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SOVIET ATOMIC ENERGY

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ARTICLES

TWENTY-FIVE YEARS OF NUCLEAR POWER

A. G. Meshkov

Twenty-five years ago, on June 27, 1954, the first nuclear power station was started up. Nuclear energy was being used for the first time for a peaceful purpose, the generation of electricity. That this occurred here in the Soviet Union was vivid evidence of the peaceful aspirations of our country and of the desire to make nuclear energy serve the welfare of mankind as a new, practically inexhaustible energy source. This noteworthy event was the beginning of the extensive development of nuclear power in the whole world. In the second half of the 1950s the first nuclear power stations in Great Britain, the U.S.A., and France were opened.

The power of the first nuclear power station was 5000 kW (electrical). The present installed power of nuclear power stations over the world has already exceeded 100 million kW and continues to grow at a rate considerably greater than the rate at which electrical power generation is growing as a whole. Despite several objective circumstances which have caused a reduction in the absolute value of nuclear generated power compared to the values projected several years ago, the share of nuclear power stations in the electrical generating power of the world should be about 45% by the end of the century [1]. The extensive development of nuclear power is due primarily to the limited supplies of organic fuel and their nonuniform distribution over the earth. This is especially true of liquid fuel, whose share in the energy balance of most countries has grown rapidly. According to estimates [2], a growth in the demand for oil at the rate established over the last 20 years will lead to the exhaustion of extractable reserves by the end of the century. The technically extractable reserves of coal are not much greater than those of oil [2, 3]. Despite the increased cost of equipment and prolonging of construction periods due to stricter safety requirements, nuclear power stations are successfully competing in most countries with organically fueled power stations, a competition which is, of course, aided by the rising cost of the organic fuels. In addition, the use of organic fuels may be limited in the future because of pollution of the atmosphere by sulfur dioxide and solid particles and the expected long term ecological effects of the accumulation of carbon dioxide in the atmosphere. Nuclear power stations substantially reduce the harmful effect on the environment. The vast potential reserves of energy included in nuclear fuel, the ecological aspects, and the possibility of locating nuclear power stations at the most convenient points without territorially linking them to the fuel supply bases make nuclear power the most realistic energy source in the future.

The Soviet Union is one of the countries that has assured reserves of organic fuels. However, the energy resources are not uniformly distributed over the country's territory: about 90% of the fuel and 80% of the hydroelectric resources are in the Asian part [4]. At the same time the bulk of the demand for electrical energy is in the European part of the USSR and this results in a need to transport large volumes of fuel there and to construct expensive, very long electrical transmission lines to carry electricity from the large hydroelectric and thermal power stations in Siberia. Technicoeconomic studies of the competing ways to cover the energy deficit in the European part of the country have demonstrated the economic practicality of building nuclear power stations there. It is in that part of the country that they are now being constructed. The directives of the Twenty-Fifth Congress of the Communist Party of the Soviet Union [5] provide for the "rapid development of nuclear power in the European part of the USSR" in the Tenth Five-Year Plan. These plans are successfully being brought to life. At the end of 1978 nuclear power plants with a total installed power of about 10,000 MW (electrical) were in operation. The advance of nuclear power in this country since the start up of the first nuclear power station has proceeded with the development of two types of thermal power reactors: channel, with graphite moderator (along the lines of the first nuclear power station), and vessel, with pressurized water (the VVÉR) [6]. The use of the two types of reactor provides a variety of experience and, because of the substantial design differences, enhances the prospects for industrial production.

Based on the experience gained in designing and using the first nuclear power station, as well as on the scientific research carried out on the cooling of the reactor channels by boiling water and steam, a channel reactor with nuclear superheating of steam has been built that has a power of 100 MW (electrical). This reactor was the first unit of the Beloyarsk nuclear power station. It was put into operation in 1964 and followed (in 1967) by the second unit with a power of 200 MW (electrical). Later on, a design for a powerful commercial power

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Deputy Chairman of the State Committee for the Use of Atomic Energy of the USSR. Translated from Atomnaya Énergiya, Vol. 46, No. 6, pp. 371-374, June, 1979.

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reactor was worked out, that for the RBMK-1000 channel reactor with a power of 1000 MW (electrical) [7] which uses new design features that permit better realization of the advantages of channel reactors. As opposed to the Beloyarsk nuclear power station reactors, a zirconium alloy (instead of stainless steel) is now used as the material for the channels and the fuel element cladding. In the RBMK heat transfer is by boiling water and separated dry steam is fed to two turbines with a power of 500 MW each. Reactors of this type are operating successfully at the Leningrad, Kursk, and Chernobylsk nuclear power stations and will be installed at other nuclear power stations. Operating experience at the Leningrad nuclear power station has demonstrated the feasibility of increasing the power of the RBMK-1000 reactor and constructing a higher power reactor based on it. The RBMK-1500 was created in this way. Further improvement of this type of reactor in raising the unit power, increasing the thermodynamic parameters, and standardizing the units has led to the design of the RBMKP-2400 with nuclear superheating of steam [8]. One feature of this reactor is unit construction which allows the power of a single reactor to be raised by adding standard structural components and equipment that are used in smaller reactors. From this standpoint channel reactors have significant advantages over the other widely used type of power reactor, the pressurized water vessel reactor. The choice of refuelling regimes is also simplified in the RBMK since it does not involve stopping the reactor. In addition, channel reactors have more material volume than the VVÉR and have a strongly branched piping network in the first loop.

Pressurized water vessel reactors are the most widespread type of power reactor in the world. They are very compact, have a fairly simple engineering arrangement which makes it possible to isolate the radioactive loop from the steam power portion, and have a high specific heat release rate. The first power reactor of this type (with a power of 210 MW (electrical)) was started up in the USSR in 1964 at the Novovoronezh nuclear power plant. This was followed by a second unit of power 365 MW (electrical) (in 1969) and a third and fourth with powers of 440 MW (electrical) each (in 1971 and 1972). A fifth unit is being installed which will employ a prototype VVÉR-1000. Standard units with VVÉR-400 reactors are used in the Kolskaya and Armyanskaya nuclear power plants which have been constructed in the past few years and in nuclear power plants constructed with Soviet aid overseas. After 1980 a transition to a new size and type VVÉR-1000 units will be made [9]. Later on units will be installed with the VVÉR-1000 at the Yuzhno-Ukrainskaya, Kalininskaya, Rovenskaya, and other nuclear power stations.

Further increases in the unit power of the VVÉR is made difficult by rail transportation of reactor vessels. This problem can be solved either by seeking new methods of transportation or by delivering the vessels in parts and assembling and welding them directly at the construction site. The use of fundamentally new technologies for fabrication of vessels at the assembly site is not excluded in the future.

At the present time and until the end of the century the basis of nuclear power are and will be thermal reactors of these two types. Use has demonstrated their radiation safety for personnel and the environment as well as their economic competitiveness relative to organically fuelled thermal power plants. However, thermal reactors have a significant drawback: a large consumption of natural uranium, which impedes the development of large scale nuclear power using these reactors alone that could significantly expand the fuel base. Thus, in the program for expanding the country's power production, an important place is assigned to the development and construction of fast reactors. Their principal advantage consists of a fundamental improvement in the efficiency with which natural resources of uranium are used and thereby ensuring broad possibilities for the development of nuclear power. In addition, combined systems of fast and thermal reactors offer more manifold possibilities for the satisfaction of different sorts of energy demands. The development of fast breeder reactors and their use for power generation began in this country about 30 years ago under the direction of Academician A. I. Leipunskii of the Ukrainian Academy of Sciences. A large amount of scientific and engineering research was done which confirmed the validity of the basic idea of the fast breeder power reactor: expanded production of nuclear fuel and the practical feasibility of such a device.

In the past the scientific–technical foundations for the industrial use of fast power reactors have been laid. The efforts of Soviet scientists and engineers led to the power start up in June 1973 of the first large fast power reactor with sodium cooling, the BN-350 with a thermal power of 1000 MW. This reactor has a loop design and is intended for production of electricity and fresh water. Heat removal is by liquid sodium through a three loop scheme. Enriched uranium dioxide is used as a fuel. At present the reactor is in a continuous reloading regime and the maximum burnup of the fuel has exceeded the design value by roughly 20%. Although its arrival at full design power was delayed due to defects in the steam generators, it has been successfully used now for almost six years and has demonstrated the reliable operation of the other equipment. Its thermal power is now 650 MW, its electrical power is 120-125 MW, and its distillate production is as much as 80,000 tons/day. At the site of the Beloyarsk nuclear power station the installation of a second BN-600 fast power reactor with a thermal power of 1470 MW and an electrical power of 600 MW is now being completed. Compared

with the BN-350, this reactor has a higher energy release rate, higher steam parameters, and integral assembly, that is, the first loop is in the same tank as the reactor. An improved (faster and more sensitive) system for monitoring the formation of leaks in the steam generators is being used.

The experience of building and using the BN-350 and BN-600 make it possible to evaluate the advantages and disadvantages of their design and to gain the industrial and technical experience needed for building commercial high power fast reactors. At present work is under way on such a reactor with a power of about 1600 MW (electrical), the BN-1600, which is intended for nuclear power stations and which should later become the commercial prototype for fast power reactors. The introduction of the first standard assemblies with the BN-1600 will be an important milestone on the way to the large scale construction of fast sodium reactors. If needed, the unit power of fast reactors may be increased to 2000-2400 MW (electrical), and the problem of on-site assembly of the reactor vessel can be solved much more simply than in the case of the VVÉR since the sodium pressure in the vessel is less than 0.1-0.15 MPa as opposed to 12.5-16 MPa in the VVÉR.

The program for work on fast reactors also includes research on possible applications of helium or the dissociating gas N_2O_4 as a coolant. Gas-cooled fast reactors could have several advantages over sodium-cooled reactors including a simplified design and a somewhat higher breeding coefficient (for helium). However, there are serious problems of reliability and safety in such systems which require further study and engineering development and testing before a well-based decision can be made about building them. One important problem is to speed up the mastery of the industrial technology for reprocessing used fuel which is necessary for reuse of the original material, for isolating secondary fuel, and for realizing the plans for extensive production of fuel. At present various methods for reprocessing spent fuel are being developed which enable this operation to be carried out after the fuel is held for 6-8 months in order to reduce its radioactivity. This is especially important for breeder reactors. Although this problem is difficult, a certain amount of success has already been attained [10].

The operation of the external fuel cycle is one of the basic conditions for the development of large-scale nuclear power. In the future the relative contribution of nuclear power to the overall energy balance of this country will rise due both to the expanded sphere of applications for electrical energy in ordinary life and in industry and to the possibility of using nuclear power plants for producing industrial and space heating. The BN-350 reactor at Shevchenko and the reactors at the Bilibinsk nuclear thermal and electrical station on Chukotka (with a total power of 48 MW (electrical)) aretypical examples of dual purpose reactors that produce both heat and electricity. It seems reasonable in the future to use high-temperature graphite thermal reactors as a heat source in synthetic fuel production and in the metallurgical industry for primary reduction of iron ore [11].

The successful development of nuclear power in this country will promote the creation of the necessary machine construction base founded on the "Atommash" factory, whose first line has already gone into operation, and an expansion of the capacity of other machine construction firms. The creation and development of a new branch of the energy industry based on advanced technology such as nuclear power is one of the important conditions for realizing the plans for progress in the country's economy adopted at the Twenty-Fifth Congress of the Communist Party of the Soviet Union.

The Soviet Union shares its experience in the field of nuclear power with other countries. The A-1 and B-1 nuclear power stations in Czechoslovakia, the Reinsberg and Bruno Leuschner nuclear power stations in the GDR, the Kozlodui nuclear power station in Bulgaria, and the Lovisa nuclear power station in Finland have been built with technical help from the USSR. In the future the Soviet Union will assist in the development of nuclear power in other countries.

The principles of socialist integration among the member-countries of the Council for Mutual Economic Aid have made it possible to organize multilateral cooperation in the design and construction of equipment for nuclear power stations and in the creation of large machine construction enterprises. The international economic unions "Interatomenergo" and "Interatominstrument" have been created for cooperation among member-countries of the Council for Mutual Economic Aid in the areas of nuclear power machine construction as well as instrument manufacture and the fabrication of special equipment for nuclear power stations [12].

The Soviet Union, which now has rich experience in the design, construction, and use of power reactors of different types, fosters the development of nuclear power on a world scale. As a member of the IAEA, our country actively participates in the development and realization of an extensive program of research to promote the development of nuclear power and in offering technical assistance to developing countries for the peaceful utilization of nuclear energy.

In a welcoming message to the Twenty-First Jubilee Session of the General Conference of the IAEA, the General Secretary of the Central Committee of the Communist Party of the Soviet Union and Chairman of the Supreme Soviet of the USSR, L. I. Brezhnev, wrote that "the International Atomic Energy Agency has been called upon to play an important role in solving this vital problem and we hope that the agency will strive as hard as possible to make the atom serve only the interests of peace."

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STATUS OF THE FIRST NUCLEAR POWER STATION

V. S. Sever'yanov and V. B. Tregubov

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The failure-free operation of the first nuclear power station for 25 years was an indisputable proof that the scientific–technical problems asked as it was being built have been successfully solved, that the installed equipment is reliable, and that the operating regimes have been correctly chosen and has led to a number of important practical generalizations about the future development and use of power reactors. Besides producing power, the reactor has been the subject of numerous scientific and applied studies which have now become the principal purpose for which it is used.

Over these years the steam generator has worked in the specified regimes and the status of the graphite–water power reactor and the systems and equipment for servicing it have therefore become, after long use, of great interest for experts working on the manufacture of nuclear equipment. Only once, in 1971, was the plant shut down for major repairs of the upper communications of the reactor. At that time a detailed inspection of the metal construction of the reactor and the graphite pile was made and the state of the equipment and of the plumbing in the main and auxiliary systems was studied. In 1976 it was decided to use the steam from the steam generator for heating rather than electricity generation, and since the end of that year all the heat produced by the reactor is used in the heating system of the city.

The accident-free operation of the reactor [1], the experimental devices mounted on it [2], the improved quality and reduced duration of preventive repairs, and the precise organization of refuelling operations have made it possible in recent years to substantially increase the operating time of the reactor at power. Thus, since 1974 the first nuclear power station in the world, operating as an experimental reactor, has been at full power 80-90% of the calendar time (see Fig. 1).

The rather high operational characteristics of the first nuclear power station are primarily due to the reliable operation of the fuel elements and of the entire fuel channel. During its years of operation, several

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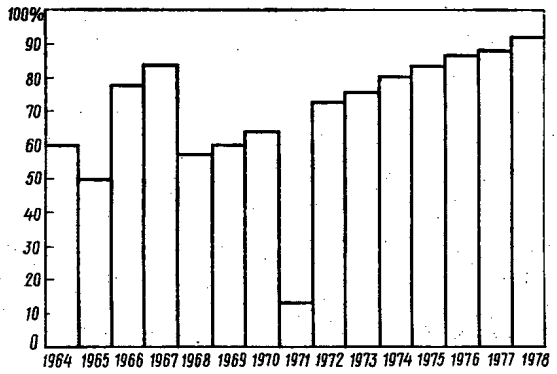

Fig. 1. Operation of the reactor at power (as a percent of calendar time).

thousand fuel elements have been used up in the reactor. When they are properly cooled there has not been a single case of destruction of the inner tube of a fuel element bearing the water pressure in the primary loop. The average burnup of the fuel composition has been 2-2.5 times above the design value. Damage to the seal of the outer shell found by the system for monitoring the intactness of fuel elements represents hundredths of a percent and, since it appears while there is still little burnup, cannot be attributed to radiation damage. Thus, it can again be said that the design and development of the technology for making cylindrical fuel elements of the dispersed type with a stainless steel shell are one of the greatest achievements in building the reactor for the first nuclear power station.

In 1971 when major repairs were being done, the reactor was unloaded for the first time and it was possible to inspect the state of the entire graphite pile at once. The reactor elements were examined with a periscope, the diameters of the elements were measured and their change along the height of the reactor pile determined. Samples of graphite were removed from different parts of the pile to study their physical and chemical properties under laboratory conditions. The maximum fluence of thermal neutrons in the core was $5.52 \cdot 10^{21}$ neutrons/sec^2. The flux of intermediate and fast neutrons in the center of the reactor is roughly equal to the flux of thermal neutrons and is about 30% of it in the outer elements.

During the examination of the graphite pile it was found that some blocks of the pile, both in the central and peripheral elements, have a crack or even several branched cracks. They are mostly through cracks, passing through the entire block, and are longitudinal. The damage was greatest in the center of the core. An examination of the inner surface of the blocks showed that the graphite in the core has an oxidized surface. The locking joints of the blocks were in good shape. The outer surface of the graphite blocks was less oxidized although its temperature was higher by about 30°C. This is explained by the fact that oxidation is mainly due to oxygen impurities in fresh nitrogen which is sent into the reactor along the central drop tube of the fuel channel and comes into contact with the inner surface of the block.

Shrinkage of the graphite, which was greatest during the first 10 years of operation, caused a reduction in the diameters of the elements along the center of the core and the structural gap between the assembly and the fuel channel. This required a mechanical reworking of the elements to their nominal dimensions, which, of course, caused additional damage to the blocks.

An analysis of the results of the inspection and selective examination of elements in later years have shown that the elements which were used throughout the years of operation to assemble the fuel channels and in which the temperature did not exceed 700°C are in satisfactory shape. These data make it possible to conclude that operation of a graphite pile for many years at temperatures of up to 700°C is feasible.

The case of the graphite pile is made of sheet carbon steel of thickness 7.5 mm with a zinc coating. The maximum temperature of the case is 360°C. It was possible to examine and evaluate the state of the outer surface of the case over a limited area through a ventilation hole in the water protection tanks. It was shown during this inspection that the zinc coating was preserved over the bulk of the visible surface, is in a good state, and has not peeled off due to mechanical action. Where the coating has been destroyed the structure is covered with a thin even layer of rust but no traces of local corrosion pitting were observed. During an inspection and testing of the sealing of the compensator gasket of the upper plate with the reactor case no cracks in either the metal or the seams were observed.

The water protection tanks for the reactor were made of 15-mm sheet stainless steel. During all the years of operation no leaks occurred in the tanks either along the welding seams or in the metal. On the outer walls of the tanks, on the side of the concrete shaft, there are insignificant traces of corrosion. The outer wall of the tanks which comes up under the protective plates of the reactor have kept their zinc coating in places and are coated with a thin layer of corrosion products in other places. Pitting was not observed under the layer of corrosion products. The temperature of the water in the tanks, measured in the upper parts, varied from 40 to 90°C depending on the reactor power and temperature of the cooling water.

A survey showed that the inner surface of the tanks as well as the reinforcing ribs were covered with a layer of iron oxides consisting mainly of magnetite. The most corrosion products are observed at the boundary where the water level oscillates. In this zone on the hotter wall nearer to the reactor traces of local corrosion in the form of pits were observed under the oxide layer. The rate of corrosion of the protection water tanks is 0.04 mm/yr, or 0.13 mm/yr including penetration by pits.

The pipes, heat-exchange equipment, pumps, and fittings of the primary loop are made of stainless steel. An examination of the main pipes of the first loop and the entire operating experience with them confirm their high reliability. Despite the extensive branching due to the large number of steam generators and circulation pumps, no defects associated with damage to the water-tightness of the metal itself were observed in any piping. Individual leaks in welding seams which appeared early in the operating period can be fully explained by the imperfect welding techniques and methods of checking the welds that were used when the station was being built.

The "weakest" point in the first nuclear power station is the small diameter pipes in the primary loop, i.e., the branching system of separate loops leading to the fuel channels and the pulsed lines to the thermal control equipment. Many of them are so located that defects cannot be repaired without major disassembly and it is this which led to the need for major repairs in 1971 due to the complete (except for the output collector) reassembly of the upper portion of the reactor.

As studies showed, the main reason for the failure of the loops was corrosion cracking under stress due to increased concentration of chlorides on the outer surface. The chlorides precipitated when water which fell on the leads during leaks evaporated; The first leaks appeared on the demountable joints (ten joints to each channel under the reactor plate) and also along the seal of the cutoff valve located on the outlet lead. During the major repairs the number of demountable joints was greatly reduced: they were replaced by welds, the outlet valve was eliminated, and at present there are no leaks in the leads to the fuel channels.

Operating experience with the heat-exchange apparatus in the first loop (steam generators, refrigerators, etc.) indicates that it is highly reliable. The four steam generators in the first nuclear power station have operated 85,000, 86,000, 100,000, and 15,000 h, respectively, in all regimes since the start up day. During this time 1100 cycles of rapid change of the thermal load on the steam generators have been completed. The customary water regime for nuclear power stations has been maintained on the first and second loops and this has prevented incrustation and deterioration of heat exchange at the transfer surfaces. Cases of loss of sealing in the steam generators have mainly occurred in the steam superheating sections. During the operation period the piping in the steam superheaters of the three steam generators has been completely replaced twice. There have been no losses of sealing in the economizer piping (similar in design to the steam superheaters). During the first years of operation there were two losses of sealing – in the vaporizers (most likely due to technical defects) along welding seams of single pipes and in the place where the bottoms are attached to the collectors. Metallographic examinations of the defective pipes of the steam superheaters showed that the cracks have a transcrystalline character and are formed due to corrosion cracking under stress. One proposal about fatigue destruction of bracket clamped pipes due to vibration was not verified.

One of the most complicated parts built for the first nuclear power station was the packing gland-type circulation pumps for the first loop with hydraulic seals to prevent the escape of radioactive water from the loop. The system which maintained the pressure drops on the packing gland seals required qualified supervision. Constant addition of water to the loop due to leaks through the packing glands required constant release of water from the loop which led to additional expense for cleanup of the excess water. In 1963 a circulation pump without a packing gland of the vertical partition type with a floating rotor was installed. This pump is reliable and convenient to use and requires minimum supervision during operation.

The equipment in the secondary loop has also shown high efficiency. To monitor the state of the carbon steel steam lines of the secondary loop, a section of the main pipe in the region of the temperature expansion compensator located in the lower point was cut out. Then it was found that the inner surface of the pipe is

covered with a layer of corrosion products. Its lower part suffered considerable pitting. In several cases the largest pits have joined and can be characterized as cavities. The state of the metal was also tested by ultrasonic defectoscopy. No cracks were found and the depth of pitting (up to 2.6 mm) agreed with the mechanical measurements. In the upper part of the tube numerous pits of diameter up to 4 mm were observed but their depth does not exceed 0.05 mm which indicates that corrosion is uniform. This sharp difference in the state of the upper and lower parts of the steam pipe indicate that corrosion occurs mainly during stops when condensate accumulates in the lower part of the steam pipe. Conservation of the steam pipe by means of short duration stops was not tried. Numerous defects in the form of honeycombs appeared on the cooling water lines and in the fire and drinking water pipes in places with high humidity and temperature. These pipes were completely replaced in 1971.

In summing up the 25 years of operation of the first nuclear power station, it should be noted that all the main components of the station have maintained their operating ability and continue to be used successfully.

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RADIATION SAFETY OF FAST-REACTOR FUEL CYCLES

O. D. Bakumenko, E. M. Ikhlov, M. Ya. Kulakovskii,* M. F. Troyanov, and A. G. Tsikunov

UDC 621.039.58

The use of plutonium fuel in fast power reactors requires a special approach to the problems of radiation safety in the manufacture of fuel elements and the handling of fuel assemblies (FA). The level of neutron and γ radiation resulting from the natural activity of plutonium fuel limits access to equipment in performing technological operations with fuel. The high specific energy release rate (250 kW per kg of fuel) and the high burnup (10% of the heavy atoms) result in an appreciable afterheat and a high fission-product activity. Irradiation of the higher isotopes of plutonium in a reactor leads to further sources of neutron and γ radiation which must be taken into account in the development of systems for transporting and reprocessing spent fuel.

The purpose of the present paper is to examine the main physical characteristics of fast power reactor fuel and to determine the radiation environment in handling fuel at various stages of the external fuel cycle, in particular in making fuel from reprocessed material and in transporting spent fuel from a nuclear power plant to a chemical reprocessing plant.

Radiation Environment in the Manufacture of Fuel Assemblies from Reprocessed Material. After an appropriate cooling period the spent fuel is subjected to chemical reprocessing to remove the fission products and to recover the unburned reusable fissionable materials. Depending on the fission-product decontamination factor, the fuel will have a certain residual fission-product activity. If decontamination is complete, the activity of the reprocessed fuel is determined solely by its natural activity. The sources of this activity are the following:

1. α activity which results from the decay of plutonium nuclides. The overall α activity of uranium–plutonium fuel used in plutonium thermal reactors with the isotopic composition ^{236}Pu, ^{238}Pu, ^{239}Pu, ^{240}Pu, ^{241}Pu, and ^{242}Pu equal to 10^{-5}, 1, 58, 23, 14, and 4% [1] is 230 Ci/kg of PuO_2 and is determined by ^{238}Pu; the α activity of plutonium of equilibrium composition (according to estimates of the authors ^{236}Pu, ^{238}Pu, ^{239}Pu, ^{240}Pu, ^{241}Pu, and ^{242}Pu comprise 10^{-6}, 0.2, 62, 27, 6, and 5%) from a fast reactor is half as large. The high α activity of reprocessed material requires leak-tight technological equipment.

2. Neutron and γ activities of the fuel, which determine the external irradiation of personnel in the fabrication of fuel elements and FA of reprocessed material. The neutron activity of the fuel results from the spontaneous fission of ^{238}Pu, ^{240}Pu, and ^{242}Pu, and from the (α, n) reaction in the oxygen of uranium and pluto-

*Deceased.

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TABLE 1. Neutron Yield from Plutonium Dioxide (neutrons/sec·g of isotope)

Source of neutrons	$^{236}PuO_2$	$^{238}PuO_2$	$^{239}PuO_2$	$^{240}PuO_2$	$^{241}PuO_2$	$^{242}PuO_2$	$^{241}AmO_2$
(α, n) reaction in oxygen	$5{,}80\cdot10^5$	$1{,}38\cdot10^4$	$3{,}30\cdot10^1$	$1{,}26\cdot10^2$	1,00	1,60	$2{,}70\cdot10^3$
Spontaneous fission	$3{,}60\cdot10^4$	$2{,}50\cdot10^3$	$2{,}20\cdot10^{-2}$	$9{,}00\cdot10^2$	$2{,}30\cdot10^{-2}$	$1{,}57\cdot10^3$	1,24
Total yield	$6{,}16\cdot10^5$	$1{,}63\cdot10^4$	$3{,}30\cdot10^1$	$1{,}03\cdot10^3$	1,02	$1{,}57\cdot10^3$	$2{,}70\cdot10^3$

TABLE 2. Intensity of Characteristic Radiation from Internal Conversion of γ Photons in the K and L Shells, (γ photons/sec·g of isotope)

Energy, keV	^{236}Pu	^{238}Pu	^{239}Pu	^{240}Pu	^{241}Pu *	^{242}Pu	^{241}Am
12—17	$2{,}15\cdot10^{12}$	$7{,}03\cdot10^{10}$	$9{,}90\cdot10^7$	$7{,}84\cdot10^8$	$1{,}08\cdot10^8$	$1{,}16\cdot10^7$	$4{,}53\cdot10^{10}$
20	$2{,}80\cdot10^{11}$	$9\cdot10^9$	$1{,}18\cdot10^7$	$9{,}54\cdot10^7$	$1{,}40\cdot10^7$	$1{,}42\cdot10^6$	$6{,}52\cdot10^9$
98	$2{,}26\cdot10^7$	$1{,}20\cdot10^6$	$2{,}50\cdot10^5$	$6{,}95\cdot10^3$	$7{,}20\cdot10^7$	—	$3{,}92\cdot10^6$
114	$6{,}80\cdot10^6$	$3{,}60\cdot10^5$	$9\cdot10^4$	$2{,}10\cdot10^3$	$2{,}28\cdot10^7$	—	$1{,}20\cdot10^6$

*Including characteristic radiation of ^{237}U which is the product of the α decay of ^{241}Pu.

nium oxides. The specific neutron yields for various isotopes of plutonium dioxide are listed in Table 1. The ^{241}Am formed in the reprocessed fuel by the β decay of ^{241}Pu has little effect on the neutron activity, although it has a high specific neutron yield. The neutron activity of plutonium from a thermal reactor is $4.2\cdot10^5$ neutrons/sec·kg of PuO_2 [1], with the contribution from spontaneous fission being approximately equal to that from the (α, n) reaction in oxygen. The neutron activity may be appreciably increased by the (α, n) reaction in admixtures of light elements such as beryllium, boron, fluorine, etc.

We list below the specific neutron yields from the (α, n) reaction in various light elements contained in the amount of 10^{-4} wt.% in plutonium with the isotopic composition ^{236}Pu, ^{238}Pu, ^{239}Pu, ^{240}Pu, ^{241}Pu, and ^{242}Pu equal to 10^{-5}, 1, 58, 23, 14, and 4%, neutrons/sec·kg of Pu:

Beryllium	$2.76\cdot10^3$	Silicon	5.30
Boron	$6.45\cdot10^2$	Carbon	4
Fluorine	$3.03\cdot10^2$	Sodium	42
Magnesium	42	Lithium	84
Aluminum	22		

In calculating dose characteristics it is necessary to take account of the fact that the average energy of neutrons from the (α, n) reaction is ~2.5 MeV higher than the average energy of neutrons from spontaneous fission.

The γ activity of uranium–plutonium fuel is determined by the γ radiation of plutonium and its daughter elements accumulated after reprocessing of the fuel. In addition to γ rays from plutonium, the characteristic x rays from the internal conversion of γ photons in the K and L shells play a significant role. For most plutonium isotopes the intensity of this radiation is two orders of magnitude higher than that of the γ radiation. In ^{241}Pu (in equilibrium with ^{237}U) and ^{241}Am the intensity of the characteristic radiation is approximately equal to that of the γ radiation (Table 2).

The radiation dose rate from unshielded plutonium fuel is determined by the low-energy characteristic radiation (13-20 keV). The dose rate is 3000 μR/sec on the surface of plutonium dioxide powder from thermal reactors, and 1000 μR/sec for plutonium dioxide powder of equilibrium composition from a fast reactor. This results from the difference in ^{238}Pu content. After storage for 1 year the buildup of ^{241}Am in reprocessed plutonium increases the γ dose rate by a factor of 1.5.

The dose rate at the unshielded surface of fast-reactor fuel pellets enriched 18% in plutonium is 400 μR/sec. In working with a small amount of plutonium, the dose rate from neutron radiation is low. It approaches the dose rate from γ radiation when working with 30–100 kg of plutonium, depending on the isotopic composition of the plutonium.

TABLE 3. Yields of Principal Plutonium Isotopes and ^{241}Am (photons/sec·g of isotope)

Energy range, keV	^{236}Pu	^{238}Pu	^{239}Pu	^{240}Pu	^{241}Pu *	^{242}Pu	^{241}Am
26—52	47 keV; $6,1\cdot10^9$	44 keV; $2,44\cdot10^8$	39 keV; $5,2\cdot10^5$ 52 keV; $8,1\cdot10^5$	45 keV; $3\cdot10^6$	26,3 keV; $2,26\cdot10^6$ 44,7 keV; $1,6\cdot10^5$ 51 keV; $2\cdot10^5$ 56,6 keV; $1\cdot10^5$	45 keV; $4,7\cdot10^4$	26 keV; $3,2\cdot10^9$ 33,2 keV; $1,4\cdot10^8$ 43,5 keV; $9\cdot10^7$
52—80	—	—	69 keV; $1,8\cdot10^4$		60 keV; $3,4\cdot10^7$ 65 keV; $1,2\cdot10^6$ 76,9 keV; $6,9\cdot10^5$	—	$4,57\cdot10^{10}$
80—130 (including characteristic X_u radiation)	$2,37\cdot10^9$	$5,35\cdot10^7$	$5,4\cdot10^5$	$7,5\cdot10^5$	$9,9\cdot10^7$	$1\cdot10^4$	$6,64\cdot10^7$
130—200	$1,3\cdot10^8$	$3,2\cdot10^7$	$2,5\cdot10^4$	$4,2\cdot10^4$	$1\cdot10^7$	$6\cdot10^2$	$4\cdot10^5$
200—300	—	$2,56\cdot10^4$	$2,2\cdot10^4$	$6\cdot10^3$	$2,27\cdot10^7$	—	$1\cdot10^6$
300—400	—	—	$7,45\cdot10^4$	$3\cdot10^3$	$1,48\cdot10^6$	—	$2\cdot10^6$
400—600	$5,3\cdot10^7$	—	$3,2\cdot10^4$	—	—	—	$2,7\cdot10^5$
600—800	$4,7\cdot10^7$	$3,2\cdot10^5$	$2\cdot10^3$	$1,5\cdot10^3$	—	—	$1,6\cdot10^6$

*Taking account of γ radiation from ^{237}U which is a product of the α decay of ^{241}Pu.

The large γ dose rates cited above necessitate special shielding when handling plutonium material. The use of a remote-control device can decrease the dose at the hands appreciably. Thus, moving away from the surface of a pellet by 10 cm decreases the dose rate by a factor of 1000. In working in glove boxes with G-13 gloves 2 mm thick the γ dose rate to the hands is halved.

Only γ and characteristic radiation from internal conversion in the K shell need to be considered in calculating the dose rate from a fuel element, since the characteristic radiation from conversion in the L shell is of such low energy that it is almost completely absorbed by the fuel element cladding. Table 3 lists the γ yields of the principal γ radiators in reprocessed plutonium fuel from which the fission products have been completely removed.

Neutron radiation does not contribute more than 7-15% to the dose rate. The principal contribution to the dose rate from a fuel element comes from the ^{241}Pu decay products ^{237}U and ^{241}Am. The dose rate at the surface of a plutonium fuel element from a thermal reactor is 50 μR/sec, and at a distance of 5 cm it is decreased to the maximum permissible value for the hands and forearms. The storage of a reprocessed plutonium fuel element for a year doubles the dose rate as a result of the buildup of ^{241}Am which has a high specific yield of 60-keV γ rays.

The dose rate from a plutonium FA used in a thermal reactor is also determined by the γ and neutron radiations. For plutonium of equilibrium composition used in a fast reactor the dose rate from neutron radiation is twice as large as that from γ radiation, and approximately equal to the neutron dose rate from plutonium of a thermal reactor.

The γ dose rate is determined by ^{241}Pu; the buildup of ^{241}Am during the storage of reprocessed material for 1 year increases the total dose rate by only 5%, since the low-energy radiation from ^{241}Am is absorbed by the FA wall. The dose rate at the surface of a plutonium FA from a thermal reactor is 80 μrem/sec, at 0.3 m from the surface it is 8 μrem/sec, and at 2 m from the surface of a FA it does not exceed the maximum permissible value.

Radiation safety of personnel during operations with fuel assemblies requires local hydrogenous neutron shielding material up to 15 cm thick along the active part of the FA. Then the head and tail of the FA, where the principal operations are performed, become freely accessible.

Radiation Environment in the Transportation of Spent Fuel. For fast reactors the length of the external fuel cycle is of fundamental importance; the shorter it is the more efficiently such reactors generate nuclear power. One of the stages of the fuel cycle is the cooling time of the spent fuel before chemical reprocessing. The length of the cooling period determines the conditions for transporting the fuel and performing subsequent operations. A decrease in the cooling time leads to an increase in the afterheat and the fission-product activity, which complicates transport operations. Using the BN-1600 reactor as an example, Table 4 lists the physical characteristics of spent uranium–plutonium fuel made of plutonium from thermal reactors.

TABLE 4. Characteristics of Spent Fuel

Parameter	Cooling time, months			
	1	3	6	12
Fission product activity:				
Ci/ton (U, Pu) O_2	$5{,}00 \cdot 10^7$	$2{,}58 \cdot 10^7$	$1{,}42 \cdot 10^7$	$7{,}70 \cdot 10^6$
Ci/FA*	$3{,}40 \cdot 10^6$	$1{,}76 \cdot 10^6$	$9{,}66 \cdot 10^5$	$5{,}25 \cdot 10^5$
After-shutdown heat-release rate of fission products				
kW/ton (U, PU) O_2	235	126	75	38
kW/FA	16,00	8,65	5,10	2,46
Neutron activity				
neutrons/[sec·ton (U, Pu) O_2]	$4{,}0 \cdot 10^9$	$3{,}1 \cdot 10^9$	$2{,}5 \cdot 10^9$	$1{,}2 \cdot 10^9$
neutrons/sec·FA	$2{,}7 \cdot 10^8$	$2{,}1 \cdot 10^8$	$1{,}7 \cdot 10^8$	$8{,}1 \cdot 10^7$

*68 kg of fuel in FA.

Table 4 shows that spent fuel from fast reactors has a high activity and a high heat-release rate. The neutron activity of spent fuel is determined by the ^{242}Cm and ^{244}Cm contents, which build up to 0.1 kg/ton of fuel, and have high specific neutron yields of $2.5 \cdot 10^7$ and $1.2 \cdot 10^7$ neutrons/sec·g of isotope, respectively. An increase in the storage time of reprocessed fuel before loading into the reactor leads to an appreciable increase in the neutron activity of spent fuel as a result of the increase in the ^{242}Cm content. Thus, 1-yr storage is approximately equivalent to doubling the neutron activity. In addition, an increase in the ^{242}Cm content makes an appreciable change in the heat release in spent fuel. If the reprocessed fuel is stored for 1 yr, the ^{242}Cm contributes 25% of the afterheat for 6 months cooling of the fuel. To eliminate the contribution of transplutonium elements to the radiation characteristics of plutonium fuel, these elements must be removed in the chemical reprocessing with a decontamination factor no smaller than 100.

The high fission-product and neutron activities require the construction of special transport facilities, and necessitate taking measures to limit the radiation to personnel. According to the safety rules, the maximum permissible equivalent radiation dose rate is 200 mrem/h at any point on the surface of a container, and 10 mrem/h at a distance of 2 m from transport facilities [2]. In addition, the rules for transporting radioactive materials [3] impose certain restrictions on the number of FA which can be transported. These restrictions amount to limiting the temperature of the surface of the container and the FA. Theoretical and experimental studies show that the optimum container is one designed for the transport of FA with a total heat-release rate of 20-40 kW [4]. For the BN-1600 reactor this amounts to 10-15 FA of spent fuel which have cooled 12 months. Such an arrangement requires a γ shield of 20-25 cm of lead, and a neutron shield of 15-20 cm of hydrogeneous material. It should be noted that the thickness of the γ shield is only slightly dependent on the number of FA being transported and on the cooling time of the spent fuel, because of the large self-absorption of γ rays in a FA.

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NEUTRON-PHYSICAL CHARACTERISTICS OF BATÉTs REACTORS (BASED ON THE RESULTS OF THE MANUAL START-UPS OF FOUR REACTORS)

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The neutron-physical characteristics of Bilibinskaya ATÉTs (BATÉTs) reactors during the period of its design have been investigated experimentally on a uranium–graphite assembly with subcritical insertion which simulated the active zone [1, 2]. The subcritical insertion was small, and its volume was less by a factor of ~20 than the volume of the active zone of the reactors. The assembly was used to check the accuracy of the methods which were utilized in connection with the calculation of some neutron-physical characteristics of the lattice of the reactors, in particular, the neutron multiplication coefficient K_∞. For this purpose the spatial distribution of the fluxes of thermal and resonant neutrons and the cadmium ratios with respect to $^{238,235}U$, ^{197}Au, ^{55}Mn, and ^{114}In were measured for all types of layouts of the active zone.

The principal neutron-physical characteristics which must be known for the operation of the reactors were determined during manual start-ups, when the effects, reactivity margin, and effectiveness of the control and safeguard system (CSS) units were measured. A large number of measurements were made to check and refine the methods of controlling the energy distribution in an operating reactor. The investigation of the physical characteristics during the start-up of the first reactor was communicated in [3]. The remaining three power units were introduced into the system in 1974–1976 (Table 1). After the introduction of the last reactor into the system it became possible to compare the neutron-physical characteristics of all four reactors.

The BATÉTs reactors have the identical electrical capacity (12 MW each), thermal power takeoff (25 Gcal/h), and essentially identical design. They belong to the channel-type of reactors with graphite moderator. Heat removal from the tubular fuel elements is accomplished with boiling water by means of its natural circulation. The layout of the reactor installation is single-loop with separation of the steam in a drum and supply of it to the central-heating turbine. Some difference of the reactors of the third and fourth power units from the first two is associated with a change in the arrangement of the four emergency protection (EP) rods in the active zone, due to which the lattice of absorbing rods has a regular appearance (Fig. 1).*

Probing of the Pile. Probing of the graphite pile (within the boundaries of the active zone) was performed in the reactors prior to charging the engineering channels (EC) with the use of a neutron source and boron counters. The aim of the probing is to check the uniformity of the pile and to estimate its diffusion properties. No appreciable nonuniformities were detected in the reactors of the first three units, and the properties of the pile corresponded to the design parameters. Defective graphite elements were detected in the pile of the fourth unit, and they were removed and replaced by new ones prior to the start of charging the EC of the reactor.

Charging of the Reactor. A complete charge of each reactor consisted of 217 EC with uranium of 3% enrichment (EC-3) and 56 EC with uranium of 3.3% enrichment (EC-3.3). Since the EC-3 differed somewhat in uranium content, they were placed in the reactor in a specified order in order to produce a certain shaping of the charge, which resulted in an increase of the uranium charge with reactor radius. The EC-3.3 were loaded into 56 peripheral cells arranged on the boundary with the reflector (Table 2).

Reactivity Margin and Compensation for It. With a complete charge the reactivity margin $\Delta k/k$ was measured for two states of the reactors: with water and without water in the EC. As a result it has been established that $\Delta k/k$ differs inappreciably for these two states (Table 3). When water is present in the EC, the average value is $\Delta k/k = 0.112$, and without water in the EC-0.107, which agrees with the design value of 0.11 [4, 5]. In the case of a uniform distribution of absorbing rods in the active zone the reactivity margin of the

*A map of the reactor of the first two power units is given in [3].

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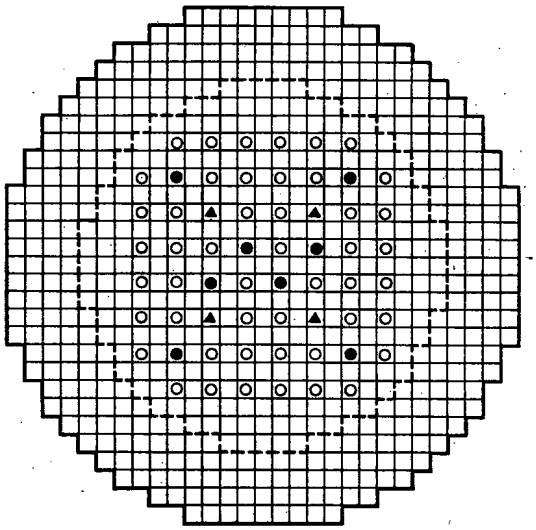

Fig. 1. Map of the reactor of the third and fourth BATÉTs units: ○, ▲, and ●) rods of the manual and automatic control and the emergency protection; – – –) boundary of the active zone.

TABLE 1. Start-up Date of the BATÉTs Units

Unit	First attainment of criticality	Duration of manual start-up, days	First switching-on of the turbogenerator under load	Duration of maintenance of design power, days
First	11.12.73	21	12.01.74	98
Second	7.12.74	16	30.12.74	47
Third	6.12.75	10	22.12.75	55
Fourth	17.12.76	7	27.12.76	14

TABLE 2. Charging of ^{235}U in BATÉTs Reactors

Unit	Amount of ^{235}U, kg			Max. scatter of ^{235}U content in EC-3, %
	217 EC-3	56 EC-3.3	total	
First	165,74	48,06	213,80	7,5
Second	164,52	46,76	211,28	5,5
Third	165,48	47,60	213,08	5,0
Fourth	165,54	47,03	212,57	4,6

reactor charged by EC with water is compensated by the complete insertion on the average of ~42 CSS rods. When water is absent in the EC, ~31 CSS rods are necessary to compensate for the reactivity margin. The appreciable difference in the number of rods necessary for the compensation of reactors with and without water in the EC is explained by the fact that due to the difference in the diffusion properties of the zones the average effectiveness of a single CSS rod in a zone loaded with EC with water is less by ~ 30% than in a zone with EC without water.

Since provision is made in each BATÉTs reactor for 52 absorbing rods in addition to the EP rods for compensation of the reactivity margin (the rods of manual (MC) and automatic (AC) control), there is a sufficient margin (~ 10 rods) even for the state of a reactor with the maximum multiplication coefficient (fresh EC with water, a cold nonfunctioning state). Subcriticality of the reactor with all the MC and AC rods (the EP rods are inserted) rods completely inserted into the zone amounts to $-1.3 \cdot 10^{-2}$ for the first two reactors and $-2.0 \cdot 10^{-2}$ for the other two. The increase in the subcriticality is caused by a difference in the position of the rods in the active zone of the reactors.

Emergency Protection. Emergency stopping of the reactors is accomplished by the rapid insertion of eight EP rods into the zone in ~1.6 sec, of which the four central ones are arranged at a distance of 0.15-0.4 $R_{a.z.}$ from the center of the active zone, and the outer four – at a distance of $0.7R_{a.z.}$ ($R_{a.z.}$ is the radius of the active zone, which is equal to 206 cm). With such an arrangement the efficiency of the EP rods depends significantly on the configuration of the radial neutron field, which is determined primarily by the charging of the EC and the arrangement of the MC rods compensating for the reactivity margin. Since excess MC rods remain when the reactivity is compensated, it is possible to change the shape of the radial neutron field by arranging the compensating rods in different ways. The effectiveness of the EP rods is ~1.8 times higher in the case of a field which has a strong rise at the center of the active zone than in the case of a decrease in the field at the center of the zone (Table 4).

TABLE 3. Experimental Data on the Reactivity Margin

Unit	Presence of water in the EC	$\Delta k/k \cdot 10^2$ for each unit	$\Delta k/k \cdot 10^2$ av.	No. of absorbing rods
First	Yes	10,5±1,5	11,2±1,0	43,0±0,5
Second	»	11,5±1,5		41,5±0,5
Third	»	11,7±1,5		41,5±0,5
Fourth	»	11,0±1,5		41,0±0,5
First	No	10,4±1,5	10,7±1,0	32,0±0,5
Second	»	10,6±1,5		31,0±0,5
Third	»	11,1±1,5		31,0±0,5

TABLE 4. Effectiveness of the Eight EP Rods, $\Delta k/k \cdot 10^2$

Arrangement of the MC rods	EC with water		EC without water	
	expt.	calc.	expt.	calc.
Uniformly throughout the active zone	1,38±0,07	1,48	2,03±0,10	1,75
All extracted rods are:				
in center	1,79±0,09	1,86	2,70±0,14	2,53
on periphery	1,08±0,05	1,05	1,57±0,08	1,43

It should be noted that positive interference is observed among the EP rods of the peripheral group: the overall effectiveness of this group of rods is higher by 20% than the sum of the effectivenesses of the remaining rods. The reverse pattern is observed for the EP rods of the central group: in this case mutual interference among the rods results in a decrease of the overall effectiveness of this group by 20%. When the MC rods are arranged uniformly, the central and peripheral groups of EP rods are identical with respect to their own effectiveness. The total effectiveness of both groups agrees with the overall effectiveness of all eight EP rods, i.e., the interference between these groups of rods is not observed. The overall effectiveness of these rods is reduced by 15% in the case in which one of the rods of the central or peripheral groups is absent ("rejected"), and by 25% when two rods (one from the central and one from the peripheral group) are absent ("rejected"). But in both cases the effectiveness of the remaining EP units exceeds the effective fraction of delayed neutrons.

In the case of an emergency shutdown of the reactors along with the rapid insertion of the EP rods four AC and six MC rods are also automatically inserted into the zone. The insertion time of the first rods is ~20 sec, and that of the second group is 110 sec. Measurements made during manual start-ups have shown that the additional insertion of these rods increases the effectiveness of the emergency protection from $1.4 \cdot 10^{-2}$ to $2.9 \cdot 10^{-2}$ (for a cold zone charged by EC with water). One should note that the latter value appreciably exceeds the sum of the power and temperature reactivity effects, which amounts to $1.2 \cdot 10^{-2}$ for BATÉTs reactors at the start of a run. The emergency protection (EP) provides for a sufficiently rapid decrease in the neutron power: it is reduced by a factor of four in 1.6 sec (Fig. 2). The effectiveness of the EP rods is approximately the same in all four reactors.

Temperature Coefficient of the Reactivity. In order to determine the temperature coefficient of the reactivity during manual startups of the reactors of the second and third units, they were heated up by means of pumping hot water ($t_{max} = 104^{\circ}C$) through all the EC and the CSS channels. During the heating up the output of the reactors was held at a level $\sim 10^{-5} N_{wom}$, and variation of the reactivity was compensated for by the AC rods. The temperature was controlled by thermocouples mounted in four EC and at two points of the graphite pile (at the center of the active zone and on the periphery, on the boundary with the lateral reflector). Variation of the reactivity was determined from the depletion of AC calibration rods and with the use of an analog reactivity meter [6]. The initial temperature of the fuel elements and the graphite pile was 24°C. Toward the end of the measurements it increased to 104°C. The overall variation of the reactivity upon heating up was $-4.5 \cdot 10^{-3}$. This corresponds to a temperature coefficient of the reactivity of $-5 \cdot 10^{-5}$ °C^{-1} for the temperature range of 24–104°C and the nonboiling operational mode of the reactor.

Neutron Field along the Reactor Radius. The shape of the radial neutron field in BATÉTs reactors is very sensitive to the arrangement of the MC and AC rods in the active zone. Since the CSS channels form the correct lattice in the reactor with a 40 × 40 cm grid, the best field is obtained with a symmetric and uniform arrangement of absorbers in the active zone. With such an arrangement of the absorbers it is possible without special difficulty to provide a radial neutron field with a nonuniformity coefficient $K_r \leq 1.4$. As experiments conducted in the reactor of the first unit have shown, a disruption of the symmetry in the absorber arrangement caused by the removal of a single peripheral MC rod results in the neutron flux in the 30-40 EC situated near the removed rod becoming higher by 20-60% than in the symmetrical EC in the other half of the reactor. An especially great difference is observed in the neutron fluxes for the peripheral EC. Therefore, symmetry is satisfied in BATÉTs reactors operating at capacity in the arrangement of the absorbers, for which the variations in reactivity (in proportion to the fuel depletion) are compensated by the shifting of four or six symmetrical MC rods.

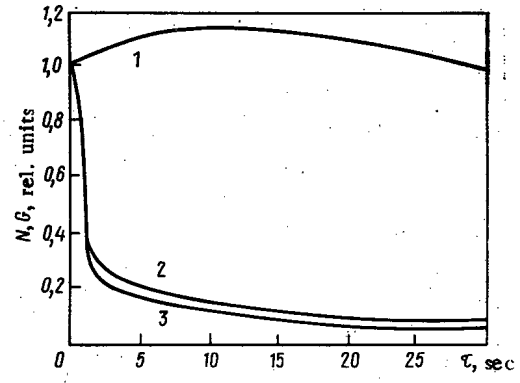

Fig. 2

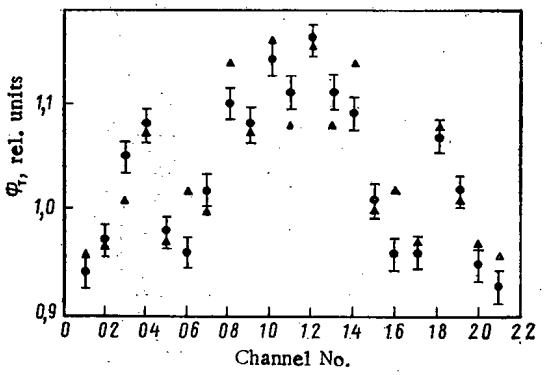

Fig. 3

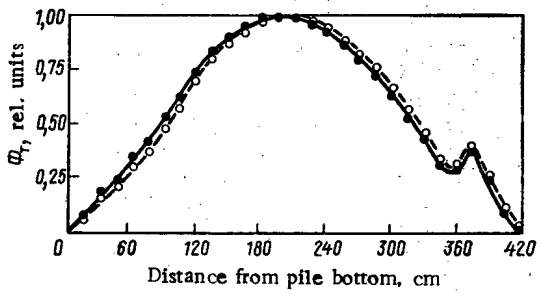

Fig. 4

Fig. 2. Variation of the coolant flow rate G (1 – experiment) the thermal power N (2 – calculation), and neutron power (3 – experiment) as a function of time upon a cut-off of the EP rods.

Fig. 3. Distribution of thermal neutrons along the diameter of a cold reactor having EC without water: ●) experiment; ○) calculation.

Fig. 4. Thermal neutron distribution with respect to height in a reactor having EC without water near the reactor center: ●) zone without CSS rods (critical assembly); ▲) in the zone of the 32 completely inserted CSS rods.

The radial neutron field during manual start-up was measured by fission chambers, which were lowered into the central tube of an EC to the level of the active zone. The calculated values of the neutron flux obtained by the double-group method [7] agree satisfactorily with the experimental values. The neutron distribution along one of the diametral series of EC is shown in Fig. 3. For this case the mean square deviation of the calculated and experimental values of the neutron flux calculated with all EC taken into account is 4.7%.

Neutron Field with Respect to Reactor Height. Since no special measures have been adopted in BATÉTs reactors with respect to equalization of the neutron flux with height, the distribution of the neutron flux with height is close to cosinusoidal only when the state of the active zone includes completely charged compensation units (AC, MC). The maximum of the distribution is shifted somewhat (by 10 cm) below the center of the active zone. This is caused by the fact that upon complete charging the compensation units fall short of the lower boundary of the active zone by 15 cm. The experimental thermal neutron distribution given in Fig. 4 is obtained by means of moving the fission chamber in the central tube of the EC. The absence of a burst of neutrons at the lower reflector is explained by the increased steel content in the lower part of the EC.

In order to reduce the nonuniformity in the energy yield with respect to height upon operation at capacity, a scheme of reactivity margin compensation has been adopted at BATÉTs according to which there are no more than 10 rods (4 AC and 6 MC) in the intermediate position. In this case the increase in the nonuniformity of the neutron flux with respect to height even in the EC located near the rods which are in the intermediate position does not exceed 16% (one should note that the capacity of these EC is less than the average).

BATÉTs reactors are identical in neutron-physical characteristics and correspond to the planned implementations. The increase in the number of compensating rods in the critical state for the first reactor in com-

parison with the rest is evidently associated with the increased ^{235}U charge in it. Therefore, one should expect that its operating period is also somewhat greater than that of the other reactors.

The experimental data obtained in connection with the start-up of the BATÉTs reactors can be used to check other methods of calculating uranium–graphite systems.

The authors thank their co-workers of FÉI and BATÉTs who took part in the manual start-ups of the reactors of this plant.

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WAYS OF ALTERING THE COEFFICIENTS OF REACTIVITY IN RBMK REACTORS

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UDC 621.039.524.2.034.3

While the RBMK reactor is being brought up to steady-state operating conditions with continuous fuel recharging, its core undergoes significant changes. Practically from the very first days of reactor operation, withdrawal of the auxiliary absorbers (AA), which compensate the initial reactivity, begins and the freed channels are charged with fuel assemblies. In the first period of operation the fresh fuel assemblies practically do not differ from those of the initial charge in respect of burn-up but gradually the difference in the burn-up of the various fuel assemblies has an increasingly pronounced effect on their individual physical characteristics, including their contribution to the integrated coefficient of reactivity for the reactor core. The combined change in the structure of the reactor core, the auxiliary absorbers, and the average burn-up of the fuel produces a quite complex picture of variation of the coefficients of reactivity with the operation of the reactor. The variation of the various coefficients of reactivity with the burn-up is shown in Fig. 1. Let us note that the real picture of the variation of the coefficients of reactivity is considerably more complicated since during operation, especially in the initial period of building up to the rated capacity of the power plant, the thermohydraulic and other parameters of the reactor may differ substantially from the nominal values and this affects the actual coefficients of reactivity.

Determination of the coefficients of reactivity by calculations or experiment is not the end in itself. The reliability of the coefficients in many ways determines the accuracy of the prediction of the dynamic behavior of the reactor and the stability of the energy distribution under steady-state and transient operating conditions. In great measure it is the job of the control system of the reactor to ensure the stability of the energy distribution. The characteristics of control systems are being steadily improved and it is proposed in future to bring in completely automated reactor control with the aid of a computer. However, attainment of the best possible dynamic parameters in the controlled system will greatly simplify the solution of this problem in the future and will at the present time facilitate the work of the operator in controlling the reactor. Hence the need to modify the coefficients of reactivity of reactors in operation.

As noted in [1], "... the uranium–graphite ratio adopted in the design (RBMK) is not entirely optimal from the point of view of reactor control under transient conditions. In subsequent reactors the uranium–graphite ratio will be close to the optimal and the control of the power distribution will be automatic." This conclusion, arrived at on the basis of the operation of the main block of the Leningrad Atomic Power Plant, can

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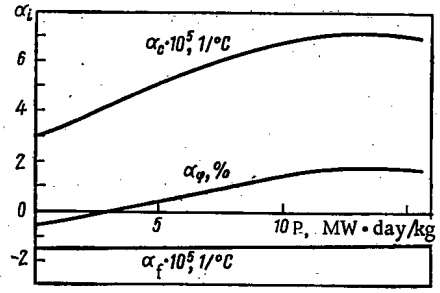

Fig. 1. Variation of coefficients of reactivity as the fuel burns up: α_C and α_f, temperature coefficients of the moderator and the fuel, respectively; α_φ, steam-void coefficient; P, average burn-up of the fuel.

also be drawn from computational investigations of the dynamic characteristics of the reactor. Indeed, as the coefficients of reactivity change the time constant τ_{01} of the development of the deformation of the energy distribution along the radius (its first azimuthal, the least stable harmonic) decreases, with operation, from several hours in the initial period to several minutes under steady-state conditions of fuel recharging (Fig. 1). The allowable decrease in τ_{01}, ensuring reliable and safe control of the reactor by an operator using means available to him for monitoring and controlling the energy distribution, can be estimated only experimentally. In particular, it has been shown that when a reactor is operated without local automatic controls the minimum allowable time τ_{01} is 15-20 min. Accordingly, an optimal set of coefficients of reactivity can be determined for the operation of the reactor: the steam-void coefficient* α_φ should lie within the interval $\pm 1\%$, the temperature coefficient of the moderator α_C should not be greater than $5 \cdot 10^{-5}$ $(°C)^{-1}$, and the temperature coefficient of the fuel α_f should lie in the interval from $-1 \cdot 10^{-5}$ to $-2 \cdot 10^{-5}$ $(°C)^{-1}$. As is seen, there is some divergence between the desired and the existing coefficients of reactivity and appropriate measures should be taken to eliminate this divergence.

Below we give the results of computational investigations to find the dependence of the coefficients of reactivity on the principal thermohydraulic and operational parameters of the reactor core as well as to choose the most effective ways of enhancing the stability of energy distribution in both functioning reactors and planned reactors. Our attention was focused on the steam-void (α_φ) and temperature (α_C) coefficients of reactivity since it is precisely these coefficients which, firstly, have the greatest effect on the stability of the energy distribution and, secondly, depend strongly on the core parameters.

In relation to functioning reactors, all the measures to vary the coefficients of reactivity can be arbitrarily divided into two groups:

employment of operating modes which ensure a set of reactor thermohydraulic characteristics for which the coefficients of reactivity are maintained within the given range;

realization of such technological measures which reduce the coefficients of reactivity and improve the stability of energy distribution.

This encompasses increasing the fuel enrichment and density, including going over to metallic fuel, increasing the auxiliary absorbers in the core, raising the operational reactivity margin, etc. In the case of reactors in the design stage, in addition to the measures mentioned above it is possible to employ measures whose effect on the coefficients of reactivity are irreversible and unchanging in the course of operation. Among these are changes in the lattice pitch of fuel assemblies and/or in the effective density of the moderator.

Let us consider the effect of these factors on the coefficients of reactivity. The calculated relations between the coefficients α_φ and α_C and the moderator temperature, mean density of the water, and the number of rods compensating the operational reactivity are quite strong dependences and must be taken into account in any comparison of calculations and experiment. This is particularly true of the steam-void coefficient of reactivity. With small deviations from the nominal parameters these relations are practically linear and can be represented as follows: increasing the graphite temperature by 100°C increases α_φ by 0.2% and decreases α_C by $0.45 \cdot 10^{-5}$ $(°C)^{-1}$; raising the mean density of the water by 0.1 g/cm^3 increases α_φ by 0.37% and reduces α_C by $0.05 \cdot 10^{-5}$ $(°C)^{-1}$; an increase in the operational reactivity margin by 10 rods reduces α_φ by 0.3%. The combined deviation of the heat-engineering characteristics of the reactor core during variation of the power (with a constant coolant flow rate) introduces a correction to the coefficient of reactivity, as shown in Fig. 2, the main factor determining the dependence of α_φ on the power being the variation in the mean density of the coolant.

*The steam-void coefficient α_φ of reactivity is defined as $\alpha_\varphi = \Delta\rho/\Delta\varphi$, where $\Delta\rho$ is the change in the reactivity when the mean volume content of steam changes by $\Delta\varphi$.

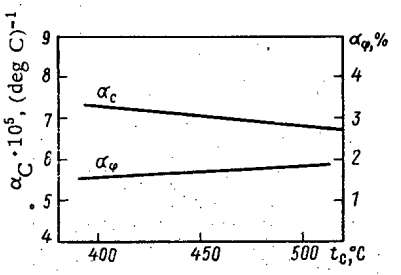

Fig. 2

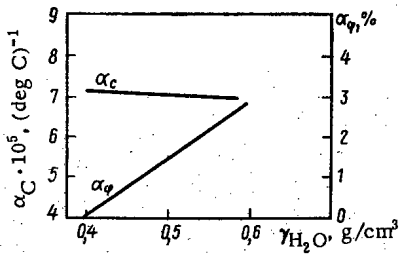

Fig. 3

Fig. 2. Dependence of coefficients of reactivity on graphite temperature.

Fig. 3. Dependence of coefficients of reactivity on mean density of water in reactor core.

Increasing the mean density of the fuel pellets proves to be important not only from the point of view of increasing the efficiency of the fuel elements but also in respect of having a favorable effect on the steam-void coefficient of reactivity: an increase of 0.1 g/cm^3 in the mean density of the pellets reduces α_φ by 0.27%. The most operative way of preventing a rise in the steam-void coefficient of reactivity above the allowable level is that of preserving auxiliary absorption in the reactor core. In this case, clearly, this is to the considerable detriment of the neutron balance as manifested by a reduction of the burn-up fraction of the fuel discharged: if the number of auxiliary absorbers in the reactor core is increased by 10, α_φ drops by 0.12% but the burn-up fraction of the discharged fuel falls by 0.54 MW·day/kg.

A more effective measure improving the dynamic characteristics of the reactor is that of increasing the enrichment of the fresh fuel. In particular, increasing the initial enrichment from 1.8 to 2.0% reduces α_φ by 0.6%. The attendant increase in the burn-up fraction of the discharged fuel by 3.8 MW·day/kg even somewhat improves the technicoeconomic indicators of the reactor.

For planned reactors whose design can be changed in accordance with optimization studies the most effective measure for reducing the coefficients of reactivity is that of reducing the ratio of graphite nuclei to ^{235}U nuclei in the reactor core. In this case the burn-up fraction changes insignificantly and because of the more efficient use of ^{238}U and the large build-up of plutonium the power of the freshly charged fuel assemblies is decreased. The most obvious way of reducing the graphite–uranium ratio is that of diminishing the pitch of the lattice of channels in the reactor. However, since a reduction of the pitch of the channel lattice is usually accompanied by difficulties with the separation of the lines bringing in and carrying away the coolant, proposals for decreasing the effective density of the graphite are also of interest. As shown by calculations, both of these factors affect the coefficients of reactivity in the same way and can be represented quantitatively by the ratio of graphite nuclei to ^{235}U nuclei, N_C/N_5 (Fig. 3).

Let us note that the discussion presented here applies to a 1.8-2.0% enrichment of the fuel charged. As the burn-up fraction increases and, consequently, the enrichment of the fuel charged is increased, say, above 3%, it may be that the lattice pitch or the effective density of the graphite must be increased and not decreased.

The steam-void coefficient of reactivity is decreased substantially in the transition from dioxide fuel to fuel with a higher density, e.g., uranium metal. This transition also possesses many other advantages [2]; therefore, the expected major difficulties of a technological and operational nature may be justified.

Of interest are the results which were obtained upon considering the influence of the shape of the distribution of the neutron flux over the height on the steam-void coefficient of reactivity of the reactor. When the dependence of α_φ on the density of the water is taken into account, it may be expected that the average steam-void coefficient of reactivity for the entire reactor core should be different for different heights of the energy distribution. According to calculations it turned out that the increase in α_φ during the transition from an energy distribution with a maximum in the upper part of the core to one with a maximum in the lower part of the core is ~1.0%, which undoubtedly affects the stability of the radial energy distribution. This type of effect must necessarily be taken into account when choosing the distribution of the absorbing properties of the auxiliary absorbers over the length of the channel in the case when, to reduce the coefficients of reactivity in the reactor core, some of the auxiliary absorbers initially charged are kept for a sufficiently long time. It is a quite complicated procedure to optimize their properties by calculation since in the process indeterminacies of a physical as well as a thermohydraulic character are superimposed upon each other. When the channel power is changed, the density of the water is redistributed over the entire length of the steam-producing seg-

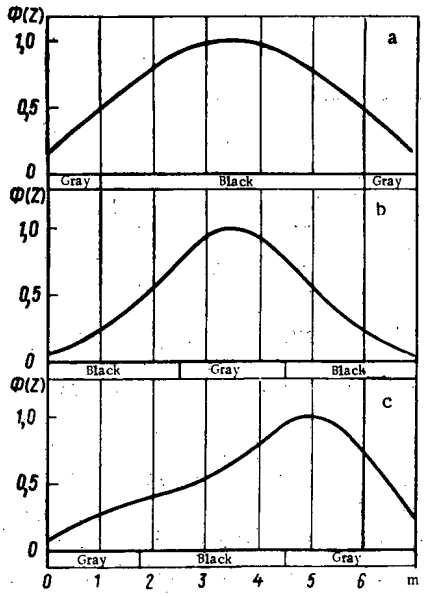

Fig. 4. Effect of properties of auxiliary absorbers over height on parameters of reactor core:

a) $K_z = 1.50$; $\Delta\alpha_\varphi = 0$; $\Delta\rho = 0$;
b) $K_z = 2.0$; $\Delta\alpha_\varphi = +0.35\%$; $\Delta\rho = +0.014$;
c) $K_z = 1.80$; $\Delta\alpha_\varphi = +0.25\%$; $\Delta\rho = +0.011$.

ment but the largest change in the steam content and water density occurs in a limited region, roughly in the middle of the channel. This makes it possible to choose the distribution of the absorbing properties of the auxiliary absorbers over the height with a view to providing the most effective action on the steam-void coefficient. It must be borne in mind in this case, however, that even a slight variation in the properties of the auxiliary absorbers over the height affects the energy distribution over the height and this in turn affects the steam distribution over the length of the fuel assembly. The effect of the auxiliary absorbers with various properties on the steam-void coefficient of reactivity and the shape of the energy distribution over the height is illustrated in Fig. 4, which shows changes in the shape of the energy distribution over the height, its coefficient of nonuniformity K_z, the steam-void coefficient $\Delta\alpha_\varphi$, and the efficiency $\Delta\rho$ of the auxiliary absorbers. The last two parameters, are given in relation to the composition of the auxiliary absorbers, are shown in Fig. 4a.

Conclusions. In RBMK reactors there are many possibilities of acting on the coefficients of reactivity, primarily on the steam-void coefficient α_φ. Some of these methods can be implemented only by building new reactors and are irreversible (e.g., changing the lattice pitch of the fuel assemblies). Other ways of great interest make it possible to operatively act on the coefficients of reactivity even in existing reactors. These include such strong, but economically acceptable, measures as keeping some auxiliary absorbers in the reactor core or increasing the operational reactivity margin as well as economically effective measures involving an increase in the density of the fuel and in the initial enrichment. An extremely great effect on the steam-void coefficient of reactivity is displayed by such operating modes as maintaining the mean water density in the reactor and the energy distribution over the height at a required level.

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CALCULATION OF COOLANT FLOW RATE BY RADIATION METHODS AND POWER IN FIRST UNIT OF ARMENIAN ATOMIC POWER PLANT

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The coolant flow rate of the primary circuit is one of the parameters which are necessary for determining the thermal power of a unit of an atomic power plant [1]. Exact and operational measurements of the flow rate make it possible to establish constant monitoring of the thermal operating conditions of a reactor and to ensure optimal operating conditions for the atomic power plant. Conversely, the lack of applicable instruments for measuring the flow rate of the coolant, e.g., in the San Onorf Atomic Power Plant [2], was one cause of the long delay in bringing the plant up to nominal operating conditions.

In operating the units of an atomic power plant the thermal power must be measured with sufficient accuracy [3]. Unfortunately, atomic power plants hitherto did not have a simple device for measuring this power operatively and accurately (with an error of 1-2%). One of the most promising methods for use in atomic power plants is the radiation method of measuring the coolant flow rate in the primary circuit [4] and the thermal power [5]. The possibility of using these methods was first shown in [6] on the basis of studies carried out on the first and third units of the Novovoronezh Atomic Power Plant.

The present paper discusses the results of measurements of the coolant flow rate in all six of the main circulating loops (ML) and the thermal power of the first unit of the Armenian Power Plant. The coolant flow rate was measured by a method based on the registration of the decay of ^{16}N activity with coolant circulating in the ML; the method used to measure the thermal power was based on registration of the neutron flux density under the reactor vessel. For practical implementation of these methods we developed and set up in the first unit of the power plant a flowmeter in the form of a system of 12 experimental channels above the pipes of the primary circuit with sensors installed in them and a power meter was set up under the reactor (Fig. 1).

Measuring the Flow Rate of Coolant in the Primary Circuit. High-stability SI-3BG Geiger–Müller counters were used as sensors. The sensors were installed above the primary-circuit pipes in special collimators to isolate a definite, strictly specified volume of coolant.

The first sensor records the intensity of the ^{16}N γ rays of the primary-circuit coolant in the segment of the ML at the outlet from the reactor (first detection point DP-1); the second sensor, after the coolant has passed through the steam generator at the inlet of the ML to the reactor (second detection point DP-2).

The coolant flow rate G in each ML was found from the ratio of the counting rates N_1 and N_2 of the sensors installed on it according to the refined formula

$$G = 2\lambda V \Big/ \left[\ln \frac{N_1}{N_2} + \ln \frac{N_{12}}{N_{21}} + 2 \ln \frac{\rho_2 G_0(\rho_2)}{\rho_1 G_0(\rho_1)} \right],$$

where λ is the ^{16}N decay constant (0.0971 sec^{-1}); V, effective volume of coolant in the ML between the pipe cross sections passing through the detection points (13.6 m^3); N_{12}, counting rate of the second sensor at the first point DP-1; N_{21}, counting rate of the first sensor at the second point DP-2;* ρ_i, coolant density at the i-th detection

*Introducing the term $\ln N_{12}/N_{21}$ in the denominator of the function given above takes account of the differences in the efficiency of γ-ray detection by the sensors.

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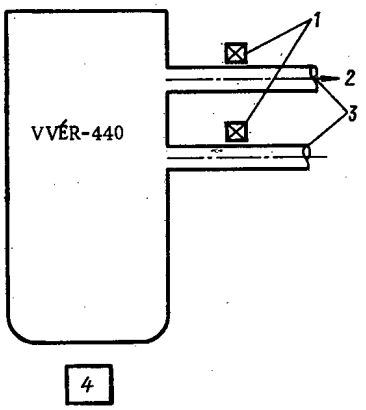

Fig. 1

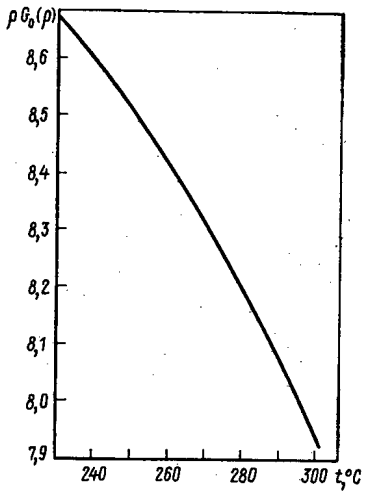

Fig. 2

Fig. 1. Positions of flowmeter sensors and power meter: 1) flowmeter sensors; 2) to steam generator; 3) main circulating loop; 4) power meter.

Fig. 2. Attenuation factor $\rho G_0(\rho)$ vs coolant temperature.

point; and $G_0(\rho)$, attenuation factor of the γ-ray source as a function of the density of the material of the source* (coolant).

The factor $G_0(\rho)$ was calculated from the NTs program [7] making it possible to calculate the γ-ray field from a set of thick-walled cylindrical layers in an arbitrary geometry. In this case the relative values of $G_0(\rho)$ are found with a mathematical error of no worse than 0.1%. The temperature dependence of $G_0(\rho)$ is shown in Fig. 2 for the range from 230 to 300°C; it follows from Fig. 2 that for the nominal power of a unit in an atomic power plant with a VVÉR-440 reactor (t_1 = 297°C for DP-1 and t_2 = 268°C for DP-2) the ratio $\rho_2 G(\rho_2)/\rho_1 G_0(\rho_1)$ is 1.04. Failure to take $2\ln(\rho_2 G_0 \rho_2/\rho_1 G_0 \rho_1)$ into account in the refined formula results in the flow rate being overestimated by roughly 7%.

Let us note that in deriving the formula given above we assumed that both sensors were set up in an identical geometry with respect to the ML pipe and that they record γ rays from equal "visible" volumes of coolant. Measures were undertaken to ensure high accuracy in setting up the sensors. Nevertheless, because of some differences in the geometry of the positioning of the sensors and collimators relative to the Du-500 pipe, a certain error was observed in the measurement of the coolant flow rate in ML. Analysis of the factors which are the sources of this error showed that they include differences in the positioning of the two sensors relative to the pipe and the sensitive volumes of the sensors relative to the collimators, different background contributions by γ rays from the ends of the pipes passing through the shields of the collimators, etc. Corrections were calculated to take account of the γ ray background. Finally, the total error in the measured values of the coolant flow rate in the ML of the first unit of the Armenian Atomic Power Plant and was found to be 6-7%.

A series of experiments were carried out using the flow meter to measure the steady-state flow rates in the ML of the first unit of the Armenian Power Plant at reactor power levels at 20-50% of the nominal value. At the same time we determined the coolant flow rate $G(\Delta p)_i$ from the head characteristics of the main circulating pumps (see Table 1). The results of the experiments are compared with data obtained from the head characteristics of the main circulating pumps (MCP).

From the data in Table 1 it follows that the maximum deviation of the flow rate G_i from the mean value $\overline{G} = \frac{1}{n}\sum_1^n G_i$ for each mode is ±7%. This does not exceed the calculated error in the flow rate. The coolant flow rate through the reactor core for various modes remains constant to within 2% and differs from the total flow rate obtained from the head characteristics by no more than 2%.

To obtain additional information about the characteristics of the flow meter we performed a special experiment. In one of the ML (ML No. 2) we gradually reduced the coolant flow rate by partially closing the main

*Source is taken to mean the volume of coolant in the primary-circuit pipe from which γ rays can enter the sensor without interacting with the material of the collimator.

TABLE 1. Coolant Flow Rates in Primary Circuit in Five Modes, m^3/h

Mode	Loop No.	$G(\Delta p)_i$	G_i	$\bar{G}$	$\bar{G}\times 6$	$\sum_1^6 G(\Delta p)_i$
1	1	7600	8280	7860	47140	46750
	2	7850	—			
	3	7590	7930			
	4	7860	7360			
	5	7870	—			
	6	7980				
2	1	7570	8330	7700	46200	46720
	2	7900	—			
	3	7600	7860			
	4	7870	7200			
	5	7870	7410			
	6	7910	—			
3	1	7660	8370	7670	46030	46750
	2	7940	—			
	3	7580	7890			
	4	7850	7100			
	5	7930	7330			
	6	7790				
4	1	7590	8440	7860	47140	46530
	2	7970	7900			
	3	7620	—			
	4	7630	7310			
	5	7900	7800			
	6	7820	—			
5	1	7620	8470	7700	46230	46480
	2	7960	7660			
	3	7580	—			
	4	7630	7270			
	5	7870	7410			
	6	7820	—			

shut-off valve. The flow meter was used to measure the coolant flow rate G_i in all the loops of the five positions of the shut-off valve. Each position was set for 10 min, the time necessary for establishing steady-state operating conditions and carrying out measurements (Fig. 3). It follows from Fig. 3 that at low flow rates the characteristics from the specifications, which were determined for the main circulating pumps at a pressure drop Δp on a test bed, differ from the real characteristics determined with the VVÉR-440 in the range under 7000 m^3/h.

It was possible with the flow meter to determine the flow rate (reverse flow rate) of the coolant in the ML arising when the MCP were cut off. For this purpose one of the MCP was disconnected, upon which the valve was opened temporarily and the time dependence of the indications N(t) of both sensors positioned on the given ML. The interval of time for which the values of the function N(t) for both sensors are constant is characterized by the constancy of the flow rate in the ML. The reverse flow rate, found by this method is 3600 m^3/h, which is 44% of the forward flow rate through this loop. There are other possible ways of using the flowmeter, especially in studying unsteady flow rates.

Measuring the Thermal Power of a Nuclear Reactor. The sensor of the power meter was placed beneath the reactor at some distance from sources of perturbation to the neutron flux emerging from the reactor. This sensor detects neutrons which emerge from the reactor vessel in the region of the core and, reflecting from the concrete shield, travel through an annular gap down through a labyrinth to the sensor of the power meter.

The neutrons are recorded by the power meter by conversion to γ rays in a cylindrical polyethylene target with a diameter of 200 mm in which Geiger–Müller γ ray counters are arranged. External γ rays are cut off from the target and the counters by a lead screen 200 mm thick. If necessary, the γ ray counters are replaced by means of a special channel which is led out into a semiservice room. The counter indications are recorded with analog and digital indicators located on the control board of the particular unit of the power plant.

For absolute calibration of the relative indications of the power meter in units of thermal power we carried out joint measurements of the thermal power W_T of the reactor (in the range from 40 to 85% nominal power) by the method of making a thermal balance, and of the power-meter indications N_{pm}. A linear dependence of W_T on N_{pm} was obtained from these data by the least-squares method. In this case the correlation coefficient is $r_{N_{pm}W_T} = 0.9997$. This value makes it possible to obtain (with calibration for $n \geq 4$ joint measurements at various levels of thermal power) a confidence interval of $\pm 0.02W_T$ for the values of the thermal

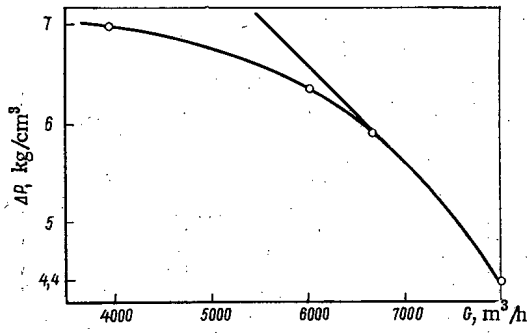

Fig. 3. Head characteristics of MCP-2 of first unit of Armenian Atomic Power Plant: O) flow meter data; ————) specifications.

power of the reactor in the range of thermal power variations from 20 to 100% (with a significance level of 95%) with the power meter. Thus, the above method of correlation measurements of the thermal power of a reactor of the water-moderated–water-cooled (VVÉR) type makes it possible to significantly refine the thermal power of the reactor of the VVÉR-440 unit.

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PHYSICAL START-UP OF IBR-2 PULSED RESEARCH REACTOR

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The physical start-up of the IBR-2 periodically functioning pulsed reactor took place in Dubna at the end of 1977 and the beginning of 1978; the reactor was designed for research on nuclear physics and the physics of condensed media in extracted beams of slow neutrons at a mean power of 4 MW and a maximum thermal-neutron flux density of $\sim 10^{16}$ neutrons/cm^2·sec [1, 2]. The investigations were carried out without coolant at an average reactor power of up to 500 W in both steady-state and pulsed modes of operation. The IBR-2 was brought up to the critical state for the first time on Nov. 30, 1977, and pulse criticality was achieved on Jan. 13, 1978. The present paper gives a review of the principal experiments performed during the physical start-up of the IBR-2.

Critical Assembly. Before the fuel assemblies were loaded all the cells of the reactor core were filled with fuel-assembly imitators which differed from the fuel assemblies only in that the fuel elements in them contained copper instead of plutonium dioxide. A Po–Be source with an intensity of 10^7 neutrons/sec was set up in the center of the reactor core. The neutron flux was monitored with the regular start-up instrumentation with three ^{235}U fission chambers set up outside the reactor at a distance of 1150 mm from its center (Fig. 1, 1) as well as two auxiliary experimental channels with replaceable detectors set up both in the Plexiglas block of the cold-moderator imitator (see Fig. 1, 11) and in the reactor core 8. The counting rate of the regular and experimental detectors with the reactor core loaded with imitators was 5-50 counts/sec. Moreover, for linear monitoring of the power we used an experimental current channel with a boron chamber in the CMI, analog reactimeter, and recording potentiometer.

Fuel was charged into the reactor core by successively replacing the imitators with fuel assemblies which were loaded in order of increasing calculated efficiency. The extrapolation of the curve of inverse counting therefore always showed a critical-mass value that was lower than the actual value; this provided additional safety in the start-up operations. The critical state was attained for two variants of reactor charge (Fig. 2): "central" (70 fuel assemblies) and "peripheral" (74 fuel assemblies). The calculated critical charge of these variants was 71 ± 2 and 75 ± 2 fuel assemblies, respectively.

Effect of Reactivity of Control and Safety Elements and Other Reactor Elements. The reactivity effects were measured by both the method of reverse multiplication of the counting rate in the start-up and experimental channels in the subcritical state (with a multiplication factor of 50-2000) and dynamic methods while varying the reactor power in the range up to 100 W. In the measurements of the reactivity according to the multiplication we took account of the different sensitivity of the detector to the neutrons from the source and the fission neutrons, the different neutron importance of the neutrons from the source and from the fission, the decay of the Po–Be source, and the contribution from spontaneous fission of ^{240}Pu. The last factor was found experimentally.

The reactivity was measured by dynamic methods with the analog reactimeter as well as by recording the signals from the chamber with a recording potentiometer or a loop oscillograph and subsequent processing of the data by computer. In some cases the reactivity was found from the established reactor riding-up time. The differential efficiency of the control and safety elements was measured by the overcompensation method.

*Deceased.

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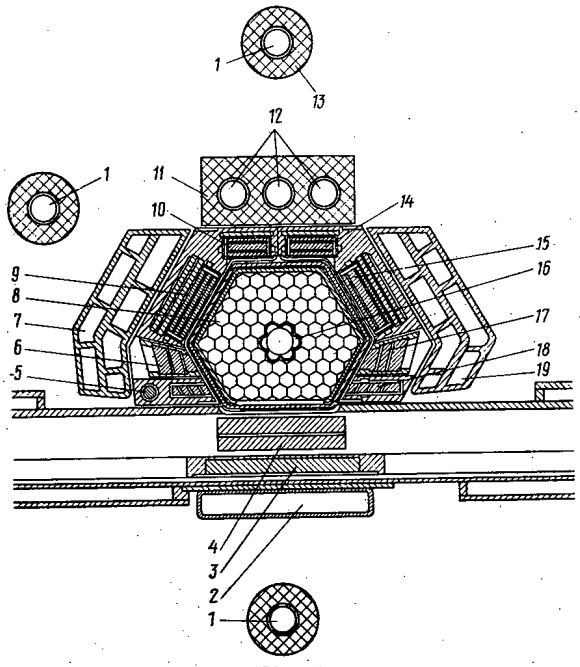

Fig. 1

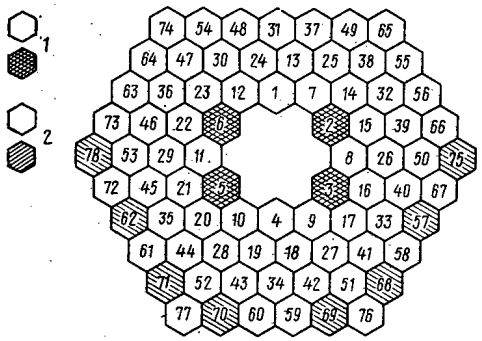

Fig. 2

Fig. 1. Cross section of IBR-2: 1) regular fission chambers; 2) water moderator behind moving reflector (MR); 3) beryllium insert of auxiliary moving reflector (AMR); 4) blade of main moving reflector (MMR); 5) rod of automatic controller (AC); 6) scram protection (SP-2); 7) water moderator behind reactivity compensator (RC-2); 8) reactor core; 9) reactivity compensator (RC-2); 10) slow emergency protection (SEP-2); 11) cold-moderator imitator (CMI); 12) experimental chambers; 13) moderator of regular chambers; 14) SEP-1; 15) RC-1; 16) experimental chamber in target channel; 17) SP-1; 18) intermediate controller (IC); 19) water moderator behind RC-1.

Fig. 2. Recorder chart of critical charge of reactor with cells containing fuel assemblies for: 1) "central" and 2) "peripheral" charges.

Analysis of the data from measurement of the efficiency of one and the same segments of the reactivity compensators (RC) by different methods reveals that the relative deviation of the results does not exceed 5%, i.e., is smaller than the error of the dynamic methods (±5%) and the error of the reverse multiplication method (±15%). It must be noted that the result of measurement of the detector position. Thus, the total efficiency of the SEP (see Fig. 1, 14) measured with a chamber in the CMI proved to be 1.5 times the efficiency measured by a chamber in the center of the reactor core. The efficiency of the RC and the moving reflector (MR) (see Fig. 1, 9, 15, 4) measured with the regular chambers situated behind these elements was below the efficiency measured by other chambers. This was caused by local perturbations in the neutron flux and spectrum.

The most complete measurements were made with a peripheral charge of fuel assemblies (Tables 1 and 2). With a central charge of fuel assemblies the efficiency of the RC and SEP units is almost 10% higher and the effect of the MR reactivity is also about as much higher than with a peripheral charge (Table 3). This is explained by the fact that with a central charge the reactor core is somewhat displaced towards the moving reflector and is further separated from the compensators and the SEP.

The measured total efficiency of the control and safety elements is lower than the design values by a factor of 1.5-1; this is explained by the fact that the calculations did not take account of some design details. Notwithstanding this, the efficiency of the control elements can be considered satisfactory whereas the efficiency of the SEP is inadequate since, should one element of the SEP fail, it would not ensure compensation of the temperature effect of the reactivity. Before the power start-up of the reactor the efficiency of the control and safety elements will be increased by changes in their design.

Figure 3 shows how the power of the reactor is affected by the tripping of one element of the SP. In 0.02 sec after a scram signal one element of the SP introduces a reactivity of $8 \cdot 10^{-4}$ k_{eff}, although its total efficiency is much higher (see Table 1). In the pulsed mode of operation the tripping of one SP element results in a decrease in the pulse energy by a factor of roughly 50.

TABLE 1. Efficiency of Control and Safety Elements

Elements	Total, 10^{-2} k_{eff}	Differential, 10^{-4}, k_{eff}/mm
RC-1	1,50±0,12	0,50±0,03
RC-2	1,55±0,12	0,52±0,03
IC	0,26±0,02	0,11±0,01
AC	0,032±0,002	0,011±0,001
SP*	0,14±0,01	—
SEP*	0,60±0,06	—

*One unit.

TABLE 2. Effect of Reactivity of Main and Auxiliary Moving Reflectors with Peripheral Charge of Fuel Assemblies*

Position of MR	Δk_{MMR}, 10^{-2} k_{eff}	$\alpha \cdot 10^4$, deg^{-2}	Δk_{AMR}, 10^{-2} k_{eff}	$\alpha_{AMR} \cdot 10^4$ deg^{-2}	Error, %	
					$\sigma_{\Delta k}$	σ_α
AMR at phys. c.†	1,76	1,01	—	—	3	1
AMR withdrawn	2,09	1,6	—	—	2	1
MMR at phys. c.†	—	—	0,39	0,1	2	20
MMR withdrawn	—	—	0,72	—	2	—

*See Eq. (1) and the relevant text in reference to the coefficient α.
†That is, in the physical center which is taken to mean the position of the AMR or MMR corresponding to their maximum efficiency.

TABLE 3. Effect of Reactivity of Main and Auxiliary Moving Reflectors of IBR-2 with Central Charge of Fuel Assemblies

Position of MR	Δk_{MMR}, 10^{-2} k_{eff}	$\alpha \cdot 10^4$, deg^{-2}	Δk_{AMR}, 10^{-2} k_{eff}	$\alpha_{AMR} \cdot 10^4$, deg^{-2}	Error, %	
					$\sigma_{\Delta k}$	σ_α
AMR at phys.c.	2,04	1,4	—	—	2	10
AMR withdrawn	2,41	—	—	—	1	—
MMR at phys.c.	—	—	0,45	—	2	—
MMR withdrawn	—	—	0,82	0,4	2	12

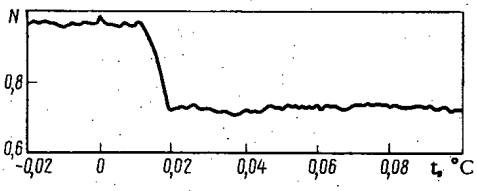

Fig. 3. Oscillogram of IBR-2 power in operation in steady-state mode and with one SP element dropped.

Experimental assessment of the effect of external hydrogeneous moderators, which are used to form the spectrum in neutron beams, on the efficiency of the control and safety elements showed that the greatest effect is exerted by the CMI (see Fig. 1). When it is moved further away, the efficiency of the SEP increases by 20% and the efficiency of the RC decreases by the same proportion. Other moderators (7 and 9 in Fig. 1) have practically no influence on the efficiency of the control and safety elements.

The MMR and AMR (Figs. 1 and 4) are used for periodic modulation of the reactivity and production of power pulses at a frequency of 50 and 5 Hz. Accordingly, particular attention is paid to measuring the effects of their reactivity. In measuring the angular dependence of the reactivity of the IBR-2 on the position of the moving reflector we rotated the reflector at a speed of 0.03-0.4 deg/sec and monitored the position with an error of no more than 0.07° for the MMR and 1° for the AMR.

Besides the dependence of the reactivity effects of the MR on the charge of the reactor core, the effect of the position of one of the reflectors on the efficiency of the other (see Tables 2 and 3) proves to be very substantial. The reason is that the AMR shades the MMR. The shadow effect has a particularly strong influence on the parabola coefficient which describes the behavior of the reactivity when the MMR is moved through small angles (± 3-4°) relative to the physical center:

$$\varepsilon(\varphi) = \varepsilon_p - \alpha\varphi^2, \quad (1)$$

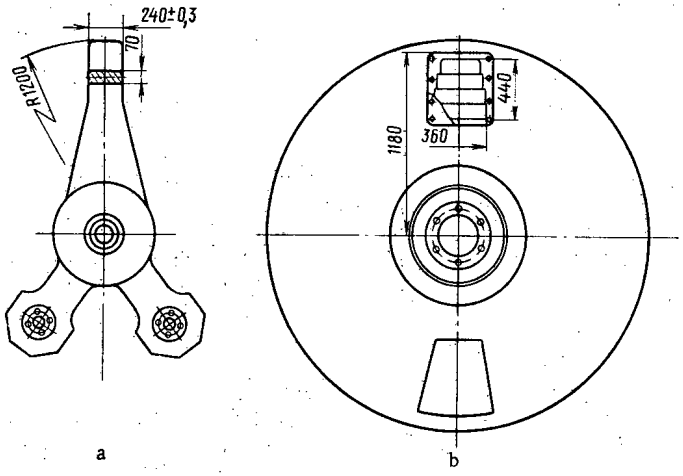

Fig. 4. Main (a) and auxiliary (b) moving reflectors of IBR-2.

where ε_p is the maximum prompt reactivity.

The value of α turned out to be smaller by a factor of 3 or 4 than the value found with allowance for the effect of the AMR. Withdrawal of the beryllium insert (especially the AMR) from the reactor, even through a small angle (~5°) results in a substantial increase in α. Indeed, in special experiments with an MMR we obtained $\alpha = 2.5 \cdot 10^{-4}$ deg^{-2} and the total efficiency proved to be practically equal to the calculated value of $2.65 \cdot 10^{-2}$ k_{eff} (Fig. 5).

Of the other measured effects mention should be made of the change in reactivity during axial displacement of the MMR, $(dk/dx)_{MMR}$; this determines the effect of the vibrations of the MMR rotor on the power fluctuations:

$$(dk/dx)_{MMR} = (4.5 \pm 0.3) \cdot 10^{-4} k_{eff}/\text{mm}$$

The measured differential efficiency when the entire MR machine is removed from the reactor core is

$$(dk/dx)_{MP} = (9.0 \pm 0.3) \cdot 10^{-4} k_{eff}/\text{mm}$$

and the total effect is ~0.06k_{eff}. The reactivity effect of the MMR counterweight proved to be substantial, i.e., 0.006k_{eff} (see Fig. 4).

In measurements of the reactivity effects of the fuel assemblies and their imitators (Table 4) the efficiency of the imitators on the periphery of the reactor core proved to be unexpectedly high. Experiments on measurement of the efficiency of metal specimens in the region of the MR confirmed the assumption concerning the comparatively high efficiency of copper in the spectrum of the IBR-2. The efficiency of specimens of steel, copper, tungsten, and beryllium with the same volume are in a ratio of 1.0:1.4:1.5:1.8, respectively.

Lifetime of Neutrons in Reactor. The mean lifetime τ of a generation of prompt neutrons in many ways determines the length of the power pulse at half maximum which can be calculated for the IBR-2 from the formula

$$\theta \approx 1.4 (\tau/\alpha v^2)^{1/3}, \quad (2)$$

where v is the speed of rotation of the MMR. The mean lifetime was measured by four independent methods. The least error was yielded by the Rossi α method [3]: 83 ± 2 nsec. The value of α measured from the shape of the power pulse turned out to be 90 ± 9 nsec whereas the values found from the power fluctuations in the pulsed [4] and steady-state [3] modes of operation were 80 ± 10 and 130 ± 10 nsec, respectively. Since the last result had a considerable systematic error due to the frequency characteristic of the measuring channel, it was not taken into account in our calculations of the mean value τ. The mean value τ found from

$$\bar{\tau} = \sum_i \tau_i \sigma_i^{-2} / \sum_i \sigma_i^{-2}, \quad (3)$$

where τ_i and σ_i^2 are the mean value and the variance of τ in the ith method coincided with the value obtained by the Rossi α method and amounted to 83 ± 2 nsec.

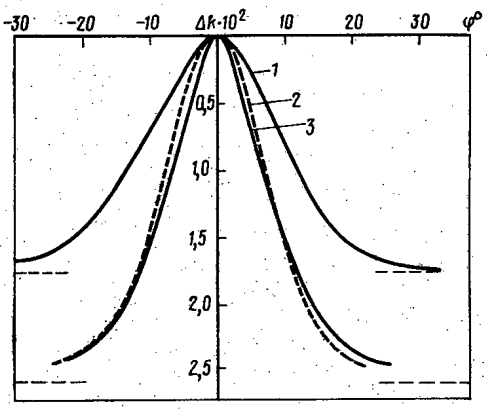

Fig. 5. Dependence of reactivity of IBR-2 on angle of rotation of MMR: 1) with regular AMR; 2) without AMR; and 3) in comparison with measurements on BF-S stand [3].

TABLE 4. Efficiency of Fuel Assemblies and Fuel-Assembly Imitators of IBR-2 Reactor, $10^{-2}k_{eff}$

Cell no.	Fuel Assembly	Fuel-assembly imitator	Cell no.	Fuel Assembly	Fuel-assembly imitador
1	1,16±0,04	0,13±0,02	58	0,96±0,03	0,31±0,04
4	1,21±0,04	0,10±0,02	59	1,02±0,03	0,30±0,03
5	1,20±0,04	0,10±0,02	61	1,07±0,03	0,30±0,03
8	1,16±0,03	—	63	0,92±0,03	0,27±0,03
14	1,07±0,02	0,09±0,02	74	0,78±0,04	0,26±0,03
20	1,28±0,05	0,10±0,02	75	0,65±0,03	0,20±0,03
31	0,99±0,02	0,28±0,04	77	0,86±0,03	0,27±0,03
35	1,18±0,05	0,15±0,02			

The great difference between the measured τ and the value calculated by the Monte Carlo method (43 nsec) [1] is attributed primarily to the effect of the AMR disk, the CMI, and the MR vessel, which was not taken into account in the calculations. Measurements by the Rossi α method, carried out without these elements, yielded τ = 47 nsec.

Neutron Spectrum and Flux in External Beams and Power Distribution in Reactor. The spectrum of fast and resonance neutrons was measured from the activation of threshold detectors with subsequent reconstruction of the initial energy distribution of the neutrons by computer [5]. In the intermediate portion of the spectrum satisfactory agreement was obtained with calculated data (Fig. 6).

The density of the thermal-neutron flux at the external surface of the moderator was found by the cadmium difference method from the absolute activity of copper and gold tracers [5]. The absolute activity of the tracers was measured by calibrated sensors and refined on a $\beta-\gamma$ coincidence arrangement. The measured density of the thermal-neutron flux, $(3.0 \pm 0.2)\cdot 10^6$ neutrons/(cm^2 · sec · W), proved to be roughly double the calculated value.

The power distribution in the reactor core was found from the γ activity of the fission products. The activity of the fuel assemblies was measured over a period of two months after irradiation. The measured distribution in the main is in agreement with the design data; the coefficients of nonuniformity of the power distribution over the height and volume of the reactor core were 1.3 and 1.6, respectively. The largest local deviations of the measured and calculated distributions are in the region of the reactor core adjacent to the MR as well as in the region in which slow neutrons from the moderator enter the core.

Pulse Criticality of the Reactor. In comparison with the attainment of delayed criticality under steady-state operating conditions, the attainment of criticality under pulsed operating conditions is characterized by distinctive features. The ratio of the mean power of the subcritical pulsed reactor to the power of the external source is given by [4]

$$y = W/S_0 = (1/\beta)(K_p/1 - K_p), \quad (4)$$

where K_p is the so-called pulsed neutron-multiplication factor which depends in an involved manner on the maximum prompt reactivity ε_p [see Eq. (1)]; $K_p = 1$ when $\varepsilon_p = \varepsilon_{p_0}$ (here ε_{p_0} is the equilibrium pulsed critical-

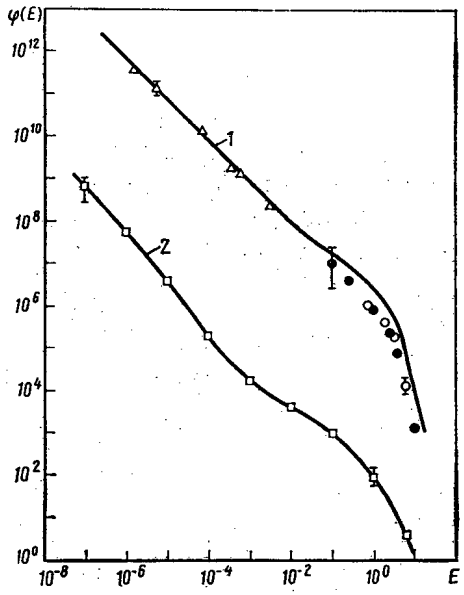

Fig. 6. Neutron spectrum (neutrons/cm^2·sec·MeV·W) in external beam of IBR-2: 1) at surface of moderator; and 2) at distance of 8 m from it; △, □, ○, ●) experiment; ———) calculation.

ity), and $K_i \ll 1$ when $\varepsilon_p < 0$. Extrapolation to criticality according to the inverse multiplication $y^{-1} = S_0/W$ of the mean power (i.e., in measuring the mean counting rate of the detectors) in the pulsed reactor would give an overestimation of the subcriticality since $y^{-1}(\varepsilon_p)$ has a negative second derivative with respect to the reactivity ε_p.

The mean power W of the reactor consists of the power W_0 liberated in pulses and the power W_b liberated between pulses. Correspondingly, we can introduce the concepts of multiplication in the pulse ($y_0 = W_0/S_0$) and multiplication in the background ($y_b = W_b/S_0$). The safest and most precise extrapolation to criticality is ensured by measuring the inverse multiplication $y_0^{-1} = S_0/W_0$. In this case W_0 is measured by the recorded counting rate of the detectors, periodically switched on for the duration of the main pulse; the time T for which the detectors are switched on should be equal to the time during which the reactivity introduced by the MMR is described by the parabola (1). In the region below the level of prompt criticality (to be more precise, when $\varepsilon_p < -10^{-3}$) the multiplication in the pulse is

$$y_0(\varepsilon_p) = \frac{1}{1-K_p} \frac{2n}{\sqrt{|\varepsilon_p|\alpha v^2}} \operatorname{arctg}\left(\frac{T}{2}\sqrt{\alpha v^2/|\varepsilon_p|}\right), \tag{5}$$

where n is the pulse-repetition frequency whereas in the region of deep subcriticality ($|\varepsilon_p| \gg \alpha v^2 T^2$) it is

$$y_0(\varepsilon_p) = \frac{nT}{1-K_p} \frac{1}{|\varepsilon_p|} \sim 1/|\varepsilon_p|. \tag{5a}$$

It follows from Eqs. (4) and (5) that W_0 depends more on ε_p than W does [$K_p(\varepsilon_p)$ is a very weak function of ε_p for $\varepsilon_p < 0$]. Moreover, for pulsed operating conditions are characterized by the following interesting features: The reactor state extrapolated from the region of deep subcriticality is one of prompt criticality: $\varepsilon_p = 0$ [see Eq. (6)].

The reactor is brought up to the critical state ($\varepsilon_p = \varepsilon_{p0} \sim 10^{-3} k_{eff}$) in the following manner. First, with the control and safety elements withdrawn, the reactivity modulator was started up. When the MR had reached the nominal number of revolutions, the RC were introduced gradually. After each rise in reactivity the counting rate in a pulse in 500 sec, i.e., the multiplication y_0, was measured and the position of the RC corresponding to the expected criticality was estimated. With a pulse-repetition frequency of 5 Hz, when both MR are rotating, and with a small multiplication y_0 we made a rough synchronization of the rotors by finding an AMR rotation phase such that corresponded to the maximum counting rate. The final synchronization was carried out in the critical state.

Good agreement was obtained between the measured and calculated values of the function $y_0^{-1}(\varepsilon_p)$ (Fig. 7). The calculated relations were obtained on the basis of Eq. (5) by using the measured values of α. A pulsed mode was achieved only with peripheral charging of fuel assemblies. At a pulse-repetition frequency of 50 Hz the equilibrium pulsed supercriticality was $0.9 \cdot 10^{-3} k_{eff}$ whereas at a frequency of 5 Hz it was $1.3 \cdot 10^{-3} k_{eff}$ (see Fig. 7).

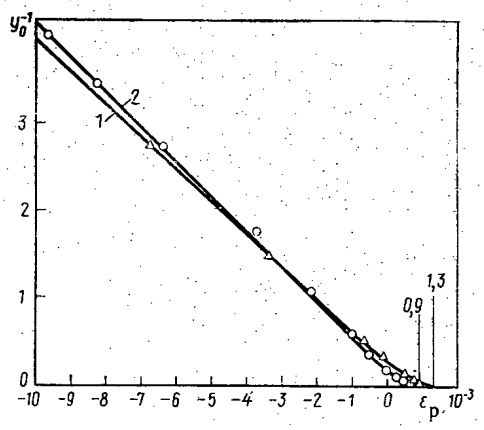

Fig. 7

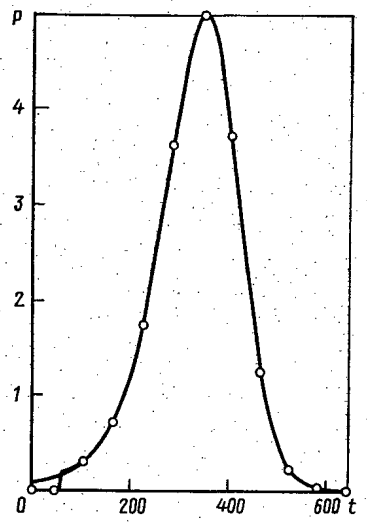

Fig. 8

Fig. 7. Dependence of inverse multiplication in pulsed mode on prompt reactivity: ———) calculation at frequencies of (1) 5 Hz and (2) 50 Hz; ○, △) experiment.

Fig. 8. Measured (○) and calculated (———) shape of IBR-2 power pulse at frequency of 50 Hz.

Shape of Power Pulse. The shape of the power pulse from the IBR-2 is one of its principal characteristics which determines, along with the intensity of the neutron flux, the efficiency of the reactor as a source of neutrons for time-of-flight neutron spectroscopy.

The shape of the spike of fast neutrons was measured in detail in different states of the reactor by the counting method with a multichannel time analyzer and by the current method by recording the detector signal from an individual pulse of power on the screen of a storage oscillograph. The fast-neutron detectors used were ^{238}U fission chambers in the reactor core and thorium fission chambers in the extracted neutron beam as well as a plastic scintillator with a photomultiplier which were also set up in the beam.

In the process of bringing the reactor up to criticality the shape of the pulse was measured by the counting method. At a subcriticality of $1 \cdot 10^{-2}$ and $2 \cdot 10^{-2} k_{eff}$ the pulse width at half maximum θ was equal to 720 and 824 μsec, respectively. In the case of prompt criticality θ = 240 μsec at maximum reactivity, which is in good agreement with the calculated value of 244 μsec obtained by using the experimental values of α and τ.

In the state of pulsed criticality the measurements were carried out mainly by the current method which in the given case is more exact than the counting method (error no greater than 2%). At a pulse frequency of 50 Hz (AMR immobile) the value of θ for fast neutrons was 220 μsec and at a frequency of 5 Hz it was 198 μsec. The fact that the measured value of θ considerably exceeds the design value (92 μsec for 5 Hz) is due to the difference between the actual values of α and τ and the calculated values. Numerical solution of the one-point kinetic equation for the reactor with experimental α and τ gives a pulse shape which practically coincides with the measured shape (Fig. 8).

Clearly, the pulse is lengthened by those elements of the motor construction which have a considerable effect on α and τ. It was established that the greatest effect is exerted by the AMR. The measured dependence of θ on the position of the AMR is plotted in Fig. 9. The AMR position at the physical center, i.e., corresponding to the highest efficiency, was taken to be zero. Withdrawal of the AMR from the core results in a marked shortening of the pulse length θ, from 190 to 150 μsec. Special experiments established that this occurs mainly because of an increase in α_{MMR} and partially because of a shortening of τ. The pulse length is also reduced considerably when the external moderators are withdrawn. The moderator behind the AMR increases by 30 ± 4 μsec and the moderator behind the blocks of the SEP, by 8 ± 4 μsec.

Power Fluctuations of IBR-2 in the Pulsed Mode. Because of the high sensitivity of the pulsed reactor to reactivity variations during physical start-up a detailed study was made of the character of the power fluctuations as well as their correlation with other random processes affecting the reactivity (MR vibrations, etc.). Data acquisition took place by recording discrete signals from the sensors on magnetic tape. The recorded

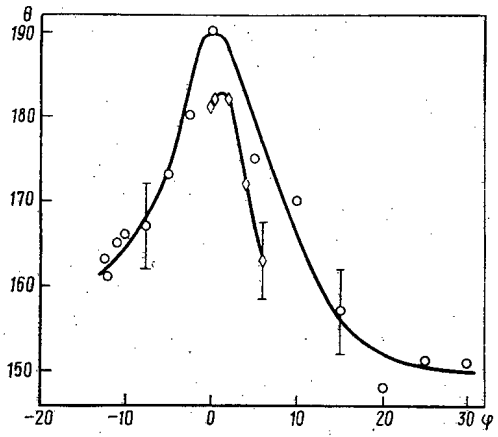

Fig. 9

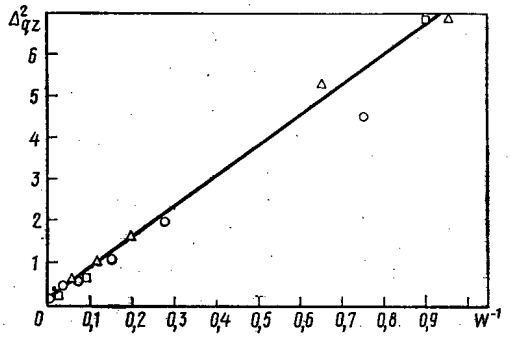

Fig. 10

Fig. 9. Dependence of length of θ of IBR-2 power pulse on AMR position φ at frequency of 50 Hz: ○) no moderator behind AMR; ✧) no moderators at all.

Fig. 10. Dependence of relative variance Δ^2_{qz} on inverse mean reactor power W^{-1} at 50 Hz: △) presence and ○) absence of water moderators; at 5 Hz in the absence of water moderators □); ——) approximation according to Eq. (6).

information was processed by computer with the aid of a special program for the analysis of steady-state random processes in a periodically operating pulsed reactor. With a low mean reactor power (under 10 W) stochastic fluctuations of the pulse energy predominate [4]. The experimental dependence of the relative variance Δ^2_{qz} of the pulse energy on W is, in accordance with the theory, linear in character (Fig. 10) and for W > 0.5 W is approximated by the function

$$\Delta^2_{qz}\left(\frac{1}{W}\right) = \Delta^2_{qz}(0) + \frac{750 \pm 50}{W} \cdot 10^{-4}, \qquad (6)$$

where $\Delta^2_{qz}(0)$ is extrapolated value of the variance, characterizing the fluctuations, which is due to the deviations of the reactivity owing to vibrations of the elements of the reactor construction. At a frequency of 50 Hz the variance $\Delta^2_{qz}(0) = (3 \pm 2)\cdot 10^{-4}$ and at 5 Hz it is $(5 \pm 2)\cdot 10^{-4}$. The stochastic fluctuations reach a maximum of in the region of prompt criticality for W ≈ 0.02 W; $(\Delta^2_{qz})_m = 0.30$. All the experiments on studying $\Delta^2_{qz}(0)$ were performed at a power of 300 W, which made it possible not to take account of the stochastic fluctuations of the pulse energy. It turned out that the dominating contribution to $\Delta^2_{qz}(0)$ is made by the transverse displacements of the MMR. It is seen from Fig. 11, which gives the results of spectral-correlation analysis, that the interrelation between the pulse energy and the MMR displacement is due to the resonance peak (in the spectrum of MMR vibration frequencies) at 16 Hz, which is exactly one-third the rotational frequency of the MMR. This is exactly the transmission ratio of the MMR reducing gear train which apparently is the source of vibrations of the rotor and, consequently, the fluctuations of the reactor power. However, the fluctuations caused in the pulse power by these vibrations are extremely small (~2%). A detailed spectral-correlation analysis was also carried out for other processes which affect the reactor power. As a result of these investigations it was established that:

a) the oscillations in the rotational velocity of the MMR and the corresponding power fluctuations are very small (the relative standard deviation does not exceed 0.2%);

b) the circulation of water in the moderators results in reactivity fluctuations not exceeding $5\cdot 10^{-6} k_{eff}$, which corresponds to power fluctuations ~2.5%. A large part of the oscillation spectrum lies in the range of low frequencies (f < 0.5 Hz);

c) with the reactor operating with a pulse frequency of 5 Hz, when the AMR also rotates, additional power fluctuations appear because of torsional oscillations of the AMR disk which are almost periodic. The spread they cause in the power-pulse amplitudes is no greater than 0.4%. The character of the MMR oscillations in 5-Hz modes differs substantially from that in 50-Hz modes.

Neutron-physical investigations with a "dry" (no coolant) start-up confirmed the possibility of realizing the principal design characteristics of the IBR-2 as a source of neutrons for physics research. The difference

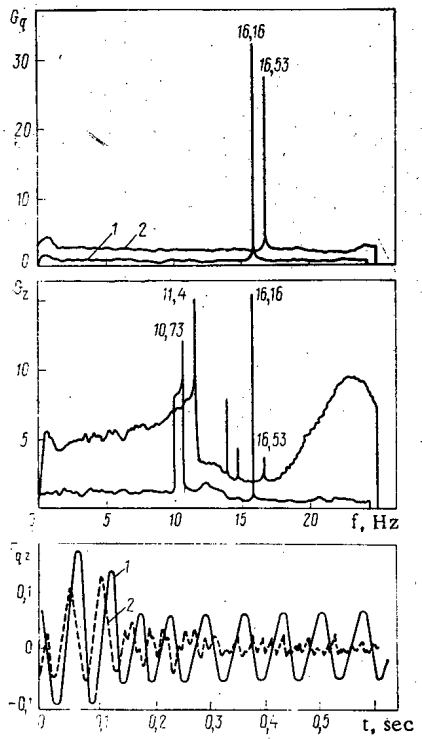

Fig. 11. Spectral densities G_q of pulse energy and transverse displacements G_z of MMR and intercorrelation function Γ_{qz} of processes q and z at frequency of: 1) 48.5 and 2) 49.6 Hz; t is the time between pulses.

of some physical parameters is due in the main to the fact that the calculations did not make allowance for auxiliary elements of the reactor construction. Measures for further improvements in the reactor characteristics have been planned on the basis of the results of the physical start-up and are being implemented.

In conclusion, the authors express their profound gratitude to all the teams and individual workers who participated in the preparation and execution of the physical start-up of the reactor.

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LETTERS

APPLICATION OF COMPUTER TOMOGRAPHY FOR FUEL-ELEMENT INSPECTION

É. Yu. Vasil'eva and A. N. Maiorov

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The technological process of fabricating rod-type fuel elements includes a number of operations of quality control and inspection of a number of parameters of the internal structure of the product. With traditional industrial radiography the two-dimensional image of the macrostructure of a product makes it difficult to interpret the results. Because of the complexity of the technique and the apparatus involved, stereoradiography has not hitherto found widespread application [1]. New possibilities are opened up by tomography, which produces sectional images of the internal macrostructure of the product. The method of transverse transmission computer tomography developed in 1971-1973 has been extensively used in medicine [2-5]. In this method the object is irradiated by a beam of radiation at various angles and then recording the information obtained and processing it by computer [3, 6]. This method makes it possible to obtain fundamentally new information about the internal structure of the object, both quantitative information and information shown on a display [3-5].

Radiographic data obtained by repeated x-raying of a rod-type fuel element with the element rotating in coordination with the motion of the film may be a basis for reconstructing an image of the internal structure. The x-ray photograph made with the fuel element turned through various angles to its longitudinal axis consists of a set of projections of the distribution function of the density of material inside the object on the plane of the film, provided that the beam of radiation can be assumed to be parallel. This condition will hold for an x-raying arrangement with $F \gg h$, where F is the focal distance and h is the height of the object in the x-raying direction. Reading the information from the film along a chosen direction AA', for one and the same object we get projections of the function from various angular directions α in the plane sought (Fig. 1). In this case the number of planes is determined by the scanning pitch of the reader. The information obtained can be written as

$$S(\alpha, p)=\tau \int_{L(\alpha, p)} \rho(x, y)\, dl, \tag{1}$$

where $S(\alpha, p)$ is the projection of the function (density of the film blackening); $\rho(x, y)$, sought function (distribution of density of material inside object); $L(\alpha, p)$, a straight line along which the projection of the function is determined; τ, a factor coupling the density of the material with the blackening of the film; α, angle of projection; and p, a parameter giving the straight line $L(\alpha, p)$ in the normal form.

Equation (1) is a Radon transformation of function $\rho(x, y)$ with respect to the hyperplane $L(\alpha, p)$. Integral geometry considers the relation between functions in space and the integrals of these functions over all possible hyperplanes [7]. Equation (1) can also be written as

$$S(\alpha, p)=\tau \int \rho(x, y)\, \delta[p-(x\cos\alpha+y\sin\alpha)]\, dx\, dy, \tag{2}$$

where $\delta(x)$ is the Dirac delta function and $p_\alpha = x\cos\alpha + y\sin\alpha$ is a parameter giving the straight line $L(\alpha, p)$ which passes through the point with coordinates (x, y).

The problem thus is to find the function $\rho(x, y)$ from the values of the function $S(\alpha, p)$ taken along any straight line $L(\alpha, p)$. The sought function can be calculated from the values of its integrals, taken over all possible hyperplanes, from the formula for the Radon transformation [6, 7] recast as

$$\rho(x, y)=\frac{1}{2\pi^2\tau}\int_0^{\pi} d\alpha \int_0^{\infty} \frac{1}{p^2}\,[2S(\alpha, p_\alpha)-S(\alpha, p_\alpha+p)-S(\alpha, p_\alpha-p)]\, dp. \tag{3}$$

Equation (3) formed the basis of an algorithm developed for reconstructing the images of the internal structure of a body with transmission computer tomography [8]. The inner integral was taken by the trapezoid method and the outer integral by the rectangle method and Eq. (3) was rewritten as

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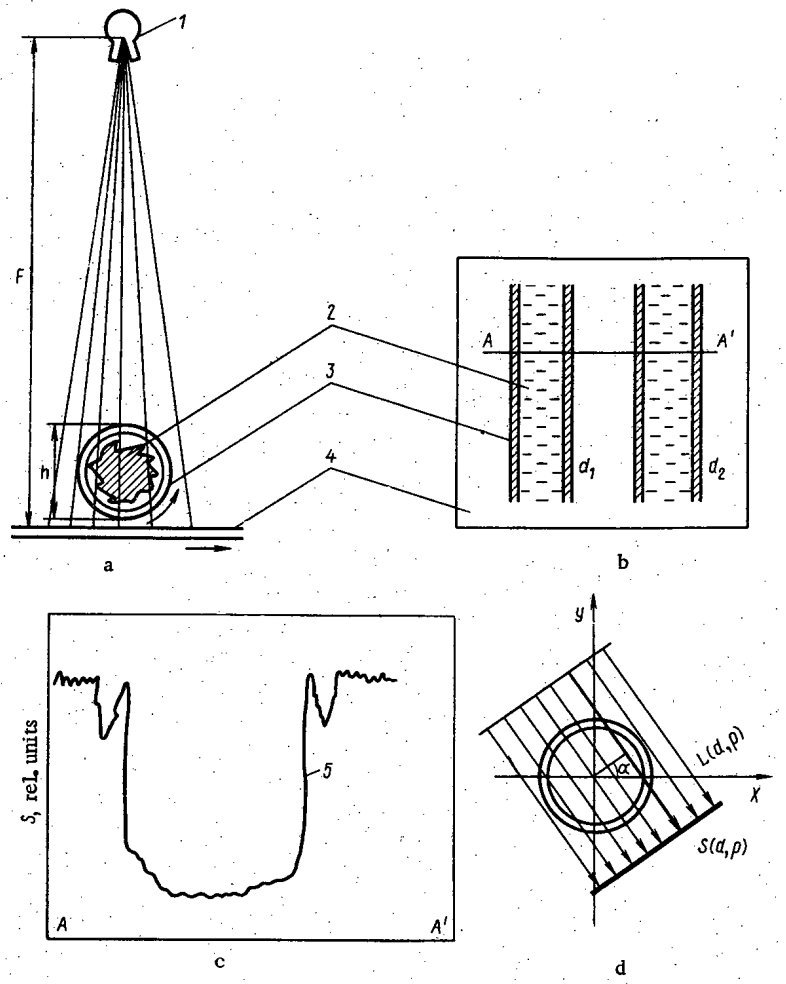

Fig. 1. Arrangement for obtaining radiographic information for reconstructing tomographic image: a) geometry of experiment; b) radiograph at various angles of rotation of fuel element [1) x-ray tube; 2) fuel; 3) fuel element; 4) x-ray film]; c) reconstruction along AA' direction for one fuel element [5) projection of $\rho(x, y)$ function]; d) explanation of Eq. (1).

$$\rho(x, y) \simeq \frac{1}{\tau 2\pi m} \sum_{j=1}^{m} \left\{ \frac{2S(p_\alpha)}{D} + \frac{1}{2h} \left[\frac{2S(p_\alpha) - S(p_\alpha + h) - S(p_\alpha - h)}{1} + \right. \right.$$

$$\left. \left. + 2 \sum_{i=2}^{n-1} \frac{2S(p_\alpha) - S(p_\alpha + ih) - S(p_\alpha - ih)}{i^2} + \frac{2S(p_\alpha) - S(p_\alpha + nh) - S(p_\alpha - nh)}{n^2} \right] \right\}_j, \quad (4)$$

where D is the diameter of the region of reconstruction; m, number of intervals with respect to α from 0 to π; h, step of the partition of the integration range; and n, number of partition points on the line of projection of the function.

A program for the M-222 computer was devised on the basis of Eq. (4). With an input array (number of counts over all projections of the function) of 4096 numbers and with reconstruction picture dimensions of 64 × 64 cells the reconstruction time is 7-10 min with a computer speed of 20,000-25,000 operations per second. Increasing the capacity is related to the computer speed and at 1 million operations per second the processing time amounts to only a few seconds.

The function is also reconstructed from its projections by other mathematical methods of processing, including Fourier analysis, solving systems of linear equations, direct iterative methods, etc. [9-12]. They have become an object of extensive discussion, the central point being that of comparing methods as to accuracy and speed. The lack of standard comparison techniques has hitherto not allowed an optimal method to be chosen. The use of the Radon transformation apparently is preferable for the following reasons:

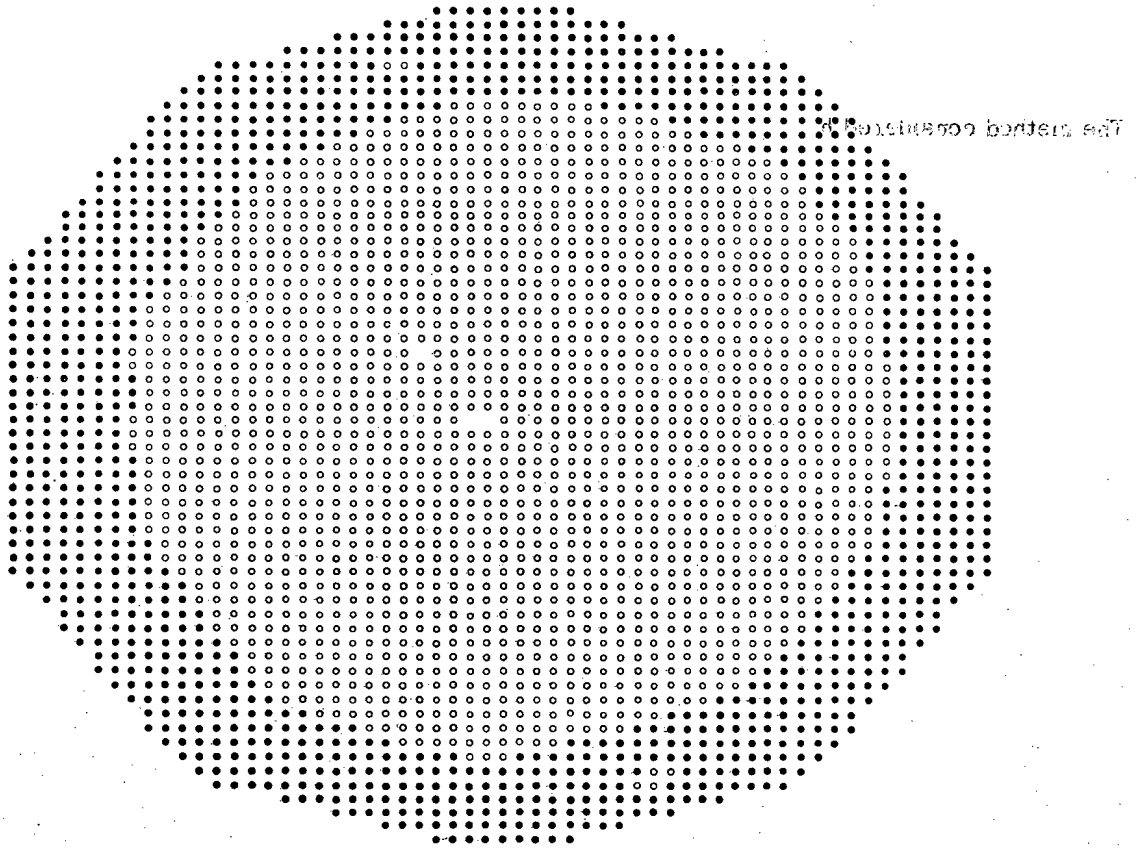

Fig. 2. Reconstructed image of cross section of rod-type fuel element with $\Delta\alpha = 30°$, $\Delta p = 0.055$ mm (pitch of reconstruction grid); ●) fuel-element can; ○) filling of fuel element.

a) being in a single-valued relation with the Fourier transformation, the Radon transformation does not require a two-fold procedure in reconstruction, i.e., a transition to the space of spectra and then back again, and consequently this algorithm is more economical in respect of computational operations;

b) the use of the method of solving systems of linear equations puts increased requirements on the computer since the computational matrix comprises $\sim 10^8$ numbers, whereas the computational algorithm based on the Radon transformation can be realized on Soviet-made computers of the M-222 type;

c) the time required for data processing with iterative methods is many times longer than the reconstruction time necessary with the Radon transformation.

The accuracy of reconstruction for all these algorithms is the same. Analytical comparison of the reconstruction methods [13] showed that the Radon transformation is the ultimate form of the solution to this problem.

The tomographic method of inspection was verified experimentally with several imitators of fuel elements, resembling the fuel elements of the VVÉR-1 water-moderated–water-cooled power reactor [14]. The diameter of the sand-filled imitators was 10.2 mm with a can thickness of 0.65 mm. Radiographic inspection was carried out by x-raying the imitators. The x-ray tube of the RUP-150/300-10-1 x-ray machine was ~700 mm above the film. The imitators were placed right next to the film and turned through an angle $\alpha = 30°$. A special reader was used to process the radiographic data [15]. The aperture of the reader in the scanning direction had an opening of 50 μm. The scanning pitch was varied from 0.2 to 0.055 m. The number of points in the collection of data for each projection of the function from the x-ray picture varied from 30 to 180 counts. Figure 2 shows the can and the filling of a rod-type fuel element in cross section with 0.1-mm cells in the reconstruction grid. The results confirm that the internal structure of fuel elements can be reconstructed by transverse transmission tomography. Further studies on the resolution and the sensitivity (contrast) to various defects are necessary, however, for practical application; moreover, it is necessary to optimize the discretization of data acquisition over angular and pitch intervals as well as the conditions for obtaining radiographic information for reconstruction of the tomographic image.

Research on computer tomography (body-section radiography) for medical purposes [3-5, 6, 8, 12, 13] permitted the conclusion that the minimum size of a detected defect can be comparable to the size of the counting window, i.e., the resolution will be 50-100 μm.

The method considered here can be used to analyze the uniformity of the distribution of fuel inside fuel elements in developing new designs or new technologies for fabricating fuel elements. Such information can also be obtained by using neutron radiography which is employed in a number of research centers [16-18].

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PROSPECTS FOR THE USE OF CARBON–CARBON-TYPE OF MATERIALS IN NUCLEAR POWER ENGINEERING

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Carbon (in the form of artificial graphite) has found wide application in nuclear reactors. Besides its traditional use as moderator and reflector of neutrons, it has been proposed to apply different technological modifications of graphite also as a construction material – in the blankets of thermonuclear reactors and in high-temperature gas-cooled reactors [1-3] for the jackets of fuel elements, matrix material, etc. Therefore, the need arises for a broader consideration of carbon materials from the point of view of the optimal choice of the kind of graphite for a specific technical problem. The principal types of carbon–graphite materials and their properties are given in Table 1 [4-6]. An appreciable effect of the original properties and peculiarities of the technology on the characteristics of the irradiated material has been noted [5-7]. The most significant alteration of the properties is observed already for a small flux of fast neutrons (~$1\text{-}2\cdot10^{20}$ neutrons/cm^2), after which the properties are relatively stabilized, and new changes arise at a higher flux (~10^{22} neutrons/cm^2).

*Deceased.

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TABLE 1. Basic Properties of Carbon–Graphite Materials

Material	Density, g/cm³	Strength limit, kgf/mm²			Modulus of elasticity E·10³, kgf/mm²	Anisotropy, $\alpha_\perp/\alpha_\parallel$	Electrical resistivity ρ, Ω·mm²/m	Thermal expansion coefficient, α·10⁶ deg⁻¹				Thermal conductivity, λ (W/m·deg K)			
		compression σ_c	bending, σ_b	tension, σ_t				300–400 °K	300–1300 °K	300–1800 °C	300–2300 °K	300 °K	800 °K	1500 °K	2000 °K
GMZ graphite	1,6—1,7	3,40 3,30	1,30 1,10	0,90 0,70	0,65 0,50	1,11	9,6 11,3	3,7 4,1	5,1 5,4	5,5 6,1	5,8 6,3	120 103	68 52	39 32	33 27
ARV graphite	1,6—1,7	4,5	1,5	1,2	0,60	1,36	14,8	4,5 3,3	5,8 4,6	6,3 5,1	6,7 5,4	86 97	58 69	42 48	36 41
MPG-6 graphite	1,72—1,85	10,0	5,3	3,3	1,05	1,11	14,2	6,6 6,0	7,9 7,2	8,5 7,7	8,8 8,1	95	62	43	37
Pyrographite (UPV-1)	2,15—2,23	30—35	10,0	5,0	2,7	25	4,3 6000	—0,4 21,0	0,9 24,0	1,4 25,0	1,7 25,0	330 3,2	220 1,7	150 1,4	125 1,45
Glass carbon (SU-1300)	1,45—1,52	26,0	8,35	5,2	2,7	1,0	4,4	2,3— 2,4	3,5	4,05	4,4	7,0	8,3	10,0	11,0
Carbon–carbon-type of material (KUP-VM)	1,40—1,45	42,0	50,0	29,0	13,5	10	15,8	—0,5 5,6	0,6 7,0	1,0 —	1,4 —	21,0 1,5	14,0 1,8	10,0 2,00	9,0 2,20
CSF (U.S.A.)	1,66	5,25 4,55			1,70 0,70	2,95		1,9 5,6							
PGA (England)	1,65—1,75	3,10 3,30	1,40 1,05		1,30 0,80	1,80		2,6 4,7							

Notes: 1. The values of the characteristics σ_c, σ_b, σ_t, E, $\alpha_\perp/\alpha_\parallel$, and ρ are given for room temperature.

2. The data of the measurement in the direction parallel to the bonding (deposition) axis are given in the numerator, and that for measurement in the perpendicular direction are given in the denominator.

As the temperature increases, the effect of irradiation decreases, although, e.g., in the case of pyrographite [8] joining of defects occurs at an irradiation temperature >800°C, and a change in shape increases. One can expect that graphite having the highest mechanical strength, a high modulus of elasticity, enhanced isotropicity of properties, and a small thermal expansion coefficient, and which is subject to heat treatment at a temperature higher than 2000°C [6, 7], will have the best durability. Thus one should assume the application of carbon–carbon-type of materials with a high-modulus fiber, pyrolytic graphite, and glass carbon to be promising.

Only pyrographite from among the indicated materials has been investigated under conditions of irradiation [5, 8-10]. Therefore, it is advisable to investigate the basic possibility of using carbon–carbon-type materials (KUP-VM) under conditions of irradiation. Other materials were irradiated simultaneously, namely: pyrolytic graphite (PGV), glass carbon (SU-850, SU-1300), and two domestic graphites, ARV and MPG-6. Tests were conducted in two temperature ranges: 100-200°C ("cold" tests) and 850-950°C ("hot" tests). The dimensions and shape of the samples, as well as the test conditions in loop channels, have been given earlier [10]. In the first stage of the investigations the values of the flux were $2 \cdot 10^{20}$ neutrons/cm², which usually corresponds to the onset of stabilization of defect formation.

After the tests the samples were visually inspected, their geometrical dimensions were measured, and they were weighed. The density of the material was determined by the method of hydrostatic weighing. In the case of measurement of the electrical resistivity of rod samples the error was not greater than 1.5%. X-ray structural analysis of the samples was performed on a DRON-1.5 apparatus with the application of a quartz monochromator. The lattice parameter was calculated from the (006) and (110) reflections. The error in the measurement of the interplanar distance was ±0.005Å. The density of all the investigated materials declines after the radiation tests. The largest density changes (~6%) are noted for pyrographite. Upon an increase in the temperature of the radiation tests the density of the majority of the materials increases again. High-temperature irradiation of PGV results in a decrease in the tetragonality of the structure: "*a*" decreases, and "c" increases by ≈1.1%. Metallographic investigations of PGV (T_{ir} = 100°C, T_{ir} = 900°C) did not allow revealing any kind of significant changes in the microstructure after irradiation. Thus, material based on a high-modulus

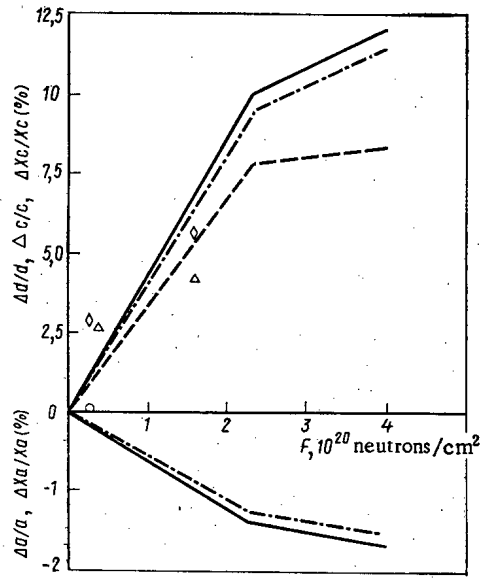

Fig. 1. Effect of radiation dose on changes in the linear dimensions Δ_a^x/a and Δ_c^x/c (–), density $\Delta^{d/d}$ (– – –), and the parameters of the crystalline lattice $\Delta^{c/c}$, and $\Delta^{a/a}$ (–·–) of pyrographite and of a carbon–carbon-type of material $\Delta^{d/d}$ in the case of an irradiation temperature of 100-150°C (data of this paper: △, ◇) pyrographite; ○) KUP-VM).

fiber obtained by the winding method showed good working qualities after irradiation both at 100°C and at 900-950°C (it preserved its integrity, stiffness, and mechanical strength with a minimal change in shape). Pyrolytic graphite exhibited a significant change in shape (see Fig. 1) [6, 8] as well as splitting and lamination failure of ring samples. Glass carbon underwent definite shrinkage, although it preserved its structure.

And so one can speak of the prospect of the use of carbon–carbon-type of materials in high-temperature and thermonuclear reactors.

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ELECTROMAGNETIC CONVERTER FOR FLOW RATE OF LIQUID METAL IN FUEL ASSEMBLIES

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UDC 681.128

In investigations on the hydrodynamics and heat exchange in models of reactor cores it is necessary to measure the flow rate in the cells of the lattice of a fuel assembly. The device described in [1] can be used to successively determine the flow rate in cells adjoining a given fuel element. To do this the fuel element must be turned about its axis along with the converter. A major disadvantage of the converter is that its output signal depends on the flow rate in neighboring cells as well as the given cell. It is difficult to take account of the contribution from neighboring cells to the output signal, especially under nonstationary conditions. A converter with a copper ring, shunting the induced voltage in neighboring cells [2], allows the influence of the neighboring cells to be eliminated, thus making it possible to measure the azimuthal distribution of the velocity in the cells by rotating the fuel element with the converter about its axis. This converter cannot be used to measure the flow rate in cells under nonstationary conditions.

A converter with a multipolar magnet (Fig. 1) has been proposed for simultaneous measurement of the flow rate in several cells [3]. The magnetic system has an alternating polarity and the number of poles is chosen to be equal to the number of cells adjoining the fuel element in which the converter has been set up. The electrodes are welded to the wall between the poles of the magnetic system. Their ends are led out beyond the limits of the experimental segment and are connected to a secondary instrument. As it moves through the space between the fuel elements the liquid interacts with the magnetic field and a potential difference is set up between each pair of neighboring electrodes, this potential difference being in a single-valued relation with the flow rate in the respective cell. The values of the flow rate in each cell at any moment of time can be assessed from the simultaneous recording of signals from each pair of electrodes.

By using such a converter we can take account of the effect neighboring cells have on each other and obtain a true picture of the distribution of flow rates over the cells. In view of the fact that an analytic solution of the problem for a given geometry is very difficult to get, let us consider the equivalent electrical circuit of the converter (Fig. 2). Here E_1-E_6 are the emf's induced in the liquid in the respective cells and are proportional to the flow rate in them, r_1-r_6 are the internal resistances of the sources of emf, i.e., the liquid metal, R_1-R_6 are the resistances of the load, i.e., of the walls of the converter and the adjoining fuel elements, shunting the emf, and U_1-U_2 are the measured voltages; the emf induced in a given cell is proportional to the flow rate and the magnetic induction in the cell, i.e., $E_i = kB_iQ_i$. If the geometric dimensions of the cells, as well as the magnetic induction in them, are the same, then the coefficient k is the same for all the cells. Its value can be found by calibration by measuring the total flow rate through a bundle (e.g., in the feed pipe) and associating it with the emf of all the cells. Then, in order to get the numerical values of the flow rate in each cell it is sufficient to know the induced emf.

Setting up and solving the system of Kirchhoff equations for the equivalent circuit, we can find the relation between the measured voltage U_1-U_6 and the sought emf:

$$E_i = U_i(1 + R_i/r_i) + \frac{(-1)^i R_i}{\sum_{i=1}^{n} R_i} \sum_{i=1}^{n} (-1)^{i-1} U_i (R_i/r_i). \qquad (1)$$

This formula is valid for any odd value of n. Since no constraints were placed in R_i and r_i in the derivation of the formula, it is also valid for cells of different shape.

The converter is intended mainly for measuring the flow rate in bundles with identical cells, i.e., can be used in all cells except for peripheral cells. If the cells in the bundle are identical, then

$$R_1 = R_2 = \ldots R_6; \quad r_1 = r; = \ldots r_6; \quad R_i/r_i = R/r;$$
$$R_6/\sum_{1}^{6} R_i = 1/6;$$

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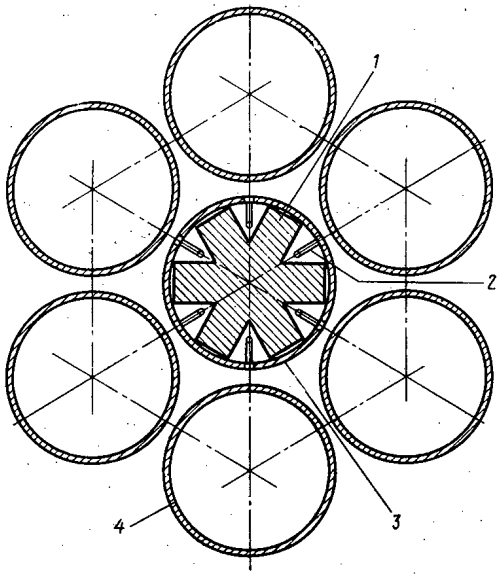

Fig. 1

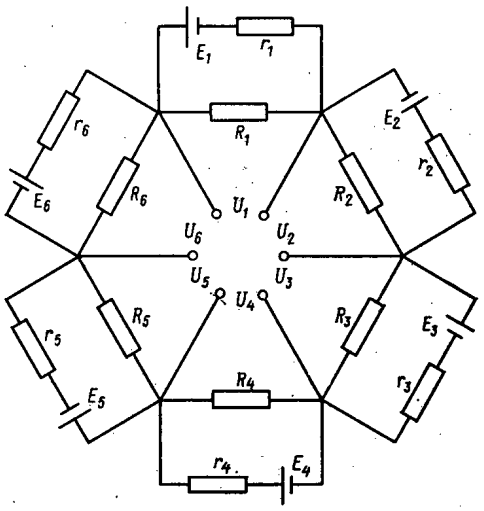

Fig. 2

Fig. 1. Schematic diagram of converter: 1) magnetic system; 2) electrode; 3) can of fuel-element mockup; 4) neighboring fuel elements.

Fig. 2. Equivalent electrical circuit of converter.

$$E_i = U_i(1+R/r) + \frac{(-1)^i R}{6r} \sum_1^6 (-1)^{i-1} U_i. \tag{2}$$

It is not possible with this method of problem solving to calculate the values of R_i and r_i, or their ratio, on the basis of the known geometric dimensions of the cells and the physical properties of the liquid and wall. The converter therefore requires calibration.

The flow rate through the cell can be expressed as

$$Q_i = E_i/kB_i = \frac{U_i}{kB_i}(1+R/r) + \frac{(-1)^i R}{6kB_i r} \sum_1^6 (-1)^{i-1} U_i = c_1 U_i + c_2 \sum_1^6 (-1)^{i-1} U_i. \tag{3}$$

The constants c_1 and c_2 are found in the course of the calibration. To this end, we measure the total flow rate through the bundle, all of whose cells are the same, and is set equal to the sum of the flow rates in the individual cells:

$$Q_{tot} = \sum Q_i = c_1 \sum_1^n U_i + c_2 \sum_1^n \sum_{i=1}^6 (-1)^{i-1} U_i. \tag{4}$$

Since the converter is calibrated at several values of the total flow rate, by choosing two values we can use Eq. (4) to get a system of two equations with two unknowns c_1 and c_2 and to determine those unknowns. Substitution of the values of c_1 and c_2 in Eq. (3) permits the flow rate through any cell to be found.

This converter with multipolar magnets can be employed to measure the flow rate in large-diameter pipes. The converters with cylindrical magnets [4] used for these purposes yield quite reliable results only in an undistorted axisymmetric flow since the flow rate is found from the known (e.g., logarithmic) velocity profile. The true profile of the velocity in the pipe, however, is not known as a rule. The multipolar-magnet converter makes it possible to determine whether the flow is axisymmetric. If the flow proves to be such, then the doubts as to the reliability of the results are dispelled. Otherwise, the converter makes it possible to assess the degree of asymmetry of the flow and the possible error of measurement.

When the multipolar-magnet converter is in a circular pipe the problem lends itself to analytic solution. In the two-dimensional approximation, i.e., with the assumption that the magnetic field extends along the axis of the pipe to a considerable distance, we can find the potential distribution in the liquid and in the wall of the

converter as well as the output voltage of the converter. The potential difference between two neighboring electrodes of the converter is found from

$$\Delta U_w = 2/3 B_m \frac{r_0^{n/2+1}(1+R_0^n) R_0^{-n/2}}{ac - bd(R/R_1)^n} \int_R^{R_1} \frac{W(r)}{r^{n+1}} dr. \quad (5)$$

In the case of the problem of a six-pole converter the radius appears in the sixth power in the denominator of the integrand and the value of the integral is determined mainly by the region of the flow around the converter. The velocity profile in the central part of the pipe is quite flat and $W(r) \simeq W_0 = \text{const}$. Then, for n = 6 we have

$$\Delta U_w = 2/3 B_m W_0 R_0 \frac{c r_0^4 R_0^2/R^6}{ac - bd R^6/R_1^6}(1 - R^6/R_1^6). \quad (6)$$

Since R/R_1 usually has a value ~ 0.1, the value of this ratio to the sixth power can usually be neglected:

$$\Delta U_w = \frac{2}{3} B_m W_0 R_0 [(r_0^4 R_0^2/R^6)/a], \quad (7)$$

where

$$a = (1 + R_0^n/R^n) + (\sigma_w/\sigma_f)(1 - R_0^n/R^n);$$

B_m is the maximum value of the magnetic induction on the surface of the magnet pole, and σ_w and σ_f are the electroconductivity of the wall and the liquid (fluid). Equation (7) is the fundamental basis for measuring the velocity (and flow rate) in a pipe.

The converter should be calibrated before being used. Since the device is sensitive only to the velocity in the region adjoining it, the calibration during which the relation is established between the output voltage and the velocity of the liquid can be carried out in a small-diameter pipe. The converter can then be used in large-diameter pipes to measure the axial velocity and, in the case of an undistorted profile, to calculate the flow rate.

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NEW METHOD FOR DETECTING BOILING OF WATER IN A REACTOR

I. I. Zakharkin

UDC 621.039.5.564.5

It is necessary not only to detect the boiling of water in the core, but also to determine where it is occurring. This is important to prevent the creation and development of a heat-removal crisis which might damage the fuel elements. Another problem involves monitoring the steam content of the steam–water mixture at the core outlet. Boiling can be detected by ultrasonic, electrical, optical, and other methods based on a statistical analysis of reactor noise power. We propose an optical method which employs the effect of Cerenkov radiation of electrons in water.*

It is known that a charged particle moving in a dielectric medium with a velocity greater than the phase velocity of light in that medium emits electromagnetic Cerenkov radiation which is propagated at an angle θ with the direction of its trajectory, where

$$\cos\theta = 1/n(\nu)\beta,$$

$\beta = v/c$ is the velocity of the particle in units of the velocity of light and $n(\nu)$ is the index of refraction of the medium as a function of frequency ν.

The intensity of the radiation in the visible region is given by the expression

$$I = 450 \sin^2\theta \text{ photons/cm}. \quad (1)$$

The threshold for the Cerenkov effect is given by the condition $\beta_t > 1/n(\nu)$. The limiting angle of radiation is reached for $\beta = 1$. In this case $(\cos\theta)_l = 1/n(\nu)$. The threshold kinetic energy of a particle E_t^k is related to the index of refraction by the expression

$$E_t^k = m_0c^2\left\{\frac{n(\nu)}{[n^2(\nu)-1]^{1/2}} - 1\right\}, \quad (2)$$

where m_0c^2 is the rest energy (0.51 MeV for an electron). Thus, the threshold energy for the production of Cerenkov radiation by a given kind of particle is determined by the index of refraction of the medium.

In an aqueous coolant having subcritical parameters the density of the liquid phase (water) is higher than that of the gaseous phase (steam). The index of refraction for each phase is given by the Lorenz–Lorentz equation

$$n(\nu) = [W + 2\rho A(\nu)/W - \rho A(\nu)]^{1/2}, \quad (3)$$

where W is the molecular mass of the medium; ρ, density of the medium; $A(\nu)$, molar refractivity ($A(\nu) = 3.7$ for sodium D light). Because of the difference in densities, the index of refraction of water is higher than that of steam. Consequently, the threshold energy E_t^k for the radiation of electrons is lower in water than in steam. The graph of Fig. 1, which was obtained from Eqs. (2) and (3), shows that for water E_t^k is practically independent of pressure, while for steam it varies rapidly.

Most of the electrons in an aqueous coolant are produced as a result of the interaction of rays with core materials and the coolant. For simplicity, we can assume that in an operating reactor the energy distribution of the flux density of the electrons which give rise to Cerenkov radiation is determined by the corresponding distribution of the flux of prompt γ rays. Above ~ 0.5 MeV the prompt γ flux falls off with increasing energy. For example, fewer than 5% of the γ photons per fission have energies above 2.5 MeV. The energy distribution of the flux of electrons producing Cerenkov radiation on the whole also falls off with increasing energy.

*I. I. Zakharkin, Inventor's Certificate No. 448770. Byull. Otkrytiya, Izobret., Promysh. Obraztsy, Tov. Zn., No. 40 (1974), p. 143.

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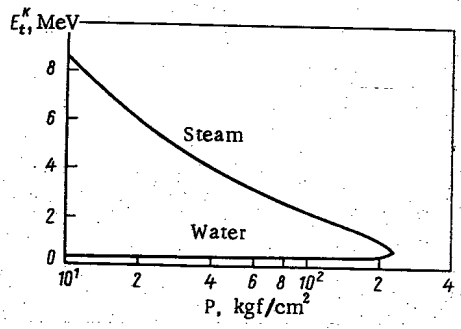

Fig. 1

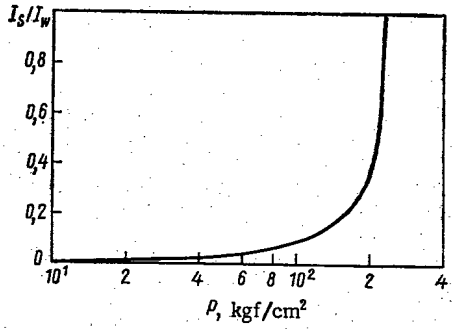

Fig. 2

Fig. 1. Threshold kinetic energy of electrons for water and steam at the saturation line.

Fig. 2. Ratio of intensities of radiation in steam and water at the saturation line for electrons with $\beta \approx 1$.

The number of electrons giving rise to radiation in water will be appreciably larger than the number producing radiation in steam. Electrons with energies below the limit for steam will radiate only in water. Thus, as a result of the difference in indices of refraction of water and steam and the shape of the energy spectrum of the electrons producing Cerenkov radiation, the total intensity of the radiation per unit path length will be higher in water than in steam.

It must also be remembered that even for electrons moving with a velocity $\beta \approx 1$, when the limiting angle of radiation is reached both in water and in steam, the intensity of the radiation will be higher in water than in steam. Actually, for $\beta \approx 1$ the ratio of the intensities of the radiation per unit path length in steam and water on the basis of Eq. (1) is determined by the expression

$$I_s/I_w \simeq (n_s^2(\nu)-1)\, n_w^2(\nu)/(n_w^2(\nu)-1)\, n_s^2(\nu), \tag{4}$$

where $n_S(\nu)$ and $n_W(\nu)$ are, respectively, the indices of refraction of steam and water. Figure 2 for I_S/I_W as a function of the pressure of the steam–water mixture at the saturation line shows that, e.g., for a pressure of 100 kgf/cm^2 $I_S/I_W \approx 0.1$, i.e., the radiation intensity is appreciably lower in steam than in water. Estimates show that for the pressure chosen, taking account of the actual electron spectrum, the radiation intensity in steam can be neglected. In this case the radiation intensity in the steam–water mixture will be proportional to the liquid phase fraction, i.e., to the water.

It should be noted that the present discussion does not take account of the slowing down of electrons in the medium, which occurs mainly because of ionization losses. The radiation intensity in the medium decreases with decreasing electron energy. The radiation vanishes for electron energies below the threshold. A more accurate analysis would require taking account of the slowing down of electrons in the medium, but the main conclusions about the possibility of using Cerenkov radiation to detect boiling are not changed.

Light from the region of the coolant being monitored can be led out through hollow metal light pipes. A measurement of the Cerenkov light emerging from a light pipe gives the necessary information about the state of the coolant. This method can be applied to an experimental thermophysical channel by using γ rays to excite the Cerenkov radiation. In addition, by choosing a γ source with energies below E_t^k in steam, it is possible in principle to eliminate Cerenkov radiation in the steam fraction of the coolant.

ENERGY DISTRIBUTION OF ^{235}U FISSION-PRODUCT γ RADIATION FOR A SHORT IRRADIATION TIME

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In connection with a detailed study of the radiation and technological characteristics of complex-power-chemical installations with uranium radiation loops (URL) as γ sources [1], and the development of the VGR-50

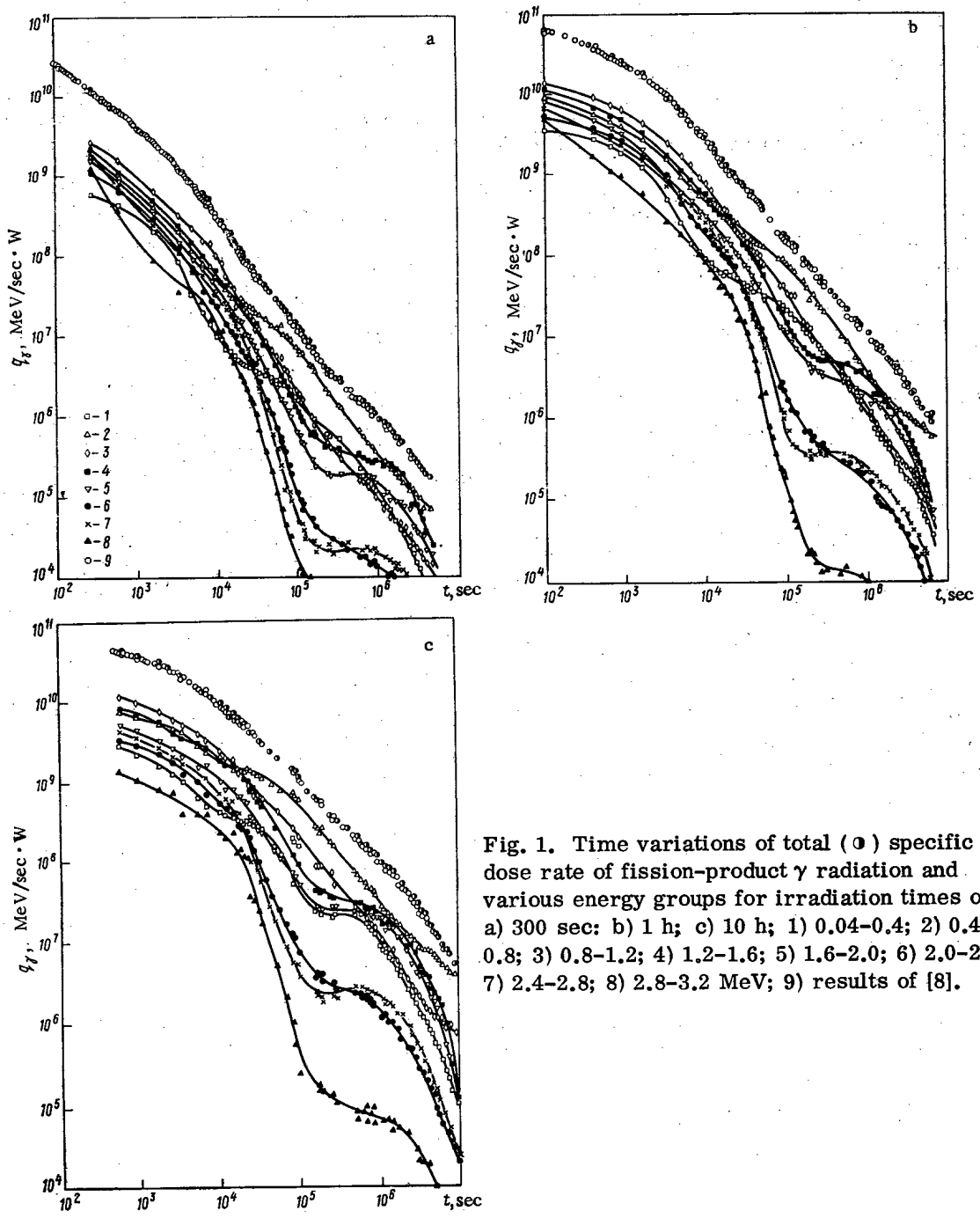

Fig. 1. Time variations of total (●) specific dose rate of fission-product γ radiation and various energy groups for irradiation times of a) 300 sec: b) 1 h; c) 10 h; 1) 0.04-0.4; 2) 0.4-0.8; 3) 0.8-1.2; 4) 1.2-1.6; 5) 1.6-2.0; 6) 2.0-2.4; 7) 2.4-2.8; 8) 2.8-3.2 MeV; 9) results of [8].

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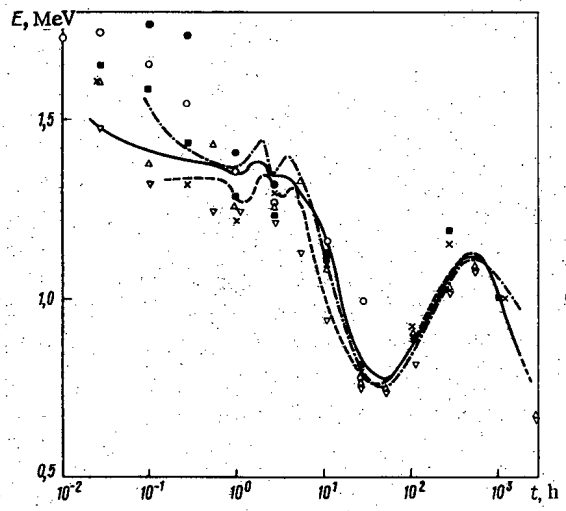

Fig. 2. Dependence of average energy of energy spectrum on cooling time for t_r = 300 sec [—·—) our data; ●) [4]]; for t_r = 1 h [—) our data; ■) [5]; ○) [4]; Δ) [7], for t_r = 10 h [— — —) our data; ×) [5]; ▽) [7]].

installation in our country [2, 3], information is required on the energy distribution of γ spectra for short (from a few minutes to several hours) fuel residence times t_r in the core.

The published time variations of γ spectra were obtained either by processing [4] known calculated [5] and experimental [6] curves for prompt fission without taking account of the burnup of fission products in the core, or by calculations [7]. The values of $t_r \leq 1$ h and cooling times $t \lesssim 2 \cdot 10^5$ sec considered in [4] are inadequate, as shown by the development of the URL [2]. An experimental study of the ^{235}U fission-product γ spectra in the ranges $t_r \leq 10$ h and $t \leq 10^7$ sec therefore became necessary. It should be noted that such information is of independent interest, since it helps to fill the gaps in the experimental information on the radiation characteristics of nuclear fuel fission products.

The experiment was performed in a horizontal channel of the VVR-Ts reactor. The samples were delivered to and removed from the irradiation zone by pneumatic rabbits. The uranium samples were carefully sealed in 0.4-mm-thick polyethylene film to prevent the escape of gaseous fission products. The spectra were measured on a single-crystal spectrometer with a 70 × 70 mm NaI(Tl) crystal and an AI-256-6 analyzer. The spectrometer was calibrated in the 60 keV-3.2 MeV energy range. The spectrograms were processed on an M-220m computer using a 31st order matrix with 100-keV steps.

Figure 1 shows some of the experimental results presented in the eight-group approximation. The values of the total specific fission-product γ dose rate (per watt of thermal reactor power) practically coincide with results [8] obtained by other experimental methods (cf. Fig. 1) and are in satisfactory agreement with data of [4, 5, 7]. For t_r = 1 h and $t = 10^5$ sec our energy distribution data differ by 200-300% from the results of [4, 5] in the fifth group, and by ~90%* from the data in [7] in the seventh group. However, the differences between our values and those of [4, 5, 7] for the variations of the average energies of the energy spectra with cooling time $\bar{E}(t)$†, which characterize exposure doses in the use of fission products as γ sources, are considerably smaller (Fig. 2).

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*The comparison was made for the energy groups used in [5].

†$\bar{E}(t) = \sum_i \bar{E}^2 N_i / \sum_i \bar{E}_i N_i$ MeV, where $\bar{E}_i$ is the middle of the energy interval and N_i is the number of γ photons in the group.

INTERACTION OF THE COOLANT FOR PROLONGED FLOW AROUND BUNCHES OF RODS

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UDC 621.039.517.3

It is important in thermohydraulic calculations of assemblies of fuel rods and pipe bunches of heat-exchange equipment to take into account the interchannel interaction of the coolant, which leads to equalization of temperatures in the transverse cross section of the rod assemblies and thereby in the possibility of overloading their capacity. We will assume (on the basis of the method of concentrated parameters) that the observed equalization of the interchannel temperature differences (concentrations) occurs due to transverse return flows of liquid through the spaces between rods with a mean composite temperature (concentration) of the original channel. For the calculations we use a dimensionless coefficient of interchannel mixing M, which is the ratio of the transverse flow rate of liquid from one channel into the adjacent one along the length of one rod diameter to the longitudinal flow rate through the channel [1]:

$$M = G_{ij}d/G_i. \quad (1)$$

Information about investigations of turbulent interchannel mixing in bunches of smooth rods is given in [1-3]. Evaluation of values of M is hindered due to the large scatter (almost an order of magnitude) of the existing experimental data. The scatter of the data in individual investigations is also great. Computational relationships generalizing these data are distinguished quantitatively and qualitatively by the nature of the dependence of M on the relative spacing of the rods x and the Reynolds number Re. The scatter of the data is caused by peculiarities of the investigated geometries, their frequent appreciable deviation from an infinite lattice of rods, and imperfection of the experimental methods. In the majority of cases the data are exaggerated by several times above the level of natural turbulent exchange due to uncontrollable deviations from regular geometry and the interchannel flux redistributions caused by them, artificial intensification of the effects of interchannel mixing due to a disturbance in the flow of the experimental procedure, scattering of flux by elements of intermediate rod spacing, and neglect of molecular effects.

An investigation is described in this article in which serious attention is turned to the reliability of the experimental method and the provision of minimal deviations from regular geometry. The investigations are carried out with the use of air on a bunch of 37 polished tubes (d = 11.9 mm, L = 1 m, x = 1.14) in the range of Re from 7660 to 30,450 on the basis of the Freon method. A description of the experimental model and the method of investigation, along with the data for Re = 14,500, are given in [1]. The descending dependence of M on Re which was obtained is shown in Fig. 1. The values of M correspond to the lower limit of the existing experimental data, although possibly slightly exaggerated.

An investigation of the drag coefficient λ and the velocity distribution along the axis of the channels of the bunch is conducted for the purpose of an additional hydrodynamical control. The values of λ are in good agreement with the computational recommendations of [4] for bunches of smooth rods. In the region of developed turbulent flow (Re from 7000 to 30,000) are located with a scatter of ± 1-2% parallel to the Blasius relation for smooth circular tubes to 5.5% above it in Fig. 1. Variations of the velocity along the length of all the channels were small. A local jump in the velocity by 2% and a small pressure drop were recorded in all channels only in the middle part of the bunch in the region where spacing bulges are located. Careful investigations carried out simultaneously with a liquid-metal coolant [3] gave similar results (see Fig. 1).

One can draw some preliminary conclusions from an analysis of experimental and computational-theoretical papers. Turbulent interchannel heat and mass exchange is less by several times than follows from the majority of the published data. The effect of secondary flows on interchannel heat and mass exchange in regular lattices of rods is rather great, but it was significantly underestimated until recently in view of the indicated causes. The component produced by secondary flows increases as x decreases below 1.4, compensating the decrease in the pulsational turbulent component. Due to this, the degree of turbulent heat and mass exchange remains approximately at the same level: M = 0.002-0.003 right up to x = 1.05, which corresponds to

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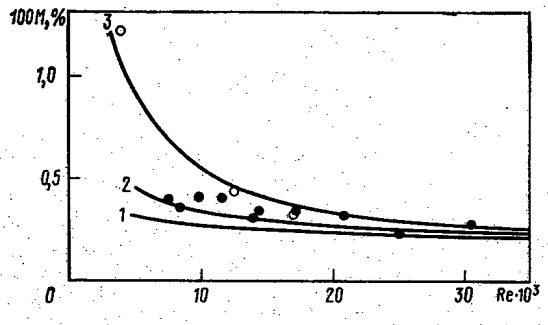

Fig. 1. Comparison of the experimental data with a calculation based on Eq. (2): ●) M_m^* (this paper); ○) M_h^* for NaK alloy, x = 1.13 [3]; 1-3) calculation of M (for x = 1.14), M_m^* (this paper), and M_h^* (for the conditions of [3], respectively.

the lower limit of the existing data. The usual neglect of molecular effects is inadmissible when experimental data in the region $Pr \ll 1$, $Sc \ll 1$, and with $Re \leq 10^4$ and $x \leq 1.1$ are used and generalized.

Let us introduce the concept of the interchannel heat- and mass-exchange coefficients:

$$M_h^* = k_h M + M_h^\lambda; \quad M_m^* = k_m M + M_m^d \tag{2}$$

where "h" denotes heat exchange, "m" denotes mass exchange, "*" denotes that molecular effects are taken into account, λ denotes thermal conductivity, and "d" denotes diffusion. The contribution of M_h^λ was taken into account in the investigations with liquid metal [3], and the contributions of M_h^λ and M_m^d were neglected in the investigations with water and air. In both cases equivalence is assumed between the interchannel exchange by moles of liquid of heat and mass of the admixture, which is strictly fulfilled only in the case of a plane profile of the temperature and concentration within the channel boundaries. The nonequivalence of the exchange in Eq. (2) is taken into account by the coefficients k_h and k_m, which evidently depend on the geometry of the bunch, the type of interchannel interaction, and the values of Re, Pr, Sc, and x. The value of M in Eq. (2) corresponds to the actual interchannel flow rate of the liquid. With the data of FÉI taken into account, let us assume in the first approximation $k_h = 0.7$ for $Pr \ll 1$. Taking the data of [3] into account, let us refine the relationship of [1] for M by means of correcting the effective distance of the interchannel heat and mass exchange $Z_{ij} = (1.08x^2 - 1)\,\delta_{ij}$, where δ_{ij} is the distance between the centers of adjacent channels. The relationships for M_h^λ and M_m^d are obtained for $Pr \gg 1$ and $Sc \gg 1$ by using the effective viscosity $(\varepsilon_\nu + \nu)$ instead of ε_ν in [1] and adopting the indicated expression for Z_{ij}. For liquids with $Pr \ll 1$ the relationship for M_h^λ according to Ginzberg and Frans [2] was adopted. Then $M \cong 0.0436\,(x-1)\,[1+(x-1)^{0.2}]^{0.5}\,x^{-1}\,(1.08x^2-1)^{-1}\,Re^{-0.2}$.

For $Pr \geq 1$ and $Sc \geq 1$ $M_h^\lambda = M_m^d = 5.66x^{-1}\,(1.08x^2-1)^{-1}\,Re^{-1.075}$, $k_h = k_m = 1$; for $Pr \ll 1$ $M_h^\lambda = 6.7\,(x-1)\cdot x^{-1}\,(Re\,Pr)^{-1}$, and $k_h = 0.7$.

Equation (2) can be used for engineering estimates of the coefficients of turbulent hydrothermal and mass exchange in regular triagonal lattices of smooth cylindrical rods in the region $1.05 \leq x \leq 1.4$ and $5\cdot10^3 \leq Re \leq 2\cdot10^5$.

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RESONANCE INTEGRAL FOR NEUTRON CAPTURE BY ^{244}Pu

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UDC 539.172.4

In order to select the optimal method of obtaining transplutonium elements by the reactor method it is necessary to know the neutron reaction cross sections for all nuclides in the cumulative chain. The goal of the present work is to measure the resonance integral for neutron capture by ^{244}Pu.

The original sample used was plutonium consisting of $\approx 6\%$ ^{244}Pu by weight. A solution of plutonium nitrate was placed in quartz cassettes (diameter, 6 mm; height, 100 mm) and was boiled down until dry. Approximately 2 mg of plutonium remained in each cassette. The prepared samples were irradiated in the reactor channel. The samples were placed in a polyethylene container (thickness equal to 18 mm, inner cavity diameter and height equal to 100 mm). Cassettes with plutonium were placed 80 mm from each other in order to eliminate any perturbations of the fluence of thermal neutrons by the cadmium during exposure of the samples with cadmium screens. Previously the absolute fluence of thermal neutrons, Φ_{th}, the effective temperature of thermal neutrons, T_n, the ratio of the thermal fluence to the epithermal fluence, Φ_{th}/Φ_{epi}, and the spectrum of epithermal neutrons were measured in the container using the (n, γ) reactions on ^{164}Dy, ^{115}Sn, ^{197}Au, ^{63}Cu, ^{23}Na, and ^{37}Cl. Methods for measuring these quantities were given in detail in [1]. The following values were determined: $T_n = 420 \pm 35°K$, $\Phi_{epi} \sim 1/E^{0.95 \pm 0.02}$, $\Phi_{th}/\Phi_{epi} = 34 \pm 2$.

The irradiation of samples containing ^{244}Pu was carried out simultaneously with monitor detectors. The (n, γ) reactions on ^{197}Au and ^{63}Cu and the (n, p) reaction on ^{58}Ni were used. The latter reaction was employed to determine the fluence of fast neutrons at the sample site. In each series of irradiations the fluence of fast neutrons with energies greater than 2.35 MeV was $\sim 7 \cdot 10^{13}$ neutrons/cm^2.

After irradiation the plutonium samples were processed radiochemically to isolate the plutonium. This was accomplished by collecting the plutonium on the anion Dowex 1 × 4 (100-200 mesh) from a solution of 10 M HCl and 0.1 M HNO_3. Subsequently, the inert components and the fragment activity were removed by washing with 7.5 M HNO_3. The desorption of the plutonium was then carried out with 0.5 M HCl.

The resonance integral for the reaction $^{244}Pu(n, \gamma)^{245}Pu$ was measured by the "cadmium ratio" method [2] by comparison with the well-known integral for the $^{197}Au(n, \gamma)^{198}Au$ reactions. The cadmium ratio was determined as the ratio of the specific activity of a given nuclide in the sample, irradiated without the cadmium shield to the activity of the same nuclide in the sample irradiated with the cadmium shield. A cadmium shield 1 mm thick was used. The cadmium ratios were measured for ^{245}Pu, formed in the radiative neutron capture by ^{244}Pu and for ^{198}Au, formed in the radiative capture by ^{197}Au.

The gold activity was measured by $\beta - \gamma$ coincidences to an accuracy of $\pm 1\%$ (the absolute activity was measured). The ^{245}Pu activity was measured on the solid-state γ spectrometer. A lead filter 113 mg/cm^2 thick was placed between the Ge(Li) detector and the plutonium sample to reduce the load on the spectrometer at the expense of detecting the lead x ray and losing low energy γ radiation ($E_\gamma < 100$ keV) from the plutonium. The measurements of γ spectra from the irradiated samples of plutonium were conducted in the standard geometry for ~40 h after the end of the radiochemical separation. The analysis of the spectra was conducted manually, with the help of a special computer program. In each case, the γ-ray energy was determined and the area of the corresponding photopeak was calculated. These measurements showed that the basic γ radiation from the irradiated plutonium samples was due to ^{243}Pu (γ-ray energies of 322, 356, 381, and 423 keV) and ^{245}Pu (γ-ray energies of 308, 327, 377, 492, 560, and 630 keV). Both ^{243}Pu and ^{245}Pu were identified by their γ-ray energies, as well as their half-lives. The cadmium ratio for plutonium was calculated as the ratio of the photopeak areas of the 327, 492, 560, and 630 keV γ rays from samples irradiated with and without the cadmium shield. Corrections were applied for decays after the time of measurements and also for the differing amounts of plutonium contained in the various samples.

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TABLE 1. Basic Experimental Results

Experimentally measured cadmium ratios		Resonance capture integral for ^{244}Pu, b
^{197}Au	^{244}Pu	
2,750±0,050	1,85±0,15	44,5±8,0
2,750±0,056	2,02±0,20	37,1±6,7
2,700±0,054	1,91±0,13	40,3±6,1

The calculation of the resonance integral was done according to a well-known formula [2] using the experimentally determined values of the cadmium ratios of the ^{197}Au standard and ^{244}Pu. The following constants and corrections were used. According to [3], the resonance capture integral for gold in an isotropic flux of neutrons and a cadmium shield of 1-mm thickness is 1543 ± 20 b. Taking into account the deviation of our neutron spectrum from the 1/E law, we obtain a correction of 1.096 ± 0.020 calculated according to [4]. Hence this integral is taken to be 1691 b in the present work. The values of the (n, γ) cross sections in the neutron energy range of 0.025 eV (2200 m/sec) are 98.6 ± 0.3 b [3] for ^{197}Au and 1.7 ± 0.1 b for ^{244}Pu [5, 6].

Corrections for the values of the cross section g(t), considering the deviation of the cross section from the 1/v law in the thermal region at a temperature of 420°K are, according to [7], 1.012 for ^{197}Au, and unity for ^{244}Pu. Corrections were calculated from the experimentally measured cadmium ratios for the absorption of neutrons by cadmium, with energies in the range between the edge of the epithermal spectrum (0.1 eV) and the absorption edge of the 1-mm-thick cadmium (0.68 eV). These were 1.059 ± 0.001 [1] for ^{197}Au and unity for ^{244}Pu. In the experimental values of cadmium ratios for gold, after computing the absorption, a further correction was applied to the gold cadmium ratio to account for self-screening of epithermal neutrons in the 4.4-mg/cm^2-thick foil. This was 0.776 ± 0.005 [8]. The analogous correction to the ^{244}Pu values was not introduced since the thickness of the irradiated samples was less than 0.5 mg/cm^2. The results of the measurement are shown in Table 1. The error in the mean value is a single standard deviation. Thus, the value of the effective resonance capture integral of neutrons by ^{244}Pu, measured in the present work, is 40 ± 3 b.

Analogous measurements were carried out in [9] where 200 mg of plutonium enriched in ^{244}Pu (99.06% by mass) were used. Gold was used as a standard, as in the present work, and In γ and $\sigma(v_0)$ were taken to be 1558 and 98.8 b. The value obtained for the resonance integral was 35 ± 4 b. The recommended value in [5] is 43 ± 4 b, obtained as the mean of the values quoted in [9, 10]. Evidently the value obtained in the present work agrees with the recommended value within the quoted uncertainties.

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ABSOLUTE MEASUREMENTS OF THE CROSS SECTION FOR THE FISSION OF ^{241}Am BY 2.5-MeV NEUTRONS

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UDC 539.173.4

In reactor engineering it is necessary to know the fission cross sections of the transuranium elements which accumulate during reactor operation. The region of the first plateau of the excitation function where the fission neutron yield is maximum is of great interest. The accuracy and reliability of fission cross-section measurements are increased as a result of a variety of experimental methods. The measurements most frequently described in the literature are made relative to $\sigma_{n,f}$ cross sections of such nuclei as ^{235}U, ^{238}U, or ^{239}Pu. Absolute measurements are more difficult, but have the advantage that the calculation of the cross sections contains fewer sources of errors. The absolute values of the cross sections for the fission of ^{238}U [1] and ^{235}U [2] by 2.5-MeV neutrons were measured earlier by recording the neutron flux by the method of accompanying particles from the D(d, n) ^{3}He reaction. This method with certain modifications was employed in the present work for the first time to determine the cross section for the fission of ^{241}Am by 2.5-MeV neutrons.

Deuterons were accelerated by a low-voltage accelerator to energies of 70-80 keV. The experimental procedure was similar to that used earlier in [1, 2] except that the accompanying helions were recorded with a silicon semiconductor surface barrier detector rather than with a gas proportional counter. The helion pulses were separated out by a differential discriminator. Deuterons scattered in the deuterium–titanium target were absorbed by an aluminum foil cemented onto a diaphragm in front of the semiconductor detector. It was found that an aluminum absorber 1 μm thick gave the best separation of the helion peak in the spectrum.

The ^{241}Am targets were made by the vacuum evaporation of americium fluoride onto rotating polished aluminum disks 0.1 mm thick. The americium was certified to contain an admixture of 0.5 wt.% ^{239}Pu. The total mass of the ^{241}Am targets used, as determined from the α activity and the published value of the half-life $T_{1/2}$ = 432.7 ± 0.6 yr [3], was 2754 ± 25 μg. The thickness of the americium fluoride layers was no more than 300 μg/cm^2.

The fission fragments were recorded by mica detectors which combine a high (96%) efficiency for counting fragments against a background of high α particle activity with an extremely small mass of neutron scattering material. Stacks of targets and mica sheets were pressed between thin cadmium disks. Targets of natural uranium and ^{235}U 20 mm in diameter were placed in the briquet with the Am targets to serve as monitors and to increase the reliability of the results.

During the measurements 2117 americium fissions and 716,205 helions were recorded. The spontaneous fission of target nuclei during the time of contact with the detector was taken into account in calculating the cross section. This correction was found experimentally to be 1.9 ± 0.2%; the correction for the insensitivity of the mica detector to fission fragments incident on it at small angles with the surface was 4 ± 1%; the correction for the separation of helions from the complete charged-particle spectrum was 1.5 ± 1.0%; the correction for nonsphericity of the target surface was 1.5 ± 0.2%; the fission of target nuclei by scattered neutrons was 3 ± 1% of the total number of fissions; the corrections for the admixture of ^{239}Pu in the americium target was 0.5 ± 0.1%; the correction for the geometry of the experiment and the angular distribution of neutrons was 0.3 ± 0.1%.

To sum up, the cross section for the fission of ^{241}Am by 2.5-MeV neutrons was found to be 1.98 ± 0.07 b. The error quoted is the mean-square value; its largest components are the statistical error of the number of recorded fragments (2.2%) and the error in determining the distance from the neutron source to the fissionable targets (1.7%).

As noted above, monitor targets of ^{239}U and ^{235}U were irradiated in the briquet along with the americium targets. Their cross sections for fission by 2.5-MeV neutrons were found to be 0.52 ± 0.02 and 1.27 ± 0.05 b,

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respectively. Within the limits of error these agree both with our previous data [1, 2] and with the average estimates of the cross sections given by Hart [4] and Sowerby [5].

The value $\sigma_{n,f}$ (^{241}Am) = 1.98 ± 0.07 b obtained from absolute measurements agrees, within the limits of error, with the known values 1.95 ± 0.2 b [6]; 1.96 ± 0.2 b [7]; 2.05 ± 0.07 b [8]; 1.9 ± 0.1 b [9], which were obtained by the relative method.

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HISTORY OF THE STARTUP OF THE FIRST NUCLEAR POWER STATION (DOCUMENTS AND INFORMATION)

After the Soviet government had taken the decision to construct a nuclear power station and the site for its construction had been selected, work on the building of the station was transferred from the Institute of Atomic Energy to the Physico-Power Institute. Under the direction of I. V. Kurchatov, the staff of this Institute, and also other scientific-research and planning organizations and establishments, had solved serious problems in the shortest time and made possible the startup of the nuclear power station.

Startup preparations commenced in May 1954 with the charging of the nuclear power station reactor, with technological channels for conducting physics experiments. Reactor criticality was achieved on May 9, 1954. Then the complicated testing of the plant and all the station systems at zero reactor power was carried out, and it was brought gradually up to its capacity. On June 26, with 75% of the reactor power, steam was fed to the turbine. Thus was made the start to the peaceful utilization of atomic energy.

In a small publication, it is not possible to represent the great contribution which was made in the construction of the nuclear power station by the staff of the many organizations, establishments, individual scientists, planners, and builders. The preparation of the fully documented history of the nuclear power station is a task for the future.

On the occasion of the 25th anniversary of the nuclear power station, we shall deal with only one of the stages in the installation of the station – its startup. The documents, published for the first time, from the archival stock of the Physico-Power Institute give some idea of the preparation, course and participants of the startup task.

Publication prepared by L. I. Kudinova

No. 1. From a letter of Comrade V. S. Emel'yanov to D. I. Blokhintsev, Director of the Physico-Power Institute concerning the appointment of the Scientific Director for work on the nuclear power station.

August 14, 1952.

...You have been approved the scientific director of Sec. 16 of the combined plan for scientific-research experimental, planning and constructional work.... in 1952-1953 (work on the AM* complex).

Your deputy is approved as Comrade A. K. Krasin and the Chief Designer Comrade N. A. Dollezhal'.

The approved subjects and the period for completion of the work are stated in the appendix.

I request you...:

1. Over a 20-day period to issue an assignment to the principal executors and to provide everyday supervision on the work according to the combined plan.

2. To present quarterly, brief accounts of the progress of the work over a 5-day period after the end of the quarter. The first account must be presented by Oct. 5, 1952.

The organizations and establishments of the Ministries and departments participating in the completion of the work plan are responsible .... for ensuring the immediate execution of the scientific-research, experimental, planning and design work, specified in the stated plan.

V. Emel'yanov

*AM –"Peaceful atom," arbitrary name for the nuclear power station.

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No. 2. Order No. 4 of the Director of the Physico-Power Institute concerning the inception and scale of organization of commissioning work on the nuclear power station.

May 6, 1954.

1. The scientific commissioning director is Comrade D. I. Blokhintsev.

2. In order to carry out the commissioning operations, duty scientific directors have been appointed: Comrades A. K. Krasin, B. G. Dubovskii, and M. E. Minashin, and assistant duty scientific directors: Comrades V. A. Konovalov, E. I. Inyutin, M. N. Lantsov, and A. V. Kamaev.

3. The duty commissioning directors are subordinate directly to the scientific commissioning director.

4. The duty scientific directors will be on duty in turns during the commissioning operations and will be responsible for the safety of equipment and personnel.

5. The operative orders of the duty scientific director are obligatory for the shift chief and chief engineer with respect to control.

6. The commissioning group of physicists is directly subordinate to the duty scientific director. The composition of the commissioning group will be approved by the scientific commissioning director.

7. The duty scientific director will compile a specific work plan for each shift, approved by the scientific commissioning director.

8. All operations on the setting up and removal of the working technological channels and other channels of equipment, measuring instruments, operations on the feeding and disconnection of water in the channels of equipment will be carried out only through the stated duty scientific director.

9. Operations on equipment will be carried out by the duty engineer through the central control room, and under the control of the head of the unit, the chief engineer or his deputy.

10. The actions of the duty scientific directors and all personnel carrying out commissioning operations, must be performed in accordance with instructions.

D. Blokhintsev

No. 3. Order No. 2 of the Head of the Nuclear Power Station, concerning the manning of the duty shift stations.

April 13, 1954

1. To man the duty shifts of the station in staff:

Item No.	Job	Surname, forename, and patronym			
		Shift No. 1	Shift No. 2	Shift No. 3	Shift No. 4
1.	Shift chief	Yu. V. Arkhangel'skii	B. B. Baturov	V. A. Remizov	G. N. Ushakov
2.	Deputy shift chief	V. B. Lytkin	R. V. Timoshenko	I. A. Sadovnikov	vacant
3.	Chief engineer in control	A. V. Karpov	V. I. V'yunnikov	V. I. Shmelev	V. A. Parfir'ev
4.	Engineer in control	Yu. V. Evdokimov	S. A. Bolonkin	G. V. Merzlikin	L. A. Kochetkov
5.	Duty technician on local observation panel	vacant	V. P. Nikonova	N. Maksimov	vacant
6.	Duty engineer, control and measuring instruments and automation equipment	vacant	I. A. Goryainov	D. I. Zhestkov	N. V. Bogdanova

7.	Duty technician, control and measuring instruments and automation equipment	V. A. Malyshev	V. A. Kopchenov	V. I. Karpov	B. A. Erofeev
8.	Duty on mass flowmeter panel	vacant	V. V. Ivanov	V. M. Kozhemyako	V. A. Osipchuk
9.	Duty mechanic, control and measuring instruments	S. S. Galushko	A. D. Torsunov	A. I. Alekseev	S. P. Aleksanin
10.	Duty electrical engineer	B. G. Sprygin	Yu. V. Gryunberg	K. V. Ermilin	A. M. Kolyzhenkov
11.	Duty electrical fitter on control panel	N. G. Ardab'ev	N. T. Balabanov	A. K. Korolev	E. I. Annenkov
12.	Duty electrical fitter on dc panel	A. G. Pershin	A. P. Gnedov	S. G. Khamidullin	A. D. Golikov
13.	Duty electrical fitter on machine hall	N. G. Ponomarenko	V. F. Kononov	I. I. Yatskevich	F. A. Podtsepilov
14.	Duty electrical fitter on ventilation system	P. G. Berelet	T. T. Akimov	B. I. Efimov	S. I. Efremov
15.	Duty electrical fitter on scale-formation equipment	P. I. Evdokimov	L. I. Troshin	A. I. Lomovatskii	vacant
16.	Duty electrical fitter on centrifugal separator-6 (TsN-6)	V. I. Rezodubov	vacant	vacant	vacant
17.	Duty master mechanic	K. I. Gulibin	B. V. Penkin	A. M. Maslenkin	A. I. Babin
18.	Duty fitter on central hall	A. M. Zvonkov	I. V. Druzhinin	F. P. Pozdnyakov	V. A. Gostev
19.	Duty fitter on pumping	S. B. Nikolaev	V. I. Kozyrev	P. A. Andreev	V. G. Semenov
20.	Duty fitter on sanitary engineering	A. I. Poletaev	V. I. Kharabarkin	I. M. Baranov	V. A. Komolov
21.	Duty fitter on ventilation system	A. K. Alimov	A. O. Panchuk	A. I. Sudakov	vacant
22.	Duty technician on "D"	D. P. Sokolov	G. F. Somov	I. I. Tikhomirov	V. F. Afonin
23.	Desk duty on "D"	E. A. Nikonov	Yu. L. Shirokovskii	T. A. Novikova	N. S. Dobrova

2. Shift chiefs to carry out work with the personnel of his shift according to the development of the working sites, solidarity of staff and strengthening of labor and production disciplines, in order to startup and become familiar with the facility in the shortest possible period.

3. In order to become acquainted with personnel, the shift chiefs will participate in committees to check the knowledge of the workers of his shift on servicing the working sections.

N. Nikolaev

No. 4. From a resolution of the State Committee concerning the preparedness of the nuclear power station for bringing into operation.

June 1954

In the execution of the order of Mar. 26, 1954, the Committee has examined the readiness of the nuclear power station for bringing into operation.

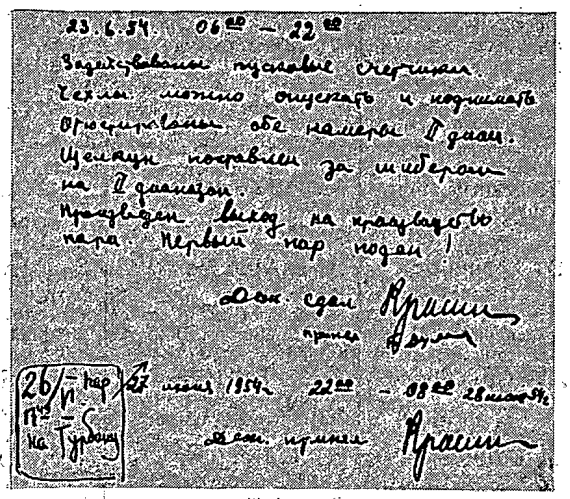

Fig. 1

Fig. 2

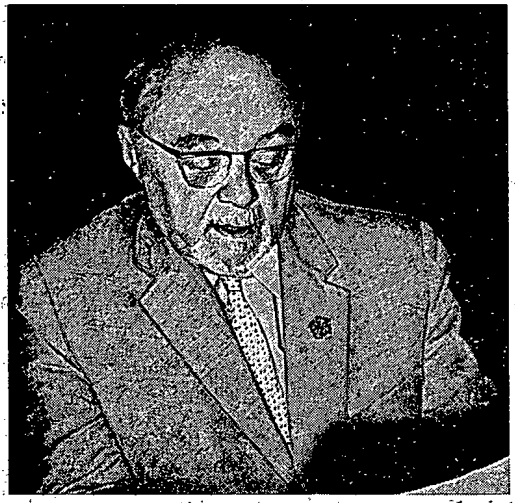

Fig. 3

Fig. 4

Fig. 1. Steam fed to the turbine! Entry in the diary on June 23-27, 1954.

Fig. 2. Scientific Commissioning Director D. I. Blokhintsev and his deputy A. K. Krasin (Photo N. Pikulin; taken in June 1974).

Fig. 3. Chief designer of the first nuclear power station N. A. Dollezhal' (Photo N. Pikulin; taken in June 1974).

Fig. 4. Director of work on fuel element production for the first nuclear power station, V. A. Malykh.

The AM reactor, with a design power of 30,000 kW (thermal) and which corresponds to a net electrical power of up to 5000 kW at the steam turbine, has been installed in the nuclear power station.

In accordance with the nuclear power station project, the steam generators also have been installed, calculated on the water circulation in the primary circuit at a pressure of 100 atm and a temperature of 290°, and on the production of steam for the turbine at a pressure of 12 atm and a temperature of 280°C, at the rate of 42 tons/h.

In order to check the individual systems and units of the facility, additional working committees of engineering-technical workers, builders and fitters... .

The working committees have carried out testing, adjustment, and running-in of all the mechanical and electrical plant, including the main pumps, steam generators, monitoring instruments and automatic control equipment in the plant, and the technological pipelines according to special programs approved by the Committee.

The AM facility is completely manned by trained and qualified specialists..... The technological and operating instructions for all working points have been prepared and approved. All operating personnel have passed a technical educational training, concluding with an examination on the operating regulations of the plant. At the work points, personnel have been assigned who are acknowledged to be trained in the work.....

The Committee, with the participation of Comrades I. V. Kurchatov, A. I. Alikhanov, and A. P. Aleksandrov, discussed and accepted the operating program for the AM plant in the first operating period.

For the purpose of ensuring the operating reliability of the facility in this period, and for a more detailed study of the heat transfer with the fuel elements, the Committee recommends that the maximum water temperature in the channels is accepted as 270°C and the maximum power as 75% of the design power. The power of the facility can be raised to the design power, as the plant is assimilated and the characteristics are studied.

To sum up, all the building and assembly work has been carried out, and also the examination of the plant, measurement monitoring systems and automatic control systems in the technological process of the facility, and the Committee considers that the facility is ready for bringing into operation and can be allowed to be brought up to power, with the operation of the steam turbine in accordance with the plan designated by the Committee.

No. 5. From the log of entries of the duty scientific director of startup operations in the nuclear power station.

May 9-June 27, 1954

.....May 9, 1907 h.

Critical mass achieved with 60.25 technological channels. For this, 6 rods of the inner manual control ring were inserted in the cells, water was filled into one of the jackets, and the rods are in the upper extreme position.

Twelve manual control rods of the outer ring – without water in the jackets, all rods – in the upper extreme position. Automatic control rods – without water in the upper extreme position. Rods of the slow scram system – without water in the upper extreme position. System damped at 1915 h. All manual control rods and slow scram rods dropped into the lower extreme position.

Krasin, Dubovskii

...June 12, 2100 h.

System brought to 1% power. Control switched over to second range of automation. Startup boost agent removed from the horizontal channel. Counter (with integrator) withdrawn to the upper level of the tube.

2200 h. Watch handed over Krasin
Watch taken over Dubovskii

...June 15, 0800 h - 2200 h.

From 0800 h to 0930 h, operated at 25%. At 0930 h the system was reduced in power for experiments at low power. At 2100 h the system was brought to 50% power.

Watch handed over Krasin
Watch taken over Dubovskii

...June 19, 2200 h-0800.

System operated at 75% power, no interruptions in operation.

Watch handed over Krasin
Watch taken over Dukovskii

...June 23, 0600-2200 h.

Startup counters functioning. Jackets can be lowered and raised. Both chambers adjusted second range. Clicker set behind gate on second range. Achieved steam production--FIRST STEAM SUPPLIED!

Watch handed over Krasin
Watch taken over Dubovskii

No. 6. From the book of testimonials by visitors to the first nuclear power station.

June 6, 1955-November 11, 1961

...June 1, 1956.

The peaceful utilization of atomic energy is a problem which rises before mankind in the second half of the century. The startup of this Soviet power station, the world's first nuclear power station, was the boundary mark on this path. Thereby, Soviet scientists have headed the struggle for the progress and well being of mankind.

A group of specialists of the German Democratic Republic

...August 8, 1956.

We are very grateful for this very interesting and instructive visit. For more than 40 years – from the time when I was concerned with science and technology – I have expected the construction of a station such as this. I was convinced that a socialist country would be the first to obtain power over atomic energy, and I know that this power will be used for peaceful purposes. I am glad that I have lived until the time when I could see the station with my own eyes, and I salute all those who have made it possible and all who will work in it.

Hewlett Johnson, Dean of Canterbury Cathedral,
N. Johnson, Kezia and Keren Johnson.

...November 18, 1957.

We have visited with great interest and admiration, the first nuclear power station, which is supplying power to the grid.

F. Perrin, Supreme Commissar of Atomic Energy, France,
and a group of French specialists.

Central Museum of Revolution of the Soviet Union. State Quality Inspection No. 34497/797.

INTERNATIONAL COLLABORATION

CONFERENCE OF THE INTERNATIONAL WORKING GROUP ON INTOR

V. I. Pistunovich

The first Conference of the Working Group on the Planning of an International Tokamak Reactor, INTOR, took place from Feb. 5 to 16, 1979, in Vienna. Delegations participated from Euratom, the Soviet Union, the U.S.A., Japan, and representatives of the IAEA. Reports were heard at the conference about national plans for a Tokamak reactor and about the efforts which will be made in support of the INTOR project. In 1979 the contribution of each country to INTOR is estimated at ~15 man·yr. By the end of 1979, the Working Group will issue a report containing recommendations in relation to the possibility of constructing an INTOR as the next step in the development of thermonuclear research after the facilities JET, T-15, JT-60 and TFTR.

The basic aims of INTOR were refined at the Conference, and which were provisionally defined by the steering Committee in Nov. 1978. Based on the previously planned developments of Tokamak reactors, the problems of which are similar to the problems of INTOR, the steering Committee defined its principal parameters, not coordinated at this stage of work. The conclusions at which the Steering Committee arrived when discussing the reactor parameters are as follows.

At the present time, the control of the amount of impurities in the plasma by means of a diverter remains unsolved. The operating cycle with short pulses may be found to be applicable without technological complications, due to the presence of the diverter. On the other hand, it may be required to operate the reactor in a cycle with long pulses and, consequently, the inverter will be necessary, or other devices for the active control of the amount of impurity. Therefore, two alternatives of the reactor parameters are being considered: with and without diverter.

The blanket of the INTOR will serve for tests of the technological systems and the production of 5-10 MW of electrical power. When defining its problem, two alternative points of view were put forward. The first consists in that the design of the reactor must be the simplest possible and the most reliable. For this, it is sufficient to have several experimental models of the blanket for the production of electric power and for testing different technological methods of breeding and extracting tritium. The requirements for tritium are given on account of the external sources. The second point of view consists in that the first stage of operation of the reactor, the blanket modules will be used for the breeding of tritium and the production of electric power. In the second stage, a solid blanket will be installed for breeding the tritium essential for operation of the reactor. Both points of view will be discussed at the next session of the Working Group. In any case, the tritium necessary for starting the operation of the reactor, must be supplied from external sources.

The Steering Committee, at the session in Nov. 1978, defined 16 basic problems of plasma physics and INTOR technology. The members of the Working Group prepared lists of questions on each problem and they distributed them to delegates as homework. Before the next conference, every country-participant must prepare answers to these questions, provide the starting data for designing the reactor, and present the requirements in additional scientific-research work. It is assumed that at the second conference, based on the data presented by the national groups of specialists, a single assessment will be made of the reactor parameters, and a rough version of the report will be compiled. It will appear in the summary account of the INTOR Working Group for 1970.

The second conference of the International Working Group will take place in June-July, 1979.

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CONFERENCE OF THE WORKING GROUP ON POWER GENERATION

M. V. Agranovich

The Fifth Conference of the Working Group on the Development of Power Generation in the Republic of Cuba, of the Intergovernmental Soviet–Cuban Commission on Economic and Scientific–Technical Collaboration, took place on Jan. 4-12, 1979, in Moscow. The achievement of the mutual commitments, associated with the development of electric power generation in Cuba was discussed. An exchange of ideas took place on the results of the achievement of the plan for scientific-collaboration in 1978, and a plan for collaboration was agreed between the Ministry of Power and Electrification of the Soviet Union and the electrical industry of Cuba in 1979.

During the period elapsed from the time of the previous Working Group Conference, with the assistance of Minénergo of the Soviet Union, successes have been achieved in the development of power generation in Cuba, in the field of construction and operation of thermal power stations and electric power transmission lines. The Working Group considered measures for collaboration in the construction of future power-generating plants in Cuba. A technical assignment was agreed for the design of the "Huragua" nuclear power station. In order to coordinate questions on the construction of the nuclear power station, a subgroup of experts on nuclear power stations was set up within the composition of the Working Group. Agreement was reached at the Conference on the exchange of technical documentation for electric power generation and nuclear power stations with water-cooled/water-moderated power reactors.

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CONFERENCES, SEMINARS, AND SYMPOSIA

CONFERENCE ON HEAT EXCHANGE AND HYDROSTATIC RESISTANCE DURING THE MOTION OF A TWO-PHASE FLOW

P. A. Ushakov and A. A. Ivashkevich

The Sixth All-Union Conference was held in Jan. 1979 in Leningrad. Nine-hundred Delegates from 180 organizations participated in its work. Some 197 reports and communications were presented. At the Plenary Session, review reports were addressed by Academician M. A. Styrikovich, Corresponding Member of the Academy of Sciences of the SSSR B. S. Petukhov, and Doctors of Technical Science N. M. Markov, V. M. Borishanskii, and É. K. Kalinin.

The following principal trends of research were considered in three sections of the Conference:

thermal hydrostatics of two-phase flows, including film motion of a liquid; condensation condensation and evaporation processes with a gradient in the direction of large two-phase flow velocities;
calculation of heat-exchange plant, increasing the heat-exchange efficiency, and processes in the near-critical region.

Great attention was paid to processes of boiling of liquids in large volumes (28 reports). The wide spectra of coolants (from water to cryogenic liquids) and the geometrical characteristics of surface heat-exchange permitted considerable factual data to be obtained. In particular, the effectiveness of porous coatings was shown. Theories have been developed concerned with complex questions of effervescing in the case of a pressure drop, the growth of vapor bubbles, simulation of boiling, models of a crisis in fissures and capillaries, etc. Many papers were devoted to heat exchange by bubble-boiling of liquids in channels. Investigations have been conducted with distilled and sea water, freons, helium, alcohols, hydrocarbons, binary and trinary mixtures, and other liquids. At the same time, the reliable calculation of heat-exchange in installations still remains an insoluble problem.

About 40 reports and communications were concerned with the hydrodynamics of gas- and steam-flows in channels. Extensive data have been obtained concerning the limits of the dispersion-cyclic regime, the behavior of liquid films at the walls, the deposition of drops from the core of the flow on the walls, flow structure, etc. Interesting data have been obtained for bundles of heat-releasing rods. Specific attention was paid to hydrostatic losses in two-phase flows, in particular the mechanisms for the two-phase flow of helium was found to be the same as for a steam–water mixture.

Part of the reports were devoted to measurements and methods of calculating the true steam content. More than 20 papers were concerned with a boiling crisis of the media in channels. The main investigations were conducted on water, but in some reports helium and solutions of salts were used. The effect of variable power release was considered, vortexing in tubes and tubular bundles, pressure, mass velocity, tilting of tubes in critical loading, and boundary steam content. As a result, empirical formulas have been proposed and models of crisis phenomena and calculation procedures have been developed, which will be useful for boiler technology and steam-generator design. Some papers were directed at the study of heat-exchange in the case of film boiling of a steam–water mixture, oxygen, nitrogen, hydrogen, freons in tubes, bending of tubes, diffusers, diaphragms, and with transverse flow around tube bundles. Interesting mechanisms have been discovered (the effect of the channel material on a crisis of second species, large thermodynamic nonequilibrium of flows, the effect of roughing on heat transfer, etc.) and methods for calculating heat transfer have been proposed. Attention was paid to heat removal in the case of transition boiling, when the heated surfaces are wetted with the coolant. This is important for cases of emergency reactor cooling. The data were generalized in the form of criterial relations, and a model of transition boiling was suggested under large volume conditions.

More than 40 papers were related with the problem of evaporation and condensation. In particular, heat transfer during the evaporation of water in porous bodies was studied, and also liquid drops in gas flows, liquid films during the condensation of steam, a mixture of steam with gases, freon vapor, ammonia, and other media.

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The subjects of the investigations were the outside and inside surfaces of tubes, tube bundles, coils, and slotted channels. The effect of corrugation, porous coatings, gas bubbling, rapid expansion of steam in the turbine, etc., were touched upon. The criterial formulas and methods of calculating heat removal reflect the practical outcome.

Papers on the hydrodynamics and heat transfer in the near-critical region (about 17 reports) were of considerable interest. They involved questions of hydroresistances in tubes, stability in parallel channels, escape of two-phase flows in the case of depressurization of high-pressure circuits, heat removal by natural convection in horizontal and vertical tubes and by the motion in tubes of water, toluene, benzene, and other liquids. Empirical formulas were given for actual cases. A new step in theory were models of turbulent exchange with variable physical properties, analysis of the resulting high-frequency flow vibrations and their positive effect on heat removal. Numerical-experimental investigations of hydrodynamics and heat exchange in power-plant components (more than 55 papers) covered very extensive problems. The hydrodynamic characteristics of two-phase flows were considered, with natural circulation and taking account of transition processes, stability, and other problems. Attention was paid to the distribution uniformity of coolant throughout the steam-generator channels. Coolant leakage through openings, flow processes of wet steam in the nozzles and grills of turbines, and the separation and removal of moisture were investigated. Heat removal in sea water evaporators, its intensification by means of aeration of the water, heat exchange in steam generators, condensers, and thermal tubes was studied. Data were presented about the development of new standards for the calculation of boiler plants. All these papers were of an applied nature.

About 20 reports and communications were devoted to the improvement of the characteristics and designs of plant. Condensation tubes with intensifiers, heat syphons, exchange hydropulsator-intensifiers, heaters for viscous media, certain types of thermal tubes, seawater evaporators, vortex condensation-separation heat-exchangers and other types of plant were investigated. In the direction of increasing operating reliability, attention was paid to turbine gaskets, supercavitation pumps, distributive collectors with a turbulence-creating mounting, etc. These papers, which were of practical value, in certain parts fell beyond the subjects of the Conference.

Individual problems concerning thermoacoustics in a fluidized bed, measurement techniques, compression waves in two-phase media, the Rank effect in two-phase flows, etc., were considered at the Conference.

The Conference showed that in the thermophysical research of two-phase flows, there have been marked successes. Much work has been carried out on a universal level and is of great scientific and practical value. However, there is still insufficient coordination of certain work.

SEMINAR ON THE RELIABILITY OF NUCLEAR POWER GENERATING FACILITIES

A. I. Klemin

In October 1978 the Seventh Conference of the Permanent Working Seminar "Engineering and Economic Aspects of Nuclear Power Generation" took place at the Scientific-Research and Design Institute of Engineering Technology. The theme of the Conference was the problems of reliability in planning and operating nuclear power facilities. Methodology of the assessment, statistics, and analysis of failures. Measures to increase reliability.

The Seminar excited great interest of the specialists. Twenty reports were presented on the principal trends of the theory and practice of reliability of nuclear power facilities. Opening the Conference, Academician N. A. Dollezhal' noted that reliability is one of the most important problems in nuclear power generation, and that this was the first seminar on this problem with such an extensive range of problems.

The reports presented at the Seminar can be divided according to subject into the following groups:

methodological problems of the quantitative assessment of the reliability indexes of the plant of nuclear power generating facilities during planning and operation;

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assessment of the reliability of pressure vessels and pipelines of nuclear power stations;
analysis of the actual reliability of nuclear power station plant according to operating data;
reliability of the core components of nuclear reactors;
monitoring, diagnosing the state of nuclear power station plant during operation, and ways to increasing reliability.

Among the reports of the first group, interest was created in the communication by E. F. Polyakov concerning the development of standard methodological data for calculating the reliability of nuclear power station facilities and their plant; V. S. Emel'yanov concerning procedure and programming for the computer processing of statistical data about the reliability of nuclear power station plant, taking account of statistical data about the reliability of nuclear power station plant, and with consideration of the specific properties of the data; R. A. Peskov, concerning the statistical assessment of the reliability of components according to current data; V. V. Postnikov, concerning the thermotechnological reliability of RBMK-1000.

The principal problem in the second group was the report of B. T. Timofeev, saturated with factual data about the statistical characteristic defects and mechanical properties of pressure vessels and pipelines of nuclear power stations and their consideration in the assessment of the reliability of components.

Among the reports of the third group should be mentioned the communications of S. S. Tulinov and B. P. Kruglova, concerning the reliability thermomechanical plant of nuclear power stations; V. D. Mikhailov, concerning the development, introduction, and experience in operating a selection system for statistical data about the reliability of nuclear power station plant.

Of the reports of the fourth group, the review by V. I. Solyanogo concerning the reliability of power reactor fuel elements was interesting; also the communication of V. V. Popov concerning methods of assessing the probability of trouble-free operation of fuel elements; V. N. Siryapin, on the statistical analysis of results of monitoring the hermeticity of fuel element claddings, and A. O. Poslavskii concerning the assessment of the reliability of the control and safety rods of the RBMK-1000 according to operating data.

Within the framework of the fifth group of reports should be distinguished the communications of A. O. Skomorokhov concerning the problem of diagnostics of the state of components of facilities, and E. K. Bezrukov concerning the organization of work to increase of reliability of nuclear power station plant in Minénergomash establishments.

SEMINAR ON STEAM-GENERATORS FOR FAST REACTORS

B. I. Lukasevich

In October 1978 at Appeldorn (Netherlands), a Soviet–Belgian–Netherlands seminar was held on steam-generators for fast reactors with sodium cooling. Twenty reports were presented reflecting the state of the program on steam generators and related problems.

It is well known that Belgium and the Netherlands, together with the Federal Republic of Germany are members of the "Interatom" organization and are participating in discussions of the SNR-300 fast reactor at Kalkar (Federal Republic of Germany). The contribution of Belgium and the Netherlands amounts to about 30%, of which 15% falls to the share of the Netherlands, where the steam generators are being developed, manufactured, and tested. It follows from the reports of the Netherlands and Belgian specialists that the extremely slow construction of the SNR-300 is having a negative effect on the development of the steam-generator program, and which is explained by the difficulty in agreeing the different problems of safety with the organs of authority and of the community. Although construction of the reactor started as long ago as 1973, the official period of startup is designated for 1983.

The Seminar and visit to establishments showed the high level of scientific–technical and industrial developments. The main interest was in the development of the reliability and safety of the steam generators for the SNR-300. Two types of steam generators will be used in this reactor: a straight-tube (in two loops) and a helicoidal (in a single loop) type, which reflects the most characteristic trends of development of sodium steam

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generators in the world. Steel, containing 2.25% chromium, 1% molybdenum, and stabilized with niobium, which is well known and widely used in steam-generator construction, has been accepted as the material. The straight-tube steam-generator, in layout and construction, is similar to the Soviet straight-tube steam-generator developed for nuclear power stations with BN-600 reactors. The assimilation of the manufacture and operation of steam generators of the two types, at present is the main problem of the Netherlands specialists and other types, including also the use of new materials, are now being developed on large scales.

There is an extensive program of research being conducted on steam generators in the following directions:

analytical investigations on the stability and safety of the steam generator. Using the ASME code, the lifetime of tubes in the zone of worse heat exchange with the water side has been calculated, and the assurance of the specified lifetime has been confirmed;

experimental investigations of the safety of the steam generators and of the sodium technology on small test-rigs. Special attention has been paid to the study of the detection and self-development of small water leakages into sodium. A system of protection of the steam generator has been formulated in the case of large water leakages, consisting of hydrogen detectors in the sodium and in the gas, a device for switching-off the steam generator and discharging the interaction products of sodium and water into special tanks;

full-scale investigations of the steam generators on a sodium test-rig at Hengelo (the Netherlands). The test-rig has the capability for testing steam generators with a thermal capacity of up to 55 MW. Thus, combined, prolonged tests of the steam generator module will be carried out, including checking of different normal and transition regimes. Here also devices will be checked which complete the steam-generator: accessories, instruments for hydrogen and acoustic leak detection, discharge and switching equipment;

development of the qualitative manufacture of the steam generators. At present steam generators of two designated types are being manufactured in two Netherlands' s factories (at Hengelo and Fliessingen). A technology and an automatic welding facility have been developed for welding the tubes with the tube plate, and also a microfocusing x-ray equipment for monitoring the weld seams. For manufacturing the helicoidal steam generators, a mill has been developed for winding the tubes of required radius and pitch.

All these problems were highlighted in the reports presented by the Netherlands specialists.

Reports of the Soviet specialists were devoted to similar problems. The main interest arose in the results of steam generator tests on the test-rig at Hengelo and the BOR-60 in the Soviet Union.

Extremely controversial in the opinion of the Netherlands specialists is the withdrawal from operation of the steam generator as a result of the detection of the first but weak indications of water leakage into the sodium. In 1976 the leak was recorded on the test-rig at Hengelo, but after taking the steam generator out of operation, its location was not found by any of the well-known methods.

Reports about the technology of manufacture of the steam generators were heard and discussed with great interest. It was noted that although the methods of joining the tubes with the turbine plates are different in the Netherlands and in the Soviet Union, the requirements for verification of the quality of the operations carried out and the manufacture of the steam generator, on the whole are identical.

An exchange of opinions also took place on different problems of sodium technology: fire-extinction methods, samplers, methods of cleansing plant from sodium and the interaction products of sodium with water. The vacuum cleansing of plant from sodium, perfected and being used in the Netherlands, was interesting.

The Seminar undoubtedly proved useful for the better understanding of the individual questions of steam-generator problems.

FRENCH–SOVIET SEMINAR ON REACTORS FOR HEAT SUPPLY

S. A. Skvortsov

The Seminar took place in December 1978 at Saclay (France). The first report of the Soviet specialists at this Seminar was devoted to the prospects of the utilization of nuclear heat for heat supply. The report cited the types of reactors which could be used in heat supply power stations for the generation of low-potential heat in the form of hot water and steam for technological requirements, the economic indices which could be achieved were assessed, and the ecological conclusions from the use of nuclear heat were demonstrated. In the four groups of reports, the development of water-cooled/water-moderated boiling-type reactors for use in nuclear boiler houses of heat supply stations was described. One report highlighted the general problems associated with the designing of these reactors and established the essential conditions for ensuring the safety and reliable operation of nuclear heat supply stations, in conditions of their location in the vicinity of the user. The special requirements for ensuring the operating safety of the reactors were defined.

One report was devoted to the assessment of the radiation safety of nuclear heat supply stations. In it, actual cases of possible discharges of activity were considered and the extent of the hazards were determined. In two reports, possible technological schemes of the reactor facility were described, and the most rational of them were selected. Two more reports were devoted to the use as coolant of organic liquids, which was recommended for use in the facilities of the far north, where the delivery of organic fuel is extremely difficult.

The French specialists presented eight reports. In one of them, the French CEA (Commissariat à l'energie atomique) organization was described, and also the long-term prospects for the use of atomic energy for heat supply were assessed. A special report was devoted to the conveyance and storage of heat in the conditions of use at heat supply stations. In other reports, a planned water-cooled/water-moderated reactor, TERMOS, was described, with a capacity of 100 MW (thermal), intended for installation in a heat supply station. In addition to a general description of the reactor, individual reports considered the equipment necessary for ensuring its safety, and also the reactor core. Highly enriched fuel in the form of laminated fuel elements will be used. The necessary experimental work associated with the use of TERMOS for heat supply was described, and also its prototype with one-half of the capacity was described.

The reports presented by both sides created great interest in those attending.

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SEMINAR ON THE WATER TREATMENT, WATER CYCLE, AND CORROSION PROTECTION IN THERMAL AND NUCLEAR POWER STATIONS

Yu. V. Balaban-Irmenin

In February-March 1979 in the "Electrification of the Soviet Union" pavilion at the Exhibition of Achievements of the National Economy of the Soviet Union, a Seminar and a School of Advanced Practice were run, devoted to the chemistry of water in power stations. More than 300 specialists from thermal and nuclear power stations, regional power directorates, design, repair, and maintenance organizations participated in the work of the Seminar and the School, which were organized by the F. É. Dzerzhinskii All-Union Heat-Engineering Institute. An exhibition operated at the Seminar, in which 146 exhibits were represented, reflecting the advanced achievements of Soviet power generation in such fields as the treatment of make-up water and condensate purification, water-chemical cycles and corrosion protection, technology of prestartup and operating flushings of plant, automation of technological processes and chemical monitoring of the quality of coolant, purification of sewage, and the treatment of cooling water.

The most interesting reports were presented on water-chemical cycles of high-capacity power units of electric power stations, where the search for new technological solutions are currently being conducted. Instead of the conventional cycle with correcting additions of ammonia and hydrazine, neutral conditions are proposed, with dosing by oxygen (reports of the Krzhizhanovskii Institute of Power Engineering), hydrogen peroxide (reports of the Beloyarsk nuclear power station and the Kostrom hydroelectric power station), and hydrazine (report of the F. É. Dzerzhinskii All-Union Heat-Engineering Institute). The first two cycles can be used for plant which does not have components of copper-containing alloys, but for the latter cycle there is no such limitation. The advantage of these cycles is the increased reliability of operation of the plant, because of the reduction of the rate of corrosion of the metal and the rate of formation of scale on heat-transfer surfaces, simplification of operation of block desalinating facilities (BDF), economy of chemical reagents used as correcting additives, and for the regeneration of the BDF filters. The neutral cycle with oxygen dosing was found to be the most widely used in thermal power stations at the present time.

In connection with the operation of power-generating plant in conditions of frequent startups and shutdowns, the papers and reports referring to conservation of the plant merited attention. In particular, the F. É. Dzerzhinskii All-Union Heat-Engineering Institute and Pavlodarénergo service have suggested and tested a quite simple facility which will allow nitrogen to be obtained from the furnace gases of boiler units operating on any type of fuel. The nitrogen produced by the facility, with a purity of not less than 99.75%, can be used for protecting plant which is kept as spare, from corrosion, and also for expelling hydrogen from power generators during maintenance, for creating a protective atmosphere in transformers, etc.

Among the new technological solutions in the field of water treatment, the purification of water by the reverse osmosis method (paper by the F. É. Dzerzhinskii All-Union Heat-Engineering Institute) and electrodialysis (report of the Institute of Thermoelectrical Design and the F. É. Dzerzhinskii All-Union Heat-Engineering Institute) may be mentioned. The use of these methods, which at the present time are passing through industrial proving, for the treatment of power station water, will reduce the amount of sewage. Both methods can be used also for treatment of technological water from nuclear power stations.

Successes in the field of automation of thermal and nuclear power stations were reflected in the exhibition and in the work of the Seminar. In particular, an assembly of automatic instruments was shown for the operative monitoring of the water-chemical cycle of high-capacity power-generating units. This assembly includes automatic oxygen- and hydrogen-measurements, conducto- and pH-meters, and hardness, silicon and sodium analyzers.

Reports were given at the Seminar and in the School on the chemical purification of the plant of thermal power station works (Boiler Cleaning, V. I. Lenin Moscow State Pedagogical Institute) and Decontamination of

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Nuclear Power Plant. New technological solutions, such as a device for the autonomous decontamination of the plant of nuclear power stations with water-cooled–water-moderated reactors, will enable plant maintenance to be simplified.

The Exhibition, Seminar, and School made it possible for thermal and nuclear power station workers to become acquainted with advanced achievements in the field of water chemistry at Minérgo power stations. The work displayed at the Exhibition will be introduced extensively into practice.

SOVIET–AMERICAN SYMPOSIUM ON HYBRID THERMONUCLEAR REACTORS

G. E. Shatalov

At the Third Symposium, held in January 1979 at Princeton (U.S.A.), the results of scientific-research work were given, which was conducted in more than 15 scientific laboratories and commercial firms of the U.S.A. The Soviet work was presented in reports mainly by contributors of the I. V. Kurchatov Institute of Atomic Energy, High-Temperature Institute, Academy of Sciences of the SSSR, and the All-Union Scientific Research Institute of Standardization in Machinery Manufacture.

Investigations of the possibilities of using hybrid thermonuclear reactors in power generation of the future have been conducted in the Soviet Union and in the U.S.A. during the last decade. The successes achieved in the field of controlled thermonuclear fusion, in particular on tokamaks, will allow the construction in the near future of demonstration thermonuclear facilities to be considered, with a power level which is characteristic for power stations. In the future, pure thermonuclear stations may be introduced into power generation, and will provide the solution of important problems of protection of the environment, etc. In the nearer long-term, however, it will be found advantageous to use the energy of thermonuclear fusion as a source of neutrons, by the capture of which in the blanket of a reactor containing nuclear raw materials (uranium-238, thorium), fissile isotopes will be made. The plutonium or uranium-233 produced, after reprocessing of the fuel elements (or without reprocessing), can be used for feeding nuclear reactors. The inclusion of hybrid thermonuclear reactors in power generation is one of the routes for solving the fuel problem of nuclear power generation, which according to the majority of forecasts will be a critical issue at the start of the next century.

At the Symposium designs of hybrid reactors were considered based on different concepts of thermonuclear facilities with magnetic and inertial plasma containment. At the present time, the best developed reactors are based on tokamaks. Designs of these reactors were presented for the American side by Princeton Laboratory and the firm of Westinghouse, and also by certain other laboratories. In contrast from the scientific developments of previous years, the designs contained detailed technical solutions on such fundamental constructional problems of the tokamak reactor as the diverter, dismountable blanket, etc. The design of a demonstration hybrid tokamak reactor with a power of 1-1.5 GW (thermal), intended for the generation of 300-400 MW (electrical) of power and the production of 250-400 kg/yr of uranium-233 or plutonium, is especially interesting. The assumed technical solutions (positioning of the blanket in the outer part of the facility, channel construction of the fuel element rods, water cooling, externally located windings of the poloidal field, etc.) are based on modern technology and will allow a facility to be built with great experimental capabilities. According to the assumptions of the authors, this facility can be built at the beginning of the 1990s.

The symposium showed a marked interest toward linear systems with magnetic plasma containment. The reason for this is their more suitable geometry, which is important for realistic engineering consideration. In the first place, this relates to reactors based on open traps with double plugs, on which papers were given from the American side by the Lawrence Livermore Laboratory, the firms of General Atomics, General Electric, Bechtel, etc. and from the Soviet side by the I. V. Kurchatov Institute of Atomic Energy and the High-Temperature Institute. The use of these systems as the basis of a hybrid thermonuclear reactor with a plasma intensification factor of 1-2 is very promising. The systems will be able to operate with the injection of a power of 300-400 MW into outside traps, in the form of deuterons with an energy of 200 keV. The use of the direct energy conversion of the plasma, flowing through the external traps with an efficiency of 0.7-0.8, is es-

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sential for the characteristics of these systems. The overall conversion factor of heat into electricity (net) may reach 0.25-0.3 with maintenance of a nuclear fuel production of 600-100 kg/GW (thermal)·yr. The advantage of open traps with double plugs consists in their linear geometry, which permits problems of disassembly and operation of the reactor to be solved relatively easily, in the possibility of achieving a continuous operating cycle, and in the requirement of a relatively small magnetic field in the main power section of the trap. An experimental facility of this type has been built in the Livermore Laboratory. Operation with the open traps is planned to extend into 1979-1980.

The characteristics of pulsed hybrid thermonuclear reactors with inertial plasma containment were considered in papers from the Los Alamos Scientific Laboratory, Wisconsin State University, the firm of Bechtel, and the I. V. Kurchatov Institute of Atomic Energy, and High-Temperature Institute. Systems were considered with a target ignited by a laser and with a beam of relativistic electrons. In the reports, quite optimistic views were expressed on the possibility of building these systems; however, the degree of their development is so much lower than systems with magnetic plasma containment. All systems require either a plasma intensification factor of 100-1000, or a considerable energy intensification in the blanket due to its proximity to criticality. The most important problems remain those of cyclic thermal stresses in the first wall and in the fuel elements with the number of cycles 10^7-10^8, and for systems with relativistic electron beams – beam transportation. At the same time, economic estimates indicate that systems with inertial containment may be no less profitable than systems with magnetic plasma containment.

Considerable attention was paid at the Symposium to the consideration of fuel cycles (uranium–plutonium, thorium) and the possibilities of enrichment of the fuel elements from natural or uranium tailings in the blanket of the hybrid reactor. An analysis, carried out with consideration of the engineering reasoning, shows that one hybrid thermonuclear reactor can provide the make-up for 4-6 water-cooled–water-moderated (VVÉR) type reactors of the same power, using the uranium–plutonium cycle, and 5-10 using the thorium cycle. However, in the case of the thorium cycle, high specific loadings are required at the first wall of the chamber, and therefore this cycle has a better prospect for use in the blankets of pulsed hybrid reactors, where the chamber dimensions are not directly related with the thermonuclear source. For reactors of the tokamak type, the production of specific loadings of 4-5 MW/m^2 is still questionable for an acceptable total power of the facility of 5-7 GW (thermal).

The development of symbiosis systems is interesting, in which the fuel elements, made of natural or depleted uranium or thorium, are enriched with a fissile isotope by irradiation in the blanket of a hybrid reactor, and are then put into a nuclear reactor without additional reprocessing and refrabrication. In principle, this cycle can be repeated several times. In this case, one hybrid reactor can feed approximately two nuclear reactors of the VVÉR type, but the cost of electricity generated by this system will be increased. Whether these economic considerations of advantage, obtained because of the absence of reprocessing and refabrication of the fuel elements in this system can be surpassed, will be shown by further research.

At the symposium, such problems were also considered as those associated with investigations and design developments of hybrid thermonuclear reactors. Among them, the following may be mentioned.

In the field of investigation of fundamental fusion systems, a tokamak reactor operating on a D–D plasma and sustained with a powerful injection was discussed. The plasma intensification factor of this reactor is not high (>1), and it can be considered mainly as a reactor for the production of fissile isotopes. In this field, its preference because of the absence of necessity to produce tritium is obvious. One of these reactors could feed 20-25 VVÉR-type reactors of similar power. However, a more detailed engineering consideration will be necessary for these conclusions.

Great attention was paid to the search for materials capable of enduring for the entire operating period of the reactor (20-30 years) without failure. In addition to the extensively considered austenitic steels, in many designs preference has returned to ferrite steels because of their low swelling and adequate plasticity with a fluence of ~10^{23} neutrons/cm^2.

A large discussion took place concerning the possibility of conducting experimental investigations on modules of the blanket in such thermonuclear facilities, under construction and planned, as TFTR (Princeton), "Shiva-Nova" (Livermore) and T-20 (Soviet Union). Plans of these modules were presented. Proposals for carrying out work on the TFTR facility, which may be completed in 1983-1984, were particularly interesting. The desirability for international collaboration when conducting these experiments was expressed.

The symposium showed that the work on hybrid thermonuclear reactors in the Soviet Union and in the U.S.A. is increasing without decline. Almost all the commercial firms in the U.S.A. engaged in the production

of nuclear reactors now have groups designing hybrid thermonuclear reactors. The conduct of regular symposia on this subject will allow the progress in the development of the problem to be assessed. In 1978 the main emphasis was placed on the engineering development of reactors and the detailed study of demonstration hybrid reactors, for which the start of technical planning is proposed in the next few years. Problems of the fundamental possibilities of hybrid reactors in solving the fuel problem of nuclear power generation, and earlier being argued, have received considerable engineering confirmation.

TWELFTH EUROPEAN CONFERENCE ON THE INTERACTION OF LASER RADIATION WITH A SUBSTANCE

V. Yu. Baranov and A. Yu. Sebrant

The Conference, held in December 1978 in Moscow, was organized by the Institute of Physics, Academy of Sciences of the SSSR (FIAN), the State Committee for Atomic Energy of the SSSR, and the All-Union Society, Znanie. About 300 specialists participated, including more than 50 scientists from Australia, Belgium, Great Britain, the German Democratic Republic, Canada, Poland, U.S.A., France, and Japan. Academician N. G. Basov opened the Conference and the Vice-President of the Academy of Sciences of the SSSR Academician E. P. Velikhov addressed a welcoming speech to the participants.

Reports were presented at the Conference on the latest achievements of the various laboratories of the world in the following principal trends: high-powered laser systems for controlled thermonuclear fusion and programs of work; the interaction of high-powered laser radiation with a substance; laser compression and thermonuclear fusion, and laser plasma diagnostics.

Several reports were presented by the Lawrence Livermore Laboratory (U.S.A.), which has available the most powerful laser system today, Shiva, and which will allow experiments to be conducted at a power of up to 30 TW in a pulse with a duration of 1 nsec and at a wavelength of 1.06 μm. It was reported that the neutron yield amounts to $3 \cdot 10^{10}$ and the ion temperature was 6 keV. Dr. McCall from Los Alamos Scientific Laboratory (U.S.A.) spoke about the results obtained on the 8-beam CO_2 laser facility "Helios." In the experiments carried out at a power of ~10 TW, a neutron yield of $\sim 2 \cdot 10^8$ was demonstrated. M. Lubin from Rochester University, U.S.A., reported on the experiments in the 6-beam "Zeta" facility, operating on neodymium glass. With a power of 3-4 TW, a neutron yield of $\sim 3 \cdot 10^8$ was observed. A 24-beam laser system, "Omega," with a power of 16 TW has been developed and built at Rochester, the special feature of which is the use in the final cascades of amplification, of active mirrors instead of the conventional disk amplifiers.

Polish scientists from the Warsaw Institute of Plasma Physics and Laser Microfusion spoke about the extensive program on laser controlled thermonuclear fusion. In the Institute, a 4-beam neodymium glass facility with a pulse energy of 4×50 J has been brought into operation, and preparations are being made for experiments on the compression of spherical targets on this facility. Here, further, a CO_2-laser system is being developed, with a calculated energy yield of 4×1.6 kJ. The basic units of the facility have been tested and experiments have been conducted on the compression of a D–T mixture in conical targets with a pulse energy of the CO_2-laser of 200 J and a duration of 100 nsec. No neutrons have been recorded. The construction of a CO_2-laser with an energy of 1.2 kJ has been completed, for the supplementary heating-up of the plasma in systems of the "plasma focus" type.

Japanese scientists spoke about the production of a neutron yield of $\sim 10^7$ on the "Gekko-IV" facility, with a power of 4×1 TW and a wavelength of 1.06 μm, with a pulse duration of 100 psec. In Osaka University, alignment of the two-beam CO_2-laser system "Lekko-II" has been completed, and in the near future experiments on the compression of spherical targets will be conducted on it.

French specialists reported on experiments in the 8-beam laser system "Octal," operating on neodymium glass at Limée. With a power of >5 TW, a neutron yield of $\sim 5 \cdot 10^7$ has been obtained. Work is being carried out at Limée on the comparison of the utilization efficiency of a different wavelength of laser radiation for

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heating up the plasma. Experiments are being conducted on the 4th, 2nd, and fundamental harmonics of the emission of the neodymium laser, and also with radiation at a wavelength of 10.6 μm.

Scientists from FIAN spoke about the experiments on the 9-beam facility "Kal'mar," in the nanosecond pulse regime. The values obtained for the neutron yield were found to be in the range 10^4-10^6. Several reports concerned the construction of the "Delfin" facility, in which specialists of the German Democratic Republic participated in the development of the optical and diagnostic sections. Specialists from the I. V. Kurchatov Institute of Atomic Energy and the D. V. Efremov Scientific-Research Institute of Electrophysical Equipment (NIIEFA) reported on a joint program for the application of pulsed CO_2 lasers in thermonuclear research. Work is being conducted in two directions: the use of pulses of radiation of nano- and microsecond duration for heating up the thermonuclear targets and the plasma in a θ pinch, respectively. In the reports about less powerful laser systems, operating in many scientific centers, programs were presented for research into the physics of interaction between laser radiation and the targets, in which the main attention is paid to refining the absorption mechanisms of a high-powered plasma, and also the mechanisms of heat transfer in the plasma, and the effect of magnetic fields and fast particles on the interaction process.

Great attention was paid at the Conference to the design of an optimum target for laser controlled thermonuclear fusion. In several reports, plans of complex shell targets were shown, which according to calculations will allow a density in the compressed nucleus to be obtained, which will exceed the density of the condensed state by a factor of ~1000, and will provide a considerable thermonuclear yield. Possible versions of cryogenic targets were also considered, and problems associated with the production and monitoring of targets for laser controlled thermonuclear fusion. In many reports, the effect of the pulse shape (presence of a prepulse) on the interaction process was discussed. It was noted that in the experiments with spherical targets, filled with a gas mixture, high contrast values are necessary, as damage to the target takes place even with an incident energy of $\lesssim 100\ \mu$J.

In the reports devoted to the interaction of laser radiation with targets, numerous experimental and theoretical results were given on the hydrodynamics of the plasma corona, modification of the density profile, resonance absorption, the generation of magnetic fields, and the transport mechanisms in the plasma. Good agreement between theory and experiment was obtained over a wide range of intensity up to 10^{16} W/cm^2, on the twisting of the density profile, the absorption coefficient, the generation of harmonics, and also fast particles. The achievements in the development of diagnostic techniques, in particular the construction of equipment with a high spectral and time resolution in both the optical and x-ray range, have enabled the fine effects to be detected, which take place in the picosecond range of duration and which do not have a quite complete theoretical explanation. In the reports concerned with laser plasma diagnostics, the wide possibilities of modern experimental procedures were shown. The time resolution of optical and x-ray equipment amounts to several picoseconds, which makes it possible to obtain detailed information about high-speed processes in the plasma, such as the time variation of the spectral composition of the reflected radiation, the dynamics of magnetic field generation, and target compression processes. Considerable attention was paid to x-ray methods of diagnostics, which are able to give accurate and multilateral information about the plasma parameters, which are essential for carrying out computer calculations.

The high neutron yield achieved on high-powered laser facilities has served as a stimulus for the development of procedures using the fusion products. In Livermore, a camera-obscura and a zone plate are being used for producing an image of the compressed target nucleus by means of α particles. Development and experiments have been conducted on the production of radiochemical targets, containing ^{63}Cu, ^{197}Au, and ^{40}Ar in the composition of the cladding or in the gas mixture.

In the opinion of some speakers, the new facilities which have been brought on stream in recent years, and the achievements in the understanding of interaction physics, have given the basis for supposing that a physical demonstration of the feasibility of creating a thermonuclear reactor based on a laser will take place within the next 5-7 years.

The next Conference is scheduled for December 1979 in Leipzig (German Democratic Republic).

NEW BOOKS

PROBLEMS OF NUCLEAR SCIENCE AND TECHNOLOGY*

A. M. Petros'yants

Reviewed by Yu. I. Koryakin

If one attempts to answer the question as to what principal reason the book under review has been successful, having been issued in four editions during the last 8 years, then the answer can resound with the paradox that there is no reason. It is more correct to speak about a combination of reasons, each of which contributes to the success to an equal extent. Undoubtedly, the subject is of great importance, but this is small. The timeliness and significance of a subject must be matched by the skillful supply of material, its freshness, style of discussion, the load distribution between numerical and descriptive data, the arrangement of accents between the particularly scientific side of the matter and its popularization, illustrations, authors' remarks and other attributes, constituting a single and purposeful ensemble of the form and content of a book.

The book possesses all these qualities, and they will ensure its success not only in the Soviet Union but also abroad. The book has been issued in the German Democratic Republic, Czechoslovakia, Romania, U.S.A., and it is being issued in Poland, Hungary, and France. To be sure, it must be said that to revise the book nevertheless is easier than to write it from the beginning. But the correct beginning was found by this author, and time has verified it.

The author of the book is well informed in the field of Soviet and international nuclear life, which is important for the book, as it reflects recent important changes, events, and special features of the rapidly running and dynamic nuclear problem. Thus, in the edition being reviewed, new sections have been introduced on nuclear central heating and heat supply, a detailed review of the reports and discussions of the Salzburg Conference on Nuclear Science and Technology and its Fuel Cycle (May 1977) has been given, and the achievements of Soviet and foreign nuclear science and technology which have taken place after the previous issue. The sections on Soviet nuclear centers have been expanded, the section on the use of nuclear engines in ships has been rewritten, the sections on the use of nuclear energy in the member-nations of COMECON and on work in this direction within the scope of COMECON have been renewed considerably, and new data and arguments of information on the safety of nuclear power stations and radioecological problems have been added. Other changes and additions have been introduced.

All this has made the new edition of the book more weighty and multisystemmatic. The latter is valuable in that readers of a different level of qualification can find in it what is appropriate and necessary for themselves. Numerous tabular and graphical data allow abundant factual data to be available to the reader. In this sense, the book possesses encyclopedic features to a certain degree. One cannot but speak about the style of presentation of the material. It is restrained, without being dry, but at the same time it is expansive without changing over to enthusiasm.

The broad panorama of the development and present-day state of human knowledge in the field of utilization of atomic energy drawn by the author, nevertheless gives rise to a certain guardedness for the reviewer. It is explained by a certain fetishism of atomic energy creeping in, his way in tending to attribute to it a coloring of "azure tones." In doing justice to the exceptional importance and undoubted prospects of atomic energy, one cannot but acknowledge that it also has shadowy aspects, and also its dramatic nature of development, which by far does not always resemble a ceremonial procession. The author, in his undoubted justified optimistic opinion, nevertheless prefers either not to notice or to draw a veil over the serious and still unresolved problems. These problems result from the successes of nuclear power generation, the expansion of its scales and the number of nuclear power stations built, i.e., they are problems of the system of its growth and problems of quantity. For example, these involve the storage of radioactive waste, the disposition of nuclear power stations, overcoming the observed psychological negativism, and the general "human factor" in the development of nuclear power generation. These problems are not simple and undoubtedly will require for their solution not only material but also intellectual efforts. The author of the book, it seems, does not attach great importance to these problems. The reviewer takes upon himself the boldness to assert that the main problems still lie ahead.

*Atomizdat, Moscow (1979), 456 pp., 3 rubles, 90 kopecks.

Translated from Atomnaya Énergiya, Vol. 46, No. 6, pp. 430-431, June, 1979.

0038-531X/79/4606-0495$07.50

Finally, the arrangement of the accents is the right of the author, and what is more is competent. But the title of the book begins with the word "Problems....", and it would be desirable that they should be discussed but not in an undertone. The book could only additionally profit from this.

SPARK CHAMBER TECHNIQUES*

A. A. Vorob'yev, N. S. Rudenko, and V. I. Smetanin

Reviewed by B. P. Maksimenko

The successes achieved in recent years in studies of high-energy particles would have been impossible without the techniques of modern experimental equipment: track detectors of the spark (trace in the form of a spark channel) and streamer (trace in the form of a chain of streamers) types of chambers operating on the principle of the initiation of an electric gas discharge by electrons, which are formed by high-energy particles during passage through the discharge space. These chambers, with a size of several meters and with quite good parameters – spatial isotropicity, trace localization accuracy, luminosity brightness of the trace, etc. – are fairly complicated equipments, both from the point of view of the physics of the discharge development, as well as in relation to their design and the high-voltage power-supply equipment. Because of this, the emergence of the book being reviewed is very timely.

When describing the processes of the formation of the tracks of charged particles in a gas, filling the spark chamber, the dynamics of the development of the streamer process have been analyzed in detail. The mechanisms of development of the positive (anode) and the negative (cathode) streamers are described and the basic condition for transition from avalanche to streamer is formulated. As applicable to localization of the spark discharge in the track of the charged particle in the track spark chamber, the mechanism of formation of an oblique spark channel and the conditions in which this type of breakdown are discussed (the time of formation of the spark, the rate of increase of brightness, etc.). The conditions of formation of the charged particle in the streamer chamber are discussed (luminosity brightness of the streamers, their size in the track and their displacement from the point of origin of the initiating electrons, their number per unit length of track, process of self-propagation) and the requirements for the nanosecond high-voltage pulse generators, used for supplying the chamber, are defined.

In considering the pulsed chamber power supply, the generalized layout of the nanosecond generator is analyzed, from which it follows that to obtain pulses with specified parameters reduces to the development of a low-inductive storage unit, the commutator, determining the leading edge and amplitude stability of the nanosecond pulse, and the clipping device, responsible for the shaping of the trailing edge and stability of the pulse duration. The basic principles of construction of controllable commutators, clipping spark gaps and actual designs are discussed, as used in nanosecond pulse techniques, and also layouts of generators with low-inductive capacity storage units and a double shaping line. As applicable to the design of generators, engineering methods of calculating gaseous, liquid, and solid insulators used in various units are explained, and the basic characteristics of insulation materials are given.

The possible improvements of the existing types of spark chambers are described in the book – the use of a multipulse power supply, mutually perpendicular arrangement of pairs of electrodes, to which voltage pulses are fed individually with a time shift relative to one another, etc.

The authors have succeeded, in a book of small volume, in systemmatizing the considerable information about spark and streamer chambers. There is no doubt that the book will be useful to physicists–experimenters and development workers on such equipment.

*Atomizdat, Moscow (1978), 1 ruble, 20 kopecks.

Translated from Atomnaya Énergiya, Vol. 46, No. 6, pp. 431-432, June, 1979.

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ATLAS OF THE DOSE CHARACTERISTICS OF EXTERNAL IONIZING RADIATION* (HANDBOOK)

E. E. Kovalova (editor)

Reviewed by Yu. V. Sivintsev

The extensive introduction of sources of ionizing radiation in different spheres of activity of mankind – from science and technology to art and criminology – has made necessary the generalization of the numerous data on the dose characteristics of ionizing radiations. The authors of the atlas being reviewed have systematized these data for seven types of radiation and for a wide range of their energies, as applicable to the conditions of external irradiation like the so-called standard geometry of the one-sided irradiation of a phantom. For the latter, a semiinfinite layer of tissue-equivalent substance (resembling water) has been chosen, simulating soft tissue with a thickness of 30 cm.

In the introductory section data are concentrated about the basic concepts and definitions and the transfer processes of the radiation energy to the medium.

The main part of the handbook is occupied by detailed data (12 tables and 36 figures about the dose characteristics of γ radiation over the energy range 10 keV to $2 \cdot 10^4$ MeV, electrons ($5 \cdot 10^2$–10^6 MeV), ions of both signs (10–$5 \cdot 10^6$ MeV), protons (2–$5 \cdot 10^6$ MeV), neutrons (0.025 eV–$5 \cdot 10^6$ MeV), and heavy nuclei with charge 2 or more (2–$1 \cdot 10^4$ MeV/nucleon).

The concluding section contains recommendations worked out by the author's staff (maximum value of the specific equivalent dose, conversion factor – the flux density of the particles producing unit intensity of equivalent dose of 1 mrem/h, and the irradiation nonuniformity coefficient).

The atlas undoubtedly will be useful to specialists working with radiation, including designers and planners of radiation protection, exploiters of new metric instruments, investigators in the field of nuclear physics, shielding physics, dosimetry, radiobiology, etc.

*Atomizdat, Moscow (1978), 660 pp., 20 kopecks.

DESIGN PRINCIPLES OF NUCLEAR REACTOR CONTROL MECHANISMS*

I. Ya. Emel'yanov, V. V. Voskoboinikov, and B. A. Maslenok

Reviewed by I. T. Gusev

The book, by eminent specialists in the field of development of nuclear power-generating facilities, is of undoubted interest, as in it for the first time, the experience in the development of these mechanisms by Soviet establishments and by the principal foreign firms has been generalized. The approach of the authors to design methodology is worthy of special attention. Several chapters of the first part of the book are devoted to the analysis of the factors defining the operating conditions of the control mechanisms (coolant, temperature, ionizing radiation, etc.) and the requirements imposed on them as a function of the type and designation of the facility. The effect of the operating conditions of the facilities on the approach to designing the control mechanisms is considered by specific examples, and it is shown how as a result of this, the formulated requirements are achieved, e.g., the assurance of nuclear safety and reliability, the quality of achievement of the basic functions of the control system, etc.

*Atomizdat, Moscow (1978), 272 pp., 2 rubles, 90 kopecks.

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0038-531X/79/4606-0497$07.50

Based on the consideration of the control mechanisms, the authors analyze the structure which will allow the design principles of different mechanisms to be developed, effecting operation according to a program defined by the control system. The study of the designs of individual structural elements of the mechanisms is the logical completion of the proposed methodology.

In the book a large number of designs of mechanisms are analyzed for almost all types of reactors: power, research, ship, and for space vehicles. Certain control mechanisms are cited in the technical literature for the first time, e.g., for the RBMK power reactor and the MIR and SM-2 research reactors. For the first time, designs of linear electromagnetic mechanisms are considered.

Great attention is paid to the electrical equipment of control mechanisms and, in particular, to special electrical machines, used in reactor construction. Numerous examples of the production of special electrical machines (asynchronous and synchronous-thrust electric motors, vacuum sealed step electric motors) and illustrations assist the reader to form a picture of the present-day level of the special electrical plant used as electric drives and means of measuring the position of the control rods of reactors.

The section of the book devoted to the planning and calculation of linear discrete control mechanisms is of considerable interest. In these mechanisms the forces of an electromagnetic field are used for moving the control organs. The fast response, absence in the primary circuit of the reactor of mechanical transmissions, frequency control of the speed over wide limits, predetermine the prospects of these mechanisms and their high reliability. The discrete form of the control pulses applied to the motors facilitate the use of digital computers for controlling the reactor facility. In this section data are also given on the calculation, research, and technology of manufacture of discrete mechanisms obtained on the basis of research which has been carried out by the authors and with their direct participation.

In a separate chapter, structural materials used for the manufacture of control mechanisms are considered, including also materials for biological shielding, welded joints, corrosion properties of materials, etc. As the control mechanisms are one of the principal components for ensuring the nuclear safety of reactors, considerable attention is paid in the book to increasing the operating reliability of mechanisms, a procedure is given for their calculation in reliability, and statistical data are given concerning the reliability of Soviet and foreign nuclear power station mechanisms. A separate chapter of the book is devoted to the estimate of the degree of technology and efficiency.

In a book, which is relatively small in volume, the construction of reactor control mechanisms is considered taking account of the prospects of development of nuclear power generation. It will be useful to engineers and technicians occupied with the designing of nuclear power facilities, students of universities and technical schools when studying a course on reactor design, and also young specialists.

FIFTH SCIENTIFIC CONFERENCE ON POWER GENERATION IN THE HIGHER SCHOOL OF ENGINEERING AT ZITTAU (GERMAN DEMOCRATIC REPUBLIC)

In honor of the 30th year since the day of foundation of the German Democratic Republic and the 10th anniversary of the formation of the Higher School of Engineering in Zittau, a scientific conference on power generation will be held from Nov. 14 to 16, 1979. The subject of the conference is "The Rational Production and Utilization of Energy." The conference will be conducted with the participation of specialists of the German Democratic Republic and foreign governments. The plenary session will be held in the first half of the day of Nov. 14, 1979. Then discussions will take place in five sessions:

SESSION 1

Control and planning of scientific–technical progress in power generation. At this session, in particular, practical problems of the subsequent improvement of control and planning of scientific–technical progress in power generation will be discussed. The center of attention of the session is the putting into practice of theory and exchange of experience.

SESSION 2

Results and problems of scientific–technical progress in power station technology. At this session data will be presented about the present-day state of technology in the field of planning, development, and operation of conventional and nuclear power stations, and also problems will be considered associated with scientific–technical progress: the assessment and development of power stations and operation.

SESSION 3

Rationalization of planning and operation of electric power-generating systems using computers. The use of computers represents a decisive prerequisite of the rational solution of the problem of development and operation of electric power-generating systems. Their efficient utilization presupposes the use of effective algorithms and mathematical methods, and also the formulation of numerical programs convenient for the users, and a suitable selection. These points of view will be considered in the following thematic directions: computers, planning, and operation.

SESSION 4

Fuel economy owing to the use of secondary power and the energy of the environment. Both the increasing cost in fuel and the problem of resources, in conjunction with urgent measures for protection of the environment, create the necessity for a more rational utilization of energy-carriers. Pertinent possibilities are, together with others, the intensified use technologically of specified secondary power and the energy of the environment. Possibilities and experience in the use of these forms of energy in order to ensure a rational total utilization of energy are the subject of the work of this session, which will be conducted according to the following theme: the utilization and assessment of the use of secondary power and the energy of the environment.

SESSION 5

Determination of the periods and volume of technical servicing of power-generating facilities. Within the scope of the session, the following thematic questions will be considered: costs in technical servicing of power-generating facilities during their operation; technical servicing using diagnostic methods; theoretical and experimental approach to the investigation of wear processes; results of technical diagnostics; use of scientific labor organization methods for the basis of costs in technical servicing.

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The Prorector for Natural Science and Technology of the Higher School of Engineering in Zittau, Professor Dr. of Econ. Sci. G. Schumann, 88 Zittau, Theodor-Kerner Allee, 16, is responsible for the preparation and conduct of the Conference. Interested persons from the country and from abroad may make application to this address to obtain an invitation to the Conference and also additional information.

As on previous occasions, so also within the scope of this Conference, evening scientific meetings will be added. More detailed information will be given in the invitation or it can be obtained from the organizer.

NEW RUSSIAN JOURNALS

IN ENGLISH TRANSLATION

BIOLOGY BULLETIN

Izvestiya Akademii Nauk SSSR, Seriya Biologicheskaya

The biological proceedings of the Academy of Sciences of the USSR, this prestigious new bimonthly presents the work of the leading academicians on every aspect of the life sciences—from micro- and molecular biology to zoology, physiology, and space medicine.

Volume 5, 1978 (6 issues) $175.00

SOVIET JOURNAL OF MARINE BIOLOGY

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Evaluates the water resources of specific geographical areas throughout the world and reviews regularities of water resources formation as well as scientific principles of their optimal use.

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LITHUANIAN MATHEMATICAL JOURNAL

Litovskii Matematicheskii Sbornik

An international medium for the rapid publication of the latest developments in mathematics, this quarterly keeps western scientists abreast of both practical and theoretical configurations. Among the many areas reported on in depth are the generalized Green's function, the Monte Carlo method, the "innovation theorem," and the Martingale problem.

Volume 18, 1978 (4 issues) $150.00

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Programmirovanie

Reports on current progress in programming and the use of computers. Topics covered include logical problems of programming; applied theory of algorithms; control of computational processes; program organization; programming methods connected with the idiosyncracies of input languages, hardware, and problem classes; parallel programming; operating systems; programming systems; programmer aids; software systems; data-control systems; IO systems; and subroutine libraries.

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