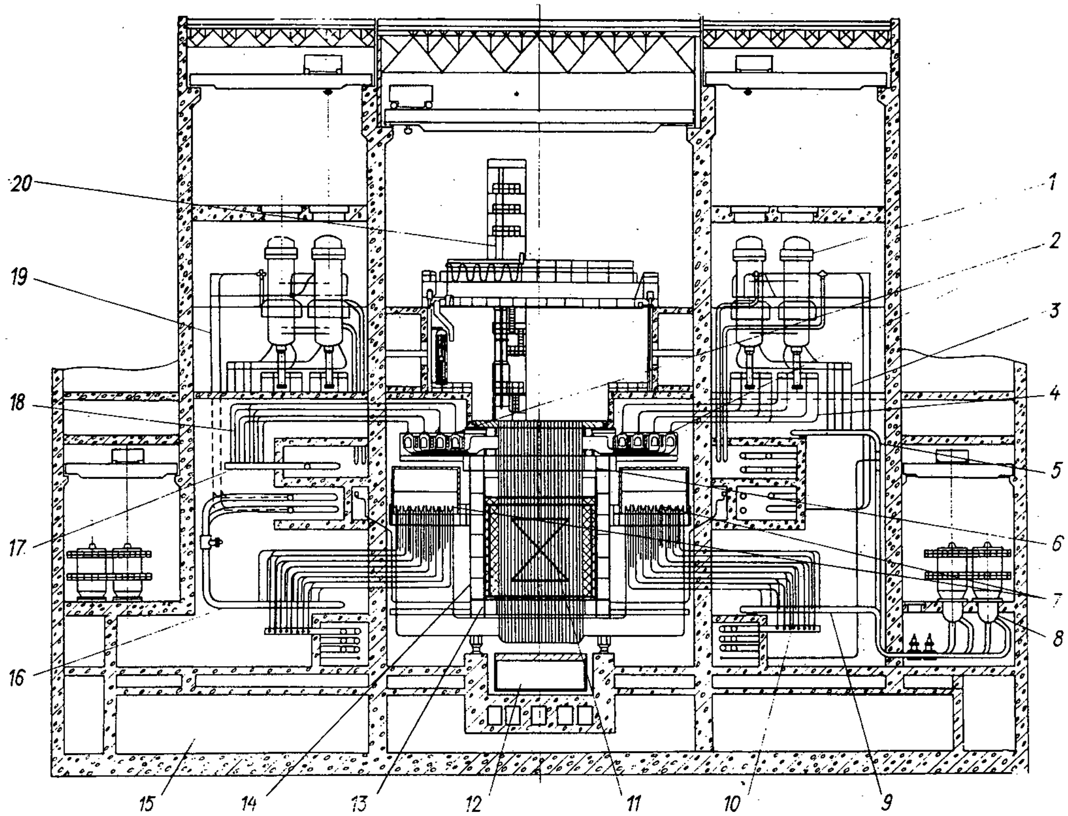


Fig. 3. Cross-sectional view of main RBMKP-2400 equipment: 1) steam separator; 2) sectional headers; 3) upper water pipelines; 4) steam-water pipelines; 5) intake manifold; 6) upper unit; 7) distributing headers; 8) main circulating pump; 9) pressure header; 10) feeder water drum; 11) reactor core; 12) lower repair machine; 13) lower unit; 14) lateral unit; 15) tank-bubbler; 16) saturated steam drum; 17) superheated steam drum; 18) superheated steam pipelines; 19) saturated steam drum; 18) superheated steam pipelines; 19) saturated steam pipelines; 20) refueling machine.

The RBMK-1000 reactors went on line and were successfully operated at the V. I. Lenin Leningrad nuclear power plant (LNPP) (Fig. 1) and at the Kursk and Chernobyl plants. They are being built at the Smolensk NPP, and are planned for a number of other plants. The experience with startup adjustment experiments and the operation of the first two RBMK-1000 reactors confirmed the validity of the basic technical solutions, permitted the development of reserves and the necessary changes to further increase the reliability and economy of the operating reactors and those under construction.

The first unit of the LNPP went on line in Dec. 1973 and was brought up to the design power of 1000 MW on Feb. 1, 1974, ≈ 10 months after startup. The second unit required 5 months (from July 1975 to Jan. 1976). At the present time both units are operating in accord with demand at 800-900 MW. The first unit was in operation for 65, 70, and 77% of the time in 1974, 1975, and the first nine months of 1976, respectively. The main plant and systems have operated completely satisfactorily during the last period. The zirconium alloys and the hermetic joints of the zirconium and steel pipes stood up well.

The first stage of operation of the pilot unit of the LNPP showed that the uranium—graphite ratio assumed in the design was not quite optimum for reactor control under transient conditions. In subsequent reactors the uranium—graphite ratio will be approximately optimum, and control by the power distribution will be automatic.

In constructing reactors of this type, it is important to be able to predict a 30-yr operating life of the zirconium channels and their replacement without reactor shutdown in case of damage. More than 8 years of experience in the operation of zirconium test channels in the BNPP confirmed the long operating life of these channels. The zirconium tubes of the LNPP fuel channels failed twice, once because of the breakdown of the system for cooling the fuel assembly, and the second time because of excessive vibration of nonrigid structure in the channel. The damaged tubes were replaced within 48 hours by using a special device. This is a normal repair operation.

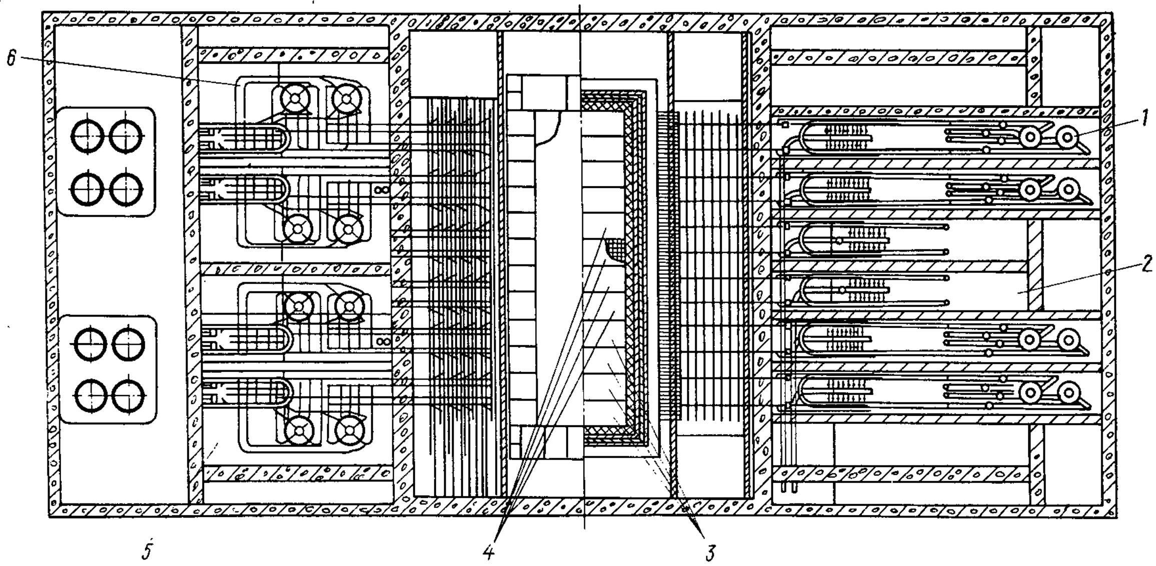


Fig. 4. Plan view of arrangement of main RBMKP-2400 equipment: 1) main circulating pump; 2) superheater loop cell; 3) evaporator sections; 4) superheater sections; 5) evaporator loop cell; 6) steam separator.

During the operation of the LNPP the main units and systems of the RBMK-1000 reactors have performed adequately and have confirmed the possibility and expediency of their further development. Experience in operation at rated power showed that there were certain reserves, in particular with respect to the pressure of the circulating pumps and the temperatures of metal structures and the graphite moderator, which indicated the possibility of increasing the RBMK-1000 power. This was accomplished in the design of the RBMK-1500 [3].

The increase in unit power in the RBMK-1500 (Fig. 2), which is installed at the Ignalina NPP was achieved without increasing the overall size of the RBMK-1000, mainly by improving the construction of the fuel assemblies, with minimum changes in the reactor and the coolant circulation loop. The problem was solved by introducing heat-transfer intensifers into the assemblies, enabling the maximum admissible power to be increased by $\approx$ 50%, and by certain improvements in the separation equipment. Increasing the RBMK-1000 power appreciably decreases the specific capital expenditures for the NPP construction and increases the average specific power which can be removed from the fuel.

The development of channel reactors is related to the main contemporary trends in power engineering including the increase in unit power, the increase in maneuverability requirements, the increase in the parameters of the working medium to lower thermal dumping, and a more extensive standardization and unification of systems and units of electric power plants. The further improvements to channel reactors to satisfy these requirements were embodied in the design of the RBMKP sectional-modular reactor with a power of 2000-2400 MW (electrical). The development of this reactor drew on the experience gained in the construction and operation of the RBMK-1000, the RBMK-1500, and the BNPP reactors with nuclear superheating of steam [3-7].

The RBMKP-2400 employs two 1200 MW turbines patterned after turbines of the same power developed for thermal electric power plants for supercritical steam parameters. A characteristic feature of this reactor is a core in the form of a rectangular parallelepiped. This permits the construction of a reactor from identical sections which can be assembled from factory-produced units. The sizes and masses of the units are such that they can be transported by railroad. Most of the assembly welding work can be done at the factory. This improves the quality of manufacture and control, simplifies and speeds up assembly, and increases the reliability of the operation of the reactor units. An important advantage of the sectional-modular reactor construction is the possibility of increasing the unit power by using identical structural units and equipment developed for a lower power reactor.

The RBMKP-2400 reactor consists of eight evaporator sections, each containing 1920 channels, and four superheater sections, each containing 960 channels. The evaporator section of the reactor is served by 16 main circulating pumps with two-speed electric motors and 16 vertical separators combined into eight independent circulation loops with two separators and two pumps per loop. The arrangement of the main reactor equipment is shown in Figs. 3 and 4. The sectional-modular construction ensures independent circulation loops which increases the maneuverable qualities of the reactor.

The introduction of nuclear superheating of steam into the RBMKP-2400 increases the overall NPP efficiency to 37%, simplifies the evaporator loop by decreasing the number of separators and circulating pumps, significantly decreases the capital investment in engine room equipment by using high-speed (3000 rpm) turbines, decreases the power of the water-supply systems, etc.

We list below the main technical characteristics of the reactors:

Electric power, MW	1000	1500	2400
Thermal power, MW	3200	4800	6500
Steam output, tons/h	5800	8800	9600
Steam parameters upstream of turbines:			
pressure, kgf/cm^2	65	65	65
temperature, °C	280	280	450
Core dimensions, m:			
height	7.0	7.0	7.0
diameter (width and length)	11.8	11.8	7.5 × 27
Number of channels:			
evaporator	1693	1661	1920
superheater	-	-	960
Uranium loading, tons	192	189	293
Uranium enrichment, %	1.8	1.8	1.8/2.3
Average burnup of uranium unloaded from channels, MW • days/kg:			
from evaporator	18.1	18.1	19.4
from superheater	-	-	18.1

The shape of the reactor makes the separate core regions relatively independent of one another. This produces better conditions for control and shaping the power distribution. This is particularly important for high-power reactors since it increases the possibility of using them not only at power close to maximum when the neutron distribution should be flattened, but also at lower power when it is expedient to decrease the power or even to shut down completely a certain region of the reactor to make repairs or to perform other operations or to refuel an operating reactor. The core shape also simplifies nuclear superheating which requires producing and maintaining definite power ratios for the vaporization of water and the superheating of steam.

A characteristic feature of large reactors of any type is the high sensitivity of the neutron distribution to reactivity perturbations of various kinds. This property can have a favorable effect on shaping the power distribution and at the same time can hamper the stable operation of the reactor. Therefore high-power reactors undergoing development always have a very complex in-pile monitoring and control system with a large number of sensors and points of control of the neutron distribution in addition to a system of local automatic power control. In a sectional-modular reactor with a large number of channels and controls symmetrically spaced in the core, it is rather simple to arrange a refueling regime to ensure minimum perturbations of the neutron distribution. The sectional-modular principle of reactor construction enabled solutions of certain problems of reactor reliability and safety to be obtained in a new way as compared with the RBMK-1000 and RBMK-1500 reactors.

By separating the circulation loop into a number of independent loops the maximum diameters of the pipelines can be reduced to 370 mm, and the equipment of each circulation loop can be disposed in separate hermetic cells. This significantly decreases the leakage of coolant in possible loop accidents, helps to localize the consequences of accidents, makes possible equipment maintenance and repair without shutting down the reactor, etc. In combination with the design of feed-water supply to the circulation loops from two independent sources (from the electrically driven feed pumps and from the turbopumps) this solution makes for more reliable core cooling in emergency situations. In this case the power of the emergency core cooling systems is appreciably lower than in the RBMK-1000 and the RBMK-1500.

The RBMKP-2400 design provides means for expanding the possibilities of its operation at varying loads, a feature which is becoming more and more typical of the operation of power systems: in particular the possibility of regulating the power of individual sections, the thermostatic control of metal structures enclosing the reactor, an adequate operative reactivity margin for compensating power-dependent xenon poisoning, etc.

By using relatively small steam-driven turbopumps built into the drum-separator, the pipelines can be shortened appreciably, their diameters decreased, and the conditions for localizing accidents in case of loop rupture improved. Since it is characteristic of a circulation loop with turbopumps that the coolant flow rate is proportional to the reactor steam capacity, the average steam content of the reactor core is practically constant under various operating conditions. This results in the necessary margin before cavitation at the pump intake occurs, and the minimum required amount of water in the separators is decreased. The turbopumps automatically ensure cooling of the reactor under all emergency conditions including complete deenergizing. Experimental turbopumps were bench tested under conditions approaching as closely as possible full-scale operation at a high estimate of working conditions.

Together with experiments on heat-transfer intensifiers in channels to postpone the crisis of boiling and to increase the steam content of the coolant, a "multistage" channel was developed. In this channel the coolant is fed into the inlet of each stage into which the channel is divided, and the straight through cross section does not vary along the channel length. A multistage channel has a high maximum power and a smaller hydraulic resistance, since it is equivalent to several shorter channels connected in parallel. All other things being equal, such a channel has an appreciably higher admissible steam content for a very much smaller variation in density of the coolant along the core height, which is very favorable for reactor physics. Such channels not only increase the power and improve the reactor physics, but can also simplify the circulation loop and improve conditions for the natural circulation of the coolant.

A promising development for channel reactors is a large diameter channel with an increase in the number of fuel elements and power to 8-10 MW. This decreases the capital investment and the cost of a fuel loading, the expenditure for piping and monitoring for technological parameters, fittings, refueling, etc. Problems of cooling the graphite stacking must be solved in converting to such channels.

Improvement of the Fuel Cycle. In operative high-power channel reactors and those being developed, the uranium—plutonium fuel cycle using slightly enriched (1.8-2%) uranium dioxide as fuel is basic. With such enrichment the burnup of discharged uranium is 18-20 MW·days/kg, which ensures a quite acceptable value of the fuel component of the cost of electric power production. The burnup of discharged uranium is obtained in continuous refuelings, since in reactors of the RBMK type the fuel is replaced through channels by a special machine while the reactor is in operation. The burnup can be increased by increasing the initial enrichment. This may give a further saving. Because of the small amount of steel and the absence of water in the superheater channels of the RBMKP a burnup of discharged uranium of 20 MW·days/kg is ensured by a trivial increase of the initial enrichment to 2.0-2.2%.

One of the advantages of channel reactors is the possibility of a relatively simple change to other fuel compositions and fuel cycles. Thus, in these reactors uranium metal fuel can be used instead of uranium dioxide. In this case in achieving burnups of 11-13 MW·days/kg for uranium metal the fuel component of the cost is decreased by about 6%, with a simultaneous decrease of the relative expense of natural uranium by 10%.

CONCLUSIONS

Twenty years experience in the development and operation of uranium—graphite channel reactors has confirmed their high reliability, safety, flexibility of cycle, possibility of constant improvement of construction, and the thermal circuit. At the present stage the use of high-power channel reactors successfully solves the most important national economic problem of the intensive accumulation of nuclear power capacity. Channel reactors correspond to contemporary and contemplated trends in power development.

Channel reactors give the most complete answer to the modern trend to increase unit power, since there are no technical or transportation limitations on increasing their power.

The transition from the construction of the RBMK-1000 to the building of the RBMK-1500 appreciably decreases the specific cost of a NPP. It is particularly important also that this transition requires hardly any increase in the operating staff of the NPP or the builders and manufacturers of equipment. This means that in the transition the productivity of nuclear power production workers is sharply increased.

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PROGRAM AND STATE OF WORK ON FAST REACTORS IN THE SOVIET UNION

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The development of nuclear power generation, based on the fission of heavy nuclei, is a unique real alternative extension of the fuel basis of power generation. At present, nuclear power stations with thermal reactors are competing successfully with conventional power stations; however, further development of nuclear power generation up to scales at which it will ensure a significant contribution to the overall development of electric and thermal power is becoming difficult in consequence of the low utilization efficiency of natural uranium by thermal reactors.

The use of fast breeder-reactors will permit this problem to be solved. Of course, the effect from the use of fast reactors can be revealed in an appreciable way only when their number in the nuclear power generating system approaches the number of thermal reactors. Because of this, the acceleration of the industrial utilization of fast reactors and their incorporation in power generation reduces to a more rational utilization of the nuclear fuel resources.

If, e.g., a period of 35 years is considered and it is assumed that in the first 15 years only thermal reactors are constructed, but after this also fast reactors (in a reasonable ratio), then at the end of the period, the economy in the consumption of natural uranium by comparison with the purely thermal model of development will amount to 50%, and approximately the same economy will be achieved in the divided operation.

As the price of natural uranium is a function of the scale of its demand, expenditure on the fuel cycle of the system must be reduced. Computational investigations show that the application in nuclear power generation of fast reactors with good breeding properties will allow not only the consumption of natural uranium to be much reduced, but will also allow the structure of nuclear power to be optimized more flexibly relative to the minimum of the derived costs, achieving a significant economy by comparison with the use only of thermal reactors.

The aim of fast reactor operation in the Soviet Union is the creation of reliable and competitive power generating fast reactors, capable in the future, together with thermal reactors, of ensuring the most rational utilization of the existing nuclear raw material resources. The program includes the construction of the first power reactors for obtaining experience in design, construction and operation; further development and improvement of the reactor plant, increasing the reliability and enlarging its dimensions; investigations of fuel and structural materials; improvement of the fuel cycle; safety investigations.

REACTOR CONSTRUCTION

Reactors Operating, under Construction, and Planned

The BR-10 Reactor. Reconstruction of the fast BR-5 experimental reactor, built in 1958, was carried out between 1971 and 1973 in order to increase the capacity. Before reconstruction a large amount of research

Translated from Atomnaya Énergiya, Vol. 43, No. 5, pp. 343-351, November, 1977.

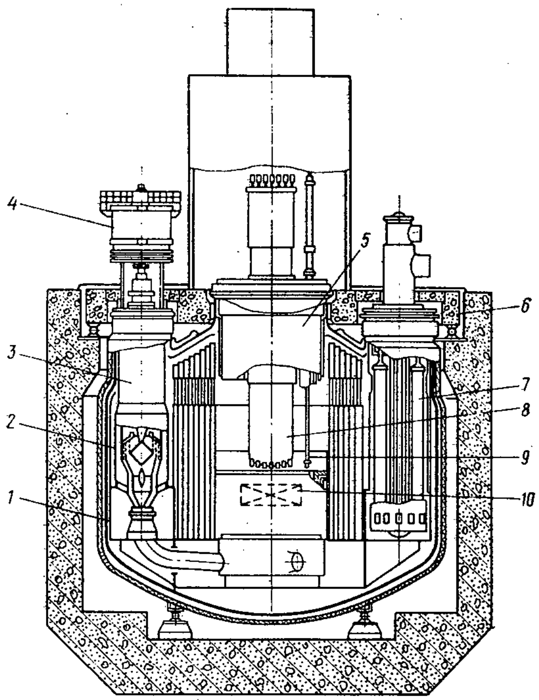

Fig. 1. Vertical section of the BN-1600 reactor (first version): 1) bearing collar; 2) reactor housing; 3) pump; 4) electric motor; 5) rotable plug; 6) upper shield; 7) heat exchanger; 8) central pillar with scram and control system mechanism; 9) recharging mechanism; 10) core.

work was undertaken on it, the most important being the first bulk tests of fuel elements with plutonium dioxide in a cladding of stainless steel and fuel elements with uranium monocarbide. For the first time, the feasibility of operating a reactor with nonleaktight fuel elements was demonstrated. Methods of purifying the sodium loops from radioactive contaminants have been developed. The reconstruction has shortened the time necessary for radiation investigations. Now, the capacity of the reactor amounts to 6.5-7.5 MW, and it is recharged with plutonium dioxide fuel elements, and has achieved a maximum burnup of 7%. Material behavioral studies are also being carried out on the reactor.

The work of reconstruction of the reactor gave much useful information about the technique of maintenance operations, purification of the loops, the behavior of structural materials which are in contact with sodium for a prolonged period, and reactor structures which receive a considerable fluence of high energy neutrons.

BOR-60 Reactor. During the past years since startup (1969), the reactor has demonstrated the reliability and operating stability of all the principal systems and components of the sodium circuits (pumps, control and safety actuators, gating accessories, etc.), which have much in common with the components and parameters of large fast reactors. The principal parameters of the reactor are given below:

Thermal capacity, MW	60
Specific capacity of core, kW/liter	1180
Linear capacity, W/cm	560
Neutron flux density, neutrons/cm^2·sec	$3.7 \cdot 10^{15}$
Max. sodium temp. at outlet from fuel assembly, °C	640
Av. sodium temp. at outlet from reactor, °C	530
Temp. of superheated steam, °C	490

A high power level (52-54 MW) is being maintained in the reactor for a large part of the time, which allows a wide circle of investigations to be carried out efficiently. One of the main assignments of the BOR-60 reactor is the testing of fast reactor fuel elements, along the following principal lines: the testing of experimental fuel elements for the BN-600 reactor, including mixed uranium—plutonium fuel; study of the operating efficiency of fuel elements with vibrocompacted fuel; research work for the choice of the optimum design and technology of fuel elements with carbide fuel; study of the behavior of the structural materials of the fuel element claddings and hexahedral tubes. Another important problem is the testing of sodium—water steam generators of different design. Investigations are also being carried out on safety, the spread of fission products throughout the circuit with the use of unsealed fuel elements, the purification of sodium from fission fragment activity, etc.

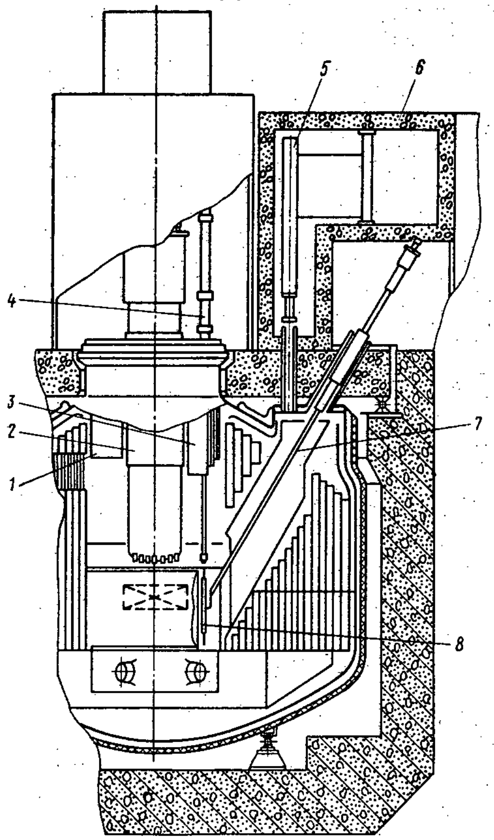

Fig. 2

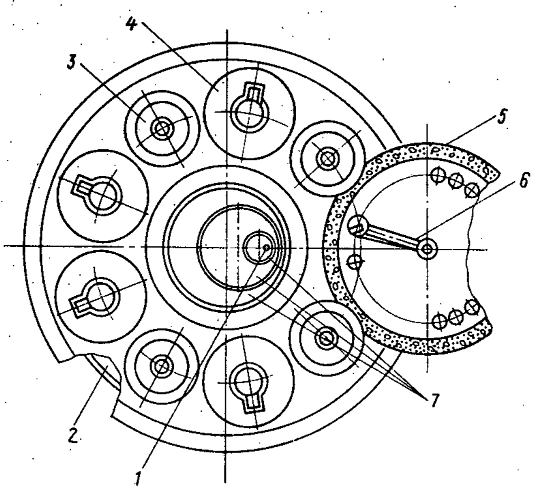

Fig. 3

Fig. 2. Main recharging plant of the BN-1600 reactor: 1-3) large, medium, and small rotatable plugs, respectively; 4, 5) fuel element assembly recharging and transfer mechanism; 6) transfer compartment; 7) elevator; 8) core fuel element assembly.

Fig. 3. Horizontal section of the BN-1600 reactor: 1) recharging mechanism; 2) reactor housing; 3) pump; 4) heat exchanger; 5) transfer compartment; 6) fuel element assembly transfer mechanism; 7) rotatable plug.

Experience in operating the reactor with defective fuel elements (up to 1% of their total number) showed that despite a significant increase of radioactivity due to release of gaseous and solid products which enter the circuit, no serious difficulties arise during discharging operations or replacement of equipment.

BN-350 Reactor. Physical startup of the reactor was achieved on Nov. 29, 1972 [1]. From the instant of the power generation start-up (July 16, 1973), operating experience has shown that all the main plant, except the steam generators, has been operating well. The failures of the steam generators, caused by defects in their manufacture, did not allow the power of the reactor to be raised above 30% until 1975. Before 1975, the steam generators were overhauled and the reactor operated at a power of 15-30%. In 1975, after completion of the overhaul, the power of the reactor was increased to 520 MW (52%). At present, the reactor power amounts to 650 MW (65%), five of the six loops are in operation and one has operated without incident since the time of start-up. Under these conditions, the reactor has provided an electrical capacity of 120 MW and the distillate production has amounted to $\sim 50{,}000$ m^3/day.

The fuel burnup achieved the design value (5%); recharging is carried out in accordance with the chart and no damaged fuel elements have been detected.

Operation of the reactor has enabled valuable operating experience to be built up, including the carrying out of maintenance work with cutting of the main sodium conduits, washing out of the steam generators and the circuits from the interaction products of sodium with water and air. The feasibility has been demonstrated for the construction of industrial nuclear power stations with fast reactors.

BN-600 Reactor. In order to justify the choice of parameters and the design of future fast power reactors, experience in the construction, start-up and operation of this reactor as the third unit of the Beloyarsk nuclear power station will be of particularly great importance. Its electrical and thermal capacities are 600 and 1470 MW, respectively [2].

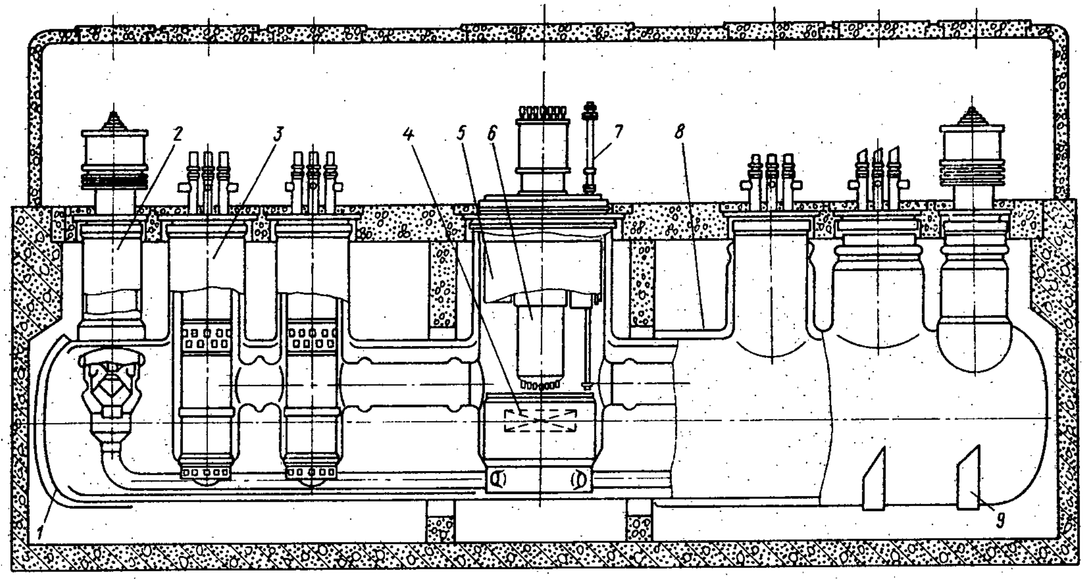

Fig. 4. Vertical section of the BN-1600 reactor (second version): 1) reactor housing; 2) pump; 3) heat exchanger; 4) core; 5) rotating plugs; 6) central pillar with the scram rod and control mechanisms; 7) recharging mechanism; 8) "insurance" casing; 9) reactor housing suspension.

In the course of planning the reactor, various alternatives for the design of the core, control and safety systems, and design of the fuel elements were studied, and also different versions of the arrangement of the primary circuit. In many structural elements, the BN-600 is similar to the BN-350, but an "integral" grouping has been chosen for it, in which the reactor and all the plant of the primary circuit are located in one tank. This allows a clearer idea to be formed about the merits and drawbacks of "integral" grouping by comparison with loop grouping. The BN-600 can be considered as a typical power reactor for nuclear power stations.

Installation of the plant was started in 1974. By the middle of 1975, welding of the reactor housing was mainly completed. Simultaneously with assembly of the reactor housing, installation of the pumps of the secondary circuit and the plant of the conveyor-technological section were carried out. Installation of the internal tank structures is now proceeding.

BN-1600 Reactor. Development is under way of an improved fast reactor with an electrical capacity of 1600 MW, which in future should be the basis of the commercial type units of nuclear power stations. This capacity has been determined from the condition of utilization of two turbogenerators of 800 MW, the delivery of which can be guaranteed even at the present time.

Comparative structural studies were carried out again of different types of grouping of the primary circuit plant with a different plant content of the primary and secondary circuits.

The principal merits of loop grouping are the small dimensions of the reactor housing, the comparative simplicity of its fabrication, favorable conditions of assembly, maintenance and replacement of plant, the absence of internal tank radiation shielding, and also the capability of a wide unification of the basic plant for reactors with different capacities, as a result of variation of the number of heat transfer loops. The disadvantages, which are revealed most clearly during the development of high capacity reactors, are the necessity for autocompensation in the divided, large diameter pipelines ($\sim$1.5 m) of the primary circuit, requiring their considerable extension, which reduces the reliability of the circuit, increases its volume and the area of the corresponding compartments. Moreover, the "cutting off" of a pipeline with large diameter by means of a nonreturn gate is very difficult. The "insurance" casings complicate the construction of the pipelines considerably. The electrical heating system of the pipelines, drainage and other auxiliary communications systems become more cumbersome and complex.

Integral-type groupings were considered in two versions. In the first version (Figs. 1-3), the heat exchangers, pumps, core and coolant of the primary circuit are enclosed in a single large-diameter vertical tank. This design, in essence, is a development of the BN-600 design. The reactor housing is a cylindrical tank with an elliptical bottom and a conical upper section. The low excess pressure ($\sim$0.4 kgf/cm^2) makes it possible to manufacture a large diameter housing ($\sim$18 m) with a small wall thickness (40-50 mm).

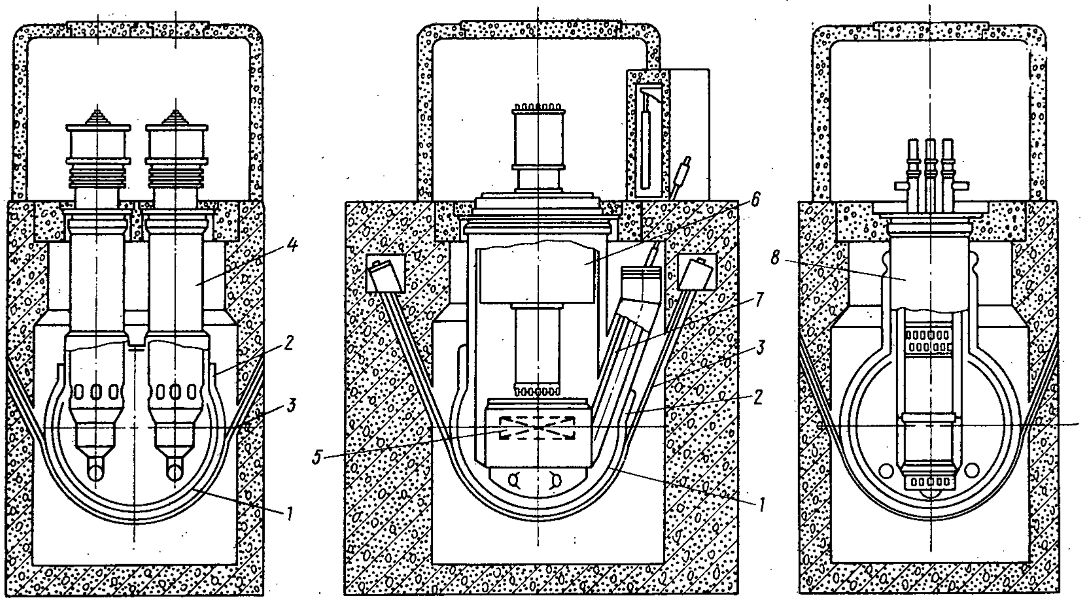

Fig. 5. Sections of the BN-1600 reactor through the main plant: 1) reactor housing; 2) "insurance" casing; 3) reactor suspension; 4) pump; 5) core; 6) rotatable plugs; 7) elevator; 8) heat exchanger.

The merits of integral grouping in a vertical tank are: succession of the grouping with the circuit-grouped solutions of the BN-600 reactor; uniform distribution of the coolant between the heat exchangers with relatively short routings, which, other conditionss being equal, facilitate the development of natural circulation in the primary circuit in the case of shutdown of the forced circulation; mounting of the reactor plant on a rigid structure in the form of a bearing collar of large mass, which is favorable from the point of view of vibration stability.

The second alternative (Figs. 4 and 5) is characterized by an "eschelon" arrangement of the core and of the main primary circuit plant in a single horizontal cylindrical tank. All the primary circuit plant — heat exchangers, pumps, core with bearing chamber — is arranged in vertical chimneys with a free coolant level. The core is mounted on a structure which bears on the upper fixed shield of the reactor stack. Inside the reactor tank, by means of a "hot box" joined to the central shell with the walls of the heat exchangers, the "hot" section of the coolant circuit is formed. The coolant from the heat exchangers is returned through nonreturn pipelines by the primary circuit pumps. An annular gap over the whole length of the tank, in which part of the "cold" coolant flows from the heat exchangers to the pumps, is provided for cooling the reactor tank. This integral grouping has a number of advantages. For example, the diameter of the main tank is commensurate with the diameter of the tanks of the loop version ($\approx$ 7.5-8 m), there is no internal tank radiation shielding, being replaced with a layer of sodium located between the core and the heat exchangers. This grouping allows reactors to be constructed with a wide range of power, by means of a set of a specified number of pumps and heat exchanger sections. The final choice of the type of integral grouping (in a horizontal or vertical tank) must be made after an intensive study of the project.

The principal characteristics of fast power reactors are given below:

	BN-350	BN-600	BN-1600
Electrical capacity, MW	150*	600	1600
Thermal capacity, MW	1000	1470	4000
Net eff., %	35	41	40
Diam. and height of core, cm	150/106	205/75	330/110
Max. neutron flux density, neutron/cm^2·sec	$0.8\cdot10^{16}$	$1\cdot10^{16}$	$1\cdot10^{16}$
Max. depth of burnup, %	5	10	10
Max. specific core density, kW/liter	730	840	710
Sodium temp. at reactor outlet, °C	500	550	530-550
Steam pressure, kgf/cm^2	50	140	140
Steam temp., °C	435	505	490-510

* Plus 5000 ton/h of distilled water.

Declassified and Approved For Release 2013/03/22 : CIA-RDP10-02196R000700100001-6

Equipment

The development and construction of reliable equipment for the sodium circuits are important problems, with which in the first place the circulatory pumps and steam generators of the sodium—water type are concerned.

Mechanical centrifugal pumps of the vertical type have been chosen as the main circulatory pumps for fast reactors in the Soviet Union. The principal characteristics of fast reactor pumps are given below [3]:

	BR-5*	BOR-60	BN-350	BN-600
Capacity, m^3/h	150	600†; 830	3200; 3700	9400; 8000
Pressure head, m	40	85; 60	110; 68	89; 58
Temp. of pumped medium, °C	450	300-500	300; 270	410; 340

* With reconstruction of the reactor to BR-10 (1971), the mechanical pumps were replaced by electromagnetic pumps.
† The first number refers to the primary circuit and the second number to the secondary circuit.

In the BOR-60 and BN-600 reactors, a hydrostatic bearing is used, and the BR-5 and BN-350 use no bearing in the coolant. In order to leak-seal the inside cavity in the BR-5, a hermetic electric pump is used, and in the other reactors a mechanical seal is used, and an electric motor installed in the servicing zone. The pumps of the BOR-60, BN-350, and BN-600 reactors passed all the complex of experimental investigations on water and sodium test-rigs. The operation of the BOR-60 and BN-350 reactors has demonstrated the faultless operation of the pumps.

During 1972-1975 eight incidents of leakage into the steam generators occurred. In all the emergencies the safety system of the steam generators operated normally. The system for the emergency discharge of the sodium and water interaction products functioned faultlessly. The system for detecting water leakage into the sodium now being used was found to be insufficiently efficient. The failures of the steam generators were caused by defects in their manufacture, and showed that serious attention should be paid to the technology of manufacture in the development of this component, as well as to the design of high-speed leakage monitoring systems for water into the sodium.

Endurance tests were carried out successfully over 20,000 h under operating conditions of the straight-through spiral steam generator in the BOR-60 reactor. It has now been dismantled and in its place a straight-tube model of the BN-600 type steam generator will be installed. During operation of the spiral steam generator and after completion of the tests, the state of the 1Kh2M steel heat-transfer tubes was checked. The operating endurance of the materials was far from being exhausted, and no significant corrosion replacements were necessary in either the water or the sodium circuits. Tests of a modular steam generator, manufactured in Czechoslovakia, also have been completed successfully on the second loop of the BOR-60 reactor.

The modular-type Czechoslovakian steam generator will be installed in one of the BN-350 loops. A modular straight-through steam generator of the sectional type with eight sections is planned for the BN-600; each section consists of three modules: an economizer-evaporator, steam superheater, and an intermediate steam superheater [4]. Modular steam generators, although somewhat more cumbersome in comparison with block steam generators, nevertheless possess great advantages during the elimination of emergency situations caused by leakage from the tubes; they have a better efficiency and therefore, probably, are preferable in any case in the initial period of construction of fast reactors.

From the point of view of confining damage of the pipe bundle in the event of water leakage into the sodium, there is interest in the concept of the "inverse" steam generator, in which the water (steam) is in the casing and the sodium flows in the pipelines. The feasibility of building this steam generator is being studied on models. The structural materials used for the manufacture of sodium—water steam generators must be investigated for resistance to corrosion of a different kind, which may occur under operating conditions. Because of this, investigations of such materials as 1Kh2M type low-alloy perlite steels, austenitic steels and high-nickel alloys have been specified.

Reactor Physics

At present, the physics of fast reactors with a uranium charge is being studied more completely.

The BOR-60, BN-350, and BN-600 reactors have been studied by simulation on the BFS physics test-rigs. The startup and investigation of the BN-350 reactor permitted the results of calculations and model studies to be compared with the actual reactor characteristics. The comparison confirmed mainly the sufficient accuracy of the physical calculations, although certain refinements of the methods of calculation and the values of the physical constants assumed appeared to be necessary, in particular for the low energy part of the spectrum.

Calculations of the physical characteristics of plutonium-fueled reactors are considerably less reliable. For the purpose of increasing the accuracy of the calculation of the physical characteristics of fast breeder-reactors, an extensive program is being carried out on the assessment of the nuclear constants and their verification by experimental data.

Critical plutonium assemblies are being studied on the BFS test-rigs in the spectral ranges which are characteristic for large fast reactors. Numerical methods also are being developed for the detailed calculation of the ideal geometry of a reactor and for the refined calculation of the neutron spectrum.

Other Types of Fast Breeder-Reactors

The development of fast reactors, cooled with sodium, is the principal trend in the Soviet Union in the program for constructing breeder-reactors, but the possibilities of using gaseous coolants – helium and the dissociating gas N_2O_4 – also are being studied. These investigations have been brought about by the tendency to provide an alternative solution; to simplify the technological circuit of the facility and, possibly, to gain somewhat in the breeding factor.

It is still too early to speak about the advantages which gas-cooled reactors may have in comparison with sodium-cooled reactors, in consequence of the large differences in the extent of the development of one or the other.

WORK ON THE DOMAIN OF THE FUEL CYCLE

Fuel and Fuel Elements

The achievement of the advantages which fast breeder-reactors have in comparison with thermal reactors owing to the high breeding factor is possible only with a well-organized closed fuel cycle. In its turn, this is related to the choice and development of a type of fuel which will permit a high breeding factor and a quite deep burnup to be obtained. At present, considerable experience has been accumulated in the operation of fuel elements based on uranium and plutonium dioxides.

In the BOR-60 reactor, 230 fuel element assemblies (~8500 fuel elements) have been irradiated, of which 158 (~5800 fuel elements) were irradiated to a burnup in excess of 10%. In some experimental fuel-element assemblies the burnup reached 15%. The overall state of the fuel elements is satisfactory. Only individual cases have been observed of fuel-element failure. A change of diameter in the case of 11% burnup (1.7-3%) occurs predominantly because of swelling of the steel, which amounted to 6-7% (fluence $7.2 \cdot 10^{22}$ neutrons/cm^2, $E > 0.1$ MeV). In the BOR-60 and BN-350 reactors a special program is being carried out to study the radiation swelling of structural materials.

A study has commenced on the BN-350 of irradiated fuel-element assemblies and fuel elements. Oxide fuel is being considered as the primary and immediate object for study and improvement.

As an alternative version, carbide, nitride, and also metallic fuel are being considered, which have better nuclear-physical and thermophysical properties than oxide fuel. In the first place, tests of carbide fuel elements are proposed. The first bulk investigation of uranium monocarbide fuel elements was undertaken on the BR-5 reactor to a burnup of 6%, which showed that carbide fuel elements in a 0Kh16N15M3B steel cladding will endure this burnup. Now, further investigations of this fuel are being carried out.

The complexity of the production of carbide with a stoichiometric composition, possessing excellent radiation stability, requires a more detailed study of the properties of nonstoichiometric carbides with a simpler production technology. In order to achieve the advantages of carbide fuel (in comparison with oxide fuel), such as a considerably higher thermal conductivity, the maximum permissible loads and temperatures will require a more thorough study, and the cladding-fuel contact layer (helium or sodium) will need to be chosen.

Tests of carbide fuel elements have been carried out on the BOR-60 reactor. Recent investigations [5] of 38 uranium monocarbide (with carbon content 4.68-5.1 wt.%) fuel elements (2 fuel assemblies), irradiated to a burnup of 3 and 7% with a maximum linear power of 550 W/cm, maximum cladding temperature 650°C and a temperature at the center of 1100-1200°C for a helium contact layer and 900-950°C for sodium—potassium, showed that the fuel elements were in excellent condition and suitable for further operation.

Carbide fuel can give (according to estimates) better economic indices than oxide fuel, if it will be possible to ensure an average linear power of 600-700 W/cm and a maximum depth of burnup of not less than 10%.

Positive experience has also been built-up on the BOR-60 in the operation of fuel elements with carbide fuel up to a burnup of 10% and with a temperature at the center of 1100-1200°C, and with a maximum linear power of 650 W/cm. Tests are being carried out in more intense conditions with a maximum linear power of 1000-1200 W/cm.

The capability of achieving deep burnups, operating reliability of the fuel elements and, to a known degree, achieving good breeding factors, depends on the properties of the fuel element and fuel assembly cladding materials as well as on the properties of the fuel. The next few years should give the answer to the question of the limiting permissible (from the point of view of swelling of the steel) burnups with the use of the existing grades of steels, and also about the routes for reducing swelling in order to ensure high fuel burnups.

Fuel Regeneration

The external part of the fuel cycle includes cooling of the spent fuel, its transportation, reprocessing, manufacture and intermediate storage of the fuel element assemblies in a storehouse before loading into the reactor. The total duration of the external fuel cycle has a considerable effect on the rate of breeding of fissile material. Taking account of such a criterion as the doubling time, it may be said for example that a shortening of the time of the external cycle from 2 years to 1 year is equivalent to an increase of the breeding factor by ≈ 0.10.

The principal contribution to the duration of the external fuel cycle may be (and at present is) the time required for cooling of the fuel after discharge from the reactor, for decay of radioactivity and residual heat release.

In the initial stage of construction and introduction into the energy generation system of fast reactors, when their number is small in comparison with the number of thermal reactors, the fuel turn-around time of fast reactors still does not play a marked role in the balance of the production and requirement of plutonium. However, in the future a long duration of the external fuel cycle can limit the rate of introduction of fast reactors and can make difficult the achievement of the possibilities of extended regeneration. Therefore, investigations and the development of regeneration methods for the fuel of fast reactors with a shortened time of cooling are being worked on, in order that the overall time of the external fuel cycle could be shortened to one year or less. It should be noted that the achievement of this time of the external fuel cycle is a complex technical problem. Water processes are being considered as the main courses, and pyrochemical processes as an alternative.

Water reprocessing of fast-reactor fuel is based on the Pyrex-process, with cooling of the fuel after discharge from the reactor over a period of 6 months [6]. A combined fluoride—salt fuel regeneration technology, with cooling for 3-6 months after discharging, is also being investigated.

An experimental facility for the investigation and development of fuel regeneration has been built in the Scientific-Research Institute of Nuclear Reactors, where the first regeneration of irradiated uranium fuel from the fuel assemblies of the BOR-60 reactor was carried out, with a specific radioactivity close to the radioactivity of the fuel from commercial fast reactors (average burnup ~10%, cooling time 3-6 months). The feasibility was demonstrated of regeneration, by this method, of uranium fuel with a high activity (up to $\approx 10^5$ Ci/kg). In this case, a total purification factor from γ-activity of more than 10^6 has been achieved, and the irreversible losses amounted to 0.1-0.5%.

Work in the field of the external fuel cycle includes the development of reliable and operationally suitable containers for the transportation of irradiated fuel to the reprocessing plants, methods of stripping the fuel elements before reprocessing, and also investigations on the removal of radioactive wastes. The program of work on fast reactors in the Soviet Union covers a wide field of investigations, design-structural developments, and the building and operation of fast reactors, which are necessary for solving the problem of creating large-scale nuclear power generation with fast reactors.

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CONTEMPORARY STATE OF RESEARCH ON CONTROLLED THERMONUCLEAR FUSION

V. A. Chuyanov

Research on controlled thermonuclear fusion began in the Soviet Union in 1950 right after the first successes in mastering the energy of uranium fission. Considered as a possible source for obtaining fissionable materials and tritium, it was started under conditions of secrecy, but already in 1956 research on controlled thermonuclear reactions were declassified on the initiative of Soviet scientists, and I. V. Kurchatov made the first open communication about this in the same year at Harwell (Great Britain). Since that time research on controlled fusion has been one of the areas of fruitful international scientific cooperation.

Research on thermonuclear reactions started out very promisingly. In 1952 neutron emission — the first sign of a fusion reaction — was detected in experiments with high-current discharges in deuterium. However, a more careful investigation conducted under the direction of L. A. Artsimovich led to the conclusion of an accelerator source for the observed neutrons: not all the plasma was heated up in the discharges to high temperature, and only a small fraction of the particles was accelerated in the strong electric fields arising as a result of the developing instabilities. Upon increasing the energy contribution, unstable oscillations rapidly carried heat out to the walls of the discharge chambers, thereby preventing the heating of the plasma to the desired temperature. The instabilities forced the investigators to give up their hopes for quickly mastering the energy of thermonuclear fusion and to examine closely for 15 years the creation of a new science — the physics of a high-temperature plasma.

The distinguishing characteristic of the first decade of research on the physics of a high-temperature plasma was the weak correlation between theory and experiment. A plasma did not want to exist under theoretical equilibrium configurations of various kinds and always threw its own energy out to the walls in a time shorter by many orders of magnitude than the diffusion and thermal conductivity times calculated by the theoreticians. Only a deep analysis of the stability of the plasma performed by the theoreticians of the M. A. Leontovich school and the experimental mastering of stabilization methods permitted reconciling theory with experiment and obtaining an extended plasma containment time. M. S. Ioffe with his coworkers was the first to succeed in the stabilization of one of the instabilities — the flute instability — with the help of a magnetic field whose strength increases in all directions from the plasma boundary. A communication about these experiments made in 1961 at the International Conference in Salzburg became an important milestone on the path of development of thermonuclear research. Since that time special conductors with a current creating fields having a "magnetic well" in the magnetic traps have been "Ioffe bars" all over the world.

Translated from Atomnaya Énergiya, Vol. 43, No. 5, pp. 351-359, November, 1977.

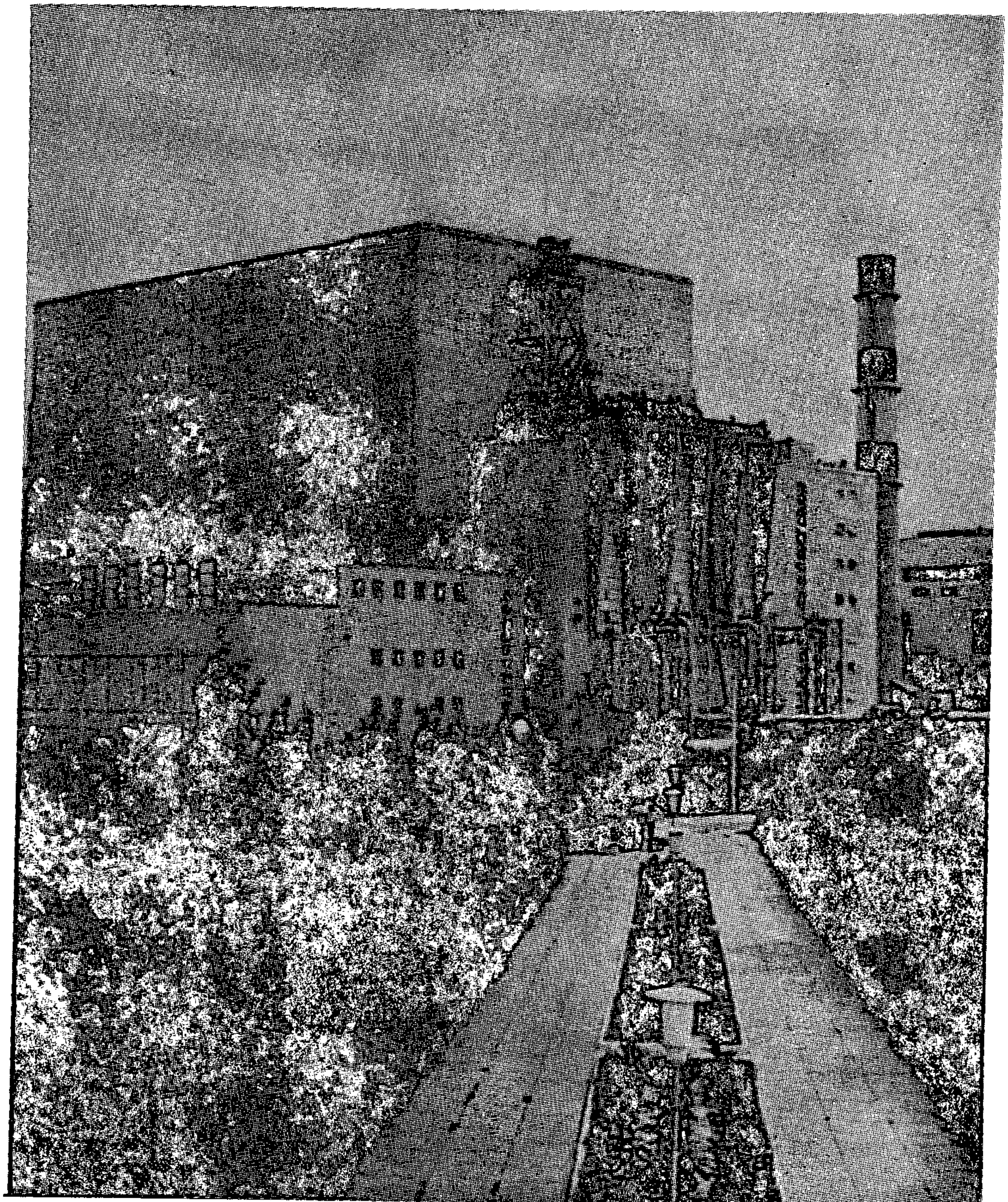

Fig. 1. The first Soviet industrial fast reactor, BN-350. It is used to provide electricity and distilled water for the city of Shevchenko and the manufacturing division of Mangyshlak. The nominal parameters are: thermal capacity — 1 million kW, electrical — 150,000 kW, and distilled water production — 120,000 m^3/day.

At present it is clear that in the descriptive expression of B. B. Kadomtsev the sea of instabilities which appeared boundless at one time has turned out to be only a lake, whose boundaries and even depth are already sufficiently investigated. Such complete and accurate physical models of the instabilities were obtained in a number of cases that it proved possible to use automatic control methods for the stabilization of these instabilities. The first successful experiments of this kind were conducted at the I. V. Kurchatov Institute of Atomic Energy in 1967.

The increasing understanding of plasma physics permitted creating towards the beginning of the second decade of research magnetic configurations in which instabilities causing rapid motion of the plasma as a whole were suppressed and the containment time was determined by slower processes — diffusion and thermal conductivity. However, the latter can also lead to the inadmissibly rapid escape of heat if small-scale instabilities resulting in turbulence and intensified mixing develop.

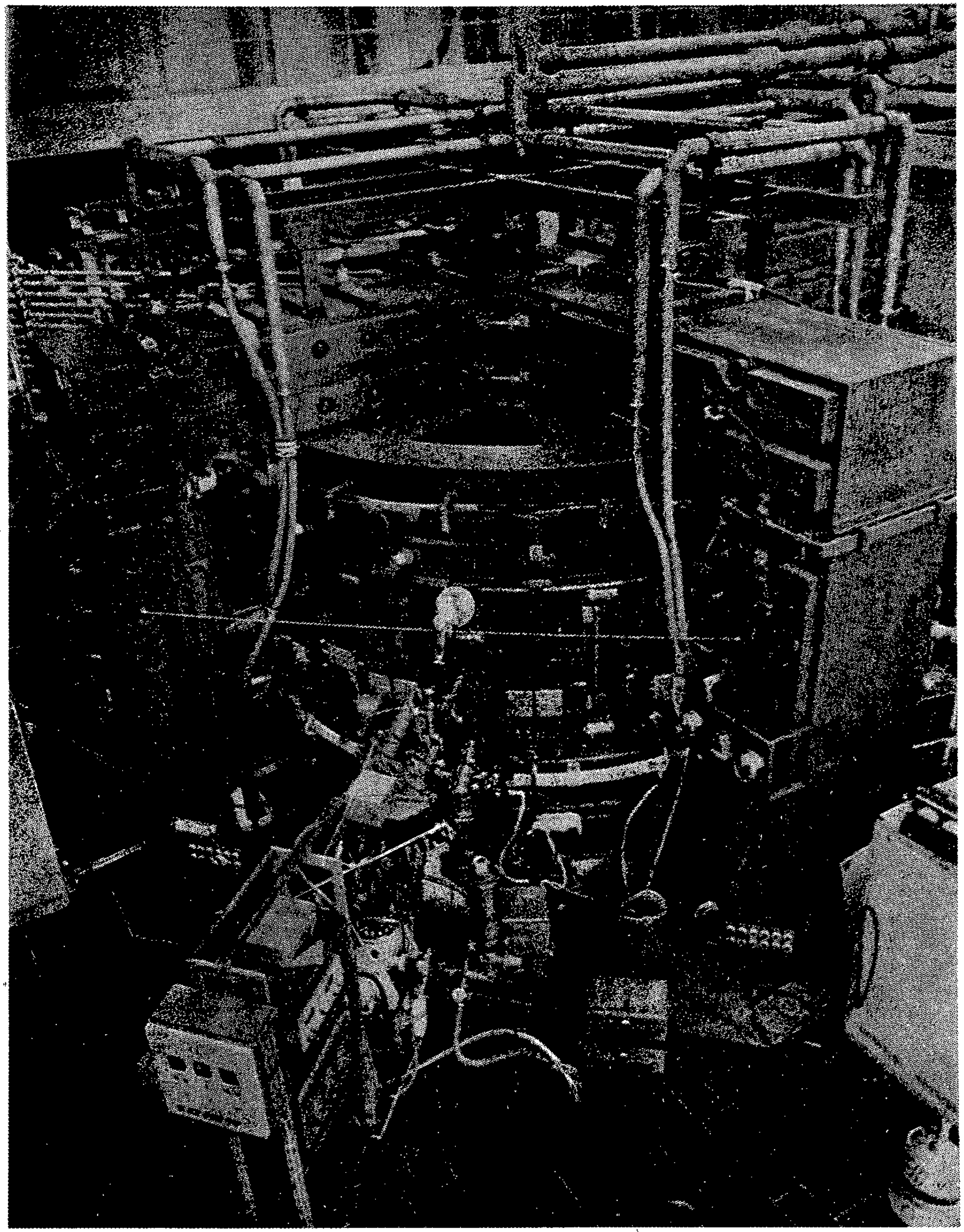

Fig. 2. Tokamak T-10 — one of the largest thermonuclear installations in the world, in which an intense stable thermonuclear reaction has been observed. The plasma temperature has reached ~ 1 keV, the energy lifetime of the plasma is 0.06 sec, and the maximum density is $8 \cdot 10^{13}$ particles/cm^3.

In the U.S.A. experiments were conducted over an extended time on the largest thermonuclear installation, Stellarator-C. Intensified diffusion of plasma in a magnetic field was regularly observed in them, which was first detected in investigations of arc discharges and described by the American physicist D. Bohm in 1949. From that time it has been called Bohm diffusion. Bohm diffusion is so great that it makes the development of a steady-state thermonuclear reactor of reasonable size impossible.

On the basis of their own research on Stellarator-C, the American physicists concluded that Bohm diffusion was universal. Investigations carried out in the USSR under the direction of L. A. Artsimovich on tokamaks disposed of the myth of the universality of Bohm diffusion.

The containment in a T-3 tokamak of a plasma with a temperature of hundreds of electron volts for 10 msec, i.e., 30 times longer than predicted by the Bohm formula, was communicated in papers presented in 1968 at a conference in Novosibirsk. This result initially caused some mistrust overseas, but the combined investigations of Soviet and English physicists in 1969, who employed Thomson scattering of the radiation of a rubidium laser, eliminated all doubt. The effect of the success achieved with tokamaks was so great that in the U.S.A. they completely halted research on stellarators and switched to the development of the Soviet idea— tokamaks. However, as subsequent research on stellarators conducted now in the USSR and European countries showed, such haste was not quite justified: the causes of Bohm diffusion in the American stellarators did not consist of a defect in the stellarator idea itself but in the low accuracy of the production of the magnetic fields in the American equipment. On the stellarators being investigated at the present time at the P. N. Lebedev Institute of Physics and the Kharkov Physicotechnical Institute the Bohm limit has been surpassed, and the results are not inferior to those obtained with tokamaks of similar sizes.

Thus the outlines of the new science — the physics of a high-temperature plasma — became clearly evident towards 1968. Not all the problems were solved, but the growing agreement of theory with experiment and the progress in the achieved parameters showed that the research was on the right path. All this permitted returning again to the sources of the thermonuclear problem and to that idea for the sake of which the investigations of a hot plasma were begun.

Addressing a conference at Novosibirsk in 1968 with his concluding remarks, G. I. Budker appealed to the physicists to return to the problem of a thermonuclear reactor — an inexhaustible source of energy for mankind — without discontinuing their plasma research. The 9 years which have passed since then show that this appeal was actually listened to, and a significant concentration of research has occurred in directions which are more promising from the standpoint of the creation of a thermonuclear reactor.

In order to evaluate the contemporary state of research, let us recall the requirements which the properties of the elementary cross sections of fusion place on the parameters of a thermonuclear plasma. As is well known, if one wishes the energy liberated in a plasma in the case of the easily accomplished reaction of the fusion of deuterium and tritium nuclei to be greater than that expended on its production, then the Lawson criterion must be fulfilled:

$$n\tau > 10^{14}\ \mathrm{cm}^{-3}\ \mathrm{sec\ for}\ T > 10\ \mathrm{keV}\ ,$$

where n is the density in particles/cm^3 and τ is the energy containment time in seconds. As is evident from the Lawson criterion, it is possible to satisfy thermonuclear conditions with various combinations of density and containment time: the lower the density, the longer the containment time should be. Different approaches to the solution of the problem of controlled fusion assume the use of various densities. The successful approach to the Lawson criterion in almost all directions is an outstanding characteristic of the progress being observed at the present time. Let us discuss them successively. We will start with low density.

A plasma density much lower than $5\cdot10^{13}$ cm^{-3} will scarcely be of technical interest at all due to an energy liberation density which is too low (less than 1 W/cm^3). At a density of 10^{14} cm^{-3} the thermal energy of a plasma is significantly less than the energy of the external magnetic field in the case of technically realistic fields of tens of kilogausses; therefore, the motions of a plasma cannot strongly perturb the magnetic fields, and it is contained by the field of a practically fixed configuration. The containment of charged particles moving along the force lines of a magnetic field can be accomplished if the force lines do not leave the volume, i.e., they form a toroidal configuration. The idea of magnetic thermal isolation of a plasma was advanced in the Soviet Union in 1950. The tokamaks are the principal installations where research on toroidal containment has been and is being carried out. A tokamak is a toroidal chamber with a strong longitudinal magnetic field in which a plasma is created by an electrodeless discharge and is contained by the combination of an external toroidal field and the field of the internal current; as the theoretical investigations of V. D. Shafranov have shown, the magnetic field of the current should be significantly less than the toroidal field in order to provide for magnetohydrodynamic stability. Investigation of tokamak-type systems was begun at the I. V. Kurchatov Institute of Atomic Energy in 1954 by I. N. Golovin and N. A. Yavlinskii, and it has been continued by L. A. Artsimovich and his students since the tragic death of N. A. Yavlinskii. The ideas embodied in the tokamaks have proved to be so profound and so vital that until recently practically only technological refinements have occurred: improvement of the vacuum conditions by means of conversion to all-metal heated systems, reduction of the scattered fields, and an increase in the sizes of the equipment and the strength of the magnetic field. What is more important, careful experimental and theoretical investigations were simultaneously carried out on equilibrium, stability, and transport processes in tokamaks. A large contribution to the investigation was the so-called neoclassical theory, developed by A. A. Galeev and R. Z. Sagdeev, of transport in toroidal systems, which took into account the characteristics of charged particles in such systems and the deviations of the distribution function from the equilibrium one which arise at low collision frequencies. A comparison of the results of investigations on tokamaks with the Galeev—Sagdeev theory has shown that at the temperature reached (still not very high) the ion component of the plasma behaves in a classical (i.e., the best possible) way, and the escape of heat from the electrons occurs approximately an order of magnitude more rapidly than follows from the classical formulas, and consequently it is associated with some kind of small-scale turbulence not taken into account by the classical theory. (By classical theories we mean theories which only take into account pair interactions among the particles.) However, the heat losses by the electron channel are of a diffusion nature, and thus the containment time should increase with an increase in the sizes of the equipment as the square of the plasma radius. This conclusion, obtained in 1968, was checked on the largest Soviet tokamak, the T-10 (Fig. 2), where the small radius of the plasma torus has been increased to 37 cm and the energy containment time has reached the record value of 0.06 sec.

The T-10 installation, neglected in 1975, has still not reached the limit to its own results — the investigations are continuing, but the fact that it has been obtained today completely confirms the theoretical concepts developed earlier. Unfortunately, this installation cannot give comprehensive information on the behavior of a plasma under reactor conditions. The neoclassical theory predicts a significant decrease in the transport coefficients as the plasma temperature increases, but the plasma temperature in the T-10 (1 keV) is still insufficiently high for the detection of this effect. In the T-10, as in all the previous tokamaks, the plasma is heated by a current flowing through it. As the temperature increases, the resistance drops, and along with this the power of the ohmic heating is reduced. The necessity arises of supplementing the tokamak scheme with new elements — equipment for supplemental heating. Although various methods are physically possible and are being investigated, actual successes have been achieved with the use of two methods: the injection of fast atoms and heating by millimeter waves. At the present time injectors of fast atoms are being produced which give fluxes of atoms with an intensity of tens of equivalent amperes, an energy of tens of kiloelectronvolts, and an injection pulse duration of tens of milliseconds. The production within the very near future of injectors with a current of hundreds of amperes and an energy of hundreds of keV operating under steady-state conditions appears completely realizable. Experiments with the injectors already produced on tokamaks of average sizes have shown the promise of this method and have permitted obtaining a record ion temperature of 2 keV. The application of powerful neutral injection is being planned in large tokamaks constructed in different laboratories around the world, and it should provide the possibility of investigating processes at a thermonuclear temperature.

The heating of a plasma by microwaves at the frequency of electron cyclotron resonance has become possible due to the creation in the Soviet Union of powerful millimeter-region generators—masers at cyclotron resonance. The first experiments on the small TM-3 tokomak showed the large efficiency (no less than 50%) of the introduction of additional energy by this method.

The approach to thermonuclear parameters has forced the introduction of another series of additions to the classical tokamak scheme. It has proven necessary in the case of the extended plasma existence time of 0.5 sec attained in the largest tokamaks to introduce special automatic systems which center the position of the filament in the discharge chamber. Such a system for controlling the equilibrium position of the plasma filament was created at the I. V. Kurchatov Institute of Atomic energy together with the Kiev Cybernetics Institute, and it was first successfully used on the TO-1 tokamak. The rising plasma temperature and the increasing duration of its interaction with the walls of the discharge chamber forced special attention to be paid to the interaction of the plasma with the wall and to the investigation of the processes of accumulation of heavy impurities. According to classical ideas, heavy multiply charged ions should be contained especially well in the plasma filament, accumulate in it, and consequently quench the thermonuclear reaction in a short time. Many methods have been suggested in the struggle with the accumulation of impurities, and among them are choosing materials of the first wall with a low sputtering coefficient and a low atomic number, the creation of special magnetic device—diverters which shunt the external plasma layer off from the toroidal chamber, and a restriction on the artificial carrying out of impurities from the plasma column. These methods are being investigated intensively at the present time.

Now let us compare the parameters obtained on tokamaks with the Lawson criterion. The maximum containment parameter $n\tau$ attained in experiments at a high plasma temperature on the Alcator tokamak in the USA is $2 \cdot 10^{13}$ $cm^{-3} \cdot sec$ at a plasma temperature of around 0.5 keV. The maximum ion temperature of ~ 2 keV was attained in experiments with additional heating by the injection of fast atoms with $n\tau \ll 10^{12}$ $cm^{-3} \cdot sec$.

One can expect combinations in a single setup of a high temperature and high containment parameters after the introduction of powerful methods of additional heating in the largest contemporary tokamaks. Thus it is evident that a factor of five in the value of the temperature separates us at present from the Lawson criterion. During the last 15 years the containment parameter has increased by an order of magnitude every 5 years, and the ion temperature has increased during the same time by approximately a factor of two. There are no reasons at the present time to expect a decrease in the tempos of this growth. On the contrary, the existing physicotechnical levels permit one to hope for their acceleration. Therefore one can expect with certain assurance that the Lawson criterion can be attained sometime in 1985, and the so-called demonstration experiment in which the generation of thermonuclear energy in a plasma exceeds the expenditures on its creation and maintenance will be conducted on a tokamak.

Tokamaks are the most advanced and promising but far from the only direction of research on magnetic containment.

The development of another type of toroidal system — the stellarators — began simultaneously with the tokamaks. Plasma containment was accomplished in them by the combined field of an outer toroidal winding and inner helical conductors, whose field replaces the field of the current flowing through the plasma of the tokamaks. Since the field of the outer conductors is stably and uniquely defined, in contrast to the field of the plasma current, stellarators are more convenient for physical investigations. However, they are more complicated, and therefore the radius of the plasma filament is smaller in them than in tokamaks for equal expenditures, and therefore the maximum containment time is less. Technical simplicity is an important quality of an engineering system, but the significant complication of the classical tokamak scheme is gradually bringing the tokamaks closer to the stellarators, so that a partial combining of both principles is completely possible at the stage of producing a thermonuclear reactor.

The successful containment of a plasma in tokamaks and stellarators has been accomplished due to the rigidity of the magnetic configurations, which is possible, however, only for small ratios of the plasma pressure to the pressure of the containing magnetic field. This ratio has been designated in plasma physics as β. In contemporary tokamaks β is less than 1%. Since the magnetic field is one of the main expensive components of a future reactor, small values of β are necessary in strong fields and disadvantageous from the engineering point of view. There are possibilities for some increase of β within the tokamak scheme itself, and they are being investigated. However, attempts have been made for a long time to find more radical solutions and to obtain stable containment for $\beta \approx 1$. Different types of Z- and θ-pinches have mainly been investigated, i.e., systems with rapid compression of the plasma by an external magnetic field, but up until now the results of the research have not caused special enthusiasm. Rapid compression heats up the plasma well, but the containment is extremely low. Nevertheless one should not conclude on the basis of these data that good containment is impossible in systems with a large value of β. New data obtained with the aid of quite different systems — traps with magnetic plugs — show that there are good prospects for the creation of containment systems with high values of β.

Traps with magnetic plugs were proposed by G. I. Budker in 1953. The lines of force in these systems are not closed into a torus, but regions of intensified field — magnetic plugs — are created to prevent the escape of the plasma along the field lines at the ends of a rectilinear trap. Not all the particles are reflected from the plugs but only those which have sufficiently large velocity in the direction perpendicular to the lines of force. Collisions among the particles — Coulomb scattering — lead to their exiting through the plugs; however, collisions are rare at high temperature, and containment is possible. The higher the ion temperature, the longer they are contained in the trap. Since the distribution function of the particles in traps with plugs is far from an equilibrium one — there are no particles in it with a low transverse velocity — the plasma is more unstable in open traps, and specific instabilities can develop in it. M. S. Ioffe and his coworkers proved experimentally the possibility of stabilization by means of "adjusting" the distribution function by the addition of an uncontained cold plasma, Simultaneously in the U.S.A. powerful injectors of fast atoms were developed which permit filling the traps with a hot plasma. It proved possible at the Lawrence Livermore Laboratory to obtain with the use of these injectors and the stabilization method mentioned a plasma in a trap with magnetic plugs which had a density of 10^{14} cm^{-3}, an ion temperature of about 10 keV, and the parameter $n\tau$ equal to 10^{11} cm$^{-3}\cdot$sec. And what is especially important, the value of β approached unity. Although the containment in traps with plugs is still worse than in tokamaks, the results are still sufficiently interesting, and clear ways for their improvement are being suggested.

One should note that due to the rapid escape of ions through the plugs as a result of pair collisions, the prospects for the creation of a reactor based on a trap with magnetic plugs does not seem especially optimistic in regard to the complete suppression of all possible instabilities, and even small additional losses due to instabilities completely destroy the energy balance of such a reactor. It is necessary to find methods which improve the containment of a plasma by plugs. A search for such methods has been conducted for a long time, but only about 2 years ago G. I. Dimov suggested a solution which is actually promising — to use a combination of three traps in which two of extremely small size serve for improvement of the plasma containment in the central part, which is the source of thermonuclear energy. The escape of plasma from the central trap is thus prevented not by magnetic plugs but by the electric fields which automatically arise in such a configuration if the density in the outer traps exceeds its value in the central part. Theoretically this idea appears rather attractive, and its experimental verification is being planned for the immediate future.

Another interesting modification of a trap with plugs is the use of the intrinsic diamagnetic currents in a high-pressure plasma for variation of the magnetic configuration and the conversion of an open trap into a toroidal one. The possibility of the creation of such configurations has already been demonstrated on small scales with the aid of the injection into the trap of pulsed beams of relativistic electrons. The use of the neutral injection technique opens up new possibilities here.

Thus the combination of experimental results with the powerful development of injector technique and new ideas on the improvement of classical containment has evolved the trend of open traps — acknowledged earlier as a dark horse in the thermonuclear race — into one of the most promising (at least in the long-term plan).

Simultaneously with the successes in magnetic plasma containment great progress has been made in recent years in another direction: the creation of a thermonuclear microexplosion. As follows from the Lawson criterion, if the plasma density exceeds the density of a solid body, then the need for containment is eliminated: the time of free dispersion exceeds the necessary containment time. The problem thus reduces to the concentration of large energy in a short time in a small volume of material. Exceedingly powerful energy concentrators have been produced during the last decade: lasers and relativistic electron beams. The application of this technology has permitted approaching closely the accomplishment of a thermonuclear microexplosion.

The possibility of the creation of a plasma of thermonuclear temperature with the aid of a laser was first shown theoretically in 1962 by N. G. Basov and O. N. Krokhin. Experiments conducted under their direction at the P. N. Lebedev Institute of Physics led in 1968 to the detection of neutron emission from the laser target.

In 1972 E. Teller announced research of a Livermore group headed by G. Nuchols in which the possibility of the intense compression of a target under the action of a laser pulse of special profile was theoretically shown. This communication caused increased interest by the world scientific community in laser thermonuclear fusion. Since upon compression the dispersion time is shortened only as the radius, i.e., as the cube root of the density in the case of spherical compression, the intense compression permits satisfying the Lawson criterion for a small-sized thermonuclear plasma and, consequently, for a small energy contributed to it by the laser. A thousandfold compression lowers the required energy of the laser pulse by a factor of a million and allows reaching the Lawson criterion at a pulse energy of several tens of kilojoules. It is possible to accomplish the compression by means of the pressure which arises upon the evaporation of the outer layer of the target under the action of the laser radiation. One can obtain intense compression, however, only in the case in which the compressed center of the target remains cold until the very last moment. To accomplish this one should select such a temporal dependence of the laser power that all the shock waves excited by the external radiation converge simultaneously at the center of the target. Most of the energy is contributed at the very last moment, for which a laser of very great power is required. One can reduce the power requirements if one uses a target which is not continuous but consists of distinct shells. If, e.g., one encloses a D—T-target in a heavy hollow shell, then it is possible to straighten the shell with a laser of low power but with a long pulse duration and thereupon use its kinetic energy for the squeezing and heating of the D—T-target. Experiments are being developed at present in precisely this direction: the spherical squeezing of glass microspheres filled with deuterium or D—T-gas at a pressure of tens of atmospheres is being investigated. The main processes being investigated here are the physics of absorption of laser radiation, the dynamics, symmetry, and stability of compression, and the generation of fast particles and neutrons. It has proven possible in experiments with lasers permitting the attainment of an energy up to 1 kJ in nanosecond and shorter pulses to realize a volume compression of such glass spheres by a factor of 1000 and to heat the ions at the center of the target to 2.5 keV, simultaneously observing thermonuclear neutron emission which corresponds to the indicated ion temperature. Experiments are presently being conducted with a new generation of lasers having an energy of about 10 kJ in a short pulse. One can hope that it will prove possible with such systems to approach the Lawson criterion.

One should recall, however, that not electrical energy but light energy is used in the heating of the plasma with lasers, which is obtained in the contemporary superpowerful lasers from electricity with a low efficiency (less than 1%). Therefore if fulfillment of the Lawson criterion for steady-state systems denotes the eve of industrial application, then it denotes only a stage of physical research for laser systems. It is necessary for their industrial application that either the Lawson criterion be exceeded by a factor of 100 or the efficiency be significantly improved. Gas lasers in carbon dioxide gas have the best efficiency. But they generate light at a significantly longer wavelength than do the neodymium glass lasers most widely used in the experiments. Consequently, their radiation should be absorbed in the more tenuous outer layers of the plasma cloud around the target, which is not advantageous.

However, the radial distribution of the plasma density should be distorted under the action of light pressure in the case of a sufficiently large power of the laser radiation: a high step should be formed on it where the density gradient is large. In this case the point of absorption of the laser light would depend weakly on the wavelength and gas lasers would have a clear advantage over solid-body ones due to the higher efficiency. Further investigations should show whether the indicated situation is realized at large power.

Relativistic electron beams have a significant advantage over lasers from the standpoint of efficiency. Their shortcomings are more difficult transportation and focusing of energy and the problem of the contribution of energy to a small volume.

The suggestion to use relativistic electron beams for a thermonuclear microexplosion was first made in 1968 by E. K. Zavoisk. In the last five years effective methods have been developed at the I. M. Kurchatov Institute of Atomic Energy for the conversion of the energy of relativistic electrons into the kinetic energy of the shell compressing the thermonuclear target. It has proven possible at the "Triton" installation with the use of a two-shell target to realize an effective contribution of the energy of the beam to the target, to accelerate a segment of the polyethylene shell to a velocity close to that necessary for the realization of a thermonuclear microexplosion, and to obtain thermonuclear neutrons. Plasma containment has thereby reached 10^3, and the ion temperature 1 keV. An effect of an anomalous energy contribution to the target has been detected which permits significantly reducing the beam energy required for the ignition of the thermonuclear target. A focusing of the beams sufficient for the ignition of the thermonuclear target has been achieved in experiments. Thus it has proven possible in recent years to investigate in model experiments the physics of the microexplosion with ignition by relativistic electron beams having practically the required thermonuclear parameters, which allows extrapolating the results rather reliably to the conditions of a reactor. A microexplosion with an energy amplification coefficient of ~ 20 can be obtained with the application of technology which already exists with a total beam energy of about 5 MJ. The Angara-5 accelerator complex is presently being planned to realize these conditions, and it will consist of 48 identical accelerator modules which provide a cumulative current of 50 MA with a total energy of 5 MJ for a duration of 60-80 nsec. Evidently systems based on relativistic electron beams, and in particular the Angara-5, are closest to fulfillment of the Lawson criterion.

Thus the greatest progress towards the realization of the conditions of controlled fusion has occurred at low and superhigh densities. Experiments at a medium-range density of 10^{17} cm^{-3} and higher have not yet gone beyond the bounds of the physics stage but are already rather promising. The method suggested by E. P. Velikhov for plasma compression by a metal liner dispersed in advance is very promising, for example. Such a method of compression permits using in the best way the energy of the supply source and is convenient for the direct conversion of thermonuclear energy to electrical energy. It has already proven possible in such experiments to achieve a stable hundredfold compression in cylindrical geometry. The compression coefficient is 1000 in the case of three-dimensional compression by an accumulating liner. Another method of containing a dense plasma proposed by D. D. Ryutov is being investigated at present at the Institute of Nuclear Physics of the Siberian Branch of the USSR Academy of Sciences in Novosibirsk. In this case the plasma is contained in a long metal tube with a relatively weak magnetic field. The transverse pressure is contained by the walls of the tube, and the transverse thermal conductivity is lessened due to the longitudinal magnetic field, but the dispersion of the plasma along the tube is suppressed by the corrugation of the tube and the magnetic field, as a result of which magnetic plugs are formed. They directly contain part of the particles having a favorable direction of the velocity vector and retard the dispersion of the remaining particles by virtue of their friction against those being contained. The effect of the longitudinal containment has been verified in model experiments. The research is presently being conducted on a small scale but one permitting the investigation of all the phenomena involved in the model.

It is obvious from the review of research presented above that there evidently exist several ways of solving the thermonuclear problem, but there is not a simple way. There are critical parameters (sizes, power) affiliated with any approach whose attainment is possible only in very expensive installations. The application of a new physical principle only shifts, as a rule, the center of the problem to a new place but does not reduce the expenditures necessary for attainment of the goal. The closer we approach the goal, the more complicated and expensive become the experimental installations and the larger the role played by purely engineering and technological aspects. The problem acquires an ever more applied nature. It seems natural under these conditions to return again to the goal of thermonuclear research and examine it: are thermonuclear reactors necessary, what place would they occupy in the power engineering of the future, and are the expenditures which are necessary for the solution of the engineering and technological problems of creating a thermonuclear reactor justified?

Different fuel cycles of a thermonuclear reactor are theoretically possible. However, on the basis of the current level of achievements we can speak only of the reaction of a mixture of deuterium and tritium. D—D and other possible reactions do not appear realizable in the immediate future. Let us attempt to compare thermonuclear D—T reactors with practically their only competitor in the large-scale energy picture—fission reactors. On what basis can such a comparison be made? Under contemporary conditions when mankind has perceived the limited nature of terrestrial resources, one of the chief parameters for comparison is the provision of fuel. In D—T reactors tritium is necessary, which is an unstable isotope of hydrogen not present under natural conditions. It is possible to reproduce it comparatively easily in the thermonuclear reactor itself, irradiating lithium with neutrons. Thus the fuel basis of D—T reactors is limited by the reserves of lithium. An estimate of the reserves shows that there is enough of them to satisfy the contemporary energy needs of mankind for hundreds of thousands of years, i.e., the fuel basis of D—T reactors is comparable to the fuel basis of breeder reactors, and thermonuclear reactors do not have significant advantages in this area. Another important parameter is the hazard of radioactive contaminations. Unfortunately, D—T reactors are not free of radioactive contaminations. The radioactive hazard arises from two causes: first of all, due to the presence of tritium, and secondly, due to the neutron irradiation of the construction materials. In view of the diverse nature of the radioactive wastes in thermonuclear and nuclear reactors specialists disagree on the evaluation of the comparative hazard of atomic and thermonuclear stations. Optimists assume that the latter are $\approx$ 1000 times less hazardous than nuclear stations of the same capacity.

It is necessary in the first stage of development to consider thermonuclear power not as a competitor but as a component part of nuclear power, which cannot consist of just rapid breeder reactors. The latter are fine in connection with the generation of electrical energy under baseline conditions. Their utilization to cover semipeak loads, the production of high-temperature heat, and central heating is irrational. Light-water reactors can be applied with great success in these areas. D—T thermonuclear reactors with shells made of natural or depleted uranium could supply them with fuel. Such hybrid reactors could generate not only electrical energy but also plutonium — the fuel for light-water reactors; each hybrid reactor could provide 4-6 light-water reactors of the same capacity with fuel. Economic estimates show that such a symbiosis of thermonuclear and light-water reactors appears to be economically attractive, even if the cost of a hybrid reactor is two times higher than the cost of a fast breeder reactor of the same capacity. A more rational division of the power market among breeder reactors, light-water reactors, and thermonuclear reactors seems to be possible in connection with the use of hybrid reactors.

Of course, thermonuclear reactors lose their "purity" in connection with the use of uranium, but the loss is compensated by important technological advantages. Energy generation in the uranium blanket increases the total energy generation in a thermonuclear reactor by a factor of 8-10, which correspondingly reduces by just as large a factor the requirements on the containment parameter. It is sufficient to have $n\tau > 2 \cdot 10^{13} \text{cm}^{-3} \cdot \text{sec}$ for the energy balance in a hybrid reactor. This value has already been attained today in the tokamaks. Thanks to the additional energy generation in the blanket, new cheaper thermonuclear schemes become possible which are energetically unprofitable in a pure D—T reactor. Finally, the additional energy generation in the blanket makes possible a significant reduction in the flux of 14-MV neutrons through the first wall of the thermonuclear reactor, thereby significantly facilitating the solution of the problem of the radiation resistance of the materials. Summing up all that has been said, one can conclude that the construction of hybrid reactors will be technically completely realistic toward the beginning of the 1990s, since the physical conditions necessary for hybrid reactors should be achieved in the generation of thermonuclear plants now being constructed. The timely development of hybrid reactors would permit successfully solving the problem of supplying with fuel the currently rapidly growing power by light-water reactors. Together with this the creation of hybrid reactors will in no measure hinder, but on the contrary is a natural stage on the path to pure thermonuclear reactors, if the latter prove to be more attractive for mankind by virtue of their "purity."

As a result of almost 30 years of intense research the scientific basis of thermonuclear reactors has been laid — the physics of the high-temperature plasma has been created. The plasma parameters have been approached right up to those required for technical utilization in experimental installations of various types. The technology of large superconducting magnet systems and the means for additional heating of the plasma — injectors of neutral atoms and superhigh frequency generators — along with the technology of superpowerful lasers and relativistic electron beams are approaching reactor requirements and scales. The accumulated scientific and engineering potential permits proceeding today to the creation of demonstration systems, and then to industrial ones.

Performance of the appropriate developments would permit a significant lightening towards the end of this century of the problem of supplying thermal nuclear reactors with fuel and would give mankind somewhat later on a new alternative source of energy in the form of clean D—T thermonuclear reactors.

BASIC APPROACHES TO NUCLEAR POWER STATION SAFETY IN THE USSR

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A Combination of Technical and Organizational Measures is the Necessary Condition of Real Nuclear Power Station Safety

The safety of a nuclear power station is taken to mean its capacity to ensure protection against radiation effects for the technical personnel, the population at large, and the environment both in normal use and in the event of possible disturbances, including major accidents. Experience in the design and use of nuclear power stations with various reactor types in the USSR and the world shows that the problem of nuclear power station safety in normal use has practically been solved. The release of radioactive products beyond the confines of the power station corresponds to the accepted national or international norms. The trend toward more stringent norms cannot have a significant effect on the development of nuclear power, since it is technically possible to reduce or entirely eliminate such release.

Experience in nuclear power station development shows that the main emphases of safety work should be as follows:

1) high-quality manufacture and installation of the equipment, as the basis for safety and for reducing the probability of accidents and disturbances;

2) monitoring of the condition of the equipment in all stages of use;

3) the development and implementation of effective protective measures and equipment for eliminating the causes of accidents and compensating for any disturbances that arise (switching on reserve equipment, emergency reactor shutdown, etc.) or for limiting the consequences of disruptions (emergency active-region cooling);

4) the development and implementation of measures for either complete localization of radioactive materials in the event of accident or limitation of the consequences of an accident (collection systems for radioactive products, hermetic chambers, choice of power station location);

5) the implementation of all technical and organizational safety measures at all stages of design, equipment manufacture, startup, use, and repair of nuclear power stations;

6) the standardization of all technical and organizational safety measures;

7) an effective system of state supervision for nuclear power station safety, enforced by law.

The present paper considers technical measures and organizational structures used to ensure nuclear power station safety in the USSR.

Translated from Atomnaya Énergiya, Vol. 43, No. 5, pp. 360-365, November, 1977.

0038-531X/77/4305-1009$07.50©1978 Plenum Publishing Corporation

Declassified and Approved For Release 2013/03/22 : CIA-RDP10-02196R000700100001-6

Quality and Monitoring. The development of nuclear energy imposes more rigid requirements than ordinary energy technology on the quality and the corresponding quality-control standards in the manufacture and installation of equipment. New materials and technological processes are being developed and particular attention must be paid to the radiation stability of the equipment, the monitoring of welded joints, the reliability of components, etc. The necessary quality and quality-control standards rise with the importance of the component considered.

The most significant mistake made in the early stages of development was to underestimate the probability that defects would appear as a result of factors such as vibration, temperature oscillations and fatigue, corrosion (in particular, under stress), etc. In addition, experience has shown that the remote monitoring of the condition of equipment while it is in use can involve unexpected difficulty. Accordingly, the first reactors were underequipped with monitoring equipment, and their design and construction was often such as to impede access to the important points of the reactor for monitoring and repair. Subsequent experience has allowed these deficiencies to be partially removed: the amount of monitoring has been increased, weak points have been reinforced, and removable structures have been substituted for fixed, nonreplaceable structures.

The basic measures implemented in current power stations permit periodic monitoring during shutdown and repair. Procedures and instruments are being developed for continuous or periodic monitoring of the equipment while it is in use, including the analysis of acoustic and neutron noise, stress-wave emission, etc., i.e., a multiple approach to monitoring is being developed.

Disturbance and Safety Limits. The development and implementation of measures intended to maintain nuclear power station operation within safe limits necessitates a choice of the appropriate approach and the definition of a "safety limit" for each specific initial disturbance. Nuclear power station safety measures for "small" disturbances (disturbances that are entirely neutralized or those whose consequences may be avoided) are very largely identical with the measures that ensure the reliability of a nuclear power station as an energy source. To ensure uninterrupted use, the design of any nuclear power station must include adequate provision for the complete neutralization of the disturbance and the maintenance of continued operation in the event of any of a number of initial disturbances (e.g., interruption of essential supplies, turbounit failure, breakdown of the power system, isolated disturbances in the reactor-control system).

Between the first safety limit (complete neutralization of the disturbance and continued operation of the plant) and the ultimate limit arising from the definition of safety (the prevention of injury to the population at large or contamination of the environment) it is usual, in practice, to employ a whole series of intermediate safety limits, each of which corresponds to a specific initial disturbance (or a set of simultaneous but independent disturbances) and to formulate appropriate specifications for the protection and localization systems.

Maximum Accident and Quantitative-Probability Method. In considering the possible disturbances at a nuclear atomic power station, the upper limit for which the design must provide adequate safety measures is determined by the concept of the maximum possible accident (MPA). Indeterminacy in defining the MPA at present arises from the natural subjectivity in interpreting the problem. Throughout the world there is controversy between those nuclear power specialists who believe that the measures adopted to counter potential dangers at atomic power stations are too rigorous and those who think them too lax. The present lack of reliable statistical data on the probability of failure necessitates a deterministic approach in standardizing the potential nuclear power station dangers, i.e., specification of the maximum possible (limiting) accident, safety limits and corresponding disturbances, etc.

The MPA included in the design differs for different stages of nuclear power development and for power stations with different types of reactor. In the first generation of stations, the protection and backup (secondary protection) measures included in the design were determined on the basis of the limiting MPA assumed for the station and the distance from centers of population. Stations introduced today are designed to accommodate more serious disturbances; the level of technical provision for the neutralization of disturbances is such that the distance from population centers is a less significant factor. Designs have now been developed for power stations (primarily for heat supply to towns) a minimum distance from population centers or close to large concentrations of population. Of course, such stations require a further intensification of technical safety measures and an appropriate change in approach to safety limits and the corresponding disturbances.

Protection and Localization Systems. Experience in the design and use of nuclear power stations with thermal reactors confirms the possibility of implementing safety limits on fuel-element damage for practically all disturbances that do not involve loss of sealing of the primary heat-carrier loop. Disturbances associated with major rupture of the primary loop are a special case and at present determine the MPA for thermal reactors. For these disturbances it is necessary to consider safety limits characterized by the yield of radioactive products beyond a certain level.

Analysis of the occurrence of accidents shows that hazardous consequences of MPA with heat-carrier loss are almost completely eliminated by the use of reliable and powerful means of cooling the active region. If emergency cooling prevents fusion of the fuel elements, external safety barriers in the form of hermetic chambers and shells play a less significant role. Both in early nuclear power stations and in current stations with water-cooled—water-moderated reactors and RBMK, reliable emergency cooling is an extremely important part of the safety systems. The protection afforded by the localization system provides a backup capacity, in cases where the cooling system is less than totally effective.

If instantaneous transverse rupture of a primary-loop channel of maximum diameter is the limiting accident, particular attention must be paid to the localization systems, since for such major disruptions of the loop less than total effectiveness of the emergency cooling system may have hazardous consequences, with the escape of radioactive products from the active region. What is required is an optimum technical solution incorporating both primary and secondary protection systems. In new nuclear power station designs, some of which are already being implemented, the emergency-cooling systems employed are such that in the case of primary-loop rupture fusion of the fuel elements is prevented, although partial unsealing of the fuel elements may occur. Structurally, the localization systems may take different forms: e.g., hermetic ferroconcrete protective shells (fifth block of the Novovoronezh nuclear power station) or systems with hermetic boxes, without sealing of the reactor chamber, in conjuction with systems for the condensation of any vapor formed.

Particular mention should be made of a development in protection technology which allows the emergency-cooling and localization systems to be simplified, while retaining the same level of safety: the creation of collapsible power structures which maintain the whole of the potentially safe main structure of the first loop (including, possibly, the reactor body) in the event of rupture and limit the rate of heat-carrier escape.

The development of such structures would allow practically complete elimination of the indeterminacy in the idea of the MPA which is the cause of so much discussion.

Organizational Structure and Standardization of Nuclear Power Station Safety Provisions

State Regulatory Bodies. State supervision of nuclear power station safety is the responsibility of the following bodies:

the state committee for the supervision of working safety in industry and final supervision in the USSR Council of Ministers (control over supervision in nuclear power) of the observance of technical-safety standards in the design, construction, and use of a nuclear power station and its equipment;

the state inspectorate on nuclear safety (Gosatomnadzor SSSR) within the state committee of the Institute of Atomic Energy of the USSR responsible for the observance of nuclear safety standards and regulations in the design, construction, and use of nuclear stations;

state health supervision within the Ministry of Health (Minzdrava SSSR) of the observance of health regulations and radiation-safety standards in the design, construction, and use of nuclear power stations, in order to prevent irradiation of power station personnel and the surrounding population.

The jurisdiction of the state regulatory bodies extends over all establishments, institutions, and organizations, regardless of their departmental affiliation.

Standardization Documents. The "General safety principles for nuclear power stations in design, construction, and use" are at present the main standardization documents outlining the basis for the consistent implementation of technical and organizational safety measures in nuclear power stations at all stages of development and use. As well as the statement of nuclear power station safety criteria and regulations that cover the monitoring of safety-measure implementation, the documents outline basic general safety requirements for various systems and for the power station as a whole, and also deal with the organizational and technical problems of safe operation. Principles developed through experience of reactor design are outlined: the provision of reserve equipment, multiple-loop design (duplication), containment of damage, etc. It is necessary to note certain requirements determining the range of disturbances that are to be considered in power-station design:

safety must be ensured for any single disturbance of the equipment in normal use, which may coincide with an unobserved prolonged disturbance of other equipment;

the protection and localization systems must be constructed to take into account not only disturbance of the system in normal use but also simultaneous disturbance or shutdown of one of the independent active protection and localization systems.

In these documents, a number of disturbances correspond to the safety limits and fix the presently adopted MPA — rupture of the primary-loop channel, the most dangerous in terms of radiational consequences.

As well as the "General safety principles" there are detailed specific normative documents (standards, regulations, procedures, etc.) in which requirements and procedures for nuclear power station safety are set out.

Ensuring Implementation of Safety Requirements. The nuclear power station safety practices developed and implemented in the USSR cover all the stages of nuclear power station design and use, and extend throughout all the institutions and organizations associated with nuclear power.

In the course of designing a nuclear power station and its components, the observance of the safety requirements is discussed at each stage of the design; it is useful to ask persons and organizations not associated with the development of the design for their point of view (appraisal, criticism, evaluation, etc.).

The manufacture of nuclear power station equipment and components is monitored. Monitoring at the factory is carried out by special groups of personnel (technical-monitoring sections), independent of those directly responsible for production. For particularly important components, special quality-control programs are introduced. The external monitoring organizations are the state supervisory body (Gosgortekhnadzor SSSR), the clients' representatives, and the designers. Their main activities are random sampling, inspection, participation in factory testing, etc. When a particularly important component is to be accepted, a state acceptance commission is instituted, including representatives of all interested parties. Production samples carry a plate bearing information on the factory or state tests.

Roughly similar procedures apply to construction and startup, and throughout the nuclear power station operation itself. In the case of startup, the main role is that of the state acceptance commission, appointed by government agencies from representatives of the regulatory bodies, the clients, and the designers.

Specific Safety Problems for Nuclear Power Stations with Various Types of Reactor

Water-Cooled—Water-Moderated Reactor. Operating experience of nuclear power stations with reactors of this type in the USSR and COMECON countries amounts to more than 50 reactor-years. Their development from the first block of the Novovoronezh nuclear power station of 210-MW power to the present fifth block of 1000-MW power indicates the evolution in the solution of safety problems.

In the event of accidents with the primary-loop flow, the Novovoronezh station reactor is protected by an emergency flow-maintenance system of capacity 100 m^3/h and a system of hermetic chambers (boxes) in the primary loop, equipped with sprinkler devices for the condensation of any vapor. The emergency flow-maintenance system prevents active-region damage for tube ruptures of up to 100 mm. Special nozzles fitted at junctions between large and small tubes limit the flow rate of incoming heat carrier. The primary-loop boxes of the first block were designed for an excess pressure of 3 kgf/cm^2. Additional secondary protection in the case of unforeseen and improbable large-scale disturbances is provided in that the radiation-safety radius is 3 km and the power station is sited $\approx$ 50 km from the nearest large population center (Voronezh).

In the first group of reactors with a VVÉR-440 reactor, safety is further ensured by extensive monitoring and by ease of access and maintenance. The hermetic boxes are designed for an excess pressure of 1 kgf/cm^2. From the boxes there is provision for a spray of vapor—air mixture at the level of the reactor-structure breakdown through 9 adjustable valves. This spray moderates the accident process in the case of rupture of small heat-carrier tubes when the sprinkler system fails to operate or is less than totally effective. Its maximum capacity is calculated for the release of water in the rupture of tubes of 200-mm diameter, although this accident is more extreme than the maximum design accident.

The emergency-cooling system currently in use is able to deal with rupture of heat-carrier tubes of maximum diameter (500 and 850 mm for VVÉR-440 and VVÉR-1000) without fusion of the fuel elements. The main elements of the system are water stores and high- and low-pressure emergency pumps. There are several versions of the system for localizing activity within the structure of the power station in the event of the maximum accident. The VVÉR-1000 design envisages the construction of a hermetic ferroconcrete shell above the reactor chamber and all the primary-loop chambers, designed for an excess pressure of 4 kgf/cm^2 and capable of retaining all the heat carrier released from the primary loop. A sprinkler system is provided for vapor condensation and cooling. The VVÉR-440 design includes a system of localization chambers, based on a typical composition of series nuclear power stations, with a hermetic steam-generator box designed for maximum excess pressure of 1.5 kgf/cm^2 and without sealing of the reactor chamber. Total condensation of the heat carrier released is ensured in an ascending series of boxes of special construction.

RBMK. Such reactors will provide approximately half of the power in the USSR in the next 10-15 years. Pilot units of this type of 1000-MW power have been operating at the V. I. Lenin Leningrad nuclear power station since 1973 and 1975; the next units are to be introduced at the Kurskii and Chernobyl'skii power stations. Previously, this type of reactor was represented in the USSR by the world's first nuclear power stations, the reactors of the Beloyarskii and Sibirskii nuclear power stations.

Safety measures in the event of accidents associated with primary-loop rupture are affected by the following physicotechnical features of reactors of this type:

the positive vapor coefficient of reactivity or, at least, the absence of a strong negative inverse relation on dehydration, which places increased demands on the reliability and efficiency of the emergency protection systems;

the large volume of saturation vapor and water leading to a relatively low rate of pressure drop (therefore, solutions for the system of emergency cooling for the active region different from those for vessel reactors are expedient);

the small accumulated activity of the water in the primary loop because of the removal of volatile products with the ejector gases and the possibility of detecting and replacing defective channels "on stream";

the large reserves of water for the emergency-cooling system and the possibility of reliable direct transfer of the water to the active region;

the possibility, in principle, of having a "fractional" channel of narrow autonomous cooling loops, which decreases the volume and rate of flow;

the large size, because of which it is difficult to use a general protective shell and more expedient to employ other localization systems, e.g., a bubble tank.

At present in RBMK designs, the MPA assumed is the worst of two accidents in terms of radiational consequences:

total rupture of the general pressure or suction collector of the main circulation pump or of the connection to the return valves at the inlet to the communal collector;

rupture of the communal collector.

On the basis of past experience of the design and use of nuclear power stations with RBMK it may be assumed that the specific features of these reactors may fairly simply and cheaply be made to satisfy the safety requirements for practically all convenient nuclear power station sites.

Fast Reactors. The scientific—technical techniques for fast-reactor safety in the USSR are based on experience obtained in the development and use of the BR-5 (BR-10), BOR-60, and BN-350 reactors, the construction of a nuclear power station with a 600-MW (electrical) BN-600 reactor, and the development of designs for a 1500-MW (electrical) power station.

From a safety point of view, the main features of fast reactors with sodium heat carriers are associated with the active-region properties (the absence of a moderator, high thermal stress, short lifetime of the retarded neutrons for plutonium and mixed fuel, swelling of the constructional materials) and the consequences of the use of a sodium heat carrier (large heating in the reactor, ambiguity of the hydraulic characteristics of the heat-extraction channel, sodium vacuum reactivity effect, large induced activity of the heat carrier, and the chemical activity of sodium with oxygen and water).

Operating experience confirms the important properties of sodium-cooled reactors from the safety point of view: the absence of contamination and problems of spatial instability, negative temperature reactivity effects, and also favorable conditions for the extraction of residual heat liberation. Because of the large margin to the boiling point (400-500°C) and the large heat-transfer coefficient of sodium at very low rates it is possible, without hazardous consequences, for heat to be transferred from the active region and accumulated in sodium and the constructional elements of the first and second loops in conditions of natural circulation, even in the case of prolonged shutdown of the system for the discharge of heat to the atmosphere. The rate of increase in heat-carrier temperature in these conditions is no more than 20-50 deg C/h.

The probability of an accident analogous to the MPA in reactors with water under pressure (rupture of heat-carrier tube of maximum diameter) is significantly lower in a sodium reactor than in thermal reactors, because of the low pressure in the first loop and the low corrosional activity of sodium. Hazardous consequences of such an accident are easily eliminated in the integral composition (as in the BN-600).

A more dangerous and less studied possibility for fast sodium reactors is instantaneous-neutron runaway as a result of rapid destruction of the active region and the appearance of a positive sodium reactivity effect (other possibilities of the rapid introduction of positive reactivity may be excluded by constructional and design measures). For modern nuclear power station designs with the BN-600 similar dangers are absent, since in this case the integral sodium vacuum effect is negative. In future series nuclear power stations of power 1000-1500 MW (electrical) it is positive and may reach large values. Note that, given current knowledge, the only conceivable situation that might lead to the appearance of a positive sodium effect is the switching off of the primary-loop pumps and simultaneous failure of all the control-rod emergency protections. Evidently, the probability that such a situation will arise is extremely small, and may be reduced to required values by increasing the reliability and duplication of the protection systems and equipment. All other cases (the development of local damage, the passage of large gas bubbles or retarding materials through the reactor, etc.) may be eliminated by appropriate constructional and design measures.

In the light of current requirements regarding possible disturbances at nuclear power stations with thermal reactors, it may be said that the design solutions adopted in fast reactors are adequate for safety and impose no additional restrictions on the choice of power station sites.

ECONOMIC ASPECTS OF THE DEVELOPMENT OF NUCLEAR POWER AND FUEL CYCLE PLANTS IN THE USSR

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Thermal vessel-type reactors, viz., the water-cooled—water-moderated (VVÉR) and uranium—graphite channel-type RBMK, will be the most typical for nuclear power stations in the next 20-25 years in the Soviet Union, as in other countries.

An assessment of the present state of the art with respect to nuclear power stations with fast reactors, particularly with respect to the experience from the operation of the Shevchenko nuclear power station with the BN-350 reactor, and the rate of plutonium build-up in thermal reactors permit the conclusion that nuclear power stations with fast reactors will not be commissioned before 1985. The development of the nuclear power industry over the next 20-25 years presents itself as a combination of nuclear power stations with thermal reactors and with fast breeder reactors. Studies have shown that nuclear power stations with fast reactors have somewhat higher costs (20-30%) which, however, are compensated by lower electricity generating costs.

Rate of Development of Nuclear Power. According to the Directives of the Twenty-Fifth Congress of the CPSU, electrical generating capacity of $(13-15)\cdot 10^3$ MW will be added by nuclear power stations in the USSR in 1976-1980, i.e., the total capacity of nuclear power stations in 1980 will reach $(19-21)\cdot 10^3$ MW [1].

If 5-6 years is taken as the period for doubling the capacity of nuclear power stations, for making predictions it is convenient to use a function determining the rate of introduction of nuclear power stations in the form

Translated from Atomnaya Énergiya, Vol. 43, No. 5, pp. 365-369, November, 1977.

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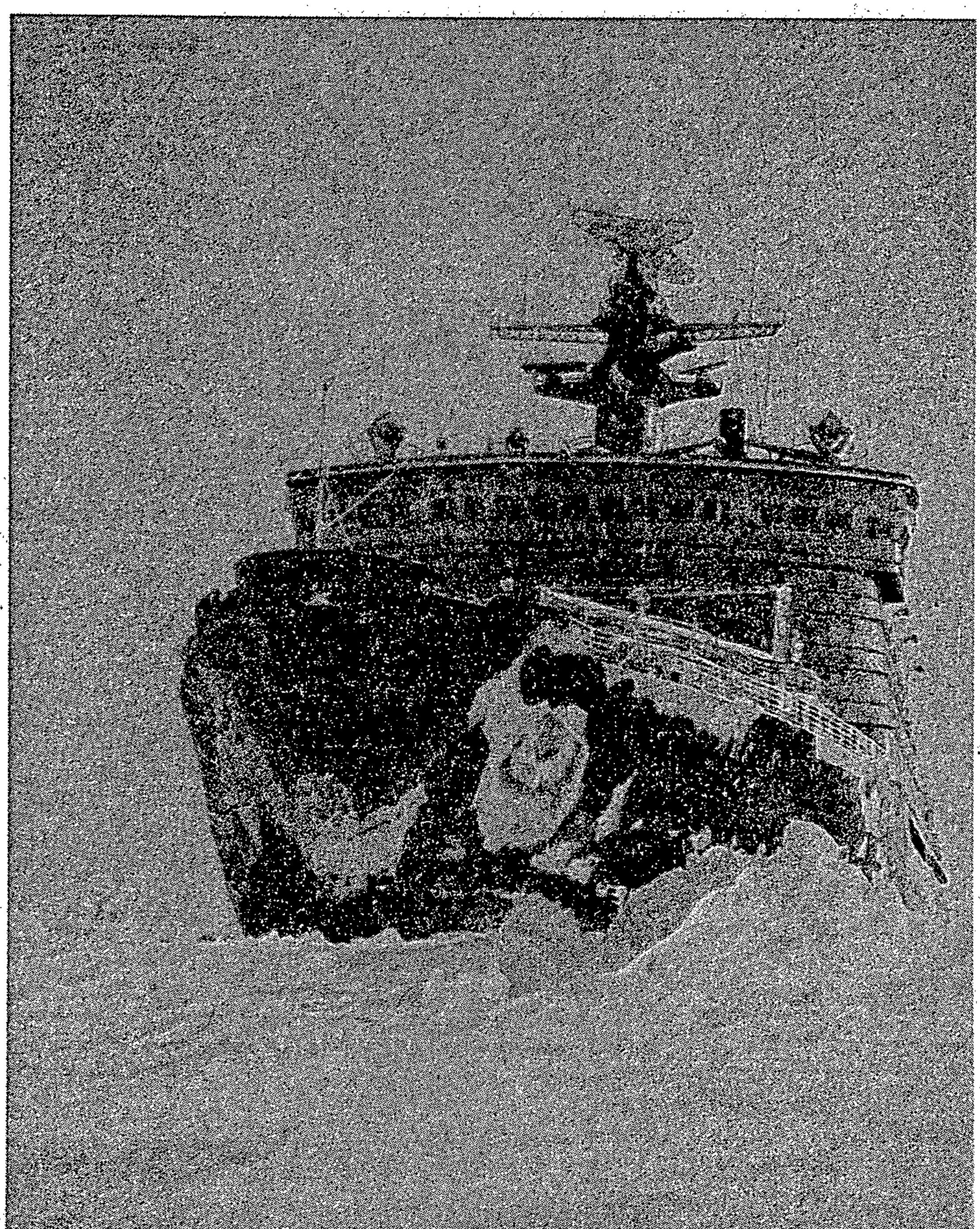

The icebreaker Arktika, put into service in 1975. The largest nuclear-powered ship in the world; shaft power 75,000 hp. The Arktika does important work for the national economy by convoying caravans of ships in the Arctic Ocean. In August 1977 the ship reached the geographical North Pole, the first surface vessel in history to reach this geographical place by active sailing.

$$N_E(t) = N_{0E} \exp[\alpha(t - t_0)] - N_1. \tag{1}$$

Here $\alpha = 0.1\ g^{-1}$; $N_{0E} = 40 \cdot 10^3$ MW(electrical); $N_1 = 19 \cdot 10^3$ MW(electrical); $t_0 = 1980$ (yr chosen by normalization conditions).

Equation (1) makes it possible to simplify the analysis and in a certain way to predict the development of the power of the fuel-cycle plants beyond the limits of the time interval specified in the Directives.

Below are some characteristics of the reactors for nuclear power stations that constitute the program's basis:

	VVÉR-100 [3]	RBMK-1000 [2]	BN-1600 [4]
Thermal capacity, MW	3000	3200	3750
Electrical capacity, MW	1000	1000	1600
Initial charge, ton	65	180	3(Pu_f)
Enrichment of initial charge, %	2.6	1.1	–
Annual recharging, tons	21	50	2.2(Pu_f)
Enrichment of stationary charge, %	4.4	1.8	–
Time to first unloading, yr	1	1	1
External fuel cycle, yr	1.3	1.3	1.3
Load factor	0.8	0.8	0.8

In the structure of the power industry, in order to obviate possible difficulties in placing orders with industry for reactor equipment, a 1:1 ratio is adopted for nuclear power stations with VVÉR and RBMK reactors. A significant role in the structure of the nuclear power industry will be played by nuclear power stations with fast reactors in view of the build-up of a considerable quantity of plutonium in the fuel of thermal reactors.

The demand for enriched uranium for the given growth rate of the power industry can be found from

$$G_0(t) + G_E(t) = \sum_i^2 G_{0i} N_{Ei}(t) + \sum_i^2 G_{Ei} \int_{1975}^{t} N_{Ei}(t - \tau_1)\, dt, \tag{2}$$

where G_0 is the fuel of the initial charge; G_E is the fuel of the stationary recharging. The quantity of enriched uranium, in tons, is easily found from Eq. (2).

<u>The Role of Nuclear Power Stations with Fast Reactors.</u> The most feasible way of using the plutonium produced in thermal reactors is in nuclear power stations with fast reactors.

With the given data taken into account, the commissioned capacity of nuclear power stations with fast reactors ($N_{F.E}$) can be described by the equation

$$\frac{dN_{F.E}(t)}{dt} = \sum_i^2 \frac{P_i}{G_F} \frac{dG_{Ei}(t - \tau)}{dt} + (C_0 - 1)\, q N_{F.E}(t - \tau). \tag{3}$$

Here P_i is the build-up of plutonium (Pu_f) in the fuel; G_F is the charge of plutonium in the complete fuel cycle of the fast reactor, in tons [$G_F = (2.1\text{–}2.2)G_0$]; C_0 is the conversion factor of the fast reactor (1.3); q is the quantity of plutonium burned in the reactor in a year, ton /yr*; G_{Ei} (t) is given by Eq. (2); τ is the plutonium residence time in the cycle for thermal and fast reactors ($\tau = \tau_1 + \tau_2 = 2.3$ yr). The solution of Eq. (3) is of the form

$$N_{F.E}(t) = \sum_{i=1}^{2} \frac{\beta_i P_i G_{Fi}}{G_F(1 + \gamma\tau)} \left\{ \frac{N_{0E}}{\alpha - \left(\frac{\gamma}{1 + \gamma\tau}\right)} \times \left(e^{\alpha(t - \tau - t_0)} - e^{\alpha(t_1 - \tau - t_0) + \frac{\gamma}{1 + \gamma\tau}(t - t_1)}\right) - \frac{N_1(1 + \gamma\tau)}{\gamma}\left(e^{\frac{\gamma(t - t_1)}{1 + \gamma\tau}} - 1\right) \right\} + N_2 e^{\frac{\gamma(t - t_1)}{1 + \gamma\tau}}, \tag{4}$$

where β_i is the fraction of reactors of the i-th type; t_1 is the time required to commission nuclear power stations with BN-1600 reactors, t_1 is the year — 1985; N_2 is the number of nuclear power stations with fast reactors which can be commissioned by 1985 in accordance with the size of the plutonium pile-up, $N_2 = 4.5 \cdot 10^3$ MW(electrical); and $\gamma = [(C_0 - 1)\, q]/G_F$.

With a burn-up of more than $20 \cdot 10^3$ MW·days/ton the plutonium pile-up ($^{239}Pu + {}^{241}Pu$) in VVÉR fuel is more than 6 kg [5, 6]. In the RBMK reactor, with a burn-up of $18 \cdot 10^3$ MW·day/ton, the plutonium pile-up ($^{239}Pu + {}^{241}Pu$) exceeds 2.5 kg [7].

Calculation by Eq. (4) shows that 13 GW(electrical) of capacity in nuclear power stations with fast re-

* It has been assumed that the fission of ^{238}U — ^{240}Pu and ^{242}Pu compensates for the radiation capture of neutrons on ^{239}Pu and ^{241}Pu.

actors can be commissioned by 1990 and 64 GW(electrical) by the year 2000.

If in Eq. (4) we set $\gamma = 0$, which corresponds to $C_0 = 1$, the capacity of nuclear power stations with fast reactors by the year 2000 will be 47 GW(electrical), i.e., will decrease by only 25%. It thus follows that the effect of only fast reactors on the rate at which capacity is commissioned in the next 20 years is not decisive. Therefore, it is not an indisputable assertion to say that effects must be concentrated over the next 10-15 years on the development of fast reactors with a maximum conversion factor to the detriment of the simplicity of design and operating reliability.

Let us consider how changes in the length of the external fuel cycle τ_2 for thermal and fast reactors affect the rate of commissioning of nuclear power stations (from the point of view of ensuring fuel). If τ_2 is reduced to 0.8 yr (fuel water-cooling time $\sim$0.5 yr) the capacity of nuclear power stations with fast reactors by the year 2000 will increase by 30% ($N_{F.E} = 85$ GW, $G_F = 5.4$ tons), whereas if τ_2 is increased to 3 years, $N_{F.E} = 37$ GW, $G_F = 11$ tons, i.e., the capacity is roughly halved.

Plutonium resources in the next 20-25 years practically will not limit the growth of nuclear power stations with fast reactors. Construction of such nuclear power stations will alow an approximate 25% decrease in the demand for natural uranium.

Basic Principles for the Development of Fuel Reprocessing Plants. The structure of the nuclear power industry poses a number of requirements upon the development of fuel reprocessing plants.

1. If plutonium is to be involved in power generation by being used in fast reactors, radiochemical plants for reprocessing fuel elements and plants for fabricating fuel elements for thermal and fast reactors must be built over the next 10-15 years.

2. The external fuel cycle for thermal and fast reactors should be no longer than 1-1.3 years (with plutonium incorporated into power generation).

3. The number of nuclear power stations with fast reactors commissioned depends on the size of the plutonium charge in the fuel cycle.

The size of the plutonium charge in the fuel cycle of a fast reactor can be found from the expression

$$G_F = G_0\,[1 + 1/T_{fl}\ (0.3 + T_c + T_{tr} + T_{rc} + T_{fabr}], \qquad (5)$$

where T_{fl}, T_c, T_{tr}, T_{rc}, and T_{fabr} are, respectively, the fuel lifetime in the reactor, the fuel cooling time, transportation time, duration of radiochemical reprocessing, and duration of fuel-element fabrication. The fuel reserve for the first reloading or refueling is 0.3 ($T_{fl} = 1.4$ yr).

When $T_c = 1$ yr and $T_{tr} = 1$ month, Eq. (5) becomes $G_F = G_0\,[1.98 + 0.71\,(T_{tr} + T_{fabr})]$.

If we proceed from the position that the plutonium charge in the fuel cycle should not exceed (2.1-2.2) G_0, then $0.16 \le T_{tr} + T_{fabr} \le 0.31$.

The chemical reprocessing of the fuel and the fabrication of fuel elements for a 1-yr charge for fast reactors should not exceed 1-1.5 months, i.e., the plants should have sufficient capacity for reprocessing the fuel (and for fabricating it into fuel elements) from seven or eight reactors a year. Therefore, fuel reprocessing and fabrication should be centrally organized since it is difficult to imagine that it would be possible to site seven or eight reactors in one small area.

The site for a plant must be chosen by taking account of the conditions for ensuring the safety of the population and the environment in the region in which long-term storage of radioactive waste can be organized. It is also necessary to organize transportation of fuel over considerable distances.

The Economics of Nuclear Power Generation and the Fuel-Cycle Enterprises. Studies carried out in a number of countries on the structure of the capital expenditures show that about 80% are spent on the construction of the nuclear power station and roughly 20% on the development of the fuel-cycle enterprises.

According to the averages of calculations made in various countries, the capital expenditures on nuclear power stations (vessel-type light-water reactors) with an electrical capacity of 1000 MW total 400-500 million dollars and, therefore, the capital expenditures on fuel-cycle enterprises to ensure the functioning of such a power plant run to 100-125 million dollars. (For convenience in making comparisons the costs are given in dollars [8].)

The principal effort in carrying out the program for the development of nuclear power should be directed at cutting the construction costs of nuclear power stations, increasing the efficiency and the reliability of the reactor and the thermal and mechanical equipment of the power plant, and simplifying the auxiliary systems.

Below we show the distribution of capital expenditures on various fuel-cycle enterprises, in % [9]:

Mining operations	50-60
Uranium enrichment plants	25-30
Plants for fuel-element regeneration, including waste transportation and reprocessing	10-15
Fuel-element fabrication plants	2-3

Thus, the highest capital expenditures are required for the development of mining operations and uranium isotope-separation plants. By contrast, plants for the radiochemical reprocessing of spent fuel with a capacity of more than 1000 tons/yr (in the general balance of costs), regardless of the technical difficulties, require relatively small capital expenditures, as do also the plants for the fabrication of fuel elements.

A typical distribution of the contribution of the costs of the various stages in the fuel cycle (in %) to the fuel component of the cost of the electricity generated in a nuclear power station with VVÉR and RBMK reactors is as follows:

Uranium costs	45-50
Uranium enrichment costs	20-40
Costs of fabrication of fuel elements (clusters, assemblies)	10-20
Costs of radiochemical fuel reprocessing	5-10

Nuclear power stations with VVÉR reactors entail large expenditures for fuel enrichment since the enrichment costs for them are roughly double those for the RBMK reactor.

The data cited above are based on the prices and separation costs characteristic for a number of countries: price of uranium 15 dollars/lb U_3O_8 (24.8 rubles/kg uranium) [8]; separation cost, 48 dollars (36 rubles)/kg [10]; fabrication cost of fuel elements (clusters, assemblies), 70 dollars (52.5 rubles)/kg uranium in fuel elements [11]; and cost of radiochemical reprocessing of fuel elements, 35 dollars (26.2 rubles)/kg uranium [11] (the costs in rubles have been converted from the dollar costs according to the 1976 rate).

The following general conclusions can be drawn about the organization of the fuel cycle.

1. The need to economize natural uranium makes it necessary that the plutonium produced in the fuel of fast reactors be incorporated into the cycle.

2. Since the contribution of the separation costs to the cost of the electricity generated is less than that of the cost of natural uranium, it is also desirable to save uranium by reducing the concentration of the tailings with some increase in the separation work required.

3. The process of radiochemical reprocessing of fuel from nuclear power stations gives a considerable saving of natural uranium, and, therefore, leads to lower capital expenditures on the fuel cycle and to lower electricity costs. Saving of uranium depends on both the return of regenerated uranium and the inclusion of an appreciable quantity of plutonium in the fuel cycle.

CONCLUSION

Plutonium breeding in fast reactors themselves will not have any significant effect in the next 20-25 years on growth rate of the capacity of nuclear power stations with fast reactors.

The character of the distribution of the capital expenditures on nuclear power stations and on the fuel cycle shows that the bulk of the expenditures is spent on the construction of the nuclear power stations and only 20% on the development of the fuel-cycle plants. For this reason, greatest savings of capital expenditure are given by a reduction in the construction costs of nuclear power stations by means of improvements in the design and of the reactor and the thermal and mechanical equipment of the plant.

In the fuel cycle the biggest economic effect is produced by measures that lead to a reduction in the specific consumption of natural uranium since the capital expenditure on the mining operations constitutes about one-half of the total capital expenditures on the fuel cycle. The natural uranium costs also make up roughly one-half of the fuel component of the cost of electricity generated by a nuclear power station. As the price of uranium rises, this fraction of costs will also increase.

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NUCLEAR-ENERGY COMPLEXES AND THE ECONOMIC AND ECOLOGICAL PROBLEMS OF NUCLEAR POWER DEVELOPMENT

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With the rapid rise in the volume of industrial production, it is necessary to plan its further development taking into account its effects on the environment [1-3].

Since the power industry is among the most important areas of production (in terms both of the scale of its own growth and of its effect on the development of other fields) and has a considerable impact on the environment, an optimal strategy for its future development is an important precondition for the successful solution of development problems as a whole.

At present, power stations are sited in accordance with the dispersed distribution principle in regions of high population density, reflecting the economic bias toward bringing the energy source close to the centers of the demand. Such a distribution involves the appropriation of large areas of useful land, intense consumption of water resources, and unavoidable effects on the local environment as a result of thermal and toxic emissions [4]. Therefore, if the rapid growth of the nation's energy requirements continues, this principle may lead to a situation in which the "ecological capacity" of such regions is exceeded.

As regards power, the ecological capacity of a given region determines the maximum permissible number of power sources in the region, in terms of the level of environmental effects. Despite its quantitative indeterminacy, the ecological capacity has clear limits. On the one hand, there is always the possibility of increasing the ecological capacity, e.g., by additional protective measures (purification of flue gases, waste water, and other emissions, use of closed technological cycles, etc.). On the other hand, the ecological capacity is limited by unavoidable thermal emissions, the accumulation of industrial wastes that arise in any form of purification, and the adverse effect of additional protective measures on the economics of energy production.

Translated from Atomnaya Énergiya, Vol. 43, No. 5, pp. 369-373, November, 1977.

Declassified and Approved For Release 2013/03/22 : CIA-RDP10-02196R000700100001-6

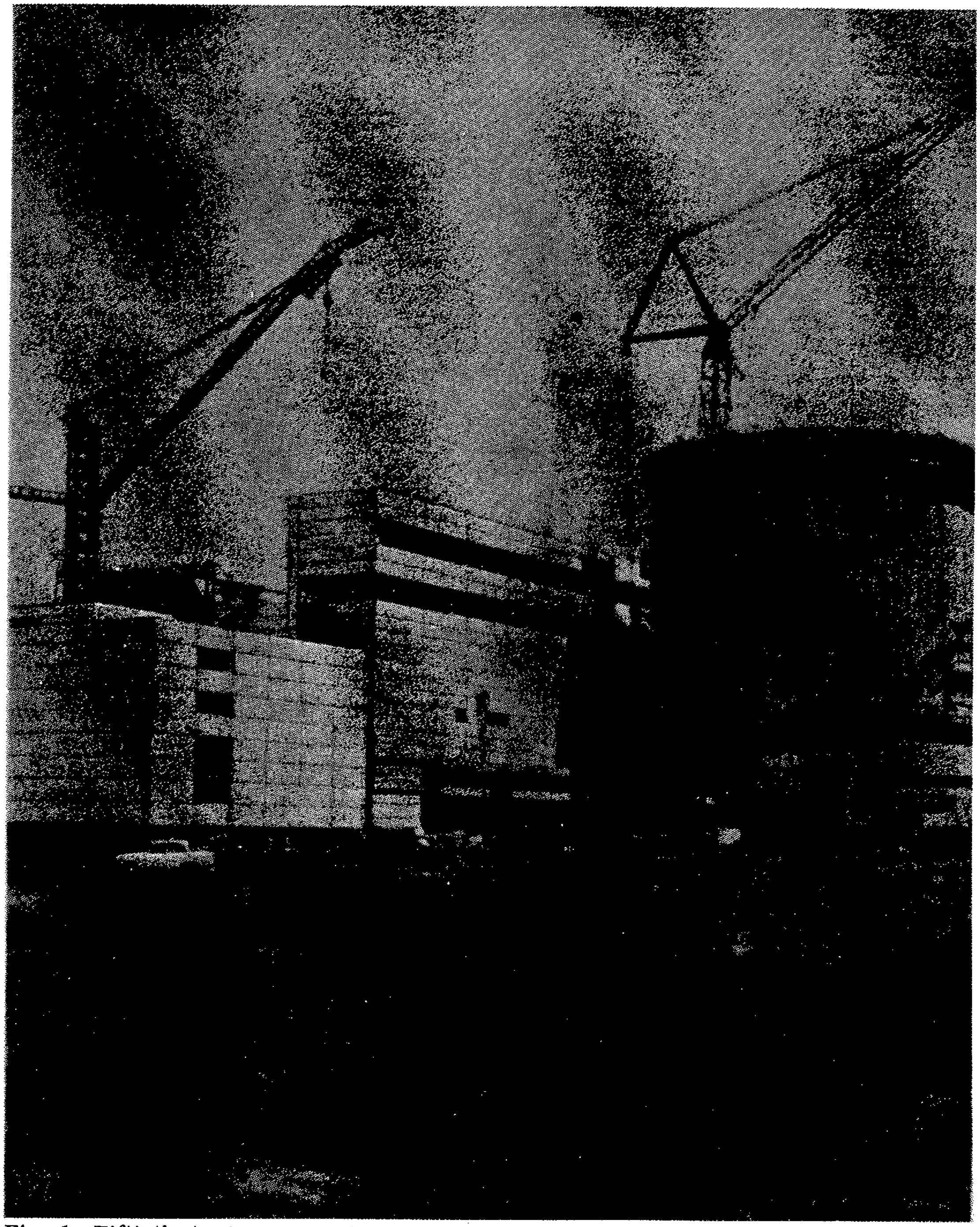

Fig. 1. Fiftieth-Anniversary-of-the-USSR Novovoronezh nuclear power station: the construction site for the fifth block with 1000-MW vessel-type reactor. Construction should be finished in 1978. In the future, blocks of this type will form the basis for the increase in power of nuclear power stations with vessel-type reactors.

The seriousness of the economic and ecological problems will evidently depend not only on the scale of development of the power industry but also on the effect of changes in its structure.

The distinctive feature of the modern power industry is the broadening of its fuel base through the introduction of nuclear energy resources (see Fig. 1). Long-term studies show that in the next 15-20 years atomic power stations will be the major electrical-energy sources in many countries and the required rate of power development will be provided mainly by the growth in nuclear power.

At present, nuclear-energy sources are used mainly for electrical-energy production, but urgent consideration is now being given to their use for centralized heat supply and in high-temperature technological processes.

In this connection, it is a matter of some importance for industrially developed countries to investigate the effect of dispersed atomic power station distribution in regions with high population density on the economic efficiency of nuclear power and also on the economic and ecological conditions for its development, taking into account the limits on the ecological capacity of these regions.

Economic and Ecological Conditions for the Development of Nuclear Power

It is known that the resource base and economic efficiency of nuclear power are entirely conditional on the achievement of a closed fuel cycle and the inclusion in the total number of electricity generating nuclear power stations of a sufficient number of stations with fast reactors [5]. This factor cannot fail to affect the economic and ecological conditions for the long-term development of nuclear power, which will depend on the territorial distribution of the individual facilities comprising a single fuel cycle.

The territorial dispersion of facilities for nuclear-fuel enrichment, fuel-element production, spent-fuel regeneration, and the treatment and storage of radioactive wastes corresponding to the dispersed distribution of atomic power stations in regions of high population density isolates the elements of the fuel cycle and necessitates the bulk transportation of unirradiated nuclear fuel to the power station and of highly radioactive spent fuel from the power station. From an ecological point of view, such transportation involves additional risks of radioactive contamination of the environment, increasing as nuclear power develops and the rate of fuel usage rises.

In addition, the dispersion of the elements of the fuel cycle and the technical difficulties of transporting irradiated nuclear fuel, associated with the current practice of prolonged holding of the fuel at the power station after its removal from the active region of the nuclear reactor, lead to increase in the duration of the external fuel cycle and reduction in the rate of introduction of atomic power stations with fast reactors and this, together with the extra cost of developing special means of transportation, reduces the economic efficiency of nuclear-power production.

It is also known that the replacement of organic fuel by nuclear fuel is accompanied by a change in the cost structure of energy production. This change is due to the reduction in specific cost of the fuel and the increase in the capital outlay in the creation of the basic atomic power station stock, which depends on the unit power of the nuclear power stations and the concentration of power per atomic power station site. Therefore the need to increase the unit power of atomic power stations is related not only to the rate of introduction of nuclear power but also to the conditions for increasing the economic competitiveness of atomic power stations.

Moreover, the need to increase the rated electrical power of atomic power stations and also the lower thermodynamic efficiency of the energy cycle of atomic power stations with thermal reactors in comparison with organically fueled power stations leads to difficulties in the choice of site, increase in the area of valuable land removed from other uses, and more rapid consumption of water resources; in short, the ecological capacity of the regions surrounding atomic-power-station sites would soon be exceeded.

Thus, in the long term, the principle of disperse atomic-power-station distribution in regions of high population density may adversely affect the economic and ecological conditions for nuclear-power development. In view of the particular requirements of nuclear fuel, and the current and anticipated levels of technological achievement, the application of this principle to nuclear power may have even more detrimental effects both on the economic efficiency of power production and on the environment than in the case of its application to organically fueled power production.

Nuclear-Energy Complexes and the Economic and Ecological Conditions of Nuclear-Power Development

Some industrially developed countries have regions which are characterized by a lower bioclimatic potential and a less acute scarcity of land and water resources.

This suggests the possibility of combining atomic power stations into large nuclear-energy complexes, far removed from densely populated regions and connected to centers of energy demand by electrical transmission lines; such complexes would include, in one site, not only a group of atomic power stations with a total power output of the order of 10-50,000 MW but also facilities for the whole of the external fuel cycle. This approach offers the possibility of a relatively simple solution to the optimal organization of the whole nuclear-power industry.

TABLE 1. Approximate Estimate of Change in Building and Running Costs of Atomic Power Station for Two Distribution Strategies

Item of expenditure	Cost referred to capital expenditure, %	
	dispersed distribution	distribution in power complexes
Storage of spent fuel at power station	1-2	0.1
Containers and transport costs for nuclear fuel	4-6	0.5-1
Compensation for value of occupied land	5-10	0.1
Water supply	10-15	3-5
General and auxiliary facilities	15-20	1-2
Housing (with supportive structures)	10-15	5-10
Losses due to time of power-station construction	10-15	1-2
Energy transmission (dc transmission lines and substations)	–	50
Total:	50-80	60-70

One factor of particular importance for the creation of nuclear-power complexes is evidently the successful completion of the development program for fast-reactor power stations, which should play a central role in such complexes. It should be stressed that the specific features of nuclear-power production by fast reactors recommend them (after the production of the first commercial reactors) for inclusion in such complexes. However, even in the case of delay in the fast-reactor program, it is worthwhile to consider the development of nuclear-energy complexes based solely on thermal-reactor power stations.

In addition to the basic arguments noted above in favor of nuclear-power complexes, their introduction would be expected to yield the following benefits:

the opening up and economic development of neglected regions and hence a more uniform distribution of industrial development;

more favorable conditions for the use of spent fuel and, possibly, for the storage of radioactive wastes;

more effective organization of the construction and use of nuclear-power facilities and their control, and also more effective use of labor resources;

the possibility of rapid regeneration of uranium and plutonium for atomic power station use;

the development of technology for the retreatment of irradiated nuclear fuel with a minimum of storage (150-200 days or less);

increased safety of storage of nuclear materials, through more rigorous monitoring and the elimination of external transportation;

the opportunity to reduce the scale of nuclear-power developments in densely populated regions without reducing the rate of nuclear-power development as a whole.

Of course, the creation of nuclear-power complexes will require solutions to many large and qualitatively new problems, the most important of which are as follows:

the organization of an economically efficient and reliable means of bulk energy transmission;

qualitative and quantitative investigations of the effects of nuclear-energy complexes on the local environment and the development and implementation of measures to maintain such effects within acceptable limits;

organizational and sociological problems associated with the establishment both of the nuclear-power complexes themselves and of the associated local infrastructures.

Therefore it is clear that the expediency of introducing nuclear-energy complexes, despite the evident advantages of such a program for the long-term development of nuclear power, is a very complicated issue and cannot be decided purely on the basis of economic estimates, without a preliminary investigation of the above-mentioned problems.

However, it may be of interest, at a preliminary stage, to carry out a comparative economic evaluation of the long-term effects of two strategies for the territorial distribution of nuclear-power facilities:

disperse distribution of atomic power stations in densely populated regions, associated with territorial separation of the elements of the closed nuclear-power fuel cycle and bulk transportation of irradiated nuclear fuel;

the location of atomic power stations in thinly populated regions within nuclear-energy complexes and the bulk transmission of electrical energy to the centers of energy demand in densely populated regions.

Approximate Estimates of the Economic Conditions for the Creation of Nuclear-Energy Complexes

As a preliminary stage in comparing the efficiency of the two distributional strategies for nuclear-power facilities, approximate estimates have been made of the economic conditions for the creation of large nuclear-energy complexes, considering as an example the use of nuclear power as a source of energy supply for the European region of the USSR [6].

In the European region of the USSR, both the energy demand and the population density are highest in the area to the west of the Volga—Volga-Balt line. Furthermore, there is a scarcity of organic forms of fuel in this area and, in the long term, it is likely to be dependent on nuclear sources of energy. The areas to the north and north east of this line are characterized by low population density and more ample land and water resources, and are comparatively close to the centers of energy demand. Therefore it is of practical interest to estimate the economic conditions for the creation, in these areas, of between one and three complexes of power output 40-50,000 MW, each with energy transmission over a radius of 1500-3000 km.

The calculation is based on the assumption that it is possible to change the building and running costs of atomic power stations by a gradual transition from disperse distribution to nuclear-energy complexes and additional expenditure on energy transmission. Allowing for a 30% rise in the building costs of nuclear-energy complexes, the two principles of power-station distribution were found to be equally economical in terms of capital outlay. The use of dc electrical-transmission lines was assumed; these are thought to be sufficiently efficient for long-range bulk energy transmission. Generalized results are given in Table 1, in which the items of expenditure are expressed as a percentage of the capital investment in the atomic power station.

The estimate shows that, in economic terms, the conditions for the creation of large energy complexes are analogous to the conditions for the construction of an equivalent number of isolated atomic power stations, even allowing for a 30% rise in the building costs of nuclear-energy complexes. This is because the additional costs of energy transmission are compensated by the economic efficiency of concentration of power sources, the saving of scarce water and land resources, the effect of reducing the number of sites, and the possible reduction in the time for atomic-power-station construction. Thus, the economic analysis of the creation of large nuclear-energy complexes may be regarded as favorable.

It is likely that many countries will now begin to study the whole set of scientific, technical, economic, organizational, sociological, and other problems, with particular reference to local conditions. Such investigations will provide a basis for more accurate economic and ecological analyses and for decisions as to the expediency, the specific locations, and the timespan for the creation of nuclear-energy complexes.

Prospects for Expansion in the Use of Nuclear-Energy Complexes

In the preceding sections, nuclear-energy complexes have been considered as a means of intensive growth in electric power production. It has been assumed that the creation of energy complexes in less developed regions will be accompanied by the continued construction of individual atomic power stations in densely populated regions.

In view of the anticipated role of nuclear power stations in supplying the variable portions of the energy-demand curve, and taking into account the need to bring nuclear reactors more broadly into use, in particular, for centralized domestic and industrial heat supply and high-temperature technological processes [7, 8], these functions may evidently be assigned in the long term to the atomic power stations located in densely populated regions with a developed infrastructure.

Electrical energy takes up a relatively small proportion (~25-30%) of total fuel consumption, industry and heat supply making up the larger fraction. Also, the basic electrical load amounts to approximately 60% of the total electrical-energy production and tends to decrease with time.

Nuclear-energy complexes will evidently play a larger role in the long term if additional functions can be found for them.

Whereas it is possible in principle for nuclear-energy complexes to supply the variable portion of the electrical-demand curve, it will require extensive grid construction to increase the carrying capacity of electrical transmission lines and also the solution of serious technical problems of the long-range transmission of variable electrical loads.

It is at present uncertain whether nuclear-energy complexes can play a role in supplying the thermal load. This will possibly become clearer with the development of a means of energy transmission in the chemically bound state, using a closed cycle of reversible chemical reactions.

It is possible in principle for nuclear-energy complexes to play a role in production technology, in particular, the large-scale production of hydrogen — either by the electrolysis of water (in particular, exploiting variations in the energy-demand curve) or by the use of high-temperature reactor technology for the development of conversional methods of hydrogen production and thermochemical processes for the dissociation of water. Possibly it will be expedient to locate thermochemical factories using highly toxic intermediate reagents within nuclear-energy complexes.

CONCLUSIONS

In the long term, serious economic and ecological problems may arise if the required scale and rate of growth of power production is to be ensured while retaining the principle of dispersed location of ever larger power stations in regions with high population density and a developed infrastructure, since the ecological capacity of these regions is limited. This is particularly true of nuclear power, which in the long term will become the main source of electrical energy.

Prospects for a solution to the economic and ecological problems of nuclear power and for an increase in its economic efficiency are offered by the construction of nuclear-energy complexes — large industrial units that contain within a single site a group of atomic power stations of total rated power output of the order of 10-50,000 MW and also the facilities of the external fuel cycle, the complexes being remote from densely populated regions but connected to the centers of energy demand by electrical transmission lines.

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ATOMIC ENERGY AND THE ENVIRONMENT

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Protection of the environment against all forms of pollution is attracting great attention in the USSR. In recent years, the Central Committee of the Soviet Communist Party and the Council of Ministers of the USSR have issued a number of regulations and pronouncements, intended to provide increased safeguards against the pollution of the atmosphere, reservoirs, the soil, and mineral resources.

At the introduction of any new industrial enterprise, the whole purification system must be completely operational. It is necessary at the design stage itself to develop plans for the prevention of environmental pollution by air-borne emissions and waste water. All these requirements apply also to nuclear-power installations and other undertakings which may be a source of radionuclide contamination of the biosphere.

Nuclear power has been developed in conditions when modern industry, with all its wastes, has already had, and is continuing to have, serious adverse effects on nature. It is sufficient to note that every year more than 200 million tons of carbon dioxide, more than 50 million tons of various hydrocarbons, almost 150 million tons of sulfur dioxide, more than 50 million tons of nitrogen oxides, and 250 million tons of finely dispersed air-borne particles are released into the earth's atmosphere. In the last few decades, the concentration of carbon dioxide in the atmosphere has risen by 10-12% and the content of dust particles has increased by 12% in the last ten years alone. The proportion of the atmospheric content of sulfur due to human activities is 93% in the northern hemisphere and 47% in the southern hemisphere. The total emission of chemicals as a result of human economic activity is presently comparable in quantity with natural processes in the biosphere. This burden on the environment is no longer a matter of purely local or regional significance but is becoming a global problem. These developments have lent particular significance to Engels' warning: "We should not, however, carry too far our conquests of nature. Nature takes its revenge for every such conquest. In the first place, it is true, each one has the consequence that we intend but in the second and third places it has other, unpredictable consequences, which very often negate the effect of the first." (Marx and Engels' Collected Works [Russian translation], Vol. 20, pp. 495, 496).

The major source of environmental pollution by dust and oxides of sulfur and nitrogen is the operation of thermal power stations (TPS). In particular, in the U.S.A. TPS contribute up to 70% of the total emission of sulfur dioxide, up to 20% of the dust, and up to 40% of nitrogen oxides.

Damage due to environmental pollution amounts to billions of dollars; according to the data of the US National Agency for Environmental Protection, the figures for 1969 were 20 billion dollars in the U.S.A., around 1 billion dollars in England, and 3 million dollars in France.

Evidently, the pollution of the biosphere by industry is a long-term problem. Therefore the need is not to make the whole of industry nonpolluting (which is unrealistic at present) but to limit environmental pollution to a level at which natural systems do not lose their capacity for self-purification, recovery, and development, and to prevent harmful effects of industrial wastes on human health.

The analysis of environmental pollution by industrial wastes, emissions, and discharges should not be restricted within an isolated field, in particular, nuclear power. Correct solutions may only be obtained against the background of a general picture of existing environmental pollution, together with scientifically based predictions for the future. The present paper gives the broad outlines of such a picture. In addition, the load on the population due to atomic power station emissions is compared with that from other sources, including the emissions of coal-fired TPS.

Radioactive emissions and wastes from atomic power stations (APS) have a number of distinctive features as compared with conventional power-industry wastes: firstly, the process of radioactive contamination is insidious and imperceptible to humans, even in the case of very high, lethal radiation doses; secondly, the wastes

Translated from Atomnaya Énergiya, Vol. 43, No. 5, pp. 374-384, November, 1977.

0038-531X/77/4305-1025$07.50 ©1978 Plenum Publishing Corporation

TABLE 1. Total Emission of the Two NVAPS VVÉR-440 Reactors in Comparison with LDP, Ci/yr

Nuclide	LPD	1972	1973	1974
RIG	$1,3 \cdot 10^6$	400	2140	2056
^{131}I	36	$1,6 \cdot 10^{-3}$	$6,6 \cdot 10^{-3}$	$5,9 \cdot 10^{-3}$
^{89}Sr	0,36	—	$3,7 \cdot 10^{-5}$	$1,4 \cdot 10^{-5}$
^{90}Sr	—	$2,2 \cdot 10^{-5}$	$1,2 \cdot 10^{-5}$	$1,1 \cdot 10^{-5}$

forming in the operation of a nuclear reactor are of high specific and total activity. One further characteristic of radioactive wastes is that they are of much smaller volume and mass than conventional power-industry wastes, which means that it is easier to prevent harmful effects on the environment and human health. It is no exaggeration to say that the problem of nuclear power is above all the problem of radioactive wastes.

By 1976, 19 countries were producing nuclear energy, from 140 APS of total power 74,200 MW and electrical output 380 billion kW-h (140 million tons of conventional fuel), and the number of stations under construction or completed throughout the world was more than 300. Between 1970 and 1976 the total APS power increased by a factor of 5. This increase was the result not only of the overall increase in energy demand but also of the trend in many countries to independence from supplies of fossil fuel. In this respect, the rates of APS construction in Europe are significant: in the EEC countries, the APS power increased by 5 GW in 1975 alone, giving a total of 18 GW, and the intention is to increase APS power to more than 200 GW by 1985; the corresponding figure for the U.S.A. is ~300 GW. An equally rapid growth of nuclear power is planned for the USSR. In the period 1976-1980 it is intended to introduce APS of power 13-15 GW, increasing the present total by a factor of 2.5.

Before considering the radioactive contamination of the environment due to nuclear power, it is expedient to study the existing radiation background from other sources of ionizing radiation and to examine current ideas on the biological effects of small doses.

Biological Effects of Small Doses

At present, authoritative international bodies assume that the effect of ionizing radiation is thresholdless, i.e., that any additional radiation is associated with a definite risk of tumor-formation and genetic damage. It is also assumed that the extent of such injury is directly proportional to the dose although even the smallest of doses is theoretically capable of producing such effects. The actual ability of small doses (at the level of the natural radiation background and below) to cause cancers and genetic damage has not been proven. Equally, however, the converse has not been proven. Therefore, to ensure that safety provisions are adequate, it is considered expedient to assume a thresholdless effect of radiation and a linear yield of injuries. At present, quantitative dose—effect relations for the total-dose range $10\text{-}10^3$ rem have been fairly well determined. These relations have been extrapolated to the region of low doses, where statistical difficulties hinder direct observation. A collective dose of 10^6 man-rem/yr on the population would be expected to lead to 150 additional deaths per year from cancer and about 40 from genetic injuries in the first and second generations [1-4]. Naturally, the extrapolation of relations obtained at $10^2\text{-}10^3$ rem to small doses must be subject to some reservations. Essentially, these figures reflect the maximum unfavorable outcome.

Existing Levels of Radiation

Natural Background. Modern man suffers the effects of very different sources of ionizing radiation, ranging from the natural background to various domestic appliances. Universally, as is known, the largest factor is the effect of cosmic radiation and natural radionuclides in the soil and the human organism. The bulk of the population of the USSR lives in lowland territories up to 500 m above sea level, where the mean radiation dose over the whole body is about 60 mrem/yr. The radiation of the population due to radioisotopes in the soil depends not only on their concentration but also significantly on human living conditions. Thus, housing notably reduces the γ-radiation intensity (by a factor of 3-10 depending on the building materials). Taking account of this factor and also the mode of existence of the population, the weighed mean γ-radiation dose from the soil for the population of the USSR is about 30 mrem/yr. Thus, the total external-radiation dose is 90 mrem/yr.

TABLE 2. Normalized Annual Air-Borne Emissions of Soviet Series APS, Ci/MW (electrical)-yr

APS	Radionuclide	1974	1975
NVAPS, VVÉR-440	RIG	3,28	42,4
	^{131}I	$9,4\cdot10^{-6}$	$4,5\cdot10^{-5}$
	^{90}Sr	$5,3\cdot10^{-5}$	—
	^{137}Cs	$5,1\cdot10^{-6}$	$2,6\cdot10^{-5}$
	$^{141,144}Ce$	$2,7\cdot10^{-6}$	$1,0\cdot10^{-5}$
	^{60}Co	$3,0\cdot10^{-7}$	$7,5\cdot10^{-6}$
KAPS, VVÉR-440	RIG	3,4	5,1
	^{131}I	None observed	$3,6\cdot10^{-9}$
	^{137}Cs	"	$4,0\cdot10^{-12}$
	^{110m}Ag	"	$4,3\cdot10^{-9}$
LAPS RBMK-1000	RIG	137	328
	^{131}I	0,01	0,019
	$^{89,90}Sr$	$3\cdot10^{-4}$	$8,3\cdot10^{-5}$

The internal-radiation dose arises because natural radioactive elements gain access to the organism with food, water, and air. Predominant among them are ^{40}K and radionuclides of the uranium and thorium family. The mean internal-radiation dose for the population of the USSR is about 20 mrem/yr. Thus, the radiation dose for the human organism as a whole due to natural sources of ionizing radiation amounts to 110 mrem/yr. The irradiation of the U.S. population is approximately the same (130 mrem/yr) [5]. This indicates that the dose due to the natural radiation background averaged over a large human population is roughly the same for different regions of the earth (except for certain specific areas).

Medical Irradiation. In terms of the levels of radiation dose, medical x-ray treatments rank second after the natural background and first among all forms of radiation due to human activity. The general improvement in the medical services to the population is accompanied by a rise in the number of treatments and a corresponding increase in the radiation. In the USSR the mean annual increment in the number of x-ray diagnostic treatments over the last decade has been about 3%, i.e., more than the increase in population. The rate of growth of radioisotope treatments is even more considerable (up to 10% per year). There is considerable scope for a reduction in the radiation load due to such uses of ionizing radiation by changing the procedure and the structure of the treatment and also by using technically improved equipment. The UN Scientific Committee on the Effect of Atomic Radiation (SCEAR) points out, with good reason, that a decrease, for example, of the genetically significant dose due to medical treatments by only 10% (which is quite possible) would balance the irradiation due to nuclear power, taking into account its probable development in the next few years.

On the average, each person in the USSR is exposed to one x-ray diagnostic treatment per year, with contributions from radiography, radioscopy, and fluorography (0.44, 0.19, and 0.37 of a treatment per year, respectively). The dose over the whole body due to such treatments is about 72 mrem/yr, more than 50% of which is due to radioscopy. These figures are roughly comparable with those for other industrially developed countries, for example, the U.S.A. [5, 6]. A higher dose (up to 130 mrem/yr) is formed in the bone marrow and up to 86 mrem/yr in the gastroenteritic tract (GET). The radiation due to the medical-diagnostic use of ^{131}I, ^{57}Cr, ^{191}Au, ^{32}P, etc., is significantly less (no more than 1 mrem/yr), except in the case of the thyroid gland, for which the dose is 100 mrem/yr, i.e., approximately equal to the dose due to the natural radiation background. Note that in large towns with a higher level of medical service the dose due to x-ray and radioisotope diagnostics is $\approx$ 2-2.5 times higher than the national average and the radiation dose for the thyroid gland in Moscow, for example, is nine times higher than the national average.

Radiation in Stone Buildings. As shown by recent research [7-9], natural radionuclides in building materials of mineral origin play an important role in the irradiation of the human organism. The dose rate is ~12 μrad/h in brick and concrete buildings and ~6 μrad/h in wood. Taking into account the size of the urban and rural populations in the country, this level gives 60 mrem/yr over the whole body due to external radiation. Internal radiation is due to the emission of ^{222}Rn and ^{228}Tn from building materials and their entry into the organism with inhaled air. In this case, of course, the maximum dose appears in the lungs and, in particular, the basal cells of the bronchial epithelium. Thus, whereas the mean radiation dose of the lungs due to ^{222}Rn and ^{228}Tn and their decay products is about 400 mrem/yr, the radiation dose in the bronchial epithelium may be higher by a factor of five. The radiation dose of other organs, for example, the gonads and the bone marrow, is only 2-4 mrem/yr. Up to 10% of all lung-cancer deaths are apparently caused by radionuclides released into the atmosphere of human residences from building materials of mineral origin.

TABLE 3. Radiation Dose of Population due to Air-Borne APS Emissions

Organ	1976			1980		
	individual dose, mrem/yr		population* dose, man-rem/yr	individual dose, mrem/yr		population† dose, man-rem/yr
	VVÉR	RBMK		VVÉR	RBMK	
Whole body	1,1	0,78	272	2,9	3,1	1380
Bone tissue	1,48	0,80	342	4,0	3,1	1420
Thyroid gland	3,2‡	35,3‡	1751	8,6‡	140‡	$1,74\cdot10^4$
Lungs	$1,1\cdot10^{-3}$	$1,5\cdot10^{-5}$	$5\cdot10^{-2}$	$3\cdot10^{-3}$	$6\cdot10^{-5}$	0,16
GET	0,93	$9,0\cdot10^{-3}$	76	2,5	$3,6\cdot10^{-2}$	237

*At the actual total APS power of 3100 MW (electrical).
† At the planned total APS power of 17,000 MW (electrical).
‡ Dose in children of age up to one year.

Radiation due to TPS. Another important source of radioactive contamination is the production of thermal and electrical energy from organic fuel (coal, petroleum, etc.). Despite the development of electrical-energy production at APS, power stations based on organic fuel (especially coal) will remain the major source of electrical energy in the next few decades. In 1974 645 billion kW-h of electrical energy were produced in the USSR, about 300 billion kW-h of which (i.e., about 50%) was from coal TPS. In the U.S.A. also, coal TPS accounted for approximately 50% of the energy produced [10], while in western Europe the figure was as much as 70% [11]. By 1980, in accordance with "Main Items in the Development of the Soviet Economy in 1976-1980," issued by the Twentieth Session of the Soviet Communist Party, electrical-energy production will have practically doubled in comparison with 1974, reaching 1340-1380 billion kW-h, mainly from coal TPS.

In these conditions it is naturally of great importance to estimate the effect of TPS emissions on the environment and human health. Without studying all the aspects of this problem, consider the emissions of natural radionuclides. Until recently, analyses of the effect of TPS emissions on the environment and human health concentrated on the chemical components (CO, SO_2, etc.) and dust. However, as shown in [7, 12-15], the dispersal of natural radioisotopes (^{226}Ra, ^{228}Ra, ^{232}Th, ^{210}Pb, ^{210}Po, ^{40}K) released in smoke leads to additional external and internal radiation of the population living close to the TPS. The amount of smoke produced depends on the form of fuel, the combustion conditions, the efficiency of dust collection, and other factors, and may vary considerably. In Soviet TPS the efficiency of collection is practically 70-80%, which is less than the value in a number of other countries (up to 97-98%). Increasing the efficiency of dust collection is an obvious means of reducing the environmental load. In the USSR modern coal-fired TPS require for 1000 MW(electrical) 2.8 million tons of coal and release ~ 0.1 million tons of solid particles into the air. Assuming that this dust is uniformly distributed around the TPS over a radius of 15-20 km, the radiation of the population in this region will be characterized by the following approximate mean doses: 0.5 mrem/yr over the whole body; 15 mrem/yr for the bone marrow; 41 mrem/yr for the lungs. The last two figures are very high, and cannot be disregarded. The mean radiation dose of the whole national population, normalized to 10^3 MW(electrical), is $5.2\cdot10^{-5}$ mrem/yr over the whole body; $2\cdot10^{-3}$ mrem/yr for the bone marrow; $7.5\cdot10^{-3}$ mrem/yr for the lungs. As the total TPS power in the USSR is $70\cdot10^3$ MW, the actual dose is 70 times higher, i.e., $3.6\cdot10^{-3}$, 0.18, and 0.53 mrem/yr. These values are obtained by calculations taking parameters typical for modern TPS.

Global Fallout. Nuclear-weapons tests have resulted in a stratospheric accumulation of long-lived artificial radionuclides, fallout from which produces general global radioactive pollution of the biosphere. This leads to radiation of the human population by external radiation sources from the earth's surface and also by internal sources due to the migration of radionuclides in biological and food chains. As is known, despite the treaty prohibiting nuclear-weapons tests in the atmosphere, in space, and in the sea (1963), test detonations continue even today, though on a much smaller scale. Thus, the problem of global fallout has lost none of its importance. Altogether, past nuclear tests amount to 530 Mtons, more than 90% of which is for the period up to 1963. As a result, the biosphere has been exposed to up to 3000 MCi ^{3}H, 3000 MCi ^{95}Zr, 600 MCi ^{144}Ce, 360 MCi ^{106}Rh, 30 MCi ^{137}Cs, up to 16 MCi ^{90}Sr, up to 6 MCi ^{14}C, and ~350 kCi ^{239}Pu [16]. In addition, immediately after a nuclear explosion, short-lived radionuclides (especially ^{131}I and ^{89}Sr) are an important source of radiation. The fallout of nuclear-fission products tends to be latitudinal, although this is truer of external radiation than of internal radiation since food products — the main means of access of radionuclides to the human organism — are as a rule not consumed in the place where they are produced in a centralized supply system. Therefore the fluctuations of the mean dose for large population groups over the territories of the Soviet Union are negligible, the only exceptions being certain regions (the subarctic region, the Polesie), where natural and social factors lead to higher levels of ^{137}Cs and, to a lesser extent, ^{90}Sr in the food intake.

What is the present extent of the radiation of the Soviet population as a result of global fallout? The external radiation of the human organism is determined almost totally by ^{137}Cs, although immediately after a nuclear test short-lived nuclides (especially ^{95}Zr and ^{95}Nb) make a significant contribution. The internal-radiation dose is mainly due to ^{90}Sr and ^{137}Cs. Recently, the mean figures for the USSR have been 1.3-1.4 mrem/yr for the external-radiation dose over the whole body and 0.6 mrem/yr for the internal dose, i.e., altogether 2 mrem/yr. The total radiation of the bone marrow is 4.2 mrem/yr (3.2 mrem/yr due to ^{90}Sr) and 1.6 mrem/yr for the lungs. In the U.S.A. the radiation dose over the whole body due to global fallout is 2-4 mrem/yr, according to various estimates [5, 6, 17]. In the inhabitants of certain specific regions the internal radiation due to incorporated ^{137}Cs is found to be 10-100 times higher than the national average.

Other Radiation Sources. The radiation due to domestic clocks results from the use of ^{226}Ra, ^{3}H, and recently ^{147}Pm in the luminescent material on the faces; the dose from ^{147}Pm is negligible, while the dose from the other two nuclides may be of order 1 mrem/yr [17]. The radiation due to television sets is produced mainly by the tube itself and, despite the measures taken to reduce the radiation from the tube, with the development of color television the mean individual and collective dose will evidently increase. At present, the dose is less than 0.1 mrem/yr [17].

High-altitude airplane flights expose passengers and baggage to an increased level of cosmic radiation. At a height of 9000 m in a modern jet, the absorbed dose rate may reach 0.7 mrem/h [17, 18]. Taking into account the volume of jet passenger flights in the USSR, the mean radiation dose over the whole body is evidently no more than a few tenths of a mrem/yr.

Overall Radiation Doses. The mean individual radiation doses, mrem/yr, of the population of the USSR in 1975-1976 from various sources of ionizing radiation are as follows:

Natural background	110
Medical x-ray diagnostics	72
Building materials	60
Global fallout	2
Luminescent clock faces	1
Air transport	< 0.5
Television sets	< 0.2
TPS	$3.6 \cdot 10^{-3}$

Radiation due to Nuclear Power

Fuel Cycle. The nuclear fuel cycle includes the mining and processing of uranium ore, the extraction of fuel materials and the preparation of fuel elements, the treatment of spent fuel, the removal and storage of wastes, and also the transportation of radioactive material in various forms between individual stages of the fuel cycle.

The negligible spread of radionuclides in the area around the uranium mines leads to practically no additional radiation of the population distinguishable from the natural radiation background. According to estimates of the mean radiation dose of the whole body for the population living within a radius of 100 km from the mine, the consequence of inhalation and oral intake of radioactive material does not exceed 10^{-2} mrem/yr [17].

The radiation of the population due to plants for uranium-ore processing and fuel-element production is slight, because the majority of uranium compounds are solids whose wastes are relatively easy to collect. Liquid wastes are stored in reservoirs or tanks, and there is practically no migration. The radiation dose of the population of the U.S.A. living within 100 km of such a plant for an annual air emission of 0.01 Ci has been estimated at $1.7 \cdot 10^{-6}$ mrem/yr for the whole body and that for a liquid discharge of 1 Ci/yr at $4 \cdot 10^{-4}$ mrem/yr. In processing nuclear fuel large quantities of radionuclides are isolated and some of these (e.g., ^{3}H, ^{14}C, ^{85}Kr, ^{129}I) are scattered over large distances. Therefore not only populations living in the vicinity of such plants are exposed to radiation but also those in remote areas (though to much smaller doses). Detailed calculations of the radiation dose for the population living within 100 km of a processing plant for the nuclear fuel of the LWR and FBR reactors in the U.S.A. of output 300 tons/yr show that the radiation dose is 0.2 and 0.5 mrem/yr, respectively, for the whole body and 11.0 and 4.0 mrem/yr for the lymph nodes of the respiratory system. Averaged over the total U.S. population, the whole-body radiation dose from the discharge of such plants is $\approx 10^{-3}$ mrem/yr [17].

TABLE 4. Additional Risk of Tumors due to Radioactive Emissions of APS and TPS (theoretical), cases/yr

Injury	APS		TPS	
	for 10^3 MW	for actual power in 1976	for 10^3 MW	for actual power in 1976
Leucosis	$1{,}2\cdot10^{-3}$	$3{,}7\cdot10^{-3}$	$2\cdot10^{-2}$	1,3
Bone tumor	$1{,}1\cdot10^{-4}$	$3{,}4\cdot10^{-4}$	$5{,}1\cdot10^{-3}$	0,34
Lung tumor	$0{,}7\cdot10^{-3}$	$2{,}2\cdot10^{-3}$	$7{,}7\cdot10^{-2}$	5,1
Various tumors on irradiation of whole body	$1{,}3\cdot10^{-2}$	$4{,}1\cdot10^{-2}$	$1{,}7\cdot10^{-3}$	0,11
Genetic damage	$2{,}9\cdot10^{-3}$	$9{,}0\cdot10^{-3}$	$4{,}2\cdot10^{-4}$	$2{,}8\cdot10^{-2}$

In the transportation of nuclear fuel from factory to APS and of spent fuel from APS to reprocessing plant, the population living along the transportation routes may be exposed to radiation. Existing calculations are very approximate. It is assumed that the population living in a strip of width 1 km from the route may be exposed to radiation and that the level of the radiation is 10-100 times less than in the vicinity of APS sites. Accidents in transporting radioactive materials may lead to further radiation. The probability of an accident has been estimated at 10^{-6} per thousand journeys, only 1% of which will be serious accidents. Evidently, therefore, this factor cannot significantly change the radiation dose of the population in normal transportation.

Air-Borne Discharges from APS. Consider finally the consequences of environmental pollution by APS. In the course of APS use, gaseous, liquid, and solid radioactive wastes are formed and there is a possibility of radionuclide contamination of the atmosphere, the soil, and reservoirs in the vicinity of the power station. Air-borne wastes form the main actual source of biospheric contamination. The radionuclide composition and extent of the discharges from APS into the air depend considerably on the type and power of the reactor and, in particular, on the degree of sealing of the fuel-element shells.

Water-cooled—water-moderated reactors (VVÉR), as is known, release mainly radioactive inert gases (RIG), a certain amount of tritium, gaseous activation products (^{41}Ar, ^{14}C, ^{16}N), halogens, and suspensions of certain radionuclides, e.g., ^{131}I. According to UN SCEAR data [19], the release of gases from these reactors is 10-20 Ci/MW(electrical)-yr. On the average, the RIG isotopic composition is, %: ^{133}Xe 68; ^{135}Xe 11; ^{85}Kr 6; ^{41}Ar 5.5; ^{85m}Kr 5.3; ^{88}Kr 2.3; ^{87}Kr 1; ^{135m}Xe 0.8; and ^{133m}Xe 0.06. Of these only ^{133}Xe and ^{85}Kr ($T_{1/2}$ = 5.3 days and 9.2 yr) are of regional or global significance. When gasholders are used at the APS, short-lived radionuclides do not contaminate the atmosphere.

As a result of the purification systems employed for air passed through APS tubes, the content of ^{131}I, ^{89}Sr, and ^{90}Sr in the emissions of VVÉR reactors is reduced to safe levels, amounting to only a few percent of the established limiting permissible discharges (LPD). This is also entirely true of RIG emissions. Thus, the actual RIG discharge at the Kol'sk APS (KAPS) was 300 Ci/yr in 1973 and 1050 Ci/yr in 1975 or no more than 0.001% of the LPD. The content of ^{131}I, ^{89}Sr, and ^{90}Sr in air-borne emissions was at such low levels that it is impossible, using current modern monitoring methods, to distinguish them against the background of global fallout. Practically all the air-borne emissions consisted of the short-lived nuclides ^{88}Rb and ^{138}Cs.

At Novovoronezh APS (NVAPS) the VVÉR-440 reactor has been found to be completely safe for the population and the environment, especially the series reactors established in the third and fourth blocks (Table 1). The actual RIG discharge in 3 years of use (the fourth VVÉR-440 block) was only 95,000 Ci, or about 2.5% of the LPD. The bulk of the air-borne emission is associated with the operation of nonseries reactors (the first — the VVÉR-210 — and the second — the VVÉR-365). Altogether, in regions between 0.5 and 50 km from NVAPS the fallout density of ^{90}Sr, ^{137}Cs, ^{144}Ce, ^{95}Zr + ^{95}Nb, and $^{103+106}$Ru does not differ from the global level. In addition, at the monitoring point (Voronezh), 50 km from NVAPS, the fallout density of all these radionuclides is approximately twice as high as at the power station itself. This is because of the precipitation of radionuclides with the usual industrial air-borne emissions.

Direct confirmation of radiation safety in the vicinity of APS is provided by the annual integral γ-radiation dose, measured in the open using an IKS-type integral dosimeter. In the vicinity of NVAPS in 1974 at distances of 0.2, 2.5, 4, 4-6, 9-12, and 50 km, the γ-radiation dose was 107, 78, 70, 74, 76, and 97 mrad/yr, respectively.

TABLE 5. Mean Individual Radiation Dose of Population due to APS and TPS Radioactive Emissions in 1976, mrem/yr

	Close to power station		Whole population			
	APS	TPS	APS		TPS	
	10^3 MW(elect)	10^3 MW(elect)	10^3 MW(elect)	actual power	10^3 MW(elect)	actual power
Whole body	0,73	0,53	$3{,}5 \cdot 10^{-4}$	$1{,}1 \cdot 10^{-4}$	$5{,}2 \cdot 10^{-5}$	$3{,}43 \cdot 10^{-3}$
Bone tissue	1,0	114	$4{,}2 \cdot 10^{-4}$	$1{,}3 \cdot 10^{-3}$	$2 \cdot 10^{-2}$	1,32
Lungs	$7{,}3 \cdot 10^{-4}$	41	$6{,}5 \cdot 10^{-5}$	$2{,}0 \cdot 10^{-4}$	$7{,}5 \cdot 10^{-3}$	0,5
Bone marrow	0,32	14,5	$1{,}5 \cdot 10^{-4}$	$0{,}45 \cdot 10^{-3}$	$2{,}6 \cdot 10^{-3}$	0,17

Air-borne emissions from NVAPS hardly contaminate the soil around the power station. This is confirmed by actual monitoring data for the radioactivity of grass and vegetable foodstuffs and milk produced in the area. The calculated radiation dose for individuals is comparable with the fluctuations of the natural background and does not exceed 2 mrem/yr.

The contribution of particular radionuclides to the total dose depends on their physical properties and the pattern of land use around the power station, and also on the distribution of emissions in the atmosphere. On this latter criterion, radionuclides in the environment may be divided into global (^3H, ^{14}C, ^{85}Kr, etc.), regional (a mixture of krypton isotopes, ^{60}Co, ^{58}Co, etc.), and local (^{131}I and other short-lived radionuclides).

Data characterizing the emissions of Soviet APS are given in Table 2. Note that these figures are comparable with those for non-Soviet APS: about 11 Ci/MW(electrical)-yr for VVÉR reactors and up to 1400 Ci/MW(electrical)-yr for boiling-water reactors [20-22]. While all the given values are significantly lower than the LPD, it has nevertheless been decided to build activity-suppression units (ASU) at RBMK reactors, which should reduce emission even further.

In 1976 the total nominal power of APS operating in the USSR was $3.1 \cdot 10^3$ MW(electrical), of which VVÉR reactors provided 2100 MW and RBMK reactors 1000 MW. Table 3 gives the radiation dose of the population due to the emissions of these reactors when the safety radius is 1.5 km, the effective tube height is 100 m for the WCWMR and 170 m for the RBMK reactor, the population density is 50 people/km^2, the nominal power of a single APS is 1000 MW(electrical) for an RBMK reactor and 1500 MW(electrical) for a VVÉR reactor. Table 3 also shows the predicted doses for the planned APS of 6500 MW (VVÉR) and 10,500 MW (RBMK), the power of a single APS with any type of reactor being 4000 MW(electrical). Note that the conditions assumed in the calculation are strict, since population and agricultural activity within 1.5 km of the power station are assumed [23]. Table 3 shows that, even for a population close to the APS the radiation dose is extremely small, except for the case of the thyroid gland in children. The mean individual dose for the whole population of the USSR is negligibly small. For example, in 1976, the radiation dose over the whole body was $\sim 10^{-3}$ mrem/yr.

Comparative Radiation Risks of TPS and APS Emissions. It is of interest to compare the radiation dose of the population from the air-borne emissions of APS and coal-fired TPS (Table 4) [13]. These results clearly demonstrate that only the radiation dose over the whole body per unit power output is comparable. The radiation of certain other organs — the bone tissue and lungs — due to TPS is higher by a factor of 100 or more, both for an individual living close to the power station and for the population as a whole.

Comparative data (Table 5) clearly indicate the low level of risk from APS and TPS. In addition, it is of fundamental importance that the overall risk of death from cancer as a result of radiation is 30 times larger for the population close to a coal-fired TPS than in the vicinity of an APS of similar output [13]. Thus, even in terms of the level of radiation, APS are preferable to TPS; this does not take into account the many harmful effects of the large quantities of other chemical material released from TPS, many of which are carcinogens (benzopyrene, heavy metals, etc.).

It is often said that the possibility of accident at a reactor poses a grave danger. However, even today modern reactors are characterized by high reliability and in the future this reliability will evidently be increased. It is sufficient to note that in more than 1000 reactor-years of use of nuclear reactors there has been not one accident causing injury to the population or the environment. The risk of fatality as a result of accident in the use of 100 reactors for the population living within 40 km of an atomic power station does not exceed $3 \cdot 10^{-9}$ cases/yr per person [24], and for a country such as the USSR is of the order of 10^{-10} cases/yr per person, i.e., less than for normal APS operation.

TABLE 6. Concentration of ^{90}Sr and ^{137}Cs in Water, pCi/liter

Sampling time	Sampling point	^{90}Sr	^{137}Cs
July, 1974	KAPS discharge-channel outflow	0,31±0,06	0,20±0,06
Sept., 1974		0,27±0,06	0,16±0,07
Mar., 1975		0,22±0,04	0,17±0,05
May, 1975		0,25±0,05	0,16±0,07
July-Sept., 1973	Monitoring region, Lake Imandra	0,28±0,08	0,24±0,05
Sept., 1974		0,28±0,07	0,16±0,08
May-July, 1975		0,23±0,05	0,18±0,10

Thus, the existing radiation load on the population due to APS is extremely insignificant in comparison with that due to the natural background and to the major radiation sources resulting from human activity. The introduction of nuclear power offers great possibilities for improvement of the environment and for the reduction and prevention of harmful effects on humans.

Radioactive APS Wastes

In the USSR all radioactive APS wastes are processed. Waste concentrates (still residues after evaporation, ion-exchange resins, pulps, primary heat carrier after replacement) are collected for storage in special vessels. The wastes are then consolidated predominantly by bituminization and stored in the solid state in subterranean or surface concrete compartments until the decay of the radionuclides is complete; all possible measures are taken for the protection of ground waters from pollution. To this end, the vessel is fitted with special trays in which wastes are collected in the event of leakage; there are systems allowing wastes to be transferred from one vessel to another and also a system of hydrogeological monitoring holes around the stores, level gases, and other means of monitoring. At the sites of the stores of liquid and solid wastes special hygienic and hydrological requirements are imposed. In consequence, wastes of mean specific activity are not at present a real source of environmental pollution.

In favorable natural conditions liquid APS wastes should be removed to subterranean stores, for which deep-sea water-bearing layers isolated from surface waters are used. Satisfactory experience of the use of this method has been accumulating in the last few decades at the Ulyanov APS.

In Soviet APS great attention is paid to the protection of the coolant reservoir from radioactive contamination. At reservoirs with APS the discharge of purified unbalanced water is limited (within 10^4 m^3 per year). The radionuclide concentration in the water should not exceed the permissible limits in drinking water; this automatically limits the total discharge of radioactive material in the reservoir. The high efficiency of the measures adopted in Soviet APS to protect coolant-water reservoirs from radionuclide contamination is clearly evident in the example of the Kolsk APS.

The source of recycled coolant water for this APS is Lake Imandra, which, as research has shown, has suffered practically no radioactive contamination. In this reservoir, with a water surface of area more than 800 km^2, the purified unbalanced water discharged each year contains less than 1 Ci of a mixture of fission products and induced radionuclides. Note, for comparison, that global fallout carried in atmospheric precipitation and river currents would normally lead to the accumulation of up to 100 Ci ^{90}Sr and ^{137}Cs in a lake of this surface area. Because the steady discharge of radionuclides is so slight, there is no increase in the artificial radioactivity of the water, either in the lake as a whole or in the vicinity of the APS. Even in the discharge channels leading the coolant water from the turbine condensers, together with the outflow of industrial and domestic sewerage and unbalanced APS water, the artificial radioactivity of the lake is the same as in the intake channel and the monitoring region (Table 6). The radioactivity of the bottom sediment and the water plants, providing a good index of radionuclide contamination, is the same in the discharge-channel outflows as elsewhere in Lake Imandra. Fish living in the outflow region of the discharges from the Kolsk APS are not exposed to contamination by radionuclides of reactor origin.

Because the measures for the protection of Lake Imandra from radioactive contamination by APS discharge water are so reliable, it is possible to use the heat released into the reservoir for intensive artificial fish farming close to the discharge-channel outflows, where the water temperature is 5-7°C higher than elsewhere in the lake so that a surface area of around 1 km^2 remains free from ice throughout the winter. The Murman combine, under the overall management of the Sevryb combine, is operating an experimental fish farm for the rearing of trout and other valuable fish species. Three years' experience indicates that the low-temperature APD discharge water may be used for fish breeding and the commercial cultivation of fish with total radiation safety for the population.

Satisfactory radiation-safety measures are also employed on the River Don, in the region where coolant water is discharged from the first and second blocks of NVAPS. At this APS, there is practically no discharge of radioactive materials. The unbalanced water of the first and second blocks is conveyed to a specially constructed filtration field and does not enter the reservoir. The only water returned to the River Don is that used to cool the turbine condensers of these blocks, and the third and fourth blocks are equipped with closed-loop water-recirculation systems using cooling towers. Systematic observation of the radioactivity of the water, the bottom sedimentation, and also the flora and fauna in the River Don and in the immediate region of the NVAPS discharge channel shows a lack of radioactive contamination of the water course by APS discharge waters.

At the Beloyarsk water reservoir provided on the River Pyzhme for the technical needs of Beloyarsk APS (BAPS), the radiation-safety measures are satisfactory. Dosimetric monitoring over many years has shown that discharge of radioactive materials into the reservoir is practically absent. The concentrations of ^{137}Cs and ^{90}Sr in the water fluctuates about the level of global contamination. This is also indicated by the low level of contamination of the bottom sedimentation and the water plants. The ^{90}Sr and ^{137}Cs concentration in fish living in the outflow region of the BAPS thermal discharge waters is statistically indistinguishable from that in other regions of the reservoir and is 2-3 orders of magnitude below the permissible level. In view of all these factors, Gossannadzor (State Health Administration) has given permission for the use of the Beloyarsk reservoir for workers' recreation and for amateur and industrial fishing. The shores of this reservoir provide a recreational center for three industrial cities.

CONCLUSIONS

In periods of scientific and technological revolution, the protection of the environment from pollution ranks as one of the most important problems. The power industry is a major source of environmental pollution and the problem of protecting the environment from its wastes is particularly urgent because of the rapid rise in energy demand.

From the point of view of environmental conservation and human health, nuclear power has an important advantage over traditional power sources using solid and liquid organic fuels. The extensive development of nuclear power in the future will require timely scientific solutions to a number of problems. The largest of these is the problem of solid and liquid wastes, solution of which will ensure reliable protection of the environment. Although air-borne APS emissions do not constitute a danger to the environment, the endeavor to reduce them further retains its urgency. This is because in the long term it is necessary not only to ensure negligible radiation of the population living close to the APS but also to prevent a significant rise in the overall dose to the population. A particularly important problem is to find foolproof means of preventing accidental contamination of the environment, which would be a significant step toward siting APS and atomic and thermal power centers close to large cities.

The possibility of a rise in the collective-dose level in the early stages of nuclear-power development must be met in a number of ways: by reducing the existing radiation load due to irrational activity (above all, x-ray diagnostics), by investigating the biological effects of small and extremely small doses (including the combined effect of radiation and chemical factors), and by developing work on the theoretical basis for the normalization of radiation effects.

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NUCLEAR INSTRUMENT MAKING, THE MEASURING-INFORMATIONAL BASIS OF ATOMIC SCIENCE AND ENGINEERING

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When appraising the role and importance of measuring technique in scientific—technical advancement, D. I. Mendeleev said that science begins when measurements begin. The further development of science has since convincingly confirmed the accuracy of this appraisal.

Measurement is one of the fundamental methods of learning about nature, a method which provides objective information about the qualitative and quantitative relations of phenomena and the characteristics of objects studied. Without the theory and means of measuring technique it is impossible to combine results of measurements made at different times and by different researchers and to present them in a unified form and continuity of scientific knowledge is impossible. Therefore, the achievements of science are in many respects determined by the development of measuring technique which, in turn, is bound up inextricably with the general advance of science which creates fundamentally new possibilities for the development of measuring technique. An equally close interrelationship and strong reciprocal influence exist between instrument making and industry which ensures the technical realization of new measuring ideas and means. The role of the latter has been particularly great in atomic science and engineering whose advent, as is well known, necessitated the development of a new class of instruments for measuring various parameters of ionizing radiation. And the successful resolution of the so-called atomic problem was inseparably linked to a need to create a sufficiently broad nomenclature for measuring apparatus. The task became especially timely by the mid-1940s when the practical

Translated from Atomnaya Énergiya, Vol. 43, No. 5, pp. 384-390, November, 1977.

0038-531X/77/4305-1034$07.50

Areas of Science and Engineering Utilized in Nuclear Instrument Making

Nuclear physics, solid state physics, radiochemistry
Electronics and computer science
Instrument-making technology

→ Nuclear Instrument Making
Theory and methods of measuring ionizing radiation, technical means, and data-measuring and control systems →

Main Directions and Tasks

Atomic Industry	Nuclear Power	Environmental Monitoring and Protection
Prospecting for and surveying of radioactive deposits Development of deposits, ore extraction and enrichment Processing of raw material, fabrication of fuel elements Ensuring radiation safety Processing wastes	Technological inspection of reactor equipment Nuclear safety, emergency protection of reactors Leak testing of fuel elements, reactor equipment, and technological loops Radiation safety of atomic power plants and personnel, external monitoring	Atmospheric inspection and protection Monitoring of water bodies Monitoring of soil, vegetation, food products Monitoring of population

Space and Planetary Research	Geophysics and Mining Industry	Nuclear Physics and Other Research
Near-space research Analysis of planetary and lunar matter Research on composition of interplanetary space Study of stars, galaxies, and nebulae, and x-ray astronomy Dosimetry on manned spacecraft	Prospecting for exploration of useful minerals (oil, gas, lead) Survey of boreholes, blastholes, and mines Ore flotation Composition and phase analysis of matter	Spectrometry of nuclear radiation High-energy physics Reactor physics Metal physics **Related Areas of Science and Engineering** Medicine (dosimetry, radiodiagnosis, lymphography) Various measurements and analyses

Theory and Methods of Measurements, Technical Means and Systems

Measurement of Parameters and Characteristics of Particles and Quanta

Energy, electric charge, velocity, mass, time and direction of travel through detector
Measurement of characteristics of radiation fields
Interaction with matter
Characteristics of source
Energy distribution of particles and quanta in space

Technical Means

Modules for detection of ionizing radiation, samplers, sample concentrators
Electronic modules and units for signal amplification, filtration, and conversion
Analog and analog-to-digital devices for measuring signals
Digital devices for counting signals
Memories, processors for processing arrays of data.
Output devices, indicators, displays. Data channels, software. Metrological means. Constructions

Data-Measuring and Controlling Systems

In-pile monitoring systems
Systems for monitoring radiation safety
Control and protection systems
VEKTOR system of electronic control modules and other systems

Fig. 1. Interrelationships of nuclear instrument making with other branches of science and engineering.

introduction of the achievements of nuclear physics into industry came onto the order of the day. In the period from 1945 to 1952 on the initiative of I. V. Kurchatov small teams of engineers and technicians of two Moscow factories, with the participation of scientists from the Institute of Atomic Energy and some other institutes, produced the first commercial Soviet dosimetric, radiometric, and electronic-physical instruments for the developing atomic science and engineering.

In view of the rapidly expanding demand in the country for such apparatus in connection with the continuous growth of both the number and complexity of measuring tasks in the domain of ionizing-radiation monitoring and the objects of measurement where such monitoring is indispensable, a central design bureau was organized in 1952 with an experimental plant. This was the first independent organization to specialize in the development of measuring instruments of this class. This marked the beginning of systematic, centralized work on laying the scientific and technical basis for designing instruments and methods of measuring ionizing radiation.

The development of nuclear physics and electronics, the growing application of their achievements, and the much broader methodological possibilities that opened up in the process, necessitated the creation of a specialized scientific base for nuclear instrument making. To this end, in 1957 the central design bureau was transformed into the All-Union Scientific Research Institute for Instrument Making which today is a major scientific center for nuclear instrument making, engaged in the development of methods, instruments, systems, and complexes of apparatus for use in various branches of the national economy.

If one were to attempt a brief formulation of the most important results of the activity in creating instruments for measuring ionizing radiation, without which atomic science and engineering could not have reached their present levels, they would boil down to the following.

First, during this time the foundation was laid for the methodology of ionizing radiation, the theory and method of nuclear instrumentation design, generalized by the foremost specialists in many monographs, textbooks, articles, and lectures; the scientific schools, which were formed and are continuing to develop successfully, reflect not only the specific features and diversity of the areas in which the equipment is applied, but also the development of nuclear instrument making itself in many directions, organically combining achievements of such different fields of science and engineering as nuclear physics and radiochemistry, solid state physics and crystallography, radio engineering and electronics, computer science, etc. Second, the organizational and technical foundations for designing, producing, and commercially introducing the equipment were developed and put into practice; these were formulated in the form of standards whose progressive nature is confirmed by the fact that many of them have been adopted as the basis for several dozen COMECON recommendations on standardization and for many standards of the International Electrotechnical Commission. Third, an industrial base was established for nuclear instrument making; engineering and technical personnel were trained, retooling was carried out, commercial production of nuclear instruments of the first (vacuum-tube) and second (transistorized) generations was organized in plants of various departments in the country, and plants for manufacturing apparatus of the third (microelectronic) generation were designed and put into operation.

In other words, the existence of a theory, organizational and technical foundation, and an industrial base indicates that in the years that have elapsed a new independent branch of science and engineering, viz., nuclear instrument making, has been formed and developed in the country, a branch without which further acceleration of the rate of scientific—technical progress would be inconceivable. It can be said today that the basic top-priority demands of almost all branches of the national economy for equipment for measuring ionizing radiation of all forms and in all media have been successfully met. In accordance with the designs developed by the All-Union Scientific-Research Institute of Instrument Making commercial plants have mastered several hundreds of types of radiometric, dosimetric, spectrometric, and electronic-physical instruments, devices, and systems. Millions of measurements are made each year with them. The institute has done a great deal of work on developing and preparing for use special-purpose apparatus for equipping extremely important units of the national economy. At this point we could list several hundred types of instruments, systems, and complexes, including:

Various types of data-gathering, measuring, and control apparatus at all nuclear power facilities of the Soviet Union, starting with the world's first atomic power plant, as well as in the atomic power plants of COMECON member-countries and the Lovisa atomic power plant in Finland;

several dozen types of research equipment installed on artificial satellites and interplanetary stations (beginning with the second manned spacecraft) for studying the radiation in near space and outer space, the lunar surface, and the planets Mars and Venus;

automated systems of radiation monitoring for the atomic ice-breakers Lenin, Arktika, and Sibir';

measuring complexes for the major scientific-research centers of the country, etc.

In the process of developing three generations of apparatus, nuclear instrument making has traversed the path from the development of individual measuring instruments to the creation of automated systems for comprehensive monitoring and control in large atomic facilities and atomic power plants, and for real-time data collection and processing during experimental research on large physical installations.

Along with the increase in the size of the systems of apparatus and the growth of functions the methods and the products of nuclear instrument making were introduced into various branches of science, engineering and the national economy. Thus, nuclear instrumentation is used for prospecting for deposits of radioactive minerals, for inspection and control in the development of deposits, in the process of ore enrichment and in obtaining enriched uranium, fabrication of fuel elements, and, finally, for monitoring and control, and nuclear and radiation safety of reactor-generators of thermal energy. Many important experimental results facilitating the introduction of nuclear power and the development of present-day knowledge in nuclear physics have

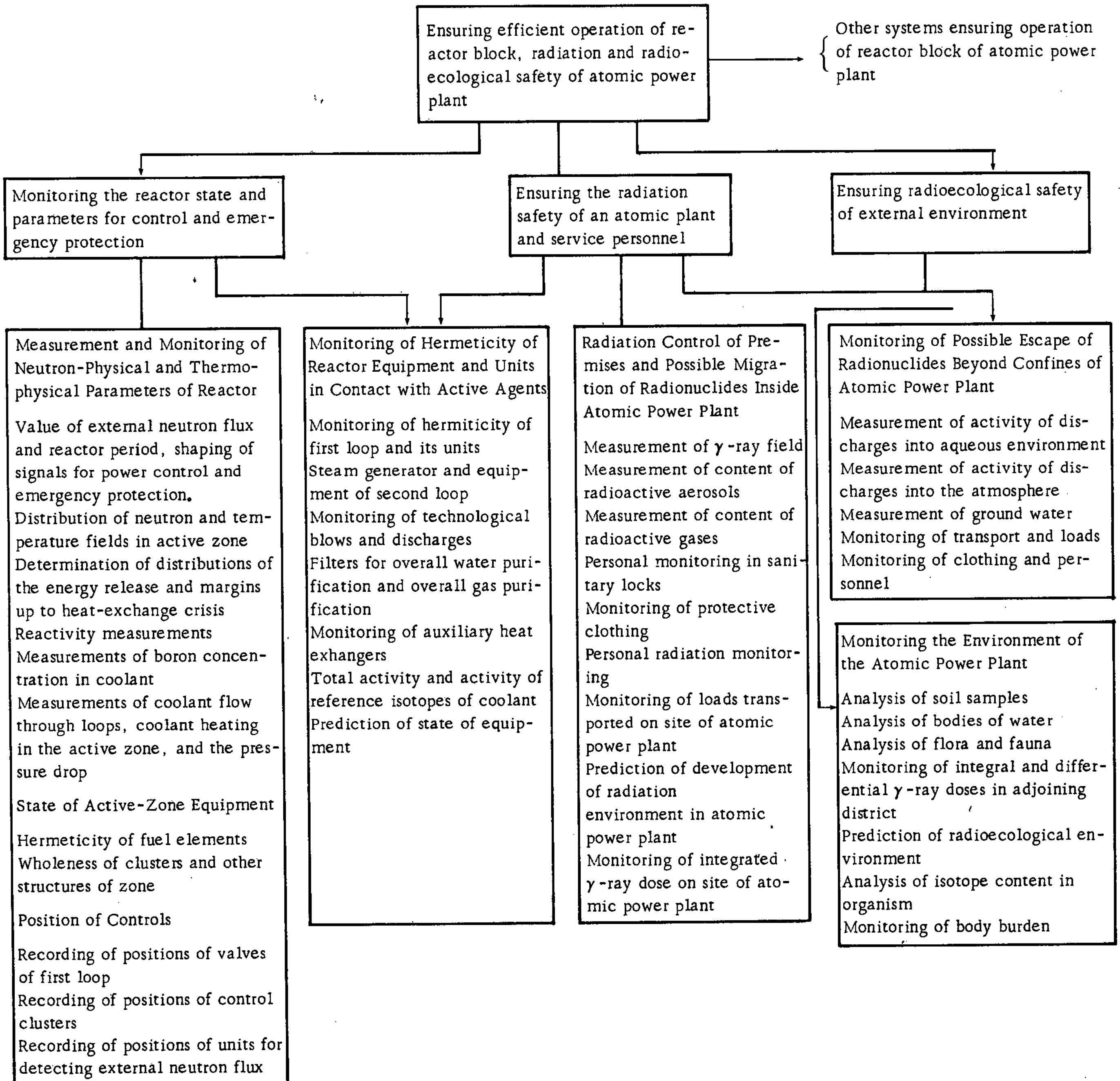


Fig. 2. Main tasks and functions performed by complex of nuclear instrumentation in atomic power plants.

also been obtained with the aid of nuclear instrumentation. As a result of the work done, nuclear instrument making today has assumed an extremely important place in the overall structure of the technical means of atomic science and engineering. Figure 1 presents a diagram that illustrates the position and role of nuclear instrument making, the problems solved with its assistance, its interrelationships with other branches of science and engineering, as well as the most important forms of measurement and data processing, carried out with the aid of the methods and means of nuclear instrument making.

We should note the following characteristic features of data-gathering and measuring devices of nuclear instrumentation, these features following from the specifics of nuclear objects:

large number of monitoring points (up to several thousand);

diverse measuring information (ordinary sequences, one- and multidimensional spectral distributions, amplitude and shape of pulsed signals, correlational measurements, etc.);

measurement in various media (solids and bulk materials, liquids, gases, aerosols, vacuum);

extremely high operating speed (up to 10^{-11} sec);

real-time processing of large volumes of data according to complex programs;

use of equipment in strong radiation fields and aggressive media;

responsibility of functions performed;

measurement of signals in the presence of a high level of noise, background, and disturbances.

Measurements of ionizing radiation have a number of singular properties which are reflected in the methods used in the equipment. The basic formula for their implementation in nuclear instrumentation, from elementary events to integral characteristics, is a manifestation of the specific quality of the processes and objects subjected to measurement.

In contradistinction to general-purpose electrical measuring instrumentation, from electronic and thermotechnical instruments whose sensors function by averaging the values of the quantity being measured over an ensemble (electronic, wave, molecular), nuclear instrumentation measures individual elementary particles, quanta, and quantum-mechanical events occurring at the level of the nucleus and the atom and on this basis synthesize the averaged characteristics of the macroprocesses and macrophenomena that take place. Having discussed the specifics of the methods, we must mention one more distinctive feature. Most constants, especially temperatures, are measured with thermocouple only by attaining thermodynamic equilibrium between the thermocouple and the medium. It may be said that the process of measuring the parameters of ionizing radiation proceeds, and the results are obtained, by disturbing the equilibrium in the volume of the detector as a consequence of its interaction with an elementary particle, nuclear fragment, or quantum of radiation. The distinctive feature consists in the wide range of quantities measured:

Intervals between signals, sec...........	10^{-11}-10^{14}	15 orders of magnitude
Particle energy, eV....................	10^{-2}-10^{14}	16 orders of magnitude
Particle-current density, $1/cm^2 \cdot sec$.....	10^{-4}-10^{14}	18 orders of magnitude

Finally, the following should be singled out. The basic carrier of the most complete, direct, and representable information about the state of the objects of atomic science and engineering is ionizing radiation, whose parameters and characteristics are measured by the methods and means of nuclear instrumentation. This is why nuclear instrument making should be considered as the measuring and data-gathering basis of atomic science and engineering. A characteristic feature of nuclear instrument making today is that the principles of a systematic approach are applied in every possible way, this approach being realized in two main ways:

as an approach that facilitates a broad grasp of the scientific and technical problems and the interrelated interdependent problems as a means of achieving the mutual consistence of technical problems;

as a way of organizing technical means and achieving the mutual consistency of their design, operating, electrical, logical, and metrological characteristics, this being indispensable for the construction of large complexes of equipment.

The first of these ways furthers the comprehensive development of nuclear instrument making, an improvement of its structure, and the creation of large complexes of equipment embracing the totality of problems but consisting of a number of apparatus systems, instruments, and units. The sequential introduction of a line of systems makes it possible to classify and interrelate functional problems. The second way directly tackles the classification and systematization of technical means. Because of this, every system of functional problems can be posed in accordance with a system of technical means capable of solving these problems. Present-day system conceptions establish a given correspondence by means of the concept of model. Each of the functional problems at any level can be presented in the form of a concrete data-gathering and measuring model, reflecting the basic laws governing the behavior of the process or particular units of the object. In the development of the model account is taken of the specifics of nuclear instrumentation — measurement of physical quantities characterizing sources and fields of ionizing radiation as well as effects due to the action of radiation on the objects being irradiated.

An example of the system organization of a data-gathering and measuring circuit is that of a complex of equipment now under development for various types of atomic power plants. An atomic power plant is a highly complex facility containing a variety of present-day equipment and units which operate under conditions of high nuclear-physical and thermodynamic parameters. This determines the diversity of function and the ramified structure of the monitoring and control points. The main functions and tasks performed by the complex of equipment measuring various types of ionizing radiation in atomic power plants are presented as the fifth group in Fig. 2.

The principal monitoring tasks of the first group are those of measuring the parameters of the active zone, the distributing the energy release and the neutron flux, monitoring the fuel-element assemblies, and determining the margin to heat-exchange crisis, and measuring the external neutron flux for controlling the reactor power. The purpose of the monitoring is to attain optimal technological parameters and to ensure nuclear safety and emergency protection. The apparatus of this group performs three main functions, i.e., identifies the reactor parameters, participates in the reactor control, and ensures nuclear safety. The state of the equipment controlling the apparatus of this group is the determining factor of the radiation environment in the atomic plant. Therefore, the functions of the apparatus of the given monitoring are somewhat contradictory: they increase the physical and thermodynamic parameters of the reactor and at the same time keep track of the safety of its operation and the radiation safety in the atomic power plant.

The tasks of the apparatus of the second group are those of monitoring and predicting the hermeticity of the equipment of the first loop, and monitoring the quality of the operating of the coolant purification filters. In this case the apparatus performs technological functions and at the same time carries out the function of providing radiation safety since it also monitors the possible escape of radionuclides from the closed loops.

The purpose of the monitoring by the third group is to ensure the radiation safety of the personnel in the service premises of the atomic power plant. This means comprehensive, thorough personal monitoring during work; monitoring of radioactive contamination of clothing, footwear, incorporated accumulations of nuclides; regular complete examination of the personnel; computation of the cumulative radiation doses; etc.

The fourth group ensures monitoring of the contact of the atomic power plant with the outside environment. The main task of this form of monitoring is to measure discharges of radionuclides that escape beyond the confines of the atomic power plant. The quality of the monitoring in the given case determines the degree to which the atomic power plant affects the environment and at the same time the data from the monitoring provide a basis for assessing the condition of the equipment and the technological loops of the atomic power plant which carry coolant and other radionuclide-containing media.

The monitoring functions of the fifth group are directed entirely at ascertaining the effect of the atomic power plant on the environment, its flora and fauna, and the condition of the atmospheric and aqueous basins.

To perform the functions discussed above, the complex of apparatus contains not only detectors and electronic equipment for measuring ionizing radiation but also sensors and instruments for obtaining data about the temperature, pressure, flow, velocity of air flow in the ventilating system, etc. A more sophisticated complex of apparatus developed for sodium-cooled fast reactors also contains technical means that permit complete monitoring of the state of the reaction and the technological operating conditions.

We should like to emphasize that the unification and concentration of various forms of measurements and measuring apparatus in one complex is not only a necessity but a present-day scientific and technical trend in instrument making, ensuring optimalization of design and a high degree of automation of such systems. However, the parameters and characteristics of fields of ionizing radiation remain the principal content of the measurements and subsequent data processing.

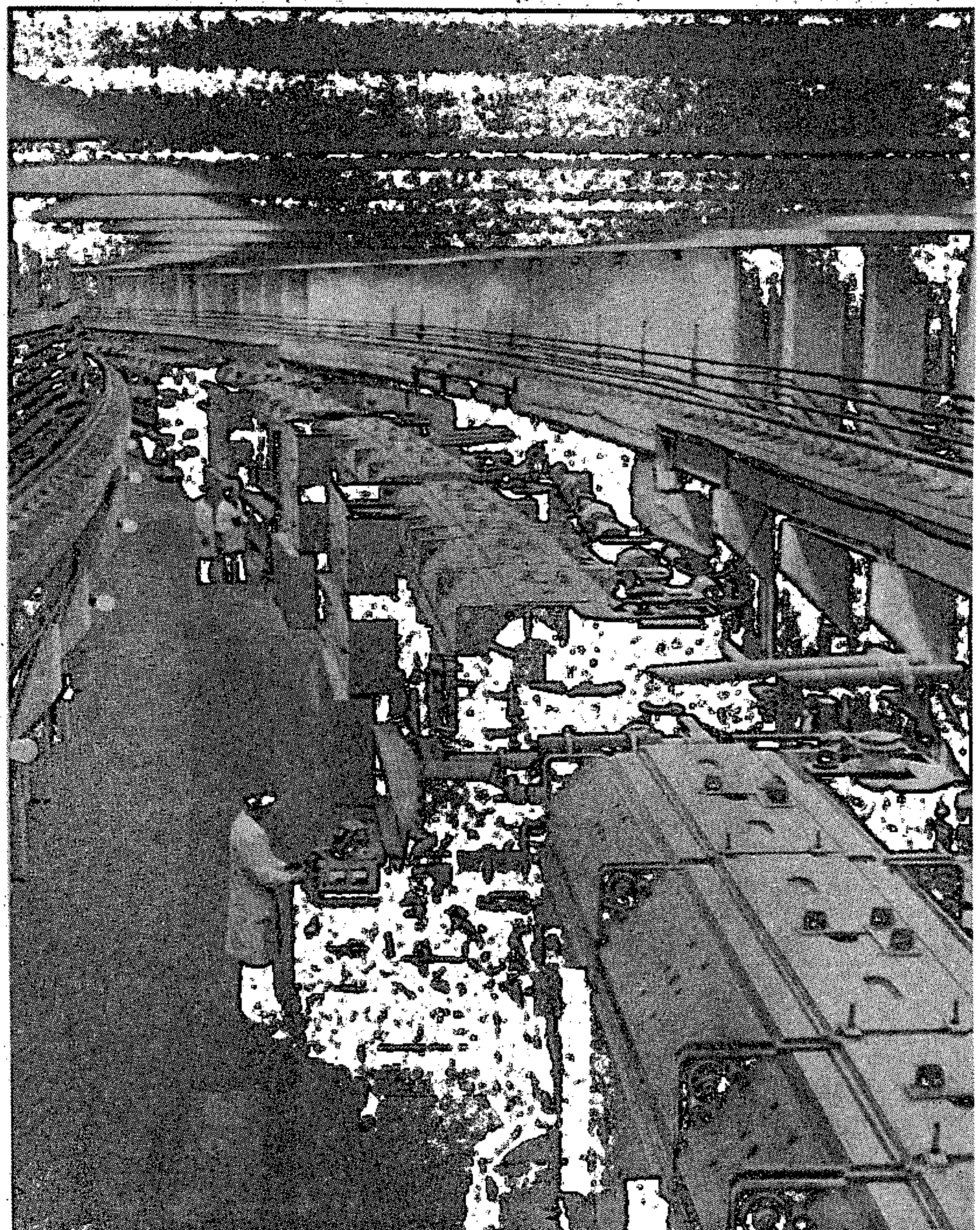
The example considered here is the fullest and most illustrative, but is far from the only one. On the basis of nuclear instrumentation other systems have also been developed for monitoring and controlling technological processes and facilities. The example described also shows that whereas in the early stages of the development of nuclear instrument making it was possible to consider separately atomic facilities as sources and the products of nuclear instrument making as meters and detectors of ionizing radiation, such an approach today is not in accord with the actual state and has no future.

The instruments and systems of nuclear instrumentation are today part and parcel of any atomic facility. The organic unity of two components is characteristic of facilities of this type at present, these components being:

the technological circuit determining the processes of the processing of materials or energy conversion;

the data-gathering and control circuit, reflecting the processes of the collection, processing, and use of data about both the basic technological process and the radiation-ecological state of the facility as a whole and of its individual parts. In other words, the stratified description of any nuclear facility should also contain data-gathering and control circuits, in addition to a technological circuit.

Thus, at the present stage of the art the efficient and safe functioning of any nuclear facility is determined by how closely the technological and the data-gathering and control circuits are interconnected. This interconnection will unquestionably be reinforced in the future and the role of the data-gathering and control circuit in nuclear facilities will be enhanced. Fundamentally new and extremely broad possibilities for the technical realization of such a trend are already available today thanks to the development of microelectronics, especially the broad introduction of microprocessors, semiconductor memory elements and other achievements of science and engineering, and the attendant "intellectualization" of the lower levels of the hierarchical structure of the systems. It is precisely in this way that it is proposed to solve the numerous new problems that confront nuclear instrument making. Therefore, in designing new atomic facilities today it is necessary to proceed to a much greater degree from the requirements of the systems approach to design, taking into account the conception of the interconnection between different circuits, and to seek new organizational forms that would ensure the simultaneous solution of the technological, measuring and data-gathering, and control problems.



Serpukhov 76-GeV proton synchrotron. The world's largest accelerating-storing complex is to be built on the basis of the accelerator.

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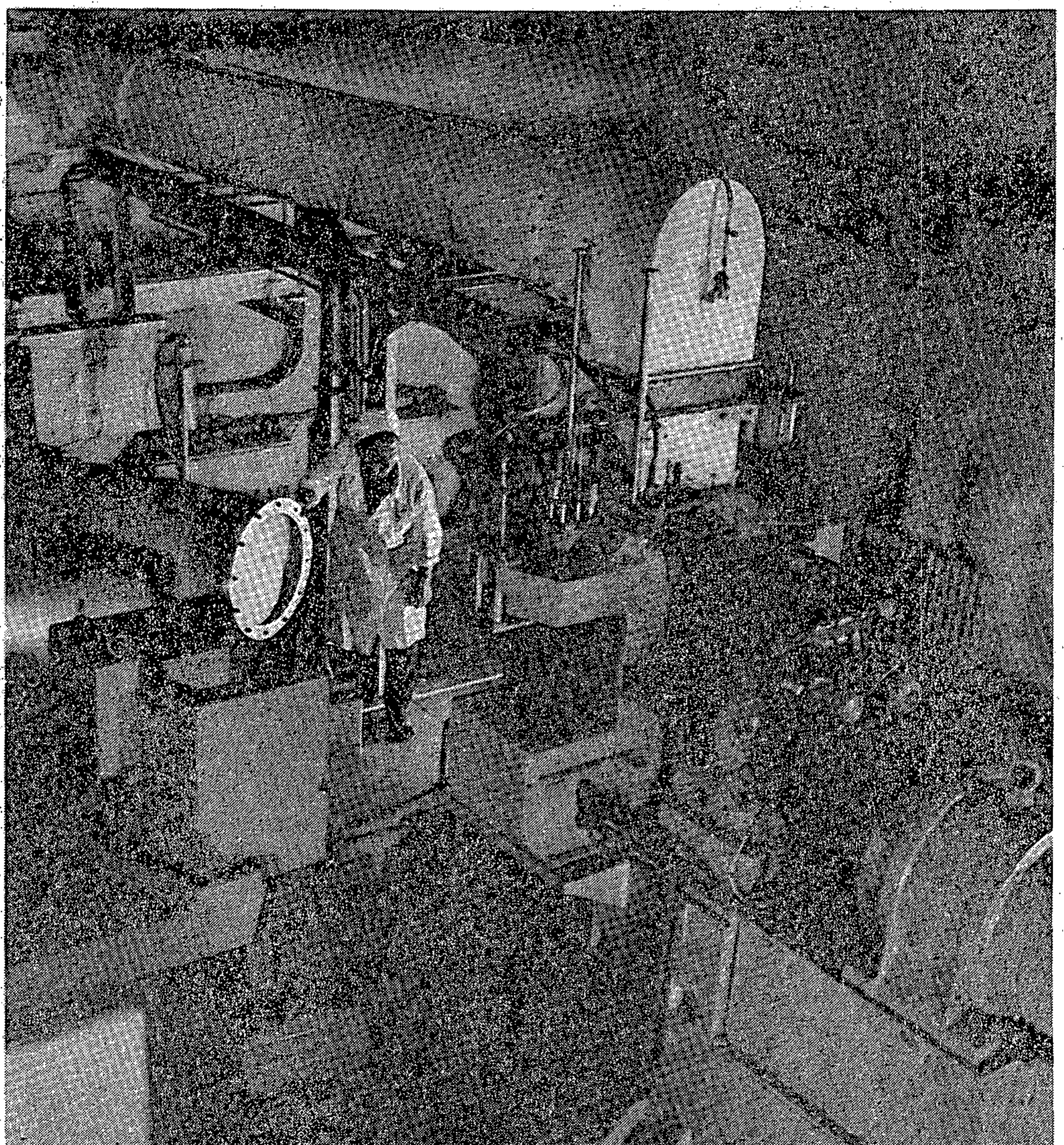


The IBR-30 research reactor, the world's only pulsed fast-neutron reactor. The reactor power reaches 150,000 kW in a pulse. A new pulsed reactor, the IBR-2, is now under construction in Dubna. Institutes of the member-countries of the Joint Institute of Nuclear Research (JINR) are participating in the construction.

DEVELOPMENT OF RADIATION TECHNIQUES AND TECHNOLOGY IN THE SOVIET UNION

A. S. Shtan' and E. R. Kartashev

One of the most important trends for the widespread use of nuclear energy is the application of artificial radioactive substances and other sources of ionizing radiations in the different branches of industry, medicine, and in agriculture. Only the production of large scales of artificial radioactive isotopes, and also the electrophysical sources of ionizing radiations, has brought about the origination and development of a new trend of nuclear science and techniques — radiation techniques. The circle of problems encompassing this trend is very wide and includes the development of nuclear-physics methods, instruments, equipment, and facilities, in which ionizing radiation is used for its effect on materials and substances for changing old, or producing new, properties, for transforming the decay energy of radioactive substances into thermal and electric power, for obtaining information about the qualitative and quantitative parameters of materials and products, for the control and guidance of technological processes, technical and medical diagnostics and therapy of the various human diseases.

At the present time, three basic trends of radiation techniques have been defined: radiation equipment construction, radiation instrument construction and radioisotope power generation.

The problem of radiation equipment construction consists in the development and construction of high-powered facilities for technological purposes, and also for radiation therapeutic techniques. Radiation instrument construction covers that region where the information carrier is a flow of ionizing radiation, correlationally related with technological parameters. Radioisotope power generation has for its purpose the creation of various power devices, in which the radioactive decay energy of radionuclides is used as the primary power source.

Each of these trends includes a complex of problems, many of which have developed into independent scientific-technical disciplines such as, for example, radiation therapy, activation analysis, radiation chemistry and defectoscopy, etc. (see Scheme 1).

Radiation Equipment Construction

The physical, physicochemical, and biological processes in substances, produced during the interaction with ionizing radiation and causing a corresponding technological or therapeutic effect, form the basis of radiation equipment construction. Work on the study of radiation processes, the development of technological processes based on these investigations, and also the development of radiation equipment construction have taken place in several stages:

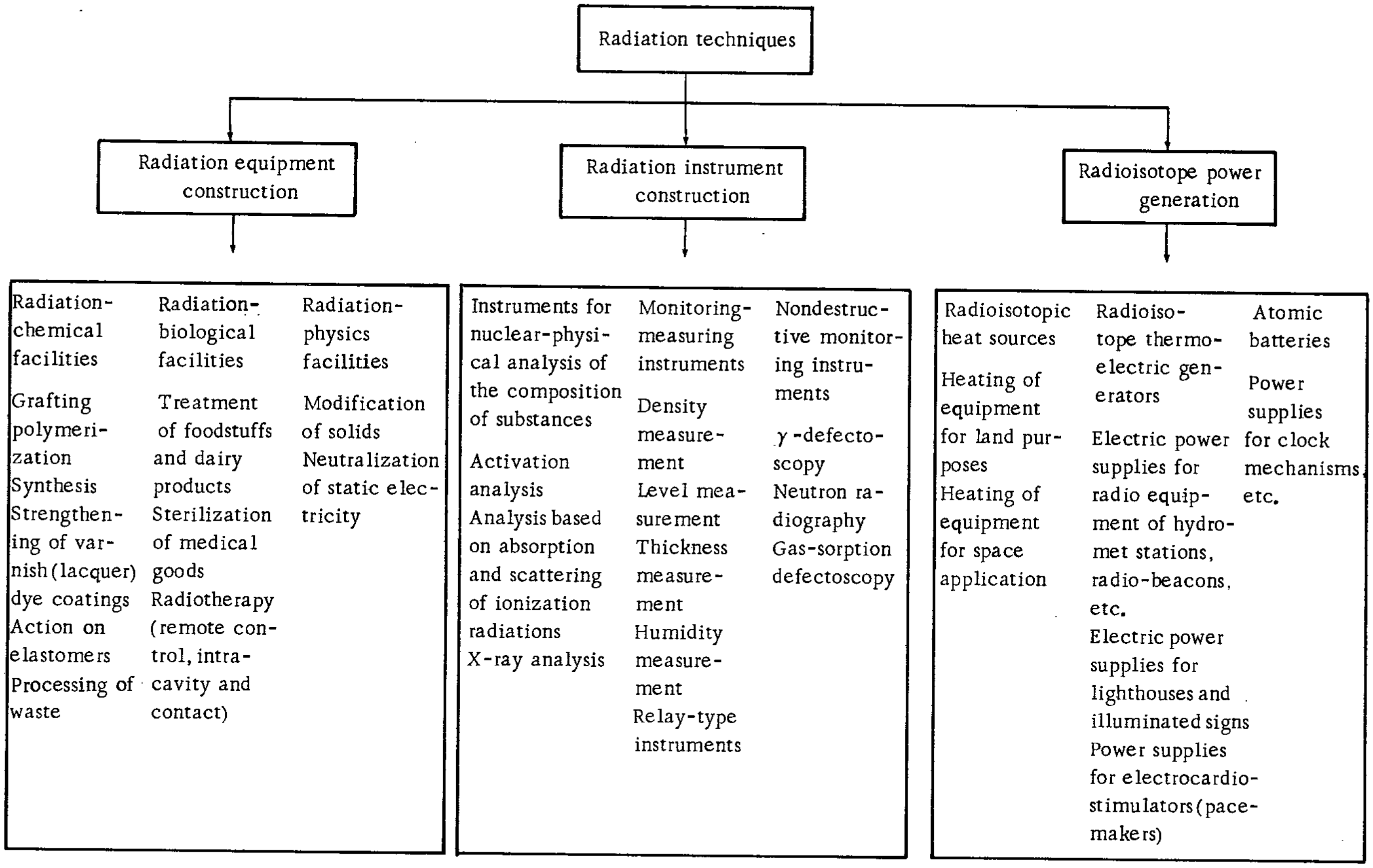
carrying out qualitative investigations, determination of the interaction mechanisms between ionizing radiation and the substance, the accumulation of data on the qualitative changes of substances; the emergence of the first experimental radiation facilities based on the application of radioisotopes (mainly ^{60}Co), electron accelerators and other types of electrophysical radiation sources; working out the demands for radiation apparatus and the scientific—technical principles of their construction;

a detailed study of the quantitative interaction mechanisms between radiation and substance; the development of radiation—technological processes for future industrial application, based on experimental and pilot-plant studies, the development on large scales of these processes, the determination of feasible technical—economic indexes, the development of methods of optimization of radiation facilities, and the working out of design and operating problems;

the creation of an industrial radiation technology, industrial radiation equipment construction; organization of radiation—technological establishments and departments.

It may be said with confidence that at the present time the scientific—technical principles of radiation equipment construction are established. For this, it was necessary to carry out a complex of investigations directed at

Translated from Atomnaya Énergiya, Vol. 43, No. 5, pp. 390-401, November, 1977.



Scheme 1

the development of methods for computing the principal parameters of radiation facilites, methods of optimization of the parameters of irradiators, to determine rational schemes of administration, monitoring, shielding and automation of radiation equipment. Methods were developed also for determining the topography of dose fields, and the utilization factor of radiation for various configurations of the objects being irradiated. Great attention was paid to the dosimetry of strong radiation fields, including high-powered beams of accelerated electrons.

In the Soviet Union, just as in other countries of the world, an intense transition has taken place from the period of research work and scientific studies, from the testing of the first pilot facilities to the wide-scale industrial introduction of the processes and plant developed. Simultaneously, investigations are continuing, which are directed at the search for new high-efficiency radiation processes and the optimization and unification of radiation equipment and sources of radiation. At the present time, in the world, about 50 different radiation—technological processes have been developed and are in various stages of experimental—industrial application. Radiation chemistry and radiation sterilization have produced the most development among them.

In radiation chemistry, having risen from the junction of atomic physics, chemical physics and physical chemistry, theories for the solution of practical problems is of specially great importance. Together with fundamental investigations of the general relationships, the special features of radiation—chemical processes, which are the most promising for industrial application, have been studied theoretically.

Radiation—chemical reactions, in comparison with reactions initiated by heat or other forms of energy, differ:

in the weak dependence (or absence) of the rate of initiation on temperature, which allows processes to be carried out at a low temperature, right down to the temperature of liquid helium;

in the possibility of carrying out radiation initiation without the use of any additives, initiators and catalysts, as a result of which specially pure products can be obtained;

in the dependence of the rate of initiation on the radiation dose intensity, which makes it possible to regulate the flow of radiation, to create a specified distribution of initiating centers in the reaction space.

On the basis of the thorough theoretical study during the last two decades, the scientific and engineering principles of radiation—chemical equipment design have been worked out. Work in this direction in the Soviet Union was started first by the L. Ya. Karpov Scientific-Institute of Physicochemical Research, and then carried on in the All-Union Research-Institute of Radiation Technology and other institutes.

TABLE 1. Radiation—Chemical Facilities with Different Radiation Sources

Facility	Radiation source	Source strength, kW	Radiation—chemical process
With long-lived radiation sources	^{60}Co, ^{137}Cs	10	Radiation vulcanization, purification of industrial and domestic sewage, radiation synthesis
With short-lived radiation sources	Radiation loops of a high-power nuclear reactor	$n\cdot 10^2$	Experimental investigations
Using a stream of electrons	Electron accelerators, ÉLT, ÉLV, "Élektron," "Avrora" and LUE-8/5V	from 0.5 to 40	Modification of polyethylene, hardening of lacquer-dye coatings, grafting polymerization, radiation purification of substances and of industrial effluents.

The investigations, carried out in the Soviet Union, are of a more fundamental nature than those carried on abroad, and at the same time have led to the publication of the handbooks which are necessary for construction and design organizations.

In the planning of radiation facilities, in the majority of cases it was necessary to carry out work on the investigation of a number of the parameters of radiation processes and to improve further the technologies such as the radiation yield of the process, determination of the kinetics, the dependence of these parameters on the dose intensity and the absorbed dose, on the concentration of starting materials, and the distribution of the absorbed doses in multilayered materials. For carrying out radiochemical processes, radiation sources (^{60}Co and ^{137}Cs), the radiation loops of a high-power nuclear reactor (e.g., VVÉR and RBMK), and electron accelerators can be used (Table 1).

Among the first radiation—chemical facilities developed under industrial conditions were the radiation sulfochlorinator RS-2.5 and a facility for the synthesis of tetrachloralkanes. At the Kazan synthetic rubber factory in 1972, the RV-1200 experimental plant was brought into operation, with a plane ^{60}Co irradiator for the production of new high-thermostable electrical insulating materials, by the radiation vulcanization method of heterosiloxane rubbers (Fig. 1). The industrial startup of a self-adhesive electrically insulated tape and rubberized glass fabric on the facility is finding a wide application in the electrotechnical and other branches of industry.

The construction of radiation—chemical facilities (e.g., for the modification of polyethylene insulation of conductors, the production of rolled Plexiglas, hardening of lacquer-dye coatings, modification of polyethylene tubes and sleeve tape, etc.) has become possible as a result of the development and organization of the production of high-power electron accelerators ÉLT, ÉLV, "Avrora" and "Élektron." Radiation—chemical facilities have been developed with accelerators in "local" shielding, which can be moved to any production location. Thus, in 1976, the "Poplin" facility was brought into operation at the Glukhov cotton combine, with an "Élektron-3M" accelerator for the radiation—chemical finishing of cotton fabrics (for giving them antimicrobe, homostatic, and preservative properties, and also noncrumpling and other desirable qualities). This facility is the first production line in the Soviet Union and in Europe for the radiation—chemical modification of textile materials, and it will allow their finishing with the high-speed passage of 10-100 m/min.

Great importance is being attached at present to the preservation of the environment and, in particular, to the radiation reprocessing of gaseous and liquid industrial and domestic effluents. In connection with this, a facility has been developed and manufactured which has an accelerator for the removal of hydrogen from electrolytic chlorine. The facility has been included in the technological line for the production of gaseous and liquid chlorine and prevents contamination of the atmosphere.

Recently, in the Soviet Union, γ-facilities have been developed for the radiation—biological purpose of irradiating foodstuffs and dairy products and for the general irradiation of agricultural livestock. Among them may be mentioned the "Stavrida" facility for the irradiation of fresh fish and other sea products, under sea swimming conditions for increasing the storage periods, a γ-facility for the preplanting irradiation of grape vine cuttings, in order to increase the yield of cultivated seedlings of the standard varieties of grape. The "Stavrida" facility is loaded with a ^{137}Cs radiation source, with an activity of 85 kCi.

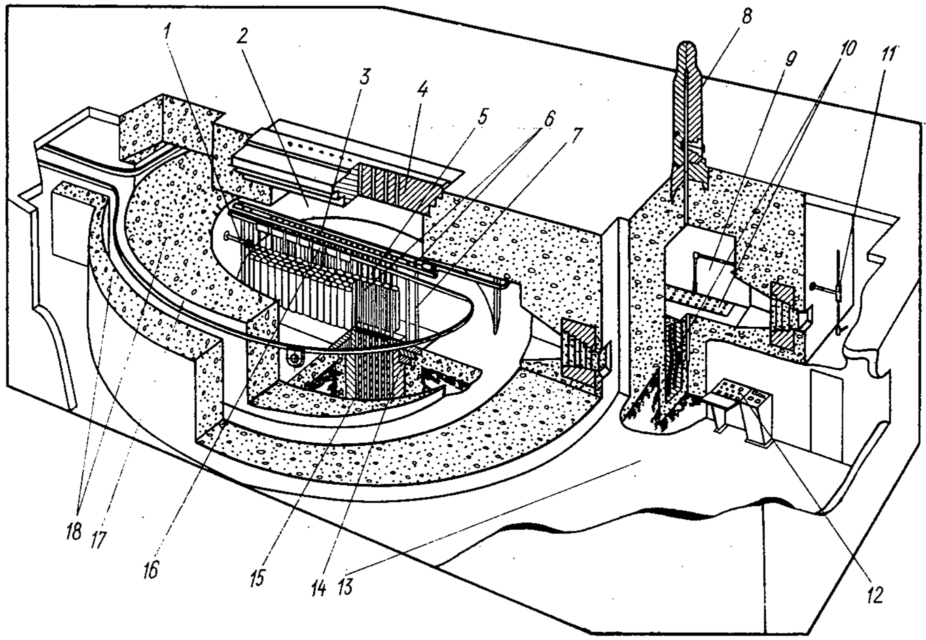


Fig. 1. RV-1200 facility for the vulcanization of electrical insulating tape and rubberized glass fabric: 1) mechanism for turning the tape blocks; 2) irradiator chamber; 3) irradiator traverse; 4) charging device; 5) tubular elements of the irradiator; 6) irradiator; 7) mechanism for moving the irradiator; 8) recharging container; 9) chamber for assembly of tubular elements; 10) assembly chamber depository; 11) master-slave manipulator; 12) control desk; 13) operations area; 14) shielding cast iron plates; 15) irradiator safe; 16) conveyor device; 17) annular conveyor channel; 18) shielding concrete walls.

Radiation sterilization has received a widespread circulation in the country, especially for materials having thermolabile properties (e.g., polymer goods for medical application — systems for the blood service, catgut, catheters, syringe-ampoules, dressings and other items for one-time usage). Radiation sterilization has been accomplished by the use of isotope radiation sources as well as by electron accelerators.

Scientific research on radiation sterilization, carried out in recent years by the institutes of the State Committee for the Utilization of Atomic Energy (GKAÉ) of the Soviet Union, Ministry of Public Health (Minzdrava) of the Soviet Union and Minmedprom SSSR, has been directed at improvement of the parameters and the constitution of the regulations for the technological process of radiation sterilization, and the construction of experimental-industrial facilities. These industrial facilities for the sterilization of blood service systems are already operating in Leningrad and Belgorod-Dnestrovsk with a ^{60}Co radiation source of activity up to 1 MCi. In Kurgan, a facility has been brought into service with two linear electron accelerators, the LUÉ -8/5V. The facility differs in the high productivity, which ensures the reliable treatment (processing) of all the output of factories with a guaranteed sterility (Fig. 2). For the Central Institute of Blood Transfusion, a facility has been developed for the radiation sterilization of bone marrow, transplants and various polymer articles. A number of other facilities have been built for establishments of the medical industry.

Great success has been achieved in radiation-therapeutic equipment construction. In order to set up the complex of equipment fittings for radiation therapy, investigations have been carried out of radiation sources and devices for shaping the dose field as the main elements of this equipment, recommendations have been devised for the optimization and justification of their physical, structural and operating parameters, and basic atlases of dosage distributions have been produced. The main equipment of a complex for the technological equipping of a remotely operated radiation therapy unit has been developed and brought into clinical practice — modern high-power static AGAT-S (Fig. 3) and rotable AGAT-R (Fig. 4) equipment, provided with an extensive set of accessories for transforming the characteristics of the radiation beam. The development and commercial issue of the unique AGAT-V equipment has had a great effect on the reequipment of one of the most widespread methods of contact radiography in onkogynecology and proctology. Remote control in the process of contact irradiation has permitted the activity of the sources to be increased and the duration of the therapeutic session to be shortened.

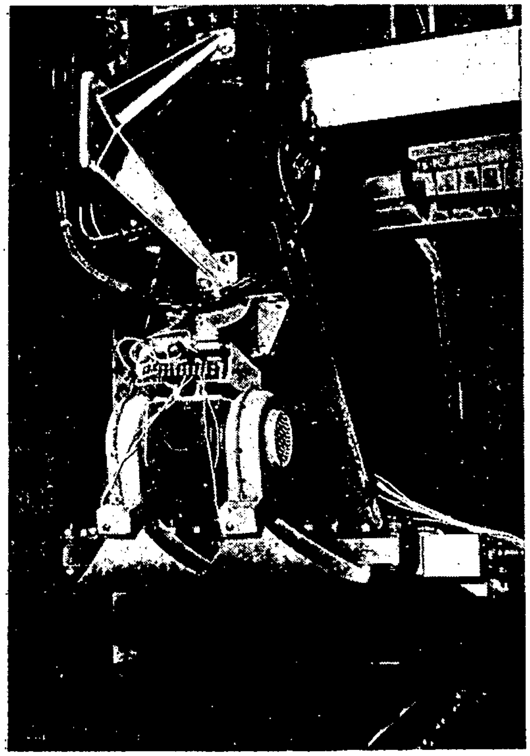


Fig. 2. "Sterilizatsiya II" industrial facility.

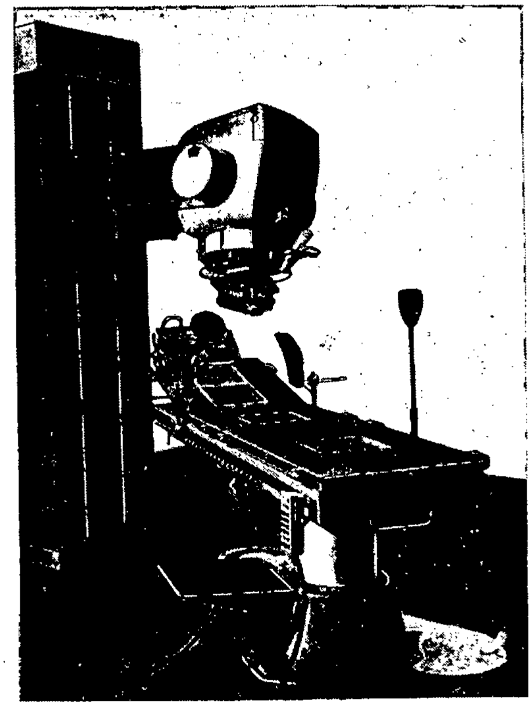


Fig. 3. Static AGAT-S remote control-irradiation equipment.

A development of this trend was the exploitation of a complex of intracavity flexing equipment, comprising equipment of three modifications: AGAT-V1, AGAT-V2, and AGAT-V3, designed for gynecology, stomatology, and proctology. The world's first AGAT-V5 experimental equipment for intracavity irradiation of malignant tumors of the bladder has been developed and manufactured.

The clinical investigations carried out have shown the advantage of intracavity γ-therapy in comparison with the well-known methods of radiation treatment of cancer of the bladder, both by the reaction of the organ and also by the effectiveness of the action on the tumor.

In addition to the development of γ-therapeutic equipments, the possibilities of using other forms of radiation for radiotherapy are also being investigated: α radiation for therapy of burn infections; β radiation for surface therapy in dermatology, ophthalmology and otorhinolaryngology, and neutron radiation for the treatment of radioresistant neoformations. Physical–biological and clinical studies of ^{252}Cf in the leading oncological institutes of the Soviet Union have shown its interesting properties as a source of neutrons for radiotherapy, and have served as the basis for the construction of neutron-therapeutic equipment with high-activity sources.

The automation of radiation-therapeutic equipment has acquired even greater urgency, especially in connection with the complexity of irradiation programs. For this, the accuracy of working out the irradiation parameters has been increased and the time on subsidiary operations has been shortened considerably. Thus, in the ROKUS-M equipment, a system of control is used for the automatic removal of a pendulum with a radiation tip. Methods have been devised for topometric investigations, giving information about the shape and location of a pathological center in an encoded form for its input into a computer, for the purpose of perfecting the most acceptable irradiation program and its subsequent automation.

In order to carry out the research work, a number of versatile radiation technological facilities have been built, with different radiation sources. Radiation-technological equipment may refer to static electricity radiation neutralizers, which over a number of years have been commercially released to industry and are being used in the textile, polygraphic, and other branches of industry for increasing the productivity of plant and for improving working conditions.

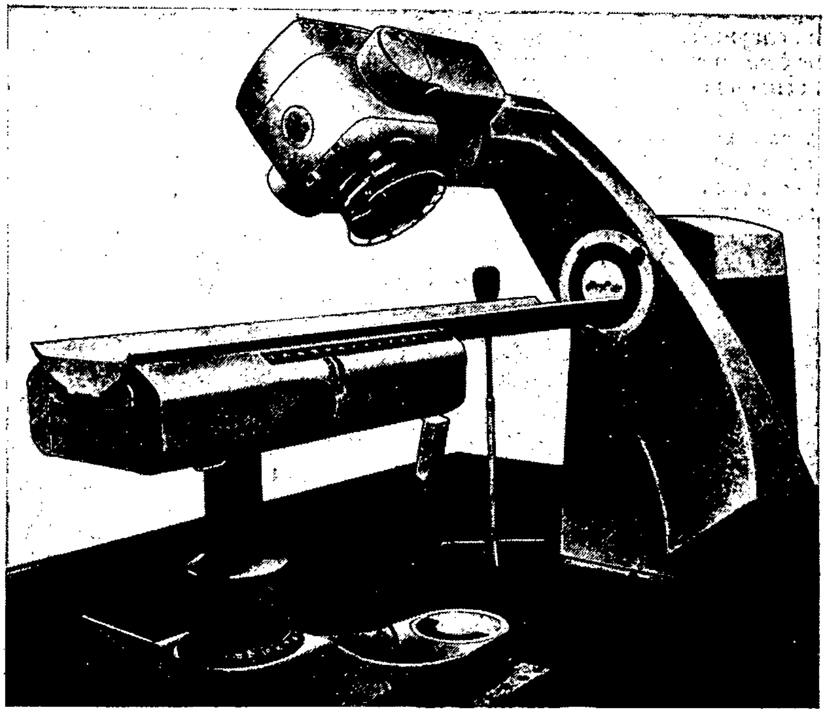


Fig. 4. AGAT-R rotatable and static irradiation equipment.

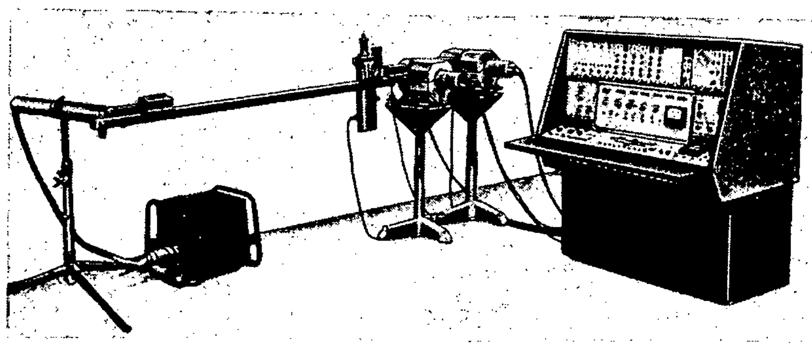


Fig. 5. The K1 facility for the rapid determination of the content of oxygen in a material by the activation method with fast neutrons.

Radiation Instrument Construction

Progress in the automation of monitoring and control in production processes depends to a considerable extent on the means of obtaining the primary information about the values which define the course of these processes. Among these means, which have made their appearance in the last two decades, an important place is occupied by radiation monitoring instruments.

The problem of radiation instrument construction consists in the development and issue of instruments based on the interaction of ionizing radiations with a substance. The information carrier — the radiation signal — is a stream of ionizing radiation (or its absence), the value of which is correlated with various physical quantities and technological parameters, such as the elemental composition of the substance, humidity, density, flow rate, level, thickness, etc.:

$$P = f(F),$$

where P is a defined parameter; F is the flux of ionizing radiation; $f(F)$ is a function of the correlated connection between the parameter and the flux.

There are dozens of interaction mechanisms between ionizing radiation and a substance. By choosing a form of radiation and its energy as a function of the substance being irradiated, a predominant value of that interaction mechanism can be ensured, which will enable the required information to be obtained to the maximum degree.

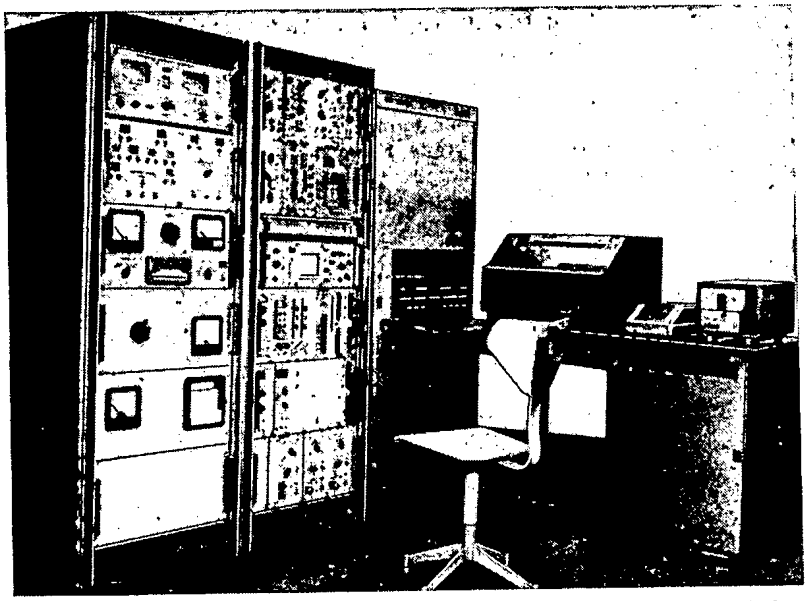


Fig. 6. The MA-1 facility for the multielement analysis of the composition of a substance by fast neutrons.

Owing to the high penetration capability of ionizing radiation, in particular γ-quanta and neutrons, information can be transmitted in the form of a radiation signal through the walls of technological equipment, pipelines, tanks, etc. This permits noncontact measurements to be made of the parameters of technological processes taking place at high temperature and pressure, in highly corrosive media, etc. Radiation-measurement devices are used where the use of measuring devices of other types is impossible or is less efficient. Soviet and foreign experience shows that radiation instruments can be used for solving important problems in the metallurgical, chemical, ore mining, coal, construction, paper and other branches of industry. It should be mentioned, however, that despite the extensive range of radiation measuring devices developed and manufactured in the Soviet Union, the possibilities for their efficient utilization are far from being exhausted.

The development, production and assembly of radiation-measurement devices are being studied in many scientific-research institutes, instrument-construction factories, base and factory laboratories.

Important results have been obtained in the Soviet Union in the field of methodical research, development and construction of analytical instruments for activation analysis (these tasks have been conducted mainly in the Vernadskii GEOKhI, Giredmet, VNIIYaGGe, IYaF AN UzSSR, SNIIPe, VNIIRTe, etc.).*

Activation analysis is being used successfully in industry and in mineral prospecting, in cosmochemistry and biology, in medicine and geology, in chemistry and in metallurgy for studying contamination of the biosphere by microimpurities, etc. In a number of cases, for the bulk analysis of products under industrial conditions, it is sufficient to determine one to three elements. For this, specialized automatic activation analysis facilities have been developed. By means of these facilities, the concentration (from 10^{-4} to 100%) of one or several elements can be found in the course of a few minutes, with an accuracy of 1 to 10%. Facilities for determining the concentration of oxygen and other elements in metals by the activation method with fast neutrons (Fig. 5) have received a large circulation. In the majority of establishments of the titanium industry, activation measuring facilities are being used, which have been developed in the VNIIRTe. For multielement analysis of the composition of a substance, using fast neutron reactions, the MA-1 facility has been constructed, where all operations for carrying out the analysis and processing the results are undertaken by the use of a computer (Fig. 6). For use in activation analysis facilities, neutron generators have been developed in the Scientific-Research Institute of Electrophysical Equipment (NIIÉFA) and in the VNIIRTe, with both continuous pumping (NG-160, NG-150I) and with the use of sealed neutron tubes (NGI-5, NGI-9, NGI-12, and 10N). The

* GEOKhI, Vernadskii Institute of Geochemistry and Analytical Chemistry; Giredmet, State Scientific-Research and Planning Institute of the Rare Metals Industry; VNIIYaGGe, All-Union Scientific-Research Institute of Nuclear Geophysics and Technology, IYaF, Institute of Nuclear Physics of the Academy of Sciences, Uzbekh SSR; SNIIPe, All-Union Scientific-Research Institute of Instrument Manufacture; VNIIRTe, All-Union Scientific-Research Institute of Radiation Technology.

TABLE 2. Characteristics of Accelerators Used for γ-Activation Analysis

Linear accelerator	Energy of accelerated electrons, MeV	Av. power in beam, kW	Current of electron beam at targets, μA	Intensity of bremsstrahlung radiation, 10^3 R/(min·m)	Use for rocks and ores		
					ore	element to be determined	determination threshold, %
LUÉ-8	8	5–7	700	14 000	Gold content	Gold	$5 \cdot 10^{-5}$
LUÉ-15	15	7–10	500	55 000	Beryllium–tungsten–molybdenum	Molybdenum	$2 \cdot 10^{-3}$

highest analysis sensitivity can be obtained by the use of a nuclear reactor as the neutron source. The RG-1M nuclear reactor, installed in the Norilsk Mining-Metallurgical Combine, has been specially developed for activation analysis. Analysis is undertaken in the analytical laboratory of the combine, using radiochemical methods for determining the noble and rare metals with a content down to 10^{-7}% in copper–nickel ores and in their reprocessed products. In addition, the majority of the noble metals with a content down to 10^{-5}% (gold to 10^{-6}%), nonferrous metals, and rock-forming elements are determined by instrumental activation methods.

A further development of the methods of technological monitoring is the analysis of the composition of a substance in a flow. These methods are of considerable interest for continuous technological processes in industry. Theoretical and experimental investigations have been undertaken in a number of organizations [VNIIRT and the Ural Scientific-Research Chemical Institute (UNIKhIM), etc.], relationships have been developed and analysis procedures have been devised. Based on these investigations, several versions of equipment have been built for the activation analysis of solutions in a flow, and also instruments for neutron-absorption analysis, including a facility for monitoring the content of boron in technological solutions and in the coolant of nuclear power stations.

In addition to neutron activation analysis, γ-activation analysis has received even more extensive development for the multielement determination of the composition of rocks and ores, having high selectivity, sensitivity, accuracy and speed. One of the most promising fields for the application of this method is the analysis of geological samples. The most suitable radiation source for activation analysis is the linear accelerator (Table 2). Electron accelerators enable intense fluxes to be obtained, not only of bremsstrahlung, but also of neutron radiation. Thus, the LUÉ -15 with a beryllium converter provides a neutron output of up to $2 \cdot 10^{13}$ m/sec, which very significantly expands the circle of analytical problems which can be solved by means of electron accelerators, thanks to the application of neutron activation analysis in a complex with γ activation.

The x-ray-radiometric method of analysis of a substance, in its physical essence, is similar to the classical x-ray spectral analysis, but differs from it in that the primary radiation source is not an x-ray tube but is a radioactive isotope. In comparison with the x-ray spectral method, the x-ray-radiometric method as yet has a worse sensitivity. However, to the positive qualities of x-ray-radiometric instruments may be attributed simplicity, small dimensions, the possibility of automating technological processes, the capability of determining light elements owing to the crystalless system for recording the low-energy characteristic radiation (emission) of these elements and the stability of the primary radiation source. These qualities, in conjunction with the sensitivity of the method which is sufficient for most practical problems, explain the wide possibilities for its use for the bulk high-speed analysis of technological and geological assays, the analysis of rocks and ores in natural deposits, and the automatic monitoring in-stream of technological media (Table 3). All these courses have been extensively developed in the Soviet Union.

An important course of work in the field of x-ray-radiometry is the analysis of light elements, where the absorption of the fluorescent radiation by air starts to have an effect. In these cases, the measuring instruments are installed in a vacuum chamber ("Mayak-V").

Broad production parameters (technological) can be measured by means of radiation monitoring-measurement instruments. The most widespread are radiation density meters, level meters, thickness meters, humidity meters, and also relay instruments. These instruments are used for the solution of problems requiring total noncontact measurements (e.g., the measurement of the density of pulp in steel tubes with a total wall thickness of up to 30 mm, the determination of the density of sulfuric acid salts in technological processes for the production of ferrites, the measurement of the level of corrosive, free-flowing, effervescent substances, crystallizing and precipitating in residues, the level of the boundary of separation of liquid media, the measurement of the thickness of sheet materials, the thickness of coatings, etc.). Radiation methods of measuring the moisture content and density of soil and earth, by comparison with thermo– and volume–weight methods, reduce the time and labor expenditure by a factor of 5-10. Therefore, in KAMAZ, BAM, and other establishments, radiation bathymetric hygrometers RVG-36 and densimeters RPG-36 are being used efficiently.

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TABLE 3. Characteristics of Some Soviet Domestic X-Ray-Radiometric Instruments

Instrument	Purpose	Operating principle	Parameter		
			range of concentration, %	error of determination, %	measurement time, min
AZhR-1	Rapid analysis of elements of the Fe group in powdered samples	Fluorescent, with compensation of the matrix effect by β-scattering	1-70	0.5	1-2
KTN-1	Monitoring the in-stream concentration of Ti and Nb in solutions	Absorption, by rapid change of absorption coefficient	0.1-20 g/liter	0.05 g/liter + 1%	-
FRAD-1	Rapid analysis of elements with 20 ≤ Z ≤ 92 in powdered samples	Fluorescent, with autostabilization (2 channels)	0.1-100	0.1 + 0.005	3
KTN-2	Rapid analysis of Ta and Nb in powdered samples	Fluorescent (spectral ratio method)	0.1-100	0.01 + 1	10
KTN-3	Rapid analysis of Ta and Nb in solutions	Absorption, by rapid change of absorption coefficient	0.1-100	0.01g/liter+1	10
FAM-3-01	Assaying of molybdenum ores in natural deposits	Fluorescent, with subtraction of background	0.02-4	0.025 + 0.05	3
"Mayak-R"	Rapid analysis of elements with 24 ≤ Z ≤ 92 in powdered samples	Fluorescent, with autostabilization (4 channels)	Threshold of detection 10^{-2}	1-7	2-15
"Mayak-V"	Rapid analysis of elements with 11 ≤ Z ≤ 24	Fluorescent, with vacuum chamber	Threshold of detection 10^{-1}	5-10	2-10
RPS-4-01	Rapid analysis of mineral raw material in natural deposits and elements with 22 ≤ Z ≤ 92 in powdered samples	Fluorescent, with differential filters	Threshold of detection 10^{-2}	5	1

Large-scale problems are being solved in industry by means of radiation relay instruments. Seventy high-sensitivity scintillation radioisotope relay instruments RRPS-1, connected into the automated system of one of the world's largest blast furnaces at the Krivorozhsk Metallurgical Combine, reliably determine the level of the charge in the bunkers of the charge feeder, and eliminate overfilling of the bunker hoppers. In many establishments, in organizations and engineering plants of the Soviet Union RUOP-1 radioisotope fire warning systems with ionization probes are installed.

Radiation defectoscopy has become a powerful tool for increasing the efficiency of production technology and the quality of the manufactured goods and structures. Radiation defectoscopes enable nondestructive monitoring to be carried out, of the quality of welded joints, plates, forgings and fabricated assemblies, over the thickness range from 1 to 200 mm. By means of the γ and neutron radiation sources used in these instruments, items of complex shape can be monitored, as well as objects located in poorly accessible places, parts and units during their manufacture, welded seams of the main pipelines, fuel elements and fuel assemblies of nuclear reactors, and power generating plants, including also nuclear power stations.

Special sharp-focus sources based on ^{192}Ir have been constructed, which allow monitoring at small focal distances to be carried out. For monitoring thin-walled products, the possibility was investigated for the use of low-energy radiation sources based on ^{147}Pm, ^{204}Tl, ^{55}Fe, ^{90}Sr, ^{241}Am, etc.

A new trend of radiation defectoscopy is the use of three-dimensional-coded sources of ionizing radiation. This makes it possible to produce three-dimensional information about the assembly being monitored. The start of development of gas-sorption methods of monitoring has been proposed for the detection of defects in the surface layer of products. Gamma-defectoscopic instruments of the series GAMMARID have been developed and brought into commercial production; these instruments replace the γ defectoscopes such as RID-21m, Gasprom, RID-11, etc. The instruments of this series differ in their high degree of unification. For monitoring the quality of the welded joints of main pipelines with a diameter of 1020-1620 mm, and with a wall thickness of up to 30 mm, under field conditions, the "Magistral'" and "Magistral'-1" γ defectoscopes have been developed.

As a result of the investigations on the use of neutrons for nondestructive monitoring of parameters and manufactured goods, rules have been developed which define the feasibility, conditions and special features of the use of neutrons for these purposes, many problems connected with the practical application of the method have been solved, permitting data to be obtained about the parameters of assemblies located behind screens of heavier materials, and enabling the inclusion of hydrogen-containing compounds, substances with a large neutron absorption cross section to be detected, etc. Work is being carried out on the creation of technical means of neutron radiography based on powerful isotopic neutron sources of ^{252}Cf, and also specialized nuclear reactors.

TABLE 4. Use of Radioisotope Power Facilities

Field of application	Purpose and type of facility	Electric power, W	Service life, yr
Hydrometeorology service	Power supply for automatic meteorological stations in difficultly accessible and uninhabited regions of the Soviet Union ("Beta-S"; "Beta-M")	8-12	More than 10
Navigation	Power supply for maritime radiobeacons with a range of action up to 200 miles, lighthouses, illuminated beacons in different climatic zones, including on the track of the Arctic Sea route ("Éfir-M")	30	up to 5
Geomagnetic service	Electric power supply and thermostatic control of magnetic stations in Antarctica ("Pingvin" = "Penguin")	2	-
Space research	Heating of equipment in space vehicles Lunokhod 1 and Lunokhod 2	-	Practical duration of operation 6-10 months
Medicine	Power supply for electrocardiostimulators (Pacemakers) ("Ritm" = "Rhythm")	$n \cdot (10^{-6}-10^{-3})$	About 10

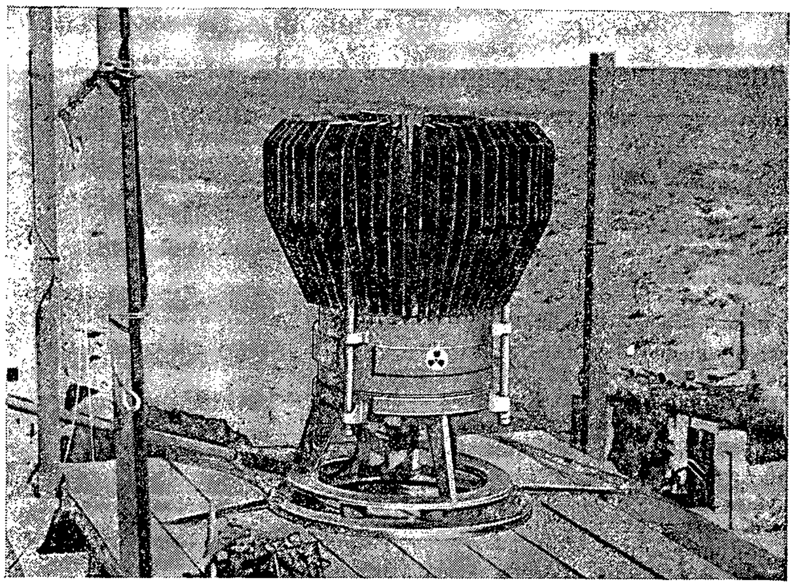

Fig. 7. Radioisotope power supply, "Éfir-M."

Radioisotope Power Generation

In the present stage of science and technology, there exists the practical possibility of converting the energy of radioactive decay into any other form of energy — heat, electrical, mechanical, light, acoustic, etc. Radioisotope sources of thermal and electrical energy usually consist of a heat source, containing ampoules with radioactive substances, a converter of the thermal energy into electrical energy, a device for transmitting the thermal energy to the consumer, and a radiator for the removal of unused energy into the surrounding space. The conversion of thermal energy into electrical energy is effected predominantly with thermoelectric converters, and also with thermoemissive and dynamic systems. There are methods for the direct conversion of radioactive decay energy into electrical energy by means of the direct collection of the charged particles. In another type of nuclear battery, by the action of irradiation, current carriers are generated in a semiconductor material, which then are separated by the electric field of a p—n junction.

Radioisotope power generation is at the junction of several sciences (nuclear physics and radiochemistry, thermophysics and electrotechnology, solid state physics, etc.). Over a comparatively short period, a complex of projects has been accomplished in the Soviet Union, in particular on the search for the most efficient methods of producing radioisotope fuel, the technology of manufacture of thermal radioisotope sources, meeting the demands of radiation safety; the principles of construction and of the mathematical modeling of the interrelated nuclear-physical, thermophysical and electrical processes have been worked out, for the optimization of the parameters of radioisotope energy sources. ^{90}Sr has found extensive use as the basic type of fuel for terrestrial thermoelectric generators; ^{238}Pu (PuO_2) (for units with a long service life) and ^{210}Po (for units which tolerate a reduction of power in the initial period, with its corresponding adjustment throughout its further operation) are widely used for high-power, power-generating facilities.

The complex of work carried out has made it possible to accomplish the commercial production of radioisotope sources for thermal and electric power (Table 4). In particular, in the Soviet Union, more than 200 automatic radiometeorological stations are operating with radioisotope power supply sources "Beta-S" and "Beta-M," developed in the All-Union Scientific-Research Institute of Radiation Technology, with the inclusion of certain specialized organizations (Fig. 7).

In 1975 at the Scientific-Research Institute of Clinical and Experimental Surgery, Academician B. V. Petrovskii carried out the first operation in vivo of a Soviet radioisotope electrocardiostimulator (pacemaker) in the body of a sick person, suffering from a total transverse blockage of the heart.

The brief and by no means complete review undertaken of the achievements of Soviet researchers in the field of radiation techniques shows that this comparatively new trend has been formed into a large-scale independent branch of nuclear science and technology, without which today further technical progress in the different fields of industry, medicine, agriculture, and also in scientific research would be unthinkable.

COLLABORATION BETWEEN COMECON MEMBERS IN POWER REACTOR DESIGN, INCLUDING SOME ASPECTS OF NUCLEAR FUEL CYCLES

I. Barbur, A. Barchenkov,
L. Molnar, A. Panasenkov,
V. Tolpygo, V. Khake,
and B. Shcherbinin

Fuel and Power Topics

A major purpose of economic and scientific collaboration up to 1990 is to meet the fuel and power requirements of COMECON member-countries. At present this is being handled under the joint program for extending collaboration and development of socialist economic integration, while in the future it will be handled within the framework of long-term programs, including ones designed to meet the needs of member countries for energy, fuel, and raw materials.

There is a tendency for the importance of solid fuel in the energy balance of member countries to increase; coal remains one of the major energy sources, and the scale of coal production will increase. Collaboration between member-countries will meet the main requirements for petroleum, natural gas, and petroleum products; in 1975, deliveries from the USSR to Bulgaria, Hungary, the German Democratic Republic, Poland, and Czechoslovakia reached about 62 million tons of petroleum and 14 billion m^3 of gas. Various agreements have been signed on multilateral collaboration in locating and exploiting new petroleum and gas deposits, including offshore ones. An important part will be played by measures for more efficient use of liquid fuel, in particular by increasing the yield from petroleum refining to 70%.

The importance of the multilateral approach to large-scale fuel and energy problems will also increase considerably. The most important integration measure is the agreement on collaboration in exploiting the Orenburg gas-condensate deposit. Under this, joint efforts during 1976-1980 will be devoted to the construction of a pipeline from Orenburg to the western border of the USSR, total length 2800 km. The construction of this will provide member-countries with a further 15.5 billion m^3 of gas per year.

The total electrical energy production in member countries in 1975 reached 14,000 billion kW-h. Electrical power is an area in which considerable success has been attained in economic and scientific collaboration, which is to be further extended. An important stage in this is provided by the general scheme for the joint power system in Yugoslavia.

Translated from Atomnaya Énergiya, Vol. 43, No. 5, pp. 402-408, November, 1977.

0038-531X/77/4305-1053$07.50 © 1978 Plenum Publishing Corporation

Declassified and Approved For Release 2013/03/22 : CIA-RDP10-02196R000700100001-6

This general scheme has been based on forecasts for the individual power systems, including definition of the optimum voltage level for transfers between systems, development prospects for nuclear power, the design of an interstate transmission system working at 750 kV, and proposals for collaboration on power stations, including large nuclear power stations and pumped-storage stations.

Considerable advantage will come from exploiting the differences between the load curves for the member-countries, which will make available considerable reserve power in these rapidly extending power systems; this may save the installation of about 4600 MW of generating capacity in the member countries, which is equivalent to about 400 million convertible rubles in terms of capital investment in generating output at the 1990 level.

Very extensive use of centralized heat supply from thermal power stations is envisaged in order to ensure more rational use of fuel, with increase in the power supplied by district-heating systems by a factor 1.8 by 1990. In addition, district heating by nuclear power stations will be extended.

Nuclear electricity is playing an increasing part within COMECON; the emphasis on this derives from the fact that the structure of the fuel and power balances of the member-countries up to 1980 must involve a considerable increase in the contribution from nuclear power stations, and the same applies in preliminary forecasts of demands for fuel and energy up to 1990: for the first data, Bulgaria requires 7700 MW, Hungary 5700, Poland over 8000, and Czechoslovakia 10,000 to 12,000.

The total output from nuclear power stations in member-countries rose in 1971-1975 from 1100 to 7500 MW. Nuclear power stations are now operating in Bulgaria, the German Democratic Republic, the USSR, and Czechoslovakia, and further such units are being built in these countries. A start has been made on nuclear power stations in Hungary, Poland, and Rumania. A decision has also been made to build a nuclear power station in Cuba. The total output from nuclear power stations in member countries should be about 30,000 MW by 1980.

This rise in nuclear power in member-countries during 1971-1975 was made possible by the routine production of the VVÉR-440 reactor type; during that period, researches were performed on optimizing the water parameters in the VVÉR-40, and also on improvement of the monitoring and control systems, as well as on reliability and safety.

The increase in the energy potential of member countries is therefore to come primarily from solid fuel, hydroelectric power, and nuclear energy. To an increasing extent, oil and gas will be redirected to other uses.

Scientific Collaboration

The plan for scientific collaboration in the use of nuclear energy for peaceful purposes for 1971-1975 envisaged joint research and development studies on 11 major problems and 50 topics. About 80 institutes and organizations collaborated. Particular attention was given to the VVÉR-1000, fast reactors, reactor monitoring and control, and power station equipment, as well as particular links in the fuel cycle.

Techniques, instruments, and systems have been developed, along with the appropriate documentation, for direct use in power stations, which has gone in hand with the production of nuclear-engineering instruments, facilities for reprocessing irradiated fuel, and radioactive-waste processing facilities. In Hungary, an instrument has been developed for measuring the boron level in the heat carrier in the first loops of reactors, while in the German Democratic Republic an instrument has been designed for determining the water content of steam in a nuclear power station, together with systems for making measurements within the reactors at the Bruno Leuschner Nuclear Power Station; Poland has developed small transducers for use within the reactor core, and Czechoslovakia has designed modular steam generators of output 30 MW for fast-reactor power stations, an electromagnetic stepping drive for control rods, and a fast pump of output 20,000 m^3/h for reactor loops. In Poland, the USSR, and Czechoslovakia there have been researches on improved designs of steam generators for power stations fitted with VVÉR reactors.

Some major researches have also been performed on the physics and hydrodynamics of fast reactors; considerable advances have been made in research methods and computational programs. The basis has been laid for further extension of collaboration in this area. The cooperation in fast-reactor research is based on joint use of physical facilities, computers, and methods.

Much work has also been done on writing new programs and improving existing ones for calculations on the cores of thermal and fast reactors, which has been carried through by the Temporary International Research Team for research in reactor physics, with critical assemblies of VVÉR type, which was set up in 1972 at the Central Physics Research Institute of the Hungarian Academy of Sciences.

There has also been extensive collaboration between member countries on research reactors, which has led to increased power levels in Hungary, the German Democratic Republic, Poland, Rumania, the USSR, and Czechoslovakia, where there have been rises from 2 to 10 MW, together with substantial extension of research facilities. Particular note should be made of the commissioning of the Marija high-flux research reactor (30 MW) in Poland, which provides the basis for an extensive research program in solid-state physics, radiation materials science, and nuclear physics.

An important place is taken by forecasts for the development of nuclear power as part of general forecasts on fuel and power requirements. The purpose of these studies is to choose the best line of advance for nuclear power engineering in member countries on the basis that nuclear power will gradually become a major factor in power production, not only with external electrical implications but also specific internal relationships arising from the fuel cycle. In this respect, nuclear power is an international system, from which the best economic return will be obtained by a balanced consideration of national nuclear power requirements of member countries.

A major task in forecasting is to define parameters for the integrated nuclear power systems of member-countries, which involves defining the differences between countries and determining the extent and scale of economic integration. Here the fragmented system may be taken as the sum of the available output powers and the demand levels for uranium, assuming independent development of nuclear power within national frameworks, while an integrated system involves joint development of nuclear power, assuming international integration on account of external fuel recycling.

Computer simulation has demonstrated, in particular, the relationship between the demand for natural uranium on the all-thermal assumption (only thermal reactors used) as against that for combined development (combination of thermal and fast reactors), as well as effects from accelerated or retarded circulation of nuclear fuel in the reprocessing cycle, which affect the structure of uranium demand in the fragmented and integrated systems.

It has been found that re-use of plutonium in thermal reactors can reduce the demand for natural uranium by about 10%. The most effective means of solving the fuel problem is the extensive use of fast breeder reactors, and it has been found that the effects of such reactors in reducing the demand for natural uranium will be comparatively small for 5-10 years after they begin to be introduced on a large scale. The major changes in the structure of national nuclear-power systems, and hence in the natural-uranium economy, are to be expected after 15-20 years. This is the period that will be required for fast breeder reactors to reach 50% of the installed nuclear power, by which point the saving in natural uranium may be about 30%.

This analysis of the structure of nuclear power engineering has shown that the integrated system, which provides the best means of utilizing the fuel, can also improve substantially the structure of the industry as well as the fuel-demand parameters, by comparison with the fragmented system. This should mean that the proportion of fast breeder reactors in the nuclear-power system could be increased by 8-12%, while the corresponding saving in natural uranium in the relevant period will be increased by 13-14%.

A very important point is the cycle time T_c in the external fuel cycle; the effect of this is larger for the integrated system because the latter assigns more significance to plutonium. If T_c increases from 1 yr to 2 yr., the integral demand for natural uranium over the entire period increases by 50%, whereas if T_c is reduced from 1 yr to 0.5 yr the demand is reduced by 20-25%. This indicates that reduction in T_c must be considered as a very important problem, whose solution may lead to a substantial increase in the contribution of fast reactors to nuclear power as well as to the saving of natural uranium.

The External Fuel Cycle

Collaboration of member-countries in this cycle is concentrated mainly on spent fuel reprocessing, together with transport, processing and storage of radioactive wastes, and radiation safety.

Spent Fuel Reprocessing. Here collaboration is directed to optimizing fuel reprocessing schemes and the design of reliable high-productivity equipment, as well as design of methods and instruments for monitoring and controlling processes and defining optimal processing. There is also considerable emphasis on researches on the isotopic composition of spent fuel by destructive and nondestructive methods.

Under the joint program, a technological scheme has already been devised for wet reprocessing of spent fuels with tributyl phosphate and a heavy incombustible diluent, which is envisaged for reprocessing VVÉR fuel rods of burnup about 30,000 MW·day/ton. It is proposed that subsequently the scheme will be improved and adapted for reprocessing fast-reactor fuel of burnup about 100,000 MW·day/ton. One of the major results is the design of equipment required for the scheme, which includes small extractors of pulsation, turbine, and centrifugal types in various sizes, a solvent-processing system, and so on.

A technological scheme has also been devised for reprocessing fuel rods by a fluoride technique, which may be used for selecting the optimum means of regenerating fast-reactor spent fuels.

Techniques and equipment have also been developed for thermal removal of fuel-pin sheaths. The process can remove stainless-steel casings from fast-reactor rods and zirconium casings from VVÉR rods. It has been found that stainless steel does not react with uranium or plutonium dioxide at 1500°C. The oxide content in the stainless steel does not exceed 0.05% by weight.

Numerous researches have also been performed on process monitoring and control; a monitoring scheme has been devised for solvent-extraction processing of spent VVÉR rods, together with a method and instruments for radiometric assay of uranium and plutonium, an instrument for simultaneous determination of uranium and free nitric acid, a system for taking small-volume samples, a neutron method of measuring solution level, an acoustic method of monitoring the position of a phase interface in chemical equipment, neutron level gauges, and thermal flow gauges working over the range from 1 to 500 liter/h.

Economic studies have also defined the optimum throughput of a processing plant, which is 1500 tons of uranium per year, and have also given criteria for siting such a plant.

Spent-Fuel Transport. This is one of the major links in the fuel cycle. The proposed future scales of nuclear power in member countries require the shipping of large amounts of spent nuclear fuel from power stations to regeneration plants. Collaboration between member countries in this area is designed to solve technical problems such as specialized transport facilities and standardized international documents having legal force and regulating transport aspects.

There has been an extensive series of studies on the design and standardization of transport facilities. A special container has been built for safe shipping of spent VVÉR fuel. Technical specifications have been drawn up for shipping spent fuel rods from nuclear power stations with VVÉR-440 reactors. These conditions have been recommended for use in a standard document such as is required in contracts for handling spent fuel rods.

A document entitled "Rules for Safe Transport of Spent Nuclear Fuel from Nuclear Power Stations in Member Countries. Part 1: Railroad Transport" has been drawn up. The definition of the second part of these rules (transport by water) is to be completed in 1978.

Radioactive Waste Disposal. A major condition for the extension of nuclear power is a solution in this area.

Experience in the industrially developed countries has shown that the levels of radioactivity are not rising, which is confirmed, in particular, by systematic measurements performed by member countries on the radiation environment in the basins of the Danube, Black Sea, and Baltic. Nevertheless, methods of disposing of radioactive wastes must still consist to a considerable extent of safe storage, not final release to the environment.

In 1971, a coordination council was set up to define a practical solution to the processing and storage of radioactive wastes, which was to deal also with equipment decontamination. The main tasks of this body were to organize multilateral collaboration, analyze the state and development trends in researches in this area, define major lines of research, examine the economic results of scientific-technical researches, and exchange information.

In 1971-1975, collaboration occurred under the program "Researches in the disposal of liquid, solid, and gaseous radioactive wastes and decontamination of contaminated surfaces." The studies under this program served to define conditions under which virtually all components of liquid radioactive wastes could be incorporated into bitumens, especially the wastes produced by nuclear power stations, and these define an optimum choice of bitumen treatment for radioactive wastes. Tests on batch-treatment systems for the purpose were based on hot mixing, which led to recommendations that such systems should operate with outputs of 20-60 liters/h.

Documents laying down methods were also devised; a list of unified analytical and economic parameters for evaluating the performance of existing equipment for processing low- and medium-level wastes, criteria for selecting methods of disposing of radioactive wastes, in accordance with the properties of the waste and the natural conditions of storage, a method for selecting the optimum mode of bitumen treatment for radioactive wastes, and a method of selecting safe storage conditions for wastes in accordance with properties and specific activity.

Much attention has been given to research on means of storing radioactive wastes of all types in geological formations in order to define the scope for storage in near-surface horizons, deep water-bearing strata, and salt formations. Extensive theoretical, laboratory, field, and pilot studies have been made on all three types of storage. Over 30 reports have been prepared on the sorption, migration, and diffusion of radioelements in soils and water-bearing horizons, as well as on the conditions for compatibility between radioactive wastes and rocks or stratal waters, techniques for preparing wastes for storage in deep water-bearing horizons, and chemical methods of adapting boreholes for such disposal.

The results are important in predicting the dispersal of radioisotopes in water-bearing horizons for various mineral compositions of the latter, and also in organizing systems for monitoring underground storage.

Researches on the temperature distribution in stored wastes in geological formations are of considerable theoretical importance, and in many instances they are decisive in selecting storage techniques.

Basic schemes have been drawn up for underground storage in deep absorbing horizons, which take account of the hydrogeological conditions, the composition and volume of the wastes, and the design of service and underground structures in the storage. Plans have also been drawn up for experimental large-scale storage of wastes in an abandoned salt mine, while a unified system of collection and transport containers has been defined for wastes of medium and low activity. On the whole, one can say that we now have the technical basis for underground storage of radioactive wastes in geological formations.

The main purposes of the program for 1976-1980 are research on optimum schemes for waste processing and storage and definition of industrial-scale equipment.

Advances in nuclear power, with its associated fuel cycle, have made it very important to design large systems in which a single site houses not only a group of nuclear power stations whose total output is some dozens of GW but also the associated plants for fuel reprocessing and waste handling. Such an approach provides for best use of the ecological capacity of the area around a nuclear power station, while minimizing the costs involved in transporting spent fuels, organizing the best waste storage, designing economically optimal plants for processing spent fuel, and creating favorable conditions for the extension of linked power systems.

Radiation Safety

Member-countries have also given considerable attention to radiation safety, which forms part of "The general expanded program for collaboration between COMECON member-countries and the Federal Socialist Republic of Yugoslavia for the period up to 1980 in environmental protection and rational use of natural resources."

The main attention has been given to engineering measures to ensure safety at nuclear power stations: monitoring systems for major equipment, dosimetric monitoring of staff and power-station buildings, as well as of the population and the environment, and sampling of air, water, and soil, as well as other forms of monitoring.

A coordination council on radiation safety has been set up, and the following major lines of collaboration have been laid down:

1. Radiation monitoring of the environment (methods of taking samples for examining water for radionuclides, recommendations on dosimetric monitoring in areas around nuclear power stations, recommendations on environmental monitoring for ^{131}I escaping into the atmosphere, etc.).

2. Definition of standard documentation on methods in radiation safety in the design, construction, and operation of nuclear power stations (initial conditions and criteria for selecting construction areas, general principles in the provision of safety at nuclear power stations during design, construction, and operation, specifications for concrete and for concrete shielding structures, and so on).

3. Definition of technical aspects of radiation safety (engineering aspects of building safety and nuclear-power station operating safety, design of remote-controlled nondestructive monitoring methods for major metal components generally, pipelines, welded joints, and equipment generally in reactor systems during operation, etc.).

4. Definition of measures for preventing emergency situations and for eliminating consequences of the latter (classification of emergency situations and methods of determining the causes of emergencies, specifications for environmental protection in radiation accidents, basic measures for preventing heat-carrier leaks, etc.).

5. Definition of measures for reducing radiation exposure of nuclear-power station staff during normal operation (recommendations on decontamination of equipment, definition of water purity standards, definition of methods for decontaminating water, etc.).

The first International Conference within the COMECON framework on radiation safety at nuclear power stations was held in Czechoslovakia in Sept. 1975; this dealt with topics such as the siting of nuclear power stations, planning against emergencies, engineering aspects of environmental protection, estimation of the effects of emergencies, radioactivity escapes, dosimetric monitoring of populations, the radiation burden of nuclear power station staff, etc.

At the present time, the available evidence indicates that nuclear power stations are highly safe, while they produce very little environmental contamination.

A large range of topics in radiation safety is to be considered in 1976-1980; collaboration between member countries in radiation safety is a major task, particularly because any disturbance in the environment may have global consequences for many countries, which can be eliminated or prevented best by the combined efforts of all governments.

International Economic Organizations

Advances in nuclear power have posed a major problem for the nuclear engineers of member countries; decisions are needed on the principles to be used on international division of labor, specialization, and cooperation in production. It is proposed that joint forecasts should be utilized, along with collaboration and coordination in planning, starting with research studies and design plans, and finishing with coordination of production plans for nuclear power station construction, in addition to collaboration in founding joint organizations, creating international economic organizations, etc.

Several bilateral agreements have been signed between the governments of the USSR, Bulgaria, Hungary, the German Democratic Republic, Poland, and Czechoslovakia on cooperation in the production of nuclear power station equipment up to 1980. These agreements envisage the manufacture and supply of reactor complexes, steam generators, and other equipment for power stations containing VVÉR-440 reactors for member countries. A program has been laid down for developing nuclear engineering in member-countries, including cooperation and specialization in production in this area, which runs up to 1990. An industrial base for nuclear engineering has been laid and extended in the USSR, Czechoslovakia, and other member countries.

The international economic organization "Interatoménergo" has been set up to advance cooperation in the production of equipment, in provision of technical assistance, and in the design, commissioning, and support of nuclear power stations, including repairs and the supply of spare parts. At the present time, this organization participates in the program for developing nuclear engineering.

The "Interatominstrument" international economic organization has also been set up in order to meet the requirements of member countries for instruments and devices for nuclear engineering; in 1975-1976, service branches of this organization were set up in Bulgaria, Poland, and the USSR to service instruments and equipment for nuclear engineering, which is a major step forward in the activity of this organization.

Agreement has also been reached on international trade in instruments and equipment for nuclear engineering in 1976-1980. These agreements have been incorporated in intergovernmental trade agreements for 1976-1980, which have been signed by the participating governments in Interatominstrument. As a result, the export volume of nuclear engineering equipment of the participating countries in 1980 will be over twice that in 1975.

International collaboration in nuclear power will also occupy a major place in COMECON activities in the future.

The developing collaboration between COMECON and the International Atomic Energy Agency will facilitate scientific and economic advance, including realization of concepts in the decisions of the Conference on Safety and Collaboration in Europe in the section concerned with the collaboration, economics, science, and the environment.

TECHNOLOGICAL ASPECTS OF MANUFACTURE OF FUEL FOR DIFFERENT POWER REACTORS

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The requirements imposed on the fuel-element cores are different depending on the intended use and conditions of operation of the power installations. For uranium dioxide cores, used in slow neutron reactors, these requirements have already been formulated to a large extent. The requirements on the cores of mixed oxide fuels for fast reactors have also been mainly formulated. It is worth nothing that as experiences in the operation of fuel elements in different countries are accumulated, the imposed requirements are also refined and they get closer.

The data on the efficiency of the active zone with fuel elements containing carbide and nitride fuels are considerably less. Therefore, the requirements on this fuel are still not formulated in the right manner.

Uranium Oxide Cores. The technological conditions imposed on fuel-element cores represent a set of diverse requirements including the geometrical sizes, the density, the mechanical hardness, the oxygen coefficient, the content of certain impurities, etc. These properties are affected by a whole series of technological factors; some of these properties are interrelated. Therefore, the problem lies in understanding this interaction, estimating the effect of different technological factors, and choosing recommendations which would insure stable indices for obtaining cores with given properties.

One of the primary factors is the density, which depends on the baking temperature and the gaseous medium, on the coarseness of the powder, and the oxygen coefficient of the original uranium dioxide.

The baking conditions of the tablets must be chosen in such a way that the main amount of gases (fluorine, oxygen, carbon monoxide, hydrocarbons, etc.) are evolved from the tablets before the beginning of the significant compacting. Otherwise different defects may appear in the tablets.

Of the gaseous media the requirements on the baking conditions are best satisfied by hydrogen. During baking in hydrogen significant compacting of the tablet begins at a temperature above 1300°C, when the main mass of the gases has already been evolved.

The coarseness of the uranium oxide powder has a significant effect on the density of the obtained tablets. The optimum lower limit of coarseness of the uranium oxide powder, determined by the method of gas penetrability at small rarefaction, comprises 0.35-0.40 μ, whereas the upper limit is 0.55-0.60 μ. The density of the tablets is important not only in respect to the amount of the nuclear fuel per unit volume; the amount of moisture and gases, in particular, hydrogen collected by the core of the fuel and hence, the reliability and efficiency of the fuel elements also depend on it. All these parameters are controlled quite rigidly.

The evolution of gases from dense tablets continues during their heating up to 1700°C. From less dense tablets having significantly open porosity the main amount of gases is evolved during heating up to 800°C. Of all the gases evolved from fuel-element cores the most hazardous is hydrogen, an appreciable amount of which goes under the casing in the form of water collected by the surface of the uranium dioxide tablets.

The dependence of the water content in the tablets on their density is shown in Fig. 1 (after N. P. Zvereva); it follows from this figure that for a tablet density of 10.3-10.4 g/cm^3, the amount of moisture in them does not exceed $(6-8)\cdot 10^{-4}$ mass %. However, a view has been expressed that a further decrease of the moisture content in fuel cores up to $2\cdot 10^{-4}$ mass % is necessary for reliable operation of fuel elements.

In order to prevent the harmful action of moisture and hydrogen on the fuel-element shells, an investigation of the possibility of extensive use of getters placed together with the fuel for preventive absorption of all the evolving active gaseous mixtures merits attention.

Translated from Atomnaya Énergiya, Vol. 43, No. 5, pp. 408-412, November, 1977.

0038-531X/77/4305-1059$07.50

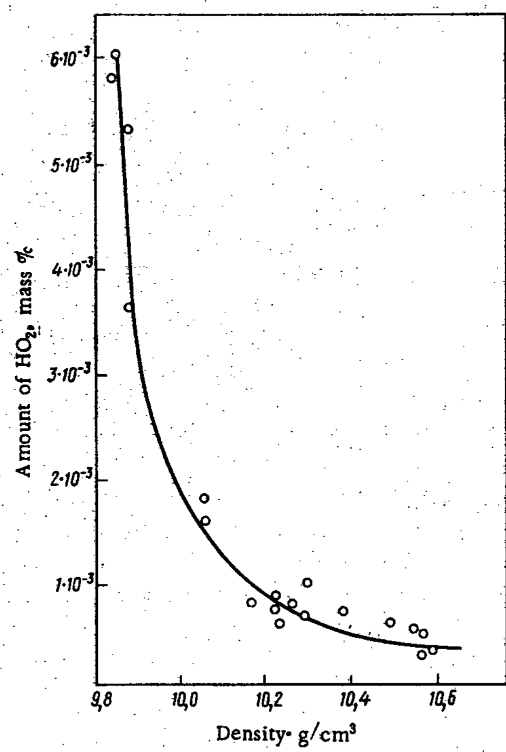

Fig. 1. Moisture content in uranium dioxide tablets as a function of their density.

TABLE 1. Carbide Uranium Cores Obtained by Different Technological Schemes

Characteristic	A*	B	C	D
Chemical comp., mass %				
C	4,88	4,78	4,90	4,83±0,07
O	0,1	0,18	0,2	0,08±0,02
N	0,04—0,05	0,01	0,01	⩽0,01
Stoichiometric comp. (C+N+O)/U, atom	1,028	1,020	1,05	1,01±0,015
Grain dimensions, mm	15—50	50—210	—	
Core dimension, μ	3—5	10—30	3—15	
Porosity, vol. %	4,8	7,3	4,2	
Swelling of the fuel(%) by depletion %	1,6	1,4	1,3	
Gas evolution, %	4,7	5,0	6,5	

*A) gaseous carbonization+pressing+baking; B) carbothermal restoration of dioxide+pressing+baking; C) carbothermal restoration of dioxide+hot pressing; D) same as in B in cleaner chambers.

The requirements on the hardness of the tablets have not still been adequately formulated. They must be directed towards obtaining tablets which would not give fissures and would not split during loading of the fuel elements, transportation, etc. So far only the hardness of baked tablets during compression is regulated.

One of the important criteria in estimating the quality of fuel-element cores is their fluorine content. Sometimes the fluorine requirements are formulated without considering the required connection with the content of other impurities, in particular, moisture. Dry fluorine in a quantity in which it can be present in the uranium dioxide cannot have adverse effects on the zirconium shell of the fuel element. However, the presence of the moisture in the core, which is evolved during heating, turns fluorine into a strong corrosion reagent. The technological processes of making fuel-element cores that are used in different countries, differ from each other in the characteristics of accomplishing different operations. For example, the methods of preparing the uranium dioxide powder for pressing differ depending on the method of production of the dioxide. There are two methods of polishing the tablets. Dry polishing has an advantage, since it eliminates the subsequent operation of drying of the tablets and formation of large amount of liquid wastes containing uranium.

TABLE 2. Amount of Uranium—Plutonium Carbide (by homogenity of composition) Depending on the Technology of Its Manufacture

Technology	Characteristics of Homogenity
Mixing of UO_2 + PuO_2 + soot in a rotating mill for two hours, carbidization at 1700-1800° C	Homogeneous, has oxide impurities
The same as above plus grounding in vibromill, baking at temperature of 1900° C	Homogeneous, 5% $(U, Pu)_2C_3$ phase
Coprecipitation of UO_2 + PuO_2, mixing in vibromill with soot, carbidization at 1700-1800° C	Homogeneous
Melting of U + Pu at 1300° C. Hydration. Carbidization by propane at 250-280°, pressing, baking at 1700-1900° C for 0.5-1.5 h	Homogeneous (a = 4.961 Å)
Production of PuC by interaction of Pu + soot, production of UC by gaseous carbidization, mixing, baking at 1650° and 1900° C for 1 h	Nonhomogeneous Has phases based on UC and PuC
The same plus homogenizing annealing for 15 h. Melting of U+Pu, hydration, mixing with soot, carbidization at 1300° C	Homogeneous "

Wet polishing insures a more smooth regime of operation of the tablets, a better purity of the surface, and a smaller amount of slips and other defects.

Using diverse technological schemes it is possible to obtain fuel element cores with specified characteristics from different original uranium dioxide powders.

Cores from Carbonide and Nitride Fuel. A large number of reactor tests of carbide fuel have already been carried out for fast reactors with sodium heat carriers. The results of tests of many experimental fuel elements and primarily of the test in BOR-60 reactors permit certain conclusions regarding the requirements on these materials.

The reliability of operation of fuel elements with carbide—nitride fuel for sufficient depletion depth are mainly governed by two factors: the intensity of chemical interaction of the fuel with the shell and the limiting deformation of the steel casing under the action of swelling fuel.

The colocation of the carbide cores with the shell depends on such characteristics of the fuel as stoichiometric composition and the content of oxygen impurities. The deformation of the shell is determined by the magnitude of swelling of fuel during combustion, which, in turn depends on its initial porosity and hardness, and by the dimension of the initial gap between the fuel and the shell.

Cores of mixed uranium—plutonium carbide fuel can be obtained by different variants of technological processes of powder metallurgy. Two methods of obtaining original powders are being investigated: the restoration of oxides by carbon and gaseous carbidization of dehydrated metal. The technological regimes of these processes permit the final characteristics of their cores to vary within wide limits. Each of the two variants has its own advantages and disadvantages. Thus, the oxides are relatively cheap raw material, while the use of metals permits apparently to simplify the equipment design of the technological process and reduce the losses of the fission material. In the latter case [1] a continuous technological process is possible with automatically controlled regime of operation of the equipment.

In order to study the effect of the composition and the technology of preparation of uranium carbide cores on their behavior during irradiation in BOR-60 reactors experimental fuels were tested with cores whose characteristics are given in Table 1.

The data obtained from these tests show that in conducting all the technological operations in chambers with dry inert atmosphere the carbothermic method enables one to obtain cores of high quality with the following characteristics: content C 4.83 ±0.07%, O 0.08 ±0.02%; N ≤ 0.01%; (C + N + O)/U = 1.010 ±0.015 (± is the rms deviation in a series of batches).

The conditions of irradiation of the test elements were: linear power, 550-700 W/cm; temperature of the shell, 685°C; depeletion up to 10.4% of heavy atoms. The cores had the shape of tablets and sleeves. The presence of an axial hole did not have any positive effect.

All the tests of the fuel element with helium sublayer retained the hermiticity up to 10.4% depletion. The investigations did not reveal any significant differences among the fuel elements with cores of different origins.

Cores coated with chromium, niobium, and other metals significantly reduced the degree of carbonization of the shell across the sodium sublayer. However, the economical feasibility of using such coatings is still not verified.

The requirements on the density of carbide—nitride cores follow from the conditions of producing maximum possible concentration of the fission elements in the active zone and insuring the necessary amount of free volume in the fuel elements for the compensation of swelling of the fuel.

An important requirement for mixed uranium—plutonium fuel is the uniformity of mixing of uranium and plutonium. Different variants of technology of obtaining cores from mixed uranium—plutonium carbide fuel can yield products of different degree of homogenity (Table 2). Until recently, it was thought that in mixed fuel the uranium and plutonium must occur in the form of solid solution of the corresponding compounds: so rigid were the requirements as to the uniformity of mixing of uranium and plutonium. This complicated the technological process of obtaining fuel by introducing the operations of coprecipitation, high-temperature, homogenizing baking, etc. But recently successful comparative experiments have been carried out in the USSR, U.S.A., and Federal Republic of Germany in which mechanical mixing of the compounds of uranium—plutonium were used side by side with coprecipitation [2, 3]. In both cases the fuel elements showed equally good efficiency. The obtained results may lead to a simplification and cost reduction of the technology of obtaining cores from mixed uranium—plutonium fuel and investigations in this direction are continuing.

Granulated Ceramic Nuclear Fuel. Its prospects are mainly related to dispersed fuel elements, e.g., for high-temperature reactors with gaseous heat carriers. The use of vibrationally compacted fuel elements in fast reactors depends to a large extent on the development of economically feasible industrial methods of producing granulates and on the results of the studies of operational characteristics of vibrationally compacted fuel elements.

Microspherical oxide, carbide, and nitride fuel can be produced by several methods. In recent years considerable success has been achieved in the development of the technology and equipment for obtaining fuel by the methods of sole-gel processes and cryochemistry. The interest in these processes has developed due to the prospects of producing remote, high-output technological schemes and the possibilities of obtaining fuel with a given composition and characteristics with high direct yield of product and minimum irreversible losses.

For the production of uranium oxide fuel the improved method of internal heleoformation is used. It has been established from physiochemical investigations that the hydrolysis of uranyl is mainly due to successive flow of the reactions of gradual protonization of hexamethyl-entethramine and not its hydrolysis with the formation of ammonia and formaldehyde as proposed earlier.

The tests of industrial equipment of continuous operation, designed on the method of internal heleoformation, insure an output of 2.0-0.5 kg/h.

The production of microspherical nuclear fuel of given granulometric composition is ensured by the construction of a unit for dispersing feed solutions. For obtaining coarse and moderate size fractions of microspheres (1000-200 μ) with the mean deviation from the normal size not more than 5-10%, vibrational feeders are used.

The production of fine fractions of microspheres ($< 100\ \mu$) present considerable difficulties not only in obtaining monodispersed drops but also in the constructional design of the continuous process.

The spherical ceramic fuel with a single inner aperture of controlled size is also of certain interest. Further development of the technology and the investigation of this type of fuel would permit determining the prospects of its utilization.

The technology of coating of granulated fuel by pyrocarbon of controlled density, silicon carbide and other materials using boiling layer equipment has been developed and perfected. The types of coating, the number and thickness of the layers, injection of getters into the fuel for reducing the diffusion of the fuel and the fission products are also being investigated. Decisions on these investigations will be taken after the completion of the reactor tests.

The sol-gel process is also applicable for preparing granulated mixed uranium—plutonium fuel. The required density of microspheres is insured by the change of the temperature regime of baking.

The preparation of uranium and uranium—plutonium oxide fuel with the density of granules close to the theoretical is accomplished with the use of the modified method of gel-preserving sedimentation and the cryochemical process. The cryochemical methods are promising for the production of granulated mixtures and for preparing uranium—plutonium fuel in tabulated form.

The production of vibrationally compacted fuel elements from granulated carbide and nitride fuel is also of interest. The technology of obtaining granulated carbide—nitride fuel with the required density and homogenity of distribution of uranium and plutonium is still in the development stage. The preliminary data enable us to hope that the sol-gel processes and the cryochemical method may ensure the production of uranium—plutonium carbide nitride fuel with characteristics necessary for the production of fuel elements for fast power breeder reactors.

The above information shows the extensive possibilities of using different technological factors for controlling the quality of obtained oxide and carbide nitride cores of fuel elements. As the experiment data accumulates, the requirements imposed on the nuclear fuel will be refined, which could certainly lead to an increase in its efficiency and reliability of fuel elements. The improvement and increased stringency of the requirements on nuclear fuel is a natural process. However, these requirements must be sufficiently justified so that the cost of the fuel elements as a whole does not increase unjustifiably and the economy of operation of atomic power plants is ensured.

The total projected power of atomic power plants in the entire world in the near future exceeds 100 MW (electrical) [4]. These are mainly installations with thermal reactors with fuel elements made of uranium oxide in zirconium casings. Experience in the operation with these fuels will facilitate subsequent transition to the mixed uranium—plutonium oxide fuel in thermal and fast reactors; a continuation of the reactor investigations of carbide—nitride fuel will permit a more complete understanding of its potentials and will accelerate widespread use of this promising fuel in fast reactors.

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PLUTONIUM FUEL AND FUEL ELEMENTS FOR POWER REACTORS

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A comparison of the growing demand of energy with the guaranteed availability of uranium ores, even considering opening of new natural resources and extraction of uranium from seawater, shows that the problem of nuclear fuel can be solved by introducing fast reactors along with the relatively cheap thermal reactors, which as a result of extensive breeding of secondary nuclear fuel would permit using practically all obtainable uranium [1]. Thus, in the complex fuel cycle of future nuclear energetics, one must provide for a widespread use of uranium—plutonium fuel both in fast and thermal nuclear reactors. One of the main requirements on atomic power plants with fast reactors and on radiochemical production is to ensure a rate of breeding of plutonium that would permit doubling the rated power output in not less than 10 years.

In nuclear energetics the use of plutonium fuel is currently being investigated along with the economic aspects of the problem in water and water—graphite reactors. In fast reactors uranium—plutonium fuel is already being used, since the use of ^{235}U in them is not promising because of the low rate of breeding. The knowledge of the physicomechanical and the technological characteristics of uranium and plutonium oxides, their chemical and radiation stability, and also the relative simplicity of technological methods of producing uranium—plutonium fuel elements led to immediate development of just this type of fuel.

Translated from Atomnaya Énergiya, Vol. 43, No. 5, pp. 412-417, November, 1977.

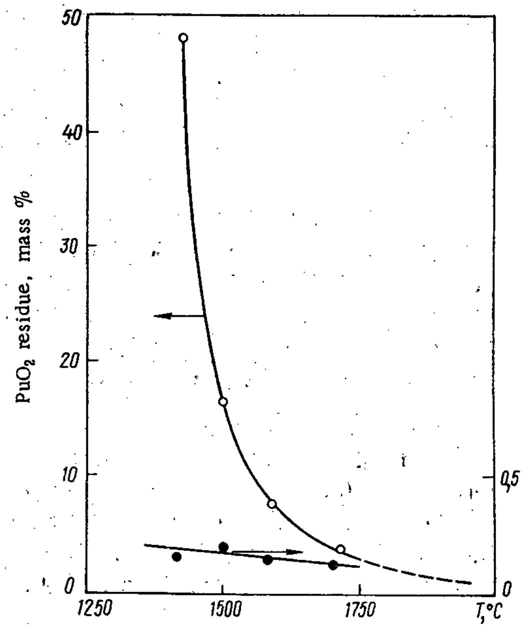

Fig. 1

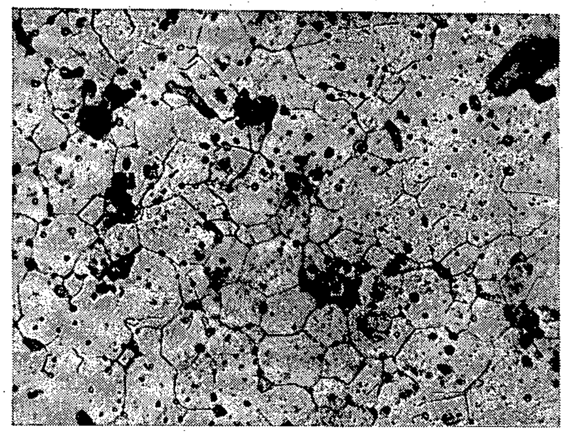

Fig. 2

Fig. 1. Dependence of completeness of dissolution of brickets of mechanically mixed powders of UO_2 and PuO_2 (○) and of chemically coprecipitated powders $(U, Pu)O_2$ (●) in nitric acid, on their baking temperature.

Fig. 2. Microstructure of core of $(U, Pu)O_2$ obtained by baking of mechanically mixed powders at a temperature of 1750°C for 3 h (× 340).

Investigations have shown that in the future the widespread use of mixed oxide fuel in thermal water-cooled reactors can be foreseen. But in fast reactors the use of such fuel is apparently limited to the first and second generations of power installations. Oxide fuel with relatively high breeding coefficients (~ 1.4) cannot ensure (considering the entire fuel cycle) the projected growth of power due to the low rate of accumulation of plutonium or large technical difficulties in attaining required rate of accumulation. However, the initial industrial use of this type of fuel in fast reactors has enormous significance for establishing large-scale production of mixed oxide fuel for thermal reactors.

Although the above question is subject to further study, the problem of developing more efficient forms of fuels is undoubtedly a pressing one; this primarily refers to the mixed uranium—plutonium carbide and nitride fuel whose physical characteristics determining the rate of breeding of secondary fuel are better than those of oxide fuel. However, the technological characteristics of these materials are worse and the operational characteristics of the fuel elements made from them have not been adequately investigated.

Characteristics of Fuel Elements of Power Reactors with $(U, Pu)O_2$ Fuel. In the development of fuel elements made of uranium—plutonium oxide fuel, a number of characteristics related to the use of this composition must be taken into consideration; among these are:

1. The decrease of melting temperature of the mixed fuel compared to uranium. This has certain significance only for fast reactors. For the fuel of thermal reactors containing 3-6% PuO_2, the decrease of melting temperature of the core can be neglected. Here the higher self-screening of the neutron field, caused by the presence of plutonium, not only considerably lowers the gas release but also permits either increasing the melting margin or increasing the thermal loading on the fuel element by 10-15%.

2. The increase of the oxygen potential in the fuel element with the increase of depletion. A consequence of this is the increased physicochemical interaction of the fission products with the fuel element shells. The solution of this problem is connected with the determination of the optimum initial oxygen coefficient of the fuel core and with the development of a technological process which would permit preparing fuel element cores with given oxygen coefficient.

3. The uniformity of distribution of plutonium in the fuel core. This condition is one of the important safety criteria for fast reactors. Physical investigations have shown that in the mixed fuel the admissible size

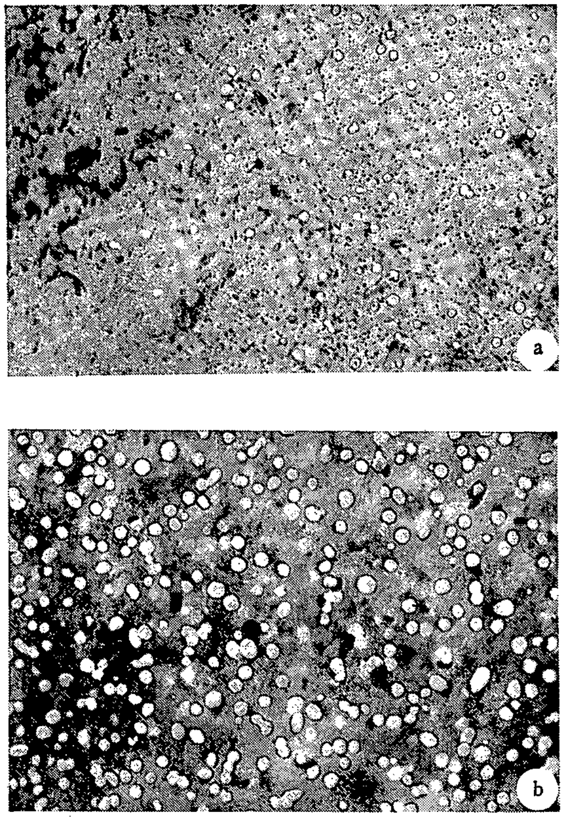

Fig. 3

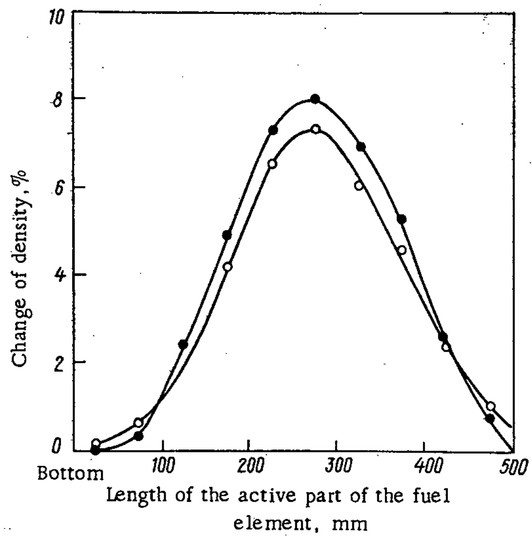

Fig. 4

Fig. 3. Vacancy pores and segregation of Nb(C, N) in the fuel-element shell made of 0Kh16N-15M3B steel for E > 0.1 MeV and $T_{irr} \approx 450$ (a) and $T_{irr} \approx 500$ (b) °C (× 200,000).

Fig. 4. Change of density of shell along the length of the fuel element: ○) experiment; ●) computation.

of PuO_2 particles must not exceed 50 μ. Therefore, on the one hand, the technology of preparation of fuel cores must ensure the required homogenity and, on the other hand, it must be considered that during the operation segregation of plutonium occurs in the fuel core due to high temperature gradients.

4. The transition to plutonium fuel in thermal reactors demands optimization of individual parameters, primarily the depth of depletion and thermal stresses of the corresponding construction of the fuel elements. In using plutonium these parameters must increase, since the cost of preparation of the fuel elements increases sharply.

5. In using plutonium fuel in power reactors, it is required to reprocess a very large amount of highly toxic nuclear fuel. This is especially true for thermal reactors (amount of reprocessed fuel in these is ≈ 10 times larger than in fast power reactors of the same capacity).

Methods of Obtaining Cores of $(U, Pu)O_2$. The cores of fuel elements for fast reactors can have the form of tablets, sleeves, or vibrationally compacted powder with the content of PuO_2 in the mixture ranging from 10 to 30%. At present the first two types of oxide cores obtained by the methods of powder metallurgy are widely used. As the initial powder, powders prepared by mechanical mixing of UO_2 and PuO_2 powders (the formation of homogeneous solid solution of $(U, Pu)O_2$ occurs during baking at a high temperature) and by chemical coprecipitation of nitric solution of uranium and plutonium, when the original powders are a solid solution of $(U, Pu)O_2$, are used.

The greatest difficulties are encountered in using mechanically mixed powders. In order to meet the specified requirements on the density and uniformity of distribution of plutonium, a preliminary treatment of the powders and a sufficiently high baking temperature of the tablets are required.

The required density for the cores of fast reactors (10.2–10.6 g/cm³) is attained by baking at 1400°C; however, this temperature is not sufficient for obtaining a homogeneous structure. The incomplete formation of the solid solution at this temperature is indicated by the phenomenon of residue of PuO_2 during dissolving of the core in nitric acid. The increase of the baking temperature to 1720°C reduces the amount of undissolved residue to 3% (Fig. 1). On using chemically coprecipitated powders, the undissolved residue does not exceed 0.2%. For cores of $(U, Pu)O_2$ made from powders obtained by the method of chemical coprecipitation, it is considerably simple to obtain a homogeneous structure of the core. A preliminary treatment of the powders is not required; the baking temperature in argon medium with 7% hydrogen does not exceed 1500°C. In the case of the use of mechanical mixtures of UO_2 and PuO_2, the homogeneous structure of cores is obtained with suitable preparation of the powders, in particular, with careful mixing (Fig. 2).

Materials of Shells for Fuel Elements with Uranium—Plutonium Fuel. The material of the fuel-element shell of test-industrial reactors 0Kh16N153B steel in austenized state has been developed, which represents a γ-solid solution hardened by molybdenum and niobium, besides carbon. The light elements in the steel have been balanced in such a way that a complex hardening is achieved and the incidence of undesirable phases (α, σ, χ) is reduced to a minimum. For this purpose the content of nickel was increased to 15%, the content of chromium was limited to 16%, the molybdenum (3%) and niobium were introduced into the steel. Depending on the temperature of austenization, its structure contains different amounts of niobium carbonitrides. Residual phases are evolved out with prolonged aging of the steel.

During the inspection of irradiated steel in transmission, electron-microscope dislocation loops of injected atoms are detected in its structure besides the holes dotted by a large number of fine extrusions of Nb(C, N) of 30 Å. It is assumed that these niobium carbonitrides block the growth and displacement of the loops; they cease to be sinks for the injected atoms, which facilitates the interaction of the latter with the holes and thus reduces swelling. The structure of 0Kh16N15M3B steel after irradiation in a BOR-60 reactor with a flux of $5 \cdot 10^{22}$ neutrons/cm^2 is shown in Fig. 3a. The swelling comprises 0.3% [2]. The 0Kh16N15M3B steel can ensure the efficiency of the fuel elements with mixed oxide fuel up to a depletion of more than 10% of the heavy atoms. However, after a certain time (at large irradiation doses) all carbon, nitrogen, and niobium get evolved out of γ-solid solution and Nb(C, N) particles begin to grow.

The structure of 0Kh16N15M3B steel after irradiation by neutrons up to a flux of $7 \cdot 10^{22}$ neutrons/cm^2, when the swelling comprises 6.5%, is shown in Fig. 3b. The large particles Nb (C, N) no longer retard swelling; furthermore, the presence of stresses around the large carbides activates sinks for the vacancies and, as shown by the electron microscope, pores form around the carbonitrides. This leads to an increase of the swelling. The tendency of the steel to high-temperature brittleness and interaction with the fission products of the field also increases. Therefore, a further improvement of the composition and the structural state of the steel was required for reactor steel; as a result, only a small radiation damage at the shells was obtained at a flux exceeding $1 \cdot 10^{23}$ neutrons/cm^2(E7 0.1 MeV).

The Investigation of Irradiated Fuel Elements with Mixed Oxide Fuel. The efficiency of fuel elements of fast reactors made from mixed fuel is currently being investigated by radiation tests in the BOR-60 reactor. In the irradiated fuel elements the density of the tablets was varied (10.2-10.6 g/cm^3, i.e., 87-96% of the theoretical); the initial gap (0.1-0.25 mm) and the temperature of irradiation (620-680°C) were also varied. The maximum linear load on the fuel element was 530 W/cm. The composition of the fuel was $U_{0.85}Pu_{0.15}O_2$. The oxygen coefficient was 1.97-1.98. Shells of 0Kh16N15M3B steel did not lose hermeticity. The fuel elements reached depletion of 11% of heavy atoms at a flux of $7 \cdot 10^{22}$ neutrons/cm^2 ($E > 0.1$ MeV). The maximum change of the diameter of the shell was 1.5-4.2%.

In order to estimate the contribution of radiational swelling of the steel and the inelastic deformation to the overall change of the diameter, the density of the samples cut from several segments of the shell was measured (Fig. 4). A comparison of the obtained results with the computed volume changes [from the ratio $\Delta V/V = 3(\Delta d/d)$] indicates that the inelastic deformation of the shell did not exceed 0.3% on the average, i.e., the contribution of the radiational creep caused by the pressure of the core to the swelling is relatively small. The gas release comprised 90%. The density of the fuel did not affect the gas release. In the upper part of the fuel element, where the temperature of the shell is the maximum, an interaction of the steel with the core is detected to a depth of 20-30 μ with local damages up to 70 μ. Corrosion interaction is caused by the effect of the fission products, especially cesium [3]. The shells still retained a certain margin of hardness and plasticity. The yield limit at 700°C is equal to 16 kgf/mm^2; the relative elongation is about 1%. These results indicate that the fuel elements can operate up to deeper depletion.

Fuel Cycle of Fast Reactors. The development of nuclear energetics with fast reactors substantially depends on the organization of production of fuel cycles, in particular, on the time required for processing of the reactive fuel and preparation of new fuel from plutonium obtained from breeding.

The amount of plutonium ensuring the operation of a single fast reactor can be determined in the following way:

$$G_f = G_0\left[1+\frac{1}{T_0}(0.3+T_h+T_{tr}+T_{rc}+T_{pr})\right], \tag{1}$$

where G_f is the total amount of plutonium in the fuel cycle of the reactor; G_0 is the plutonium loading in the active zone of the reactor; T_0 is the period of operation of the fuel in the reactor for a specified depletion; T_h, T_{tr}, T_{rc}, T_{pr} are, respectively, the time of holding, transportation, radiochemical treatment, and preparation of the fuel. The margin in the first reloading is 0.3

The efficiency of the fast reactor is usually characterized by the time of doubling of the power due to excess breeding of plutonium. The doubling time can be defined by the formula

$$T_2 = 0.693\frac{G_f}{(\mathrm{BC}-1)\,q}, \tag{2}$$

where BC is the breeding coefficient of plutonium in the reactor and q is the amount of plutonium burned in the reactor per year.

Equation (2) shows that the rate of introduction of atomic power plants with fast reactors due to plutonium breeding is inversely proportional to G_f; hence, it substantially depends on the time of the outer fuel cycle (the sum in square brackets in formula [1]).

Investigations have shown that the holding of the fuel for 1 year before processing does not cause serious technical difficulties for the process of cleaning of the fissionable isotopes of the fission products and does not lead to complex problems in transportation to the radiochemical plant. If it is assumed that G_f must not be larger than $2G_0$ then for $T_0 = 1.4$ y and $T_{tr} = 1$ month the time of radiochemical treatment of annual load of the reactor and also the time of preparation of the fuel elements for annual loading should not exceed 1-1.5 months. In this case, after substituting the appropriate values into Eq. (1) we get $G_f = (2.1\text{-}2.15)\,G_0$.

Thus, the capacity of the radiochemical plant for the preparation of fuel elements must ensure reprocessing of the fuel from 8-10 atomic power stations with fast reactors, i.e., the treatment of the fuel must be organized centrally at a large capacity plant.

Similar requirements are imposed also for a plant for preparing heat-release elements from plutonium.

The increase of the holding time of the fuel up to 2-3 years before chemical treatment, which is tempting from the point of view of simplifying transportation and reprocessing of the fuel, leads to an increase of G_f to $(2.8\text{-}3.5)\,G_0$. The rates of introduction of atomic power plants with fast reactors decreases by a factor of 1.3-1.6. The decrease of the time of holding of the fuel to 0.5 yr before chemical treatment lowers G_f to $1.8G_0$ and increases the introduced capacity of atomic power plants with fast reactors by 15-17% with appreciable complications of the transportation operation and the process of radiochemical treatment. For an atomic power plant with a reactor and electrical capacity of 1600 MW and plutonium load in the zone of 3 tons ($^{239}Pu + ^{241}Pu$) with BC = 1.3, q = 1.2 tons and $T_h = 1$ yr the time of doubling $T_2 = 0.693G_f/(\mathrm{BC}-1)q = 12$ years.

On shortening the time of holding to 0.5 yr the time of doubling decreases to 10 years, while on increasing the holding time to 3 years it increases to 20 years.

On the Use of Carbide and Nitride Uranium—Plutonium Fuel in Fast Reactors. One of the possible methods of improving the economic characteristics of power installations with fast reactors is to use such types of fuels as solid solutions of uranium and plutonium carbides and nitrides which have better thermophysical characteristics compared to the oxides. But the production of cores is somewhat more complex than in the case of oxides, since regulated highly clean inert atmosphere and high baking temperature of the cores (up to 2000°C) are required [4]. The behavior of carbide—nitride fuel during irradiation differs from that of the oxide fuel: the structural changes occur more slowly and mainly in the central "hot" part of the samples; the gas release is appreciably smaller (~20%), cracking of the core is less pronounced, and the swelling increases. The mean volumetric swelling of the core made of (U, Pu)C comprises 2.3% at 1% depletion at a temperature of 1200°C at the center and* at depletion up to 7% of heavy atoms. An appreciable accumulation of gaseous fission products in carbide fuel leads to an increase of the core also in height up to 3% (for UC) for a linear power of 535 W/cm and depletion of 10% of the heavy atoms. The behavior of the fuel-element shells with cores of carbide and oxide fuels is on the whole similar. However, the contribution of the mechanical

*Data missing from Russian Original – Publisher.

ıction of the core with the shell to the diametral deformation is noticeably larger compared to neutron ing (up to 1.7%).

A fast increase of swelling at a temperature higher than 1300°C restricts the specific energy release. An increase of the power output is possible if the fuel element is filled by sodium. However, in this case the process of carbidization and softening of the steel shell gets accelerated and uncertainties related to boiling and possible leakage of sodium appear. Computational investigations have shown [5] that fuel elements with carbide fuel can hold loads up to 750 W/cm and ensure energy release up to 300 W/g. Investigations of nitride fuel have been carried out to a lesser extent and so far the amount of experimental data on the efficiency of such fuel elements is inadequate.

CONCLUSIONS

The use of plutonium in the fuel cycle during complex utilization of thermal and fast reactors in nuclear energetics permits solving the problem of ensuring nuclear fuel for a long period. Oxide uranium–plutonium fuel facilitates the development of technology of fast reactors and so far it is considered as the basic type of fuel. At the same time, oxide fuel cannot ensure the required rate of plutonium accumulation, in view of which the investigations of more efficient fuel and constructional materials become a pressing problem. The use of uranium–plutonium oxide fuel in thermal reactors requires improvements in the construction of fuel elements and organization of large-scale completely automatic production.

Editors' Remarks. For the completeness of the discussion of the problem it is, of course, necessary to consider the possibility of using plutonium in fast and thermal reactors as done by the authors. However, it should be kept in mind that by its nuclear–physical parameters plutonium as a nuclear fuel is more suitable for use in fast reactors than in thermal reactors. The use of plutonium in thermal reactors can reduce the demands of natural uranium for the development of nuclear power in all by 10-15%, whereas its use in fast reactors reduces the demand for uranium by a factor of 10.

All this indicates the feasibility of using plutonium only in fast reactors even if its accumulation is required over a certain period.

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NUCLEAR POWER STATION CONSTRUCTION IN THE USSR

L. V. Timofeev

Nuclear power is advancing in the USSR more rapidly than power engineering as a whole; nuclear power stations are being built mainly in the European part of the USSR, where fossil fuel is scarce, with a corresponding reduction in the scale of conventional power-station construction. During the current five year plan, it is intended that nuclear power stations should contribute about 35% to the overall increase in power generation in the European part of the country. Forward plans envisage a further increase in the rate of building of nuclear power stations in this area of the USSR. The long-term plans for nuclear power station construction envisage the building of a relatively small number of stations of output 3-4 GW, which are to be built in the main from units of output 1 GW each (see Table 1 and [1, 2]).

Nuclear power in the USSR is based on two types of thermal reactor: water-cooled—water-moderated (pressurized-water) reactors of VVÉR type and uranium—graphite ones of RVMK type. A characteristic feature of progress in both types is the increase in unit power. In addition, there is a program for building and commissioning fast breeder reactors.

Pressurized Water Reactors. The first reactor of this type, the VVÉR-210, was installed in the first unit at the Novovoronezh Nuclear Power Station and commissioned in 1964. In subsequent years, the VVÉR-365 (1969) was commissioned at the same power station, which has a higher output, followed by VVÉR-440 units (1971 and 1972).

The VVÉR-440 has been adopted for standard nuclear power stations of medium output, and it is economically competitive with fossil-fuel power stations virtually everywhere in the European part of the USSR. In 1976, the cost of power at the output from the Novovoronezh station was 0.635 kopecks per kWh, while the load utilization factor was 76.3%, the corresponding figures for major fossil-fuel power stations being as follows: Krivoi Rog No. 2, 3000 MW, operating on coal, 0.895 kopecks per kWh, and Konakovo, output 2400 MW (heavy fuel oil and gas), 0.712 kopecks per kWh.

VVÉR-440 reactors have now been installed in the first and second units of the Kola Nuclear Power Station (1973 and 1974), and also in the first unit of the Armenian Nuclear Power Station (1976), and they are being built in the third and fourth units at the Kola power station, the second unit at the Armenian one, and in two units at Rovno Power Station.

During the current five-year-plan, further such units will be built and commissioned, which will be followed by transfer to the VVÉR-1000; a start was made on building the first such reactor in 1973 in the fifth unit at Novovoronezh. In the future, VVÉR-1000 units will be installed at the Kalinin Power Station, the Western and Southern Ukrainian Power Stations, and so on. While the unit power of the VVÉR series is being increased, there will also be an enlargement of the power levels of major nuclear power-station units generally, and a corresponding improvement in the efficiency.

There is also a VVÉR-500 design, which is maximally standardized; it is proposed that the VVÉR will be used in especial cases where the power-system conditions make it impossible to install VVÉR-1000 units.

As it is intended that extensive use be made of the VVÉR-1000 type in future power stations, the commissioning of the fifth unit at Novovoronezh is of particular significance, since it will contain the head reactor of this series. The pressurized shield has already been built at the construction site, and major equipment is now being installed. Work has also been done on the main vessels for the first units at the South Ukrainian and Kalinin power stations.

At the present time, the VVÉR series are the most extensively used, partly because they are compact, of reasonably good output per unit volume, require relatively small amounts of constructional materials in the core, and are reliable in operation [2-5].

Translated from Atomnaya Énergiya, Vol. 43, No. 5, pp. 418-420, November, 1977.

TABLE 1. USSR Nuclear Power Stations: Operating, Under Construction, and Planned

Site	Power, MW (electrical)	Year of commissioning	State
Pressurized water			
Novovoronezh			
Unit 1st	210	1964	Operating
2nd	365	1969	»
3rd	440	1971	»
4th	440	1972	»
5th	1000	1978	Under construction
Kola			
Unit 1st	440	1973	Operating
2nd	440	1974	»
3rd	440	before 1980	Under const.
4th	440	»	"
Armenian			
Unit 1st	405*	1976	Operating
2nd	405*	before 1980	Under construction
Kalinin			
Unit 1st	1000	1980's	"
2nd	1000	»	Planned
3rd	1000	»	"
4th	1000	»	"
South Ukrainian			
Unit 1st	1000	before 1980	Under construction
2nd	1000	1980's	Planned
3rd	1000	»	"
4th	1000	»	"
Rovno			
Unit 1st	440	before 1980	Under construction
2nd	440	»	»
West Ukrainian			
Unit 1st	1000	1980's	Planned
2nd	1000	»	"
3rd	1000	»	"
4th	1000	»	"
Ulyanovsk (VK-50 boiling-water)	50	1965	Operating

Site	Power, MW (electrical)	Year of commissioning	State
With uranium—graphite reactors			
Obninsk	5	1954	Operating
Siberian	600	1958	»
Beloyarsk			
Unit 1st	100	1964	»
2nd	200	1967	»
Leningrad			
Unit 1st	1000	1973	»
2nd	1000	1975	»
3rd	1000	before 1980	Under const.
4th	1000	»	
Bilibinsk			
Unit 1st	12	1973	Operating
2nd	12	1974	»
3rd	12	1975	»
4th	12	1976	»
Kursk			
Unit 1st	1000	1976	»
2nd	1000	before 1980	Under const.
3rd	1000	»	»
4th	1000	1980's	Planned
Chernobyl			
Unit 1st	1000	1977	Operating
2nd	1000	before 1980	Under construction
3rd	1000	after 1980	Planned
4th	1000	»	"
Smolensk			
Unit 1st	1000	1979	Under construction
2nd	1000	1980's	Planned
Ignalino			
Unit 1st	1500	after 1980	Under construction
2nd	1500	»	»
With fast reactors			
Ulyanovsk	12	1969	Operating
Shevchenko	150; 120,000 m^2/day of distillate	1973	"
Beloyarsk Unit 3rd	600	before 1980	Under construction

*On account of cooling conditions, the VVÉR-440 produces 405 MW (electricity).

Uranium—Graphite Channel Reactors. Experience acquired with the first nuclear power station and the Siberian Nuclear Power Station has been incorporated in the Beloyarsk Power Station, which has shown that a full-scale power station with one-loop coolant flow and nuclear steam superheating can be operated safely with the turbine receiving slightly radioactive steam, and also that the fuel pins can operate reliably at 500–540°C, 90–130 kgf/cm^2, heat fluxes up to 10^6 kcal/$m^3 \cdot$h, and uranium burnup in excess of 30 MW·day/kg. As a result, it has been decided to increase the burnup in a major group of steam-superheating channels in the second unit to 1200–1300 MW·day/channel (37–40 MW·day/kg) and to raise the maximum steam temperature at the outlets from certain channels to 560–565°C.

The cost of electrical power from the second Beloyarsk unit (0.92–0.93 kopecks per kWh) is equal to the mean cost of electrical energy from conventional power stations in the Ural with the same installed power.

In 1973–1976, four units of output 12 MW (electrical) were installed and operated under the conditions of the Far North, each with natural heat-carrier circulation.

The next stage in development of uranium—graphite channel reactors was the design of the standard commercial RBMK-1000, which has operated successfully at Leningrad Power Station.

In 1976 the first of four units was commissioned at Kursk power station, which is built with RBMK-1000 units, while in 1977 the first unit was commissioned at Chernobyl Power Station, and at the same time the third and fourth units were under construction at Leningrad Power Station, the second units at Kursk and Chernobyl, and the fourth unit at Smolensk. Tests at Leningrad have shown that the RBMK-1000 is capable of greater output, which has been realized in the RBMK-1500 design, which has been adopted for the first two units at Ignalino Power Station.

Further progress in uranium—graphite channel reactors is envisaged in the RBMK block-sectional reactor of output 2000-2400 MW (electrical), in which nuclear steam superheating is combined with the use of heat-resisting zirconium alloys having low neutron absorption.

Uranium—graphite channel reactors also make it possible to realize very high unit powers, with improved thermodynamic parameters in the carrier resulting from nuclear superheating (which raises the efficiency), in addition to continuous fuel exchange without shutdown, and also good flexibility during operation [6-8].

Fast Reactors. The ultimate purpose of the fast-reactor program in the USSR is to produce reliable and competitive fast power reactors capable in the future of combining with thermal reactors to make the best use of nuclear raw materials, and also to meet future demands for nuclear power throughout the country.

The scientific and engineering bases are now available for industrial adoption of economically sound fast power reactors with sodium cooling [9].

Experience with operating the BR-5 and BOR-60 has led to the design of prototype nuclear power stations with BN-350 and BN-600 fast reactors. The purpose of these systems is to evaluate major units and equipments for large fast power reactors, another object being research on the sodium coolant at full scale, in order to define optimum design parameters.

Tests on the BN-350, which was run up to power in July 1973, have shown that the major equipment has operated well, apart from the steam generators. Manufacturing faults in the latter were eliminated in 1973-1974. All major equipment, including the reactor, the central coolant circulator, the heat exchangers, the turbines, and the cooling towers have operated correctly since commissioning. Since March 1976, the reactor has been operated at a power of 650 MW (65%), with five loops out of the six operating, which has provided electrical power of 120 MW and 50,000 m^3/day of distillate [10]. During 1976, there was no instance of emergency shutdown.

The installation of the BN-600 equipment is continuing, which can be considered as the standard power reactor for nuclear stations. At present the structures within the pressure vessels are being installed.

The USSR is also undertaking design studies on the BN-1600 fast power reactor of output 1600 MW (electrical), which is intended to be highly reliable and to provide acceptable economic performance of a power station as a whole.

Nuclear power is developing in several directions in the USSR, which means that power-station construction is accompanied by the building-up of design and installation organizations, development of associated industries supplying equipment to nuclear power stations, and extension of fuel fabrication facilities. Here particular importance attaches to the construction of the Atommash plant, which will supply the growing nuclear power industry in the country. At the same time, cooperation with COMECON member countries is being extended in the construction of nuclear power station plants.

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ATOMMASH CONSTRUCTION

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The first draft plan for Atommash, the Volgodonsk plant for nuclear power engineering, was drawn up by the State Institute on the design of heavy engineering plants in order to provide equipment for nuclear power stations. Atommash will produce the larger items of reactor equipment and other associated nuclear-power station units. Designs have been drawn up for the workshops and general facilities, and general lines have been laid down for the main buildings, which are intended to handle equipment of appropriate unit size.

MAIN PRODUCTION UNITS

Section No. 1

This is the first complex of unique specialized shops, and also the largest such complex in the country. It will include a preparatory press shop, a heavy reactor-vessel manufacturing shop, mechanical treatment shops, a heat-treatment section, and so on.

The total area of this section when fully constructed will be 277,000 m^2.

The organizational structure of the section is that of a continuous production line, but with allowance for the design features of the type of component to be manufactured and the very strict specifications for guaranteed quality and reliability.

The production of such equipment on a single site and on a single production line, which combines a very large number of repeated operations, will include testing materials on arrival, and also checking out brought-in components, and will follow through to acceptance tests and despatch packing of finished units, which should minimize the volume of transport of large components.

The following are the major technical lines of such production.

Preparatory Pressing Processes. Blanks for reactor and heat-exchanger units will be produced on the most modern highly productive equipment. The unique equipment at this plant will include a hydraulic sheet press of 15,000 tons for hot pressing end sections of vessels, mounting sections for steam pipes, flanges for the upper parts of vessels, and other components of wall thickness over 250 mm. Thinner housings of wall thickness of up to 250 mm will be prepared with sheet-bending mills handling material of 250 × 5000 mm, in which the sheets will be worked in the hot state.

Pressing will be followed by subsequent correction and rolling, and ultimately finish gauging, which will provide high-precision components, particularly those that will be required by the developing power industry in the next 15-20 years.

The technology will be such as to improve labor productivity by a factor of 2-2.5 while reducing the consumption of metal in preparatory operation by 30-40%.

Assembly and Welding. It is envisaged that extensive use will be made of the most advanced techniques in welding and joining components.

These operations will be extensively mechanized and automated, including the use of automatic welding machines with numerical control and electron-optical automatic control systems, together with mechanized rolling mills, edging mills, and handling facilities of load capacity up to 1000 tons, which work with overhead conveyor systems, and so on. The scale of the equipment and the type of production technique used in this plant will mean that units of mass up to 1000 tons can be manufactured under factory conditions for subsequent water transport.

The long sheets used in casings and vessels are to be welded by electroslag techniques, with the equipment fitted with optical automatic control. The ring joints in some units and vessels will be formed on automatic multilayer welding machines with flux protection and optical monitoring. This will improve labor

Translated from Atomnaya Énergiya, Vol. 43, No. 5, pp. 420-422, November, 1977.

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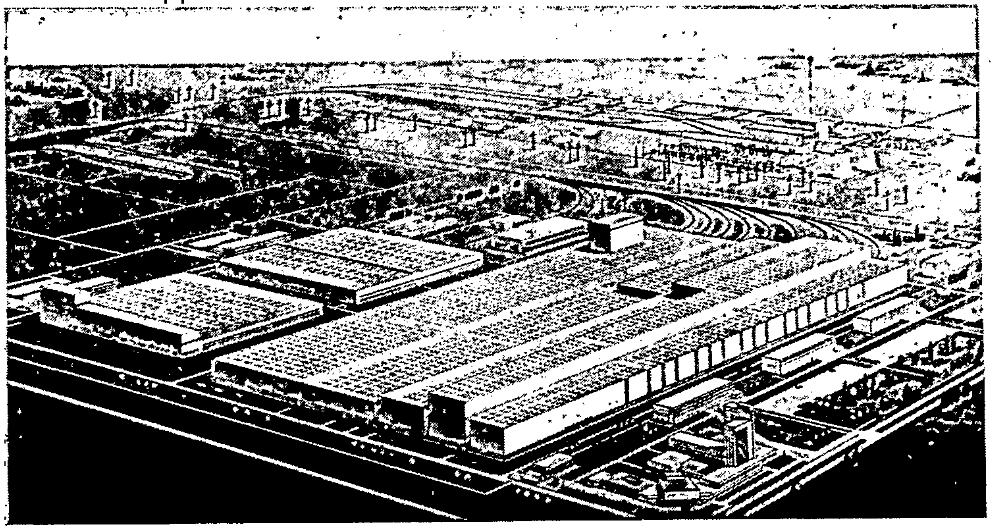

Fig. 1. Plan of the Atommash plant.

productivity by a factor of 2-4, while the welding reliability will be improved by a factor of 2-3 by comparison with existing processes. It is also envisaged that gas—electric automatic welding by the slot system will be extensively used, which improves labor productivity by a factor of 1.5, saves 20% of the welding material, and improves working conditions.

Internal anticorrosion lining of reactor vessels will be by automatic arc techniques with flux and strip electrodes of width up to 300 mm.

Heat Treatment. Good mechanical parameters in vessels and other components will be ensured by bulk heat treatment in large tunnel ovens of diameter 10,000 mm and depth 7000 mm, in which the temperature will be controlled to ±5°C, which will be used with quenching tanks and cooling chambers, which will not only provide accelerated cooling but also will prevent excessive escape of heat into the heat-treatment shop and thus provide good working conditions.

Residual stresses in welded joints will be relieved in unique recirculation gas ovens with working dimensions of 10,000 × 25,000 × 10,000 mm and temperature control to ±5°C. Local heat treatment after welding of closure joints in vessels and other components will be provided by induction systems, which will work at line frequency. Preliminary or incidental heating during welding and other such treatment will be provided by flexible and rigid inductors, as well as screened infrared radiators.

All the ovens and the heat-treatment systems will be fitted with automatic controls.

Mechanical Assembly. It is envisaged that the latest technological processes and high-output equipment will be used, including numerically controlled machines for handling very complex components without resetting.

Large and complicated housings of diameter up to 8000 mm and containing welded joints will be handled in a unique mechanical processing center, which can accommodate components of working height 8000 mm. Deep machining of large numbers of holes of diameter 18 mm in tube panels of thickness up to 1000 mm will be provided by numerically controlled three-spindle deep-drilling machines.

The collector systems in steam generators will be drilled with a special two-sided ten-spindle numerically controlled machine; the drilling and boring of holes in spherical covers will be handled by a special boring machine having a circular table of diameter 7000 mm and numerical control. A measuring machine will monitor all dimensions of components of diameter up to 7000 mm.

The plan also envisages that the internal and external surfaces of components for reactors and other equipment will be protected with special solutions between operations.

The painting sections forming part of the production line will be fitted with small systems for painting large components by parts. These will provide air suspension at appropriate points. This technique will also ensure that contaminants do not enter the surrounding space, and thus working conditions will be good. In addition, automatic fire-extinguishing systems will be fitted in the painting sections.

Section No. 1 will have maximum mechanization and automation of all processes, including transport operations, in order to maintain scheduled output. Various means of mechanization and transport are envisaged for the purpose: bridge electric cranes, power-assisted manipulators, conveyor belts, powered overhead transport lines, and mobile self-propelled transport.

Quality Control. Very rigid specifications are imposed on quality and reliability, so monitoring and test facilities will be installed within the section.

All raw metal, semifinished products, and assembled units received will be tested in a separate acceptance shop and in laboratories before being passed to production.

The latest methods and means will be employed for such testing: 100% ultrasonic acceptance testing of all metal arriving, and mechanized testing between operations, the latter with equipment fitted with a variety of flaw detectors, in addition to radiographic monitoring with high-power linear accelerators, which will provide for examining welded joints in very thick components, which will be combined with x-ray and γ-ray testing, luminescence testing, magnetic-powder examination, etc.

Finished components will pass for strength tests to special hydraulic test beds using high-pressure water or vacuum. Finished reactor vessels will in addition undergo general assembly testing.

The linear-accelerator chambers, the vacuum and hydraulic test beds, and the general assembly test system will provide guaranteed high quality in the final products, and together they will constitute unique facilities.

Transport of Finished Components. It is envisaged that lifting and transport operations will be fully mechanized with bridge cranes of capacity 1200 tons, while finished components of mass up to 1000 tons will be transported from the plant to the construction sites on special self-propelled caterpillar tractors and barges of combined marine and inland-waterway types. A special port facility will be constructed, which will be equipped with jib cranes of capacity 600 tons each.

The initial facilities of Section No. 1 were commissioned in November 1977.

Section No. 2

This section (the second section in the plant) is intended for especial-precision operations in the production of reactor control and protection systems. An area of 50,000 m^2 will contain machining, assembly, and other shops. Nuclear cleanliness in the bodies will be provided by wet cleaning of the incoming air, while precision in component production and working comfort will be provided by air conditioning and cleansing, the latter particularly intended to facilitate cleanliness.

Section No. 3

This section will supply the plant with special tools, mechanization facilities, and any nonstandard equipment, as well as providing for equipment repair. The area will be 30,000 m^2, which will include a toolmaking shop, a mechanical repair shop, an electrical repair shop, and a shop for manufacturing nonstandard equipment. The frame of the building has been constructed, and it was put into use in December 1976. A start has now been made there on the production of special equipment and facilities for Section No. 1.

EXPERIMENTAL AND DESIGN SUPPORT

A major condition for providing high quality and reliability in reactor equipment is large-scale support from theoretical, research, and trial studies. Therefore, the plant provides for high-level design and experimental support, including a research center fitted with research laboratories, design institutes, and technological institutes, as well as pilot-plant production facilities and various other supporting and service areas.

There is to be a laboratory building of area 4800 m^2 forming part of the central factory laboratory, together with laboratories for nondestructive test methods, and other facilities required in such a factory system: an administrative building, a computing center, a teaching center, polyclinics, and a thermal power station to supply the plant and a new section of Volgodonsk.

Declassified and Approved For Release 2013/03/22 : CIA-RDP10-02196R000700100001-6

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