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SOVIET ATOMIC ENERGY

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G. L. Lunin


E. S. Chernorotov


V. F. Ostashenko


V. V. Stekol'nikov


V. P. Denisov


N. I. Gorelov


N. M. Kantiev


L. M. Voronin


A. B. Sukhov


P. M. Verkhovykh


V. K. Kalashnikov


A. T. Glushkov

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ARTICLES

TEN YEARS OF OPERATING EXPERIENCE AT THE USSR FIFTIETH ANNIVERSARY NUCLEAR POWER STATION AT NOVOVORONEZH

A. N. Grigor'yants, F. Ya. Ovchinnikov, V. K. Sedov, L. I. Golubev, Yu. A. Akkuratnov, I. V. Prokopenko, I. I. Kustov, N. A. Isakov, V. D. Dobrynin, A. V. Tsybul'nik, and V. I. Skrypnikov

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The Novovoronezh nuclear electric power generating station went into operation on September 30, 1964, when the first experimental full-scale power-generating unit, rated 210 MW(e), began to feed current into the national grid.

The Novovoronezh nuclear power station is the largest nuclear electric power generating plant built in our country. The total ratings of its four nuclear power generating units total 1455 MW(e).

Construction work on the NVAÉS (Novovoronezh nuclear power station) has been proceeding at accelerated tempos. The first power generating unit with its VVÉR-210 reactor was built in the space of 7.5 years; the second power generating unit with its VVÉR-365 reactor went on the line in 1969, a mere 5.5 years since ground was broken for the plant. The principal quantity-manufactured power generating units with their VVÉR-440 reactors were installed and put into service in even shorter periods of time. The building of the third and fourth power generating units was begun in 1968, and construction work was completed within less than five years, with a one-year offset, the third being completed in 1971 and the fourth in 1972. The equipment of the fourth power generating unit was installed within one year at most. The construction program for the first echelon of the NVAÉS power station was completed successfully when the fourth power generating unit began generating power for the national grid.

Startup and adjustment operations were carried out in stages, so that adjustment of subsystems could be attended to several months in advance of the completion of installation and rigging operations, to be followed by comprehensive adjustments of overall equipment and entire systems. This method was relied on for startup of several other nuclear power stations, such as the Kola station, the Nord power station (in GDR), the Kozlodui power station (in Bulgaria), and others.

In addition to the speedup in construction, installation, rigging, and startup and adjustment operations, the lead time for installation of capacity following the physical startup and power startup was also shortened. This was particularly striking in the case of the quantity-manufactured units with VVÉR-440 reactors. The scheduled year-long period was cut down to six months when the third power generating unit went on the line, and that period was cut down to a mere 83 days in the case of the fourth power generating unit.

Engineering cost indices for the power station (see Tables 1-3) have been continually improved in the process of putting the plant into operation and expanding it.

Versatility of the Novovoronezh Nuclear Power Station. The power station operates as an integral part of the Voronezhénergo power grid, mainly at base load with a slight drop in load at the end of the week and on holidays or off-days, when the number of industrial consumers of electric power decreases.

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TABLE 1. Characteristics of NVAÉS Basic Power Equipment

Equipment	1st power generating unit	2nd power generating unit	3rd and 4th power generating units
Reactors			
type	VVÉR-210	VVÉR-365	VVÉR-440
thermal power, MW	760	1320	1375
weight of pressure vessel, tons	185,4	209,2	200,8
dimensions of pressure vessel, m	4,4×11,1	4,4×12	4,27×11,8
operating coolant pressure, kgf/cm^2	100	105	125
Steam generators			
capacity, tons/h	230	325	455
saturated steam pressure, kg/cm^2	32	33	47
heat transfer coefficient, kW/m^2·°C	4,29	4,37	4,32
heat transfer surface area, m^2	1290	1800	2500
U-tube diameter, mm	21×1,5	16×1,4	16×1,4
number of tubes	2774	3664	5146
dimensions of steam generator, m	3,06×11,57	3,07×11,57	3,27×11,99
Main circulation pumps			
type	GTsÉN-138	GTsÉN-309	GTsÉN-310
power intake, kW	1650	1530	2260
capacity, m^3/h	5250	5600	7000
head, kgf/cm^2	4	5	5
average duration of maintenance-free operation, h	5000	20 000	15 000 *
Turbogenerators			
output, MW	70	73	220
number	3	5	2

*From the time the third power generating unit was started up.

A constant shortage of electric power in the grid imposes stringent limitations on the length of reactor fuel reloading shutdown periods, which occur in spring and summer months for the most part. In some instances, particularly during the autumn and winter peak loads, a need is felt to stretch the in-pile time of the fuel loading by cutting down on power output toward the end of the reactor campaign.

Operating the reactor on power effect lowers the fuel component of the net power costs. But a decrease in the amount of electric power generated and an increase in the capital component of the net power costs restricts the time over which operation under that set of conditions can be justified economically.

When heightened wear on the equipment as the power generating unit is operated at reduced power output levels is taken into account, we see that reliance on power effect must be treated as both forced and provisional.

Dynamic tests showed the reactors exhibit excellent self-regulating properties in the face of such major disturbances as tripping out of one or several main circulation pumps, outing of the turbogenerator by disconnection from the power line, and so on. The self-stabilization capability of the reactor eases the transition to the on-power level corresponding to the new allowable operating conditions.

Some experience has already been acquired at the power station in altering the power output of the power generating unit over the course of a full day in order to cover the morning and evening peak loads in the power system. For example, the second power generating unit operated under that set of conditions throughout January of 1972. The power output of the power generating unit was maintained at the rated level every day during the morning and evening hours of peak demand, and cut back 30% at nighttime during that period.

The degree of adjustability of the nuclear power station is also characterized by the comparatively short time it takes to bring the power generating unit up to full power after a prolonged shutdown period for cooldown of the primary loop. The total time it takes to carry out a thoroughgoing complex checkout of the reactor control and emergency protection system, to attain the primary loop parameter ratings, to heat up the turbogenerators and synchronize them with the overall system, amounts to 15 h more or less. The same time is required for reactor cooldown and to adjust the reactor to the appropriate state for maintenance and repair operations.

The operating stability of the power station is of exceptional importance for normal electric power supplies to the consumers. The stability exhibited by the NVAÉS is demonstrated to be quite high by the data entered in Table 4. The total number of shutdowns experienced by the third, second, and fourth power generating units over the 1970-1973 period was 57. The bulk of these shutdowns (43 out of the 57) took place while the power generating units were being adjusted and run through their first paces. Once adjusted to power level, the power generating units have been operating with excellent stability.

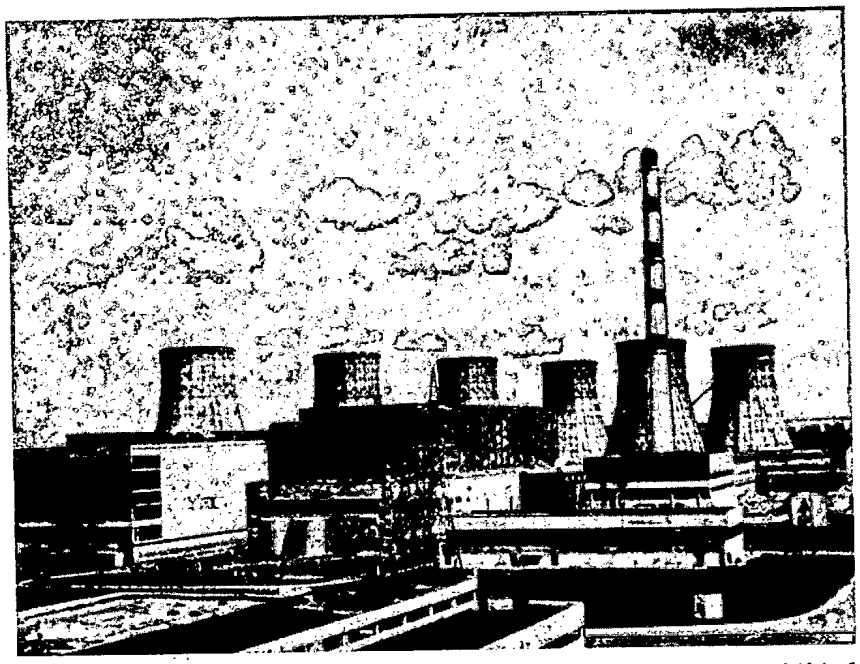

Fig. 1. Novovoronezh nuclear power station. General view of third and fourth power generating units.

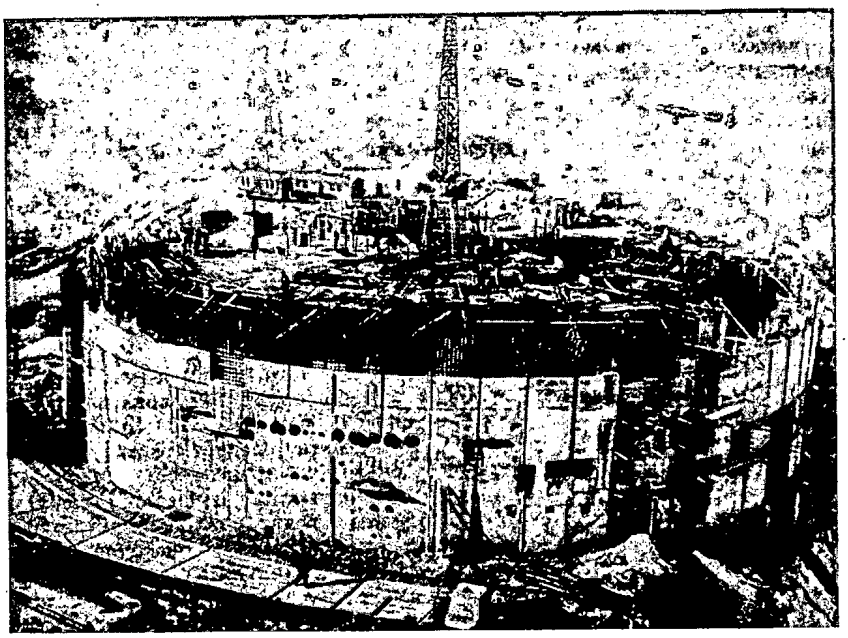

Fig. 2. Novovoronezh nuclear power station. Construction work on the fifth power generating unit.

Basic Power Equipment. Reliable performance on the part of the basic power equipment is responsible for the high operating stability of nuclear power stations. The power station staff and personnel carefully studies the performance characteristics of the equipment and the technological flowsheets, and selects optimum operating conditions.

As demonstrated by separation tests and thermal hydraulics tests carried out in collaboration with the F. É. Dzerzhinskii VTI, steam generators satisfy the necessary requirements in terms of steam quality and steam capacity. But isolated cases of tube failure were observed in two steam generators (the No. 3 steam generator in the first power generating unit and the No. 9 steam generator in the second power generating unit). The failed pipes were blanked off in order to eliminate any leakage of primary-loop radioactive water into the secondary loop.

TABLE 2. Engineering Cost Data on Power Generating Units

Indices	1st power generating unit	2nd power generating unit	3rd power generating unit	4th power generating unit
Electric power output, MW	210	365	440	440
Year construction was begun	1957	1965	1968	1969
Specific capital investments, rubles/kW	326	256	200	200
Physical startup	XII 1963	XII 1969	XII 1971	XII 1972
Power startup	IX 1964	XII 1969	XII 1971	XII 1972
Power rating achieved	XII 1964	IV 1970	VI 1972	IV 1973
Gross efficiency of power generating unit	27,6	27,6	32,0*	32,0*
In-plant power needs (% of power rating)	8,0	7,3	6,8	6,8
Total electric power generated by 1974, million kWh	9193,1	8487,7	4010,5	2790,6
Planned output of electric power in 1974, million kWh	1357	2671	2945	2577
Annual installed power utilization factor achieved by 1974	0,87	0,81	0,55	0,72
Annual electric power net costs attained by 1974, kopeks/kWh	0,925	0,643	0,834	0,716

*Turbine condenser pressure 0.035 atm.

TABLE 3. Engineering Cost Data for NVAÉS Power Station Operation 1971-1975

Indices	1971	1972	1973	1974 (planned)	1975 (planned)
Electric power generated, million kWh	2027,2	5413,4	8674,7	9550	9900
Electric power net costs, kopeks/kWh	0,948	0,810	0,752	0,66	0,65
Installed power utilization coefficient	0,633	0,607	0,681	0,753	0,776

Operating experience with the steam generating units has demonstrated clearly that deposits on the pipes, on the secondary-loop side, are of an oxide nature, are readily removable, and do not affect the heat-transfer process. But rigid standards must be imposed on secondary-loop water management in order to keep the pipes in good working condition. For example, feedwater for the steam generators was excessively high during the first reactor campaign of the third power generating unit, because of the unsoundness of some of the turbine condensor pipes. As a result, investigations carried out during fuel reloading revealed, the piping suffered extensive contamination due mainly to iron oxides, so that the steam generators had to be blown down. The water management regulations were revised to facilitate removal of contaminants from the piping. A limiting hardness of 0.1 mg-equiv. was set, with turbines outed immediately for repair and maintenance work on the pipes in the event that level was exceeded. Special condensate cleanup systems for the secondary-loop water were proposed for the future, as is the situation at most fossil-fuel electric power generating stations.

While the first power generating unit was in service, some shortcomings in the design of the main circulation pumps deemed responsible for several unscheduled shutdowns were corrected. The various subsystems of the pumps were redesigned and modernized, and the tubing arrangement entering the self-contained cooling system for the bearings was simplified, so that the impeller system could be changed with the object of minimizing fluctuations in head, and the stator blading cooling system was centralized.

At the present time, the obsolete GTsÉN-138 pumps on stream in the first power generating unit have been replaced by the more sophisticated GTsÉN-309 pumps employed in the second power generating unit.

A technique of vibratory acoustic diagnostics worked out by the NVAÉS staff has been in use to monitor malfunctions of the main circulation pumps. Acoustic transducers have been installed on the pumps, and secondary instruments have been located on the reactor control panels. The method in question aids in pinpointing typical pump malfunctions such as excessive wear on bearings, beating of the rotor on the casing, and so on.

Operation of 70 MW turbogenerator sets installed in the first and second power generating units of the power station has demonstrated the possibility of stepping up the power output of each turbogenerator set to 80 MW without any deterioration in efficiency. The unit power output of the turbines in the third and fourth power generating units has been increased to 220 MW. The use of high power rating sets and the increase in the performance parameters of the coolants has made it possible to obtain better performance per cost from the power generating units.

TABLE 4. Data on NVAÉS Operating Stability

Indices	Before attaining power rating			After attaining power rating			
	2nd power generating unit 1970	3rd power generating unit 1972	4th power generating unit 1973	2nd power generating unit			3rd unit
				1971	1972	1973	1973
Number of shutdowns of power generating units and electric power not generated (in 10^6 kWh) because of equipment malfunctions	9 / 222,4	25 / 220	9 / 55,7	6 / 81,4	2 / 1,6	1 / 19,8	5 / 86,0
Number of shutdowns of power generating units and electric power not generated (in 10^6 kWh) because of personnel judgment errors	2 / 0,7	5 / 2,0	1 / 0,4	1 / 0,3	2 / 0,5	None	None
Length of time power generating unit on power, h	5651	7120	8072	5761	7884	7849	7123

It took the combined creative efforts of the power plant operating personnel and of the personnel of the manufacturing plant to put these new turbines into service.

The operation of the nuclear power station turbines working on saturated steam demonstrated the importance of the problem of coping with erosion, a severe problem for the turbine components. Specifically, erosion wear on components of the turbine flow passages in the third power generating unit turbines was detected, and the unit had to be shut down for an additional period to effect repairs and eliminate the problem. Work was begun, in collaboration with the personnel of the turbine manufacturing plant, on improving the performance of the NVAÉS turbines by selecting and testing wear-resistant materials serviceable in the manufacture of turbine components, in plating turbine components, and in hardfacing those components when buildup is called for during repair and maintenance work.

A first-stage rotor blade of the high-pressure cylinder broke off while the third power generating unit was being run up to full power, and the nozzle guide vanes of the No. 9 turbine were ruined. The shank of a rotor blade in the No. 10 turbine was discovered overstrained. Replacement of the blades caused a two-month shutdown of those turbines. The insufficiently stable performance of the turbines was due in large measure to the frequent damage suffered by the brass piping of the condensers.

As investigations revealed, the reason for this damage was the inadequate resistance to corrosion on the part of the piping material. At the present time, preparatory studies are underway at the power station on how best to replace this piping with other piping presenting improved resistance to corrosive attack.

The design of the reactor control and protection components underwent some changes while the reactors were in service, with modifications introduced by designers varying from one power generating unit to the next depending on the experience accumulated and the uranium fuel enrichment being increased to 3.6%. The number of assemblies of electromechanically controlled control and protection systems was increased from 37 to 73 in the second, third, and fourth power generating units, and all of these assemblies combine the functions of control and scramming, so that the number of fuel assemblies could be increased from 343 to 349 with scramming speed retained, and with enrichment of the fuel charge as an added benefit. A system of fluid control of reactivity with the aid of boric acid has been instituted starting with the second power generating unit, so that the power distribution throughout the core can be improved and so that the electromechanical control and protection system can be backed up by a liquid scramming feature. A rack and gear drive is being used in the power drives of the control and protection assemblies in the third and fourth power generating units, in place of the earlier screw and ball nut kinematic pair, and that has improved the speed characteristics of the power drives while enhancing their reliability. In addition, automatic controllers (known as pulsed power controllers) tracking the parameters of the primary and secondary loops and responding to any changes in those loop parameters have been introduced into the systems of these power generating units. The pulsed power controllers also have the job of automatically lowering reactor power output when several main circulating pumps shut off.

All scramming signals were divided into four distinct categories in terms of degree of hazard, in the operation of the first power generating unit. In this protection arrangement, a complete shutdown of the reactor through the simultaneous actuation of all of the control rods released downward would come about only in instances where the heat-transfer reliability of the core deteriorated severely (e.g., tripping out of main circulating pumps, an abrupt rise in reactor power output level, total dumping of turbine loads, and the like). In any remaining cases, reactor power output would be lowered in response to a protection signal

through sequential downward displacement of groups of control rods, to be terminated with the disappearance of the protection signal. This improvement made it possible to cut down appreciably on downtime due to actuation of the reactor protection system.

On the whole, the control and protection system fulfilled its functions reliably in proper response to stimuli.

A procedure for maintenance of the equipment serving VVÉR radioactive loops has been instituted at the NVAÉS power station, and experience of great practical value has been accumulated in the process. Inspection and maintenance technology for the reactor pressure vessel has been mastered for the first time following a protracted service period, and this technology covers the pipe outlets for the principal process pipes, the sheathed tubing of the power drives from the control and protection system assemblies, and also the reactor innards (in-pile equipment and devices).

Reactor inspection in a program calling for complete unloading of the reactor core and extraction of all of the innards from the reactor of the first power generating unit was executed for the first time in the 1970-1971 period. This breakdown inspection required the creative efforts of the entire staff for working out a suitable repair and maintenance technology, methods of monitoring and inspection, devising protective devices and custom-engineered fixtures and accessories for remote-controlled in-pile operations, learning new metal-working techniques (in particular plasma cutting, electric-arc cutting, and electric erosion cutting of metals), including cutting of metals under water with the aid of carbon electrodes. A biologically shielded container equipped with handholes and windows covered with special grades of glass was designed and built to aid in inspection and repair of the reactor pressure vessel. Closed-circuit television arrangements were employed liberally in inspecting equipment sets located in inaccessible and difficult-to-get-at sites. As a rule, all of the repair, maintenance, and inspection operations were tried out first on mockups simulating the actual conditions. Before this inspection and repair work was begun, the primary-loop equipment was deactivated by applying special solutions.

Inspection of the reactor in the first power generating unit revealed some serious shortcomings in the design of the reactor innards, which exhibited inadequate strength and inadequate resistance to vibration in the coolant stream. The new in-pile equipment, like the in-pile equipment in the other new reactors of this nuclear power station, exhibit heightened ability to withstand vibrations, and feature modular design which renders the job of monitoring and inspection easier. Cracks were detected on the adapters of the main pipe connections in the process of inspecting the pressure vessel, and this led to the development of a technology for inspection and repair of pipe connections which was implemented for the first time in this program. Adapters made of 1Kh18N10T grade steel were inserted and welded in place. The adapters were press-fitted after cooling in liquid nitrogen, by using a special auxiliary fixture with a pneumatic power drive developing 3 tons of force.

In 1971, the in-pile equipment of the reactor in the second power generating unit was extracted and inspected in a similar repair and maintenance program, because of the rupturing of one of the quieting tubes in the control and protection assemblies. All of the innards of the reactor in the fourth power generating unit were extracted in April, 1974, at fuel reloading time, for a shakedown inspection of the reactor pressure vessel. Periodically scheduled inspections of reactor pressure vessels are on the agenda for all the power generating units of the power station in future practice.

Defects in the weldments joining the sheathed tubes of the power drives of control and protection system assemblies to the cover of the reactor pressure vessel were detected while the first power generating unit was in service, where the primary-loop hot water became mixed up with cold water fed into the sheathes to cool the electric motors of the power drives. Thermal stresses generated in the metal contributed to the appearance of flaws at those sites. After a special maintenance technology had been worked out in 1967, the welded joints were replaced by flanged joints. This redesigning effort was also required on the reactor of the second power generating unit in 1970. Operating experience with the mechanisms and moving parts of the control and protection system serving the first power generating unit was incorporated in the design of the VVÉR-440 reactors of the third and fourth power generating units. Flanged sheath joints were provided in those reactor assemblies, and the electric motors of the power drives are cooled in these cases with the aid of a special intermediate loop.

A change in the design of some components of the electromechanical control and protection system improved system performance. But since the presence of pulsations and temperature gradients affects the operating characteristics and the state of the metal, designers are obliged to take into cognizance the operating experience acquired with that system for attention to further improvements.

A technology developed for replacing the primary-loop main circulating pumps was first worked out at NVAÉS power station. With the active participation of the power plant personnel, a repair and maintenance machine shop was designed and built for the upkeep of main circulation pumps, with equipment for deactivation, specialized ventilation, and all necessary accessories on hand.

Equipment maintenance experience accumulated over the 1964-1971 years with the first and second power generating units made it possible to carry out successful redesign work needed while the power generating unit was being put through its paces and put into service, and in the work on the quantity-manufactured power generating units incorporating VVÉR-440 reactors.

One of the crucial operations mastered in practice at the NVAÉS power station is reloading of nuclear fuel, usually combined with inspection and maintenance of power equipment.

The power generating unit is shut down for short periods (ranging from 25 to 30 days) for fuel replacement, and involves such complex operations as failure of bonding in the primary loop, withdrawal of spent fuel assemblies, and installation of new fuel assemblies. The characteristics of the new fuel loading are calculated tentatively on computers. The loading is done in such a way that the required duration of the next reactor campaign is rendered possible, and the acceptable physical parameters limiting the reactor power output level are attained. Special attention is given to nuclear safety considerations in the refueling process and to the performance of the new core. As a rule, reloading of about one third of the fuel assemblies in each reactor is carried out once a year, during the summertime. But the presence of four functioning power generating units imposes additional time restrictions, since refueling requires serious preparation. The refueling operations proceed parallel with monitoring of the soundness and leakproofing of fuel element cans in operating fuel assemblies, with the aid of special techniques worked out at the power station. The method based on measuring the degree of radioactivity of the air in which self-heating of a fuel assembly deprived of water occurs is the method most widely relied on in practice. Air activity increases when flaws are present in the fuel element can through which gaseous and volatile fission products can exit. "Leaky" assemblies are replaced; the makeup of the fuel charge is corrected and the design characteristics of the reactor core are also corrected appropriately.

The fuel is reloaded underwater, in all of the power generating units of the nuclear power station except for the second power generating unit, by special machines designed for the purpose, and that ensures complete safety when appropriate dosimetric health physics monitoring is attended to. In the second power generating unit, the spent fuel assemblies are conveyed to a storage tank in a special shielded container. A liquid neutron absorber, boric acid added to the water of the primary loop, is relied on to maintain the deep subcriticality of the reactor during the refueling process. The neutron flux in the reactor is monitored by a special refueling auxiliary system with sensitive ionization chambers installed in the channels of the faceted belt in the reactor support cage.

Research work and pilot tests continually staged at the NVAÉS power station help, in addition to scientific-research work and planning and design work carried on by institutions dedicated to those purposes, in shortening the time required for successful adaptation and improvement of the basic equipment and improvement of engineering cost figures.

Among the most important results achieved are something little short of doubling the power rating of the newly introduced reactors without any increase in core size, and by the same token without any increase in the size of the reactor pressure vessels. This became possible through improvements in the power distribution over the core as a consequence of working out a liquid control schedule first implemented in the second power generating unit. Simultaneously, a flowsheet and a technology for normal and scrammed introduction of boric acid into the primary loop, plans for extraction of the boric acid, and plans for clean-up of loop water and proper water management, were worked out. A modification of the fuel grid with the dimensions of the fuel assemblies left unaltered made it possible to increase the number of fuel elements, and consequently to increase the average energy yield per fuel element with the maximum allowable load at which there would be no meltdown of the uranium dioxide in the center of the fuel element core retained.

There are 12 special measuring channels in the central tubes of the reactors in the third and fourth power generating units, and five such channels in the second power generating unit, for measuring variation factors in power generation over the core height. The neutron flux distribution in the channels is measured by activation detectors or direct-charging sensors. At the present time, the possibility of automating in-pile measurements with the use of direct-charging sensors and data processing through a M-6000 computer is under study at the power station.

Methods for monitoring the state of fuel elements present in the reactor core were worked out and perfected while the power generating units were in service. In addition to radiochemical monitoring based on the degree of fission-fragment activity on the part of the primary-loop water in the first power generating unit, a system designed to take continuous recordings of the background level of delayed neutrons in circulation subloops has also been worked out for the first power generating unit. The system demonstrated its feasibility and has been instituted in all of the power generating units of the nuclear power station.

Radiation Safety. Under normal operating conditions, even when several leaky fuel elements turn up in the core, the total activity of water is not greater than 10^{-4} to 10^{-3} Ci/liter. A sufficiently low level of activity and a high degree of leak-proofing of the primary-loop equipment can ensure a low level of radioactive contamination of the nuclear power station rooms, and also can place limits on the amount of gaseous and aerosol waste vented from the production rooms of the four power generating units via the two exhaust stacks. Data available support the inference that the power station exerts no harmful influence on the surrounding environment, and that the level of radioactivity of the environment is commensurate with the natural background. Various devices and protection equipment used in service, in deactivation operations, in inspection and maintenance of radioactive equipment, aid in maintaining a level of personnel exposure doses below public-health regulation levels.

Even though the probability of a serious accident associated with release of radioactivity from the primary loop is very small, the station personnel is carrying out intensive work on periodic checkups on the state of the metal in the equipment and piping of the reactor installation during the period when nuclear fuel in the reactor is replaced, and a practically complete volume of monitoring and inspection work is carried out within the space of four to five years. An especially large volume of work in investigating the metal in the pipings was carried out during the period of inspection of the first power generating unit.

Consequently, a basic summary of experience in the operation of the four power generating units over the 10-year period that the Novovoronezh nuclear power station has been in service leads to the conclusion that significant progress has been achieved in power station equipment and technology in the case of power stations using VVÉR type reactors, which are reliable sources of electric power and installations that are safe for the nuclear power station personnel and staff, for the surrounding population, and for the local environment.

A further expansion of the Novovoronezh power station is being planned with the scheduled introduction of a fifth power generating unit incorporating a VVÉR-1000 reactor, construction work on the reactor and power generating unit as a whole having begun back in 1973. The latest engineering solutions resulting in lower costs and enhanced safety will be materialized in the designs for the fifth power generating unit. Plans call for building a shielded concrete enclosure for the primary loop, which is intended to completely localize and trap radioactivity in the event of accidents involving leakage of coolant. The dimensions of the core are being increased by a factor of almost 1.5 while the pressure vessel size remains the same, and this is being achieved through modifications in the reactivity control system. The unit power rating of the turbogenerators is being increased to 500 MW. After the fifth power generating unit starts generating power at the outset of the next Five-Year Plan, the total power output of the NVAÉS will be increased by another 1000 MW. Engineering cost figures for the operation of the nuclear power station, which are even now comparable to the figures for the operation of fossil-fuel power generating stations in the central region of the nation, are undergoing impressive improvements at the same time.

EFFECT OF TEMPERATURE DISTRIBUTION ON THE SWELLING OF UO_2 AND $UO_2 - PuO_2$ CORES

V. I. Kuz'min and I. G. Lebedev

UDC 621.039.542.342

One of the processes influencing the efficiency of fuel elements, including those based on oxide fuels of the UO_2 and $UO_2 - PuO_2$ types, is the radiation-induced swelling of the fuel. Experimental data [1, 2] relating to the influence of individual parameters on volume changes taking place in these types of fuel as a result of irradiation often disagree both quantitatively and qualitatively. Calculations of volume changes in oxide fuels (especially those due to gaseous fission products) cannot as yet be carried out with adequate accuracy.

The contribution of solid fission products to swelling was determined earlier [3, 4]. In the present investigation we determined the influence of temperature distribution in irradiated UO_2 and $UO_2 - PuO_2$ fuels on the total porosity, and considered methods of calculating the volume changes taking place in cylindrical oxide-fuel cores.

METHOD

We studied UO_2 samples and mixed $UO_2 - PuO_2$ fuel containing 15 and 20% PuO_2. Sintered moldings were placed in 0Kh16N15M3B steel cans and these were then sealed hermetically by welding.

The samples were irradiated in an SM-2 reactor to various degrees of burn-up (0.4-17% of the heavy atoms) (Table 1). The can temperature was measured during irradiation with Chromel–Alumel thermocouples, and the fuel temperature was determined by calculation. The maximum error in determining the temperature of the fuel in the center of the moldings was not greater than 200°C. The burn-up was determined by γ or mass spectrometry.

The swelling of the fuel in all samples except those irradiated with a linear power of over 400 W/cm took place without any restraint on the part of the can. The gas pressure inside the samples at the end of irradiation was no greater than 30 kg/cm^2.

The samples were studied under the microscope. Using photographs of the microstructure and the method of secants [5], the total porosity was determined in several regions at different radii. The volumetric proportion of pores with a diameter of over 0.8 μ was determined in the optical microscope, and that of pores with diameters of 0.02-0.8 μ in the electron microscope by the replica method [6]. The contribution of still finer pores with diameters of 15-32 Å was calculated after studying the UO_2 by transmission electron microscopy [7].

RESULTS

The results of the porosity measurements for different degrees of burn-up are presented in Fig. 1, which also shows the porosity of the nonirradiated samples (for comparison). The porosity/burn-up relationship and the absolute values of the porosity corresponded to the existence of three temperature zones, 1000-1500, 1500-1700, and 1700-2500°C, respectively.

In zone 1 the porosity of the samples before irradiation equalled 5-9%; after irradiation to burn-up values of 1-5% it fell sharply. The fall varied in different cases. The maximum reduction in porosity occurred in samples irradiated to a burn-up of 1%, in which the grain size and the size of the original

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TABLE 1. Characteristics of the Samples and Conditions of Irradiation

Composition of sample	Porosity before irradiation, %	Temperature of fuel		Burn-up averaged over the cross section, %	Linear power, W/cm
		on the surface	in the center		
UO_2	5	700	1700	0,4	350
UO_2	5	1230	1600	1,0	275
UO_2	5	1400	2200	1,3	345
UO_2	6	920	1550	2,7	250
UO_2	7	1230	2350	2,9	440
UO_2	6	1120	1800	3,0	305
UO_2	8	800	1450	4,2	255
UO_2	6	1050	2250	4,8	510
UO_2	2,5	890	1460	5,3	320
UO_2	9	780	1860	5,6	300
UO_2	8	1000	2050	6,6	415
UO_2	9	1100	2500	8,1	500
UO_2	8	1100	2000	12,8	310
$(0,2\ Pu—0,8\ U)O_2$	2	1100	1800	14,6	245
$(0,15\ Pu—0,85\ U)O_2$	3	1150	1750	17,1	310

TABLE 2. Values of the Coefficients C_i and C_{i0}

Coefficient	Temperature range, °C			
	1000–1500	1500–1700		1700–2500
		$\frac{dT}{dr} <$ 400°C/mm	$\frac{dT}{dr} >$ 500°C/mm	
$\frac{C_i}{C_{i0}}$	$1,04 - KP_0$	$0,32 \over 15 - P_0$	$0,32 \over 7 - P_0$	$0,32 \over 5 - P_0$

pores before irradiation were minimal. The porosity of the samples originally at a level of 5-7% remained unchanged after irradiation to a burn-up of 6-7%, but after a 14-17% burn-up it increased from 2-3 to 12-15%. The porosity of the irradiated samples depended linearly on burn-up within the range of measuring error (relative 15%).

In zone 2 the porosity depended, not on the burn-up and original porosity, but only on the temperature gradient; this applied over the whole range of burn-ups (within the limits of measuring error). The absolute porosity in zone 2 for a gradient of under 400 deg/mm averaged 15%, and for a gradient of over 500 deg/mm it varied between 5 and 10%. In the zone of columnar grains (zone 3) the porosity was nonuniform. In the central part close to the cavity the porosity was usually higher, but the average value amounted to ~5%.

DISCUSSION OF RESULTS

The reduction in porosity observed in zone 1 after irradiation to a burn-up of ~1% might be asociated with two processes: sintering, and the filling of pores with solid fission products. In the samples irradiated to a burn-up of 1% the porosity diminished from 5 to 1%, while the volume of solid fission products for a burn-up of 1% was less than 0.7% of the volume of the fuel with the theoretical density [4]. It follows that the reduction in porosity was in fact due to sintering. This conclusion agrees with the results of [8]. The increase in porosity for greater degrees of burn-up associated with the increasing dimensions and volumes of the pores due to gas accumulation. It is well known that the evolution of gas from this zone in UO_2 tablets with a density of 10-10.7 g/cm^3 equals 20-50% for a burn-up of ~4% [9], while the evolution of gas from zones 2 and 3 equals ~100% for a burn-up of over 1%. Growth of the pores takes place chiefly at the grain boundaries. In samples in which the zone size equalled 15-20 μ, considerable porosity also appeared inside the grains (at the subboundaries). The increment in the porosity of zone 1 observed for fuel with the theoretical density (introducing a correction for the nonuniform burn-up over the sample cross section) equalled 0.72% /1% burn-up within the range studied. In zones 2 and 3, saturation of the porosity changes occurred even at a burn-up of ~1%. The reason for this may be understood by analyzing the structure of the fuel. The pores in these zones lie chiefly at the grain boundaries and form a network of interconnected channels, which lie mainly on the lines connecting three neighboring grains. For this geometrical disposition of the pores, a considerable proportion of the surface of the grain boundaries is linked to the free surface, and the gas passing to a grain boundary emerges freely from the fuel (Fig. 2).

The intragranular porosity in zones 2 and 3 also fails to increase on increasing the burn-up above 1%, since the rate of gas evolution at the grain boundaries is such that the krypton and xenon atoms so formed reach the grain boundaries in a time corresponding to a burn-up increment of 0.2%. This is confirmed by calculating the velocity of the gas-filled pores in the field of the temperature gradient [10].

Saturation of porosity with increasing burn-up should also occur in zone 1 owing to the transition of a considerable proportion of the isolated gas pores to the open surface. However, saturation of the porosity in zone 1 was not in fact reached for a burn-up of 17%, owing to the low mobility of the gaseous fission fragments. For the same reason it would appear that saturation in zone 1 will set in gradually, from the region of highest temperature to the region of low temperature.

The swelling of the fuel in the samples under consideration took place gradually as the gas pressure within the can increased, starting from 1-2 kg/cm^2 at the onset of irradiation to 30 kg/cm^2 at the end.

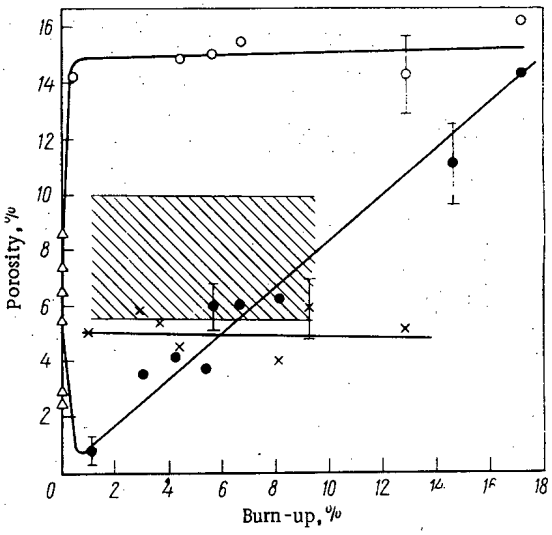

Fig. 1

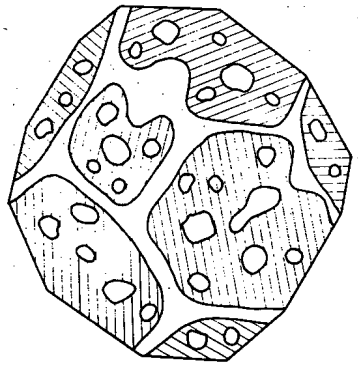

Fig. 2

Fig. 1. Porosity as a function of burn-up in various temperature zones: ●) zone 1; ○) zone 2, dT/dr < 400 deg/mm (shaded region, dT/dr > 500 deg/mm); ×) zone 3; △) original porosity.

Fig. 2. Linking of pores at the grain boundaries.

The external gas pressure on the fuel was insufficient to have any major effect on the porosity by virtue of gas-induced swelling. In samples irradiated with a linear power of over 400 W/cm the can was to some extent able to restrain the swelling of the fuel in the radial direction. However, porosity measurements revealed no marked influence of radial restraint.

Zone Model for Calculating Volume Changes in Fuel-Element Cores

The zonal character of the temperature dependence of the changes in porosity and also the quantitative data relating to changes in the volume of the fuel due to solid and gaseous fission products enable us to calculate the total volume effect in the cores of oxide fuel elements. If we consider part of the core with an external radius of R as having a length l so short that any change in the linear power and temperature along its length may be neglected, and assume that the burn-up is independent of radius, the increase in the volume of the annular layer may be expressed as follows:

$$\Delta V_i = \pi l (\gamma_i^2 - \gamma_{i+1}^2)(C_i B + C_{i0}), \tag{1}$$

where C_i and C_{i0} are coefficients characterizing the swelling in the zone limited by the radii r_i and r_{i+1}; and B is the burn-up. The radius r_i corresponding to the temperature T_i for a specified linear power q_l, core surface temperature T_R, and thermal conductivity λ is determined from the equation

$$r_i^2 = R^2 \left(1 - \frac{4\pi \int_{T_R}^{T_i} \lambda(T)\, dT}{q_l} \right). \tag{2}$$

The relative change in the volume of the selected part of the core is determined from the equation

$$\Delta V/V = 4\pi/q_l \sum_{i=1}^{n} (C_i B + C_{i0}) \int_{T_i}^{T_{i+1}} \lambda(T)\, dT. \tag{3}$$

The number of zones depends on the temperature distribution. If the temperature in the center is no greater than 1700°C, the calculation is carried out with a two-zone model. For a higher temperature a three-zone model is needed.

The coefficients C_i and C_{i0} determine the terms in (3), respectively, dependent and independent of the burn-up. The value of the coefficients C_i equals the sum of the changes in volume of the i-th zone of

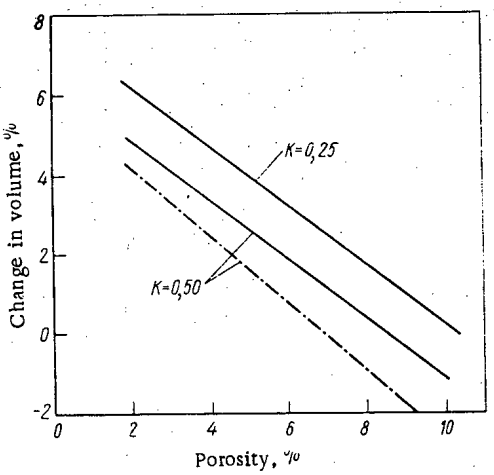

Fig. 3

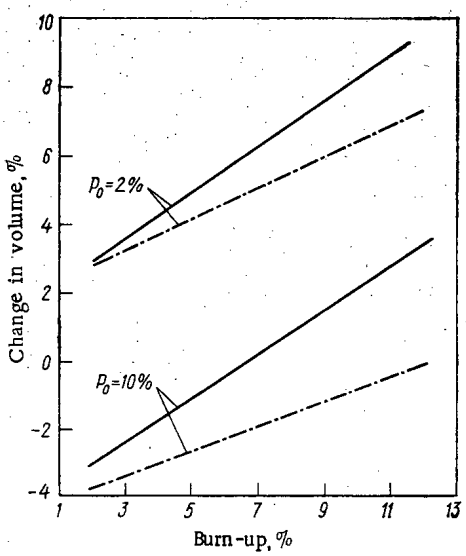

Fig. 4

Fig. 3. Increase in the volume of a uranium dioxide core as a function of the porosity of the moldings: ——— and – · – · – · –) T_{surf} = 1000 and 1200°C; q_l = 350 and 550 W/cm, respectively.

Fig. 4. Increase in the volume of uranium dioxide cores as a function of burn-up (K = 0.5). Curve nomenclature as in Fig. 3.

the core due to the solid and gaseous fission products on increasing the burn-up by 1%. The change in volume due to the solid fission products is the same for all the zones and equals ~0.4% [3] or according to refined data 0.32%. The change in volume due to the gaseous fission products is 0.72% for the first zone as already indicated and zero for the second and third zones, since the porosity is then independent of burn-up (Fig. 1). After summing the contributions of the solid and gaseous fission products, we obtained the coefficients C_1, C_2, and C_3.

The coefficient $C_{1;0}$ equals the product KP_0. The coefficient of radiation-induced sintering K determines that part of the original porosity P_0 which vanishes as a result of sintering under irradiation. The values of the coefficients $C_{2;0}$ and $C_{3;0}$ equal the difference between the porosity corresponding to saturation and the original porosity.

Table 2 gives the coefficients C_i and C_{i0} used for the burn-up range 1-17%, an original porosity of 2-10%, and temperatures of 1000-2500°C.

Substituting the corresponding values of C_i and C_{i0} we obtain the following for the two zones:

$$\frac{\Delta V \cdot 100\%}{V} = 0.32B\left(\frac{100-P_0}{100}\right) + \frac{4\pi}{q_l}\left[(0.72B-KP_0)\int_{T_{surf}}^{1500}\lambda(T)\,dT + (15-P_0)\int_{1500}^{T_c}\lambda(T)\,dT\right]; \quad (4)$$

and for the three zones:

$$\frac{\Delta V \cdot 100\%}{V} = 0.32B\left(\frac{100-P_0}{100}\right) + \frac{4\pi}{q_l}\left[(0.72B-KP_0,\int_{T_{surf}}^{1500}\lambda(T)\,dT + (7-P_0)\int_{1500}^{1700}\lambda(T)\,dT + (5-P_0)\int_{1700}^{T_c}\lambda(T)\,dT\right]. \quad (5)$$

Equations (3)-(5) may be used for a solid core if the external pressure is no greater than 30 kg/cm^2. Figures 3 and 4 give the results of a calculation of the volume changes taking place in the fuel for various initial porosities, burn-ups of 5 and 10%, and linear powers of 350 and 550 W/cm. In the calculation it was assumed that the sintering coefficient equalled 0.25 or 0.5 and the temperature on the fuel surface 1000 or 1200°C. As indicated by Eqs. (4) and (5) and Figs. 3 and 4, an initial porosity is useful for reducing the volume changes in all the temperature ranges. With increasing linear power the swelling is reduced, since the volumetric proportion of zone 1 in the core diminishes, and that of the fuel in the plastic zone 3 (less subject to gas swelling) increases. This may have important consequences, since the deformation of the can is mainly due to the "rigid" peripheral layer of fuel.

Thus we have analyzed the distribution of porosity in the cores of UO_2 and UO_2-PuO_2 fuel elements irradiated in the burn-up range 0.4-17% at linear powers of 250-510 W/cm by quantitative metallography. We have given a qualitative description of the mechanisms underlying gas-induced swelling and gas evolution in the oxide fuel. We have shown that gas swelling reaches saturation in the core zone for a burn-up of ~1% and temperatures of over 1500°C. In the core zone the porosity increases with burn-up at a temperature of <1500°C. We have proposed a zone model for calculating volume changes in the core of the fuel elements, allowing for the initial porosity, burn-up, and temperature distribution. We have shown that an increase in the initial porosity and linear power leads to a reduction in the swelling of the fuel.

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N. I. Chesnokov, A. A. Petrosov,
and A. A. Vinogradov
OPTIMIZATION OF SOLUTIONS (DECISIONS) IN THE DEVELOPMENT OF URANIUM SITES*

S. Ya. Chernavskii

The control of complex contemporary undertakings such as those involved in the mining industry may be rendered more efficient by using scientific methods of taking decisions. This is convincingly demonstrated in the book under discussion, which is devoted to a number of methodical problems of decision-taking.

Many of the methods employed in the analysis of complex systems are described in this book. Starting from popular-economic problems in the development and introduction of automatic methods of controlling processes, the authors rightly lay emphasis on mathematical approaches.

The authors are careful to illustrate each of the mathematical methods presented by examples relating to the functioning of mining processes. This is deserving of special approval, although it would have been possible to adopt a different presentation of the material by considering just one particular mathematical method as a means of solving a specific problem in a mining undertaking.

In the first chapter the authors' methodology (system of approach) is briefly outlined. Then the mathematical model of a mining undertaking is analyzed. We feel that this section devotes rather a great deal of attention to the concept of net cost. The limitations defining this aspect are not quite so vital as the authors suggest.

In the following chapters mathematical models designed for more and more complicated situations are developed in a most interesting way. Thus in the third chapter the choice of optimum decisions under conditions of determinacy is discussed, in addition to methods of selecting the best form of programming: linear, nonlinear, or dynamic.

The fourth chapter is devoted to the choice of optimum decisions under conditions of indeterminacy. Certain ideas relating to the theory of games and mass servicing are described, and so is stochastic programming. Finally the fifth chapter considers problems of multicritical optimization, as yet very little discussed in the scientific literature (from the point of view of practical applications), as well as the choice of optimization methods.

The book is written in a very readable style and is furnished with many examples; it is nevertheless not free from certain shortcomings. Thus in considering the question of choosing optimum decisions under conditions of indeterminacy it would have been better to separate the probability situation from the indeterminate situation and (in our own view) to confine attention to the class of games with Nature.

Without really sufficient basis the authors have given preference to the Wald and Savage criteria, omitting other game criteria such as those of Laplace and Gurwitz. There is an interesting attempt at making use of the methods of expert estimates in setting up criteria; however, the proposed procedure lacks a broad basis and experimental verification. In multipurpose optimization no attention is paid to the most promising way of determining optimum decisions, associated with the determination of Paretot sets.

On the whole the book is nevertheless extremely useful and will attract great interest among its readers.

*Atomizdat, Moscow (1974), 294 pp.

Translated from Atomnaya Énergiya, Vol. 38, No. 1, p. 15, January, 1975.

ARTICLES

HYDRAULIC RESISTANCE IN CHANNELS WITH SURFACE BOILING

V. A. Knyazev

UDC 532.542.4

When studying the surface boiling of an underheated liquid it is usual to distinguish a characteristic region called the stage of developed boiling. The definitions of the onset of this stage proposed by a variety of authors [1-8] give very similar results. We shall use the definition of Labuntsov:

$$(\alpha\vartheta_s/q)_v \approx 0.35, \tag{1}$$

where $\vartheta_S = T_S - T_L$ is the local underheating of the heat carrier; α is the heat-transfer coefficient in the single-phase flow; q is the thermal flux (heat flow) density.

According to [8] the volumetric vapor content $\varphi \approx 0.01$ in the cross section (1). On the other hand, the fundamental possibility of the onset of boiling may be expressed by the condition $T_W \approx T_S$, which corresponds to the cross section $\alpha\vartheta_S/q \approx 1$.

Let us call the region

$$1 \geqslant \frac{\alpha\vartheta_s}{q} \geqslant 0.35 \tag{2}$$

the region of the development of surface boiling. Properties of this stage include a comparatively weak manifestation of the ordinary effects of boiling, a high degree of deviation of the process from equilibrium, and a wide expanse of the region of development. For example, on setting out the heat-balance equation and using Eq. (2), we find that, for a cylindrical tube with uniform heating, the range of boiling development extends over a number of diameters equal to

$$\frac{L}{D} \approx \frac{0.1625}{St} \approx 7.2\,\mathrm{Re}^{0.2}\mathrm{Pr}^{0.6}. \tag{3}$$

This latter fact constitutes the basis for a special study of the law of hydraulic resistance at the stage of development of surface boiling which we shall be considering in this paper.

As a theoretical model of the flow in the region of boiling development, we take a turbulent flow affected by perturbations at the stream boundaries. The principle of stress superposition established by Millionshchikov for the case of flow at a rough surface [9] suggests that the turbulent and "perturbed" pulsating motions do not correlate with each other. The instantaneous velocity field may then be expressed in the form

$$V_i = v_i + v_i', \tag{4}$$

in which the flow averaged over the turbulent pulsations equals

$$v_i = \langle\bar{v}_i\rangle + \underset{\vee}{\bar{v}_i} + \langle\underset{\vee}{v_i''}\rangle + v_i''. \tag{5}$$

the sign $\langle\ \rangle$ expresses averaging over a certain scale of length λ; and $\vee$ denotes spatial pulsation.

On the basis of the equations of motion written in the Gromek–Lamb form we obtain the following for the average steady flow

$$-(\partial/\partial x)\langle\bar{p}/\rho\rangle = -\nu\frac{\partial}{\partial y}\langle\bar{\omega}\rangle - \overline{v_y'\omega'} - \langle\underset{\vee\ \vee}{\overline{v_y\omega}}\rangle, \tag{6}$$

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where $\langle\bar{\omega}\rangle = -\partial/\partial y < \langle \bar{v}_x \rangle$ is the vorticity of the average motion.

The first two terms on the right in Eq. (6) express the viscous and turbulent components of the resistive forces in the turbulent flow. The third term $\langle\overline{v_y\omega}\rangle$ expresses the additional vortical resistance in the flow perturbed by surface boiling.

The tangential stress will equal

$$v_*^2 \equiv \tau/\rho = -\nu\langle\bar{\omega}\rangle + \int_0^y \overline{v_y'\omega'}\,d\eta + \int_0^y \langle\overline{v_y\omega}\rangle\,d\eta. \tag{7}$$

At the same time we may write

$$v_*^2 = -(\nu + \nu_T + \nu_P)\langle\bar{\omega}\rangle, \tag{8}$$

where ν, ν_T, ν_P are the molecular, turbulent, and perturbed viscosities, respectively.

Comparing (7) and (8) we obtain

$$\nu_P = \int_0^y \langle\overline{v_y\omega}\rangle\,d\eta / \langle\bar{\omega}(y)\rangle. \tag{9}$$

Equation (9) may be considered as a definition of the effective "perturbed" viscosity. Let us find the relation between ν_P and the relative increase in the hydraulic resistance in the perturbed flow. We compare the coefficients of resistance ξ of a turbulent flow in a hydraulically smooth channel and a boiling turbulent flow subject to the same rate-of-flow (average mass-velocity) conditions. Writing the Millionshchikov equations

$$\begin{cases} [\nu + av_*(y-\delta_0)]\dfrac{du}{dy} = v_*^2; \quad y \geqslant \delta_0, \ a = 0.39; \\ u = \dfrac{yv_*}{\nu}v_*; \quad 0 \leqslant y < \delta_0; \end{cases}$$

$$\begin{cases} [\nu + av_{*P}(y-\delta_0) + \nu_P]\dfrac{du}{dy} = v_{*P}^2; \\ u = \dfrac{yv_{*P}}{\nu}v_{*P}, \end{cases}$$

we obtain for a slot channel of half-width h

$$\left[1 - \frac{\nu}{av_*h}\right]\ln\left[\frac{av_*h}{\nu}\left(1 - \frac{\nu}{av_*h}\right)\right] - 1 = -\bar{v}_{*P} - \frac{\nu - \nu_P}{\nu}\left[\frac{\nu}{\nu - \nu_P}\bar{v}_{*P} + \frac{\nu}{av_*h}\right]\ln\left[\frac{av_*h}{\nu}\left(\frac{\nu}{\nu - \nu_P}\bar{v}_{*P} + \frac{\nu}{av_*h}\right)\right], \tag{10}$$

where

$$\bar{v}_{*P}^2 \equiv \bar{\xi} \equiv \left(\frac{\bar{\xi}_P}{\xi} = \left(\frac{v_{*P}}{v_*}\right)^2\right).$$

For the conditions $h \gg \delta_0$, $v_*h/\nu \gg 1$ and $\ln \bar{v}_{*P} < 1$ Eq. (10) takes the simple form

$$\frac{\sqrt{\bar{\xi}} - 1}{\sqrt{\bar{\xi}}} = \frac{\ln(1 - \nu_P/\nu)}{\ln\left(0.195\sqrt{\frac{\xi}{8}}\mathrm{Re}\right)}. \tag{11}$$

Equation (11) enables us to separate out the intrinsic boiling effect from resultant (integrated) quantities such as the coefficient of hydraulic resistance. The analysis of experimental data relating to the hydraulic resistance associated with surface boiling in the region (2) may accordingly be conveniently carried out in the form

$$\frac{\nu_P}{\nu} = f(q,\ p,\ T_{in},\ u,\ \ldots) = \left(0.195\sqrt{\frac{\xi}{8}}\mathrm{Re}\right)^{\frac{\sqrt{\bar{\xi}} - 1}{\sqrt{\bar{\xi}}}} - 1, \tag{12}$$

where f(q, p, T_{in}, u. . .) is the unknown function of the boiling mode parameters, while $\bar{\xi}$ are the experimental values of the relative coefficient of hydraulic resistance in the boiling channel.

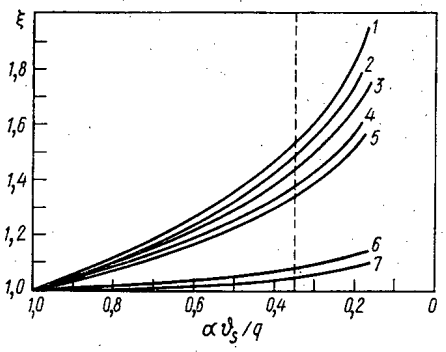

Fig. 1

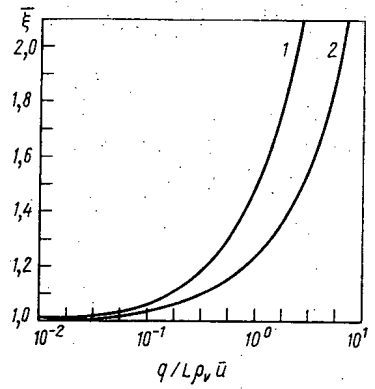

Fig. 2

Fig. 1. Change in the coefficient of resistance along the boiling channel (the broken line indicates the cross section of substantial vaporization): 1) Re = $5 \cdot 10^4$; 2) 10^5; 3) $2 \cdot 10^5$; 4) $5 \cdot 10^5$; 5) 10^6 for A = 1; 6) $5 \cdot 10^4$; 7) 10^6 for A = 0.1.

Fig. 2. Effect of the parameter $q/L\rho_v\bar{u}$ on the resistance coefficient for Re = 10^5. Cross section $\alpha v_s/q = 0.35$ (1) and 0.6 (2).

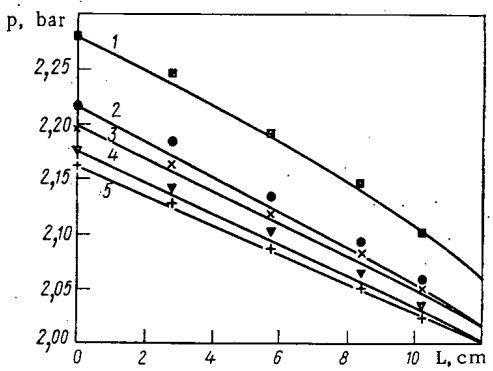

Fig. 3. Pressure distribution in the channel for the surface boiling of water (v = 6.096 m/sec; T_{in} = 27°C; P_{out} = 2.00 bar; L = 12 cm): 1) q = $6.93 \cdot 10^6$; 2) $6.18 \cdot 10^6$; 3) $5.40 \cdot 10^6$; 4) $4.63 \cdot 10^6$; 5) $3.78 \cdot 10^6$ W/m^2.

On the basis of definition (9) we may propose the following model of effective viscosity in surface boiling. First we note that the surface of the bubbles is a free surface, and their shape and frictional resistance may be neglected. A more important effect is the following. The detachment of the bubble is accompanied by the attraction of some of the boundary layer of liquid into the flow. Visual observations [10] reveal a high vorticity in the layers of liquid surrounding the risen bubble. Temperature measurements around the bubble while boiling in a large volume [11] also show it to be surrounded by a layer of superheated liquid. Of course, from the point of view of the contribution to the vortical resistance, not all parts of the liquid so carried away are equivalent. The main contribution should come from layers with a vorticity of the order of ω_w in the laminar sublayer. The vortical resistance $\langle \overline{v_y \omega} \rangle$ acts in a layer close to the wall having a certain thickness δ and a completely specific physical meaning. Let us specify a vortex with a uniform vorticity ω situated in a nonviscous liquid quiescent at infinity. An initial velocity v_y is attributed to the vortex. The vortex is then acted upon by a force of intensity $\omega \times v$, under the influence of which the vortex will move along a circle of radius $R = v_y/\omega$. The quantity R defines the size of the vortical region beyond which the flow remains potential. In order to estimate the thickness of the vortical layer δ we accept the value of R as criterion.

The scale $\bar{v}_y$ relates to the velocity of the bubble at the instant of escape $v_\perp$ and the scale $\bar{\omega}$ to the boundary vorticity $\omega_w \approx 7.8 v_*/\delta_0$. Then

$$v_b/v \approx \psi \frac{\lambda_+}{\lambda_-} (v_\perp/v_*)^2,$$

where $\lambda_+ + \lambda_- = \lambda$ is the period of the perturbations; λ_- is the region of return motion; ψ is a coefficient associated with time-averaging. The quantity λ_+/λ_- may be associated with the surface density of the bubbles N, namely, $\lambda_+/\lambda_- \approx ND_0^2$, where D_0 is the diameter of the bubble at the instant of detachment.

The quantity ψ is determined by two characteristic time scales: the lifetime τ_0 and the frequency of bubble detachment f.

Thus the effective viscosity of surface boiling is

$$\nu_b/\nu \approx \tau_0 f N D_0^2 (v_\perp \bar{u})^2 \frac{8}{\xi\bar{\xi}}, \tag{13}$$

where $\bar{u}$ is the mass-average velocity in the channel; ξ is the resistance coefficient of the unperturbed flow.

It is difficult to use this equation directly in any calculations since there are no reliable data as to the value of fN. Expressing fN in terms of the heat-transfer coefficient at boiling α_{boil} [12],

$$fN \approx \frac{q(1-\alpha/\alpha_{boil}}{\frac{\pi}{6}\rho_v L \bar{D}^3 \left(1 - \frac{p_L c_L (T_{tr} - T_L)}{\rho_v L}\right)}$$

and noting that

$$v_\perp \approx \frac{\pi a (Ja)^2}{2D_0},$$

where Ja is the Jacob number, and a is the thermal diffusivity, after some simple transformations we obtain

$$\nu_b/\nu \approx 3(q/L\rho_v\bar{u})\left(\frac{\alpha_{boil}}{\alpha} - 1\right). \tag{14}$$

Thus the coefficient of hydraulic resistance at the stage of development of surface boiling is determined by Eqs. (11) and (14).

Figure 1 shows the change in $\bar{\xi}$ along the channel for a number of values of the Re numbers and A $= q/L\rho_v\bar{u}$. Here α_{boil} was calculated from the expressions of [13].

The influence of the parameter A on the value of $\bar{\xi}$ is shown in Fig. 2. For constant values of q and $\bar{u}$ this relationship means that, for a fairly low pressure in the circuit ($A \gtrsim 0.5$), the pressure drop in the region of development of surface boiling is quite large, and may be commensurable with the pressure drop in the whole region of boiling of the underheated liquid. In particular, for $q = 10^6$ W/m^2 and $\bar{u} = 5$ m/sec, the pressure may be regarded as small if $p \lesssim 3$ bar. Under the same conditions, the contribution of the vortical resistance to the total pressure drop along the channel may be neglected if $p > 10$ bar.

Figure 3 gives a comparison between the calculated pressure profile along a boiling channel 2.34 mm in diameter and the experimental results of [14]. Cases 2-5 correspond to occasions on which the stage of development of surface boiling established itself over the whole length of the working section. For these cases the agreement between the experimental and calculated results is completely satisfactory. Case 1 is that in which developed boiling ($\alpha\vartheta_s/q < 0.35$) was established over the last quarter of the working section, in which the pressure drop was mainly associated with an increase in the volumetric vapor content.

Thus the increase in hydraulic resistance in the channel at the stage of development of surface boiling may be attributed to the creation of an additional vortical resistance in the turbulent flow perturbed by boiling. The hydraulic effect of boiling is determined by the relative value of the "boiling" viscosity, as compared with the turbulent and molecular viscosity. In low-pressure circuits it is essential to allow for the pressure drop in the section corresponding to the development of surface boiling.

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H. Müller, K. Meyersberger,

and H. Sprinz

SPECIAL METHODS OF ANALYZING STABLE ISOTOPES*

L. I. Petrenko

Stable isotopes and compounds labeled with these isotopes are widely employed in various fields of science and technology in the Soviet Union with the aim of solving a variety of important practical and scientific problems. For determining changes taking place in the isotopic composition of such materials, mass spectrometric and other analytical methods are employed (gas chromatography, optical spectroscopy, nuclear magnetic resonance, etc.). However, despite the existence of a vast amount of experimental work and numerous publications relating to the use of stable isotopes, scientists and specialists working in this field have as yet never had any practical reference book relating to methods of analyzing stable isotopes at their disposal. This is why the translation of the monograph Special Methods of Analyzing Stable Isotopes is to be welcomed.

The monograph comprises an introduction and 13 chapters giving fairly complete descriptions of the main characteristics of various methods of analyzing the isotopic composition of materials together with practical applications of these methods.

In the first six chapters the characteristics of isotope analyses based on the measurement of thermal conductivity (catharometry), density (densimetry), refractive index (refractometry), the mobility of ions in deuterized aqueous solutions (electrometry), rate of gas flow through a small aperture (effusiometry), viscosity (viscometry), freezing point (cryometry), and vapor pressure (tensometry) are considered.

Of considerable interest are the seventh and eighth chapters, which discuss methods of spectral analysis for qualitative and quantitative determinations of stable hydrogen, helium, argon, oxygen, mercury, lithium, boron, strontium, lead, uranium, plutonium, carbon, nitrogen, sulfur, and polonium isotopes. The 12th and 13th chapters constitute extremely useful supplements to these. In the 12th chapter, written by V. I. Mosichev, detailed attention is paid to the possibilities of using optical spectroscopy for analyzing stable isotopes, the demands made upon the optical apparatus required for this purpose are outlined, and additional examples of the practical use of this method are presented. In the 13th chapter (written by I. V. Domidenkov and V. I. Mosichev) the use of infrared spectroscopy for analyzing isotopically-substituted compounds is considered.

The description of quantitative and structural methods of isotopic analysis by NMR spectroscopy is left to the ninth chapter. Chapters 10 and 11 are devoted to methods of determining isotope contents by gas chromatography and activation analysis.

This book constitutes a valuable practical reference on isotope analysis and should be of great assistance to scientists and specialists working in the field of stable isotopes; it is furnished with an extensive bibliography (1182 references) and very useful reference material in the form of tables.

However, the translation and editing of individual chapters have not been conducted very carefully; there is a great deal of literal translation, the terminology customary to the Soviet Union not always being observed.

*Atomizdat, Moscow (1974), 28 pp.

Translated from Atomnaya Énergiya, Vol. 38, No. 1, p. 19, January, 1975.

ARTICLES

INFLUENCE OF LOW-TEMPERATURE IRRADIATION ON THE PHASE COMPOSITION OF URANIUM ALLOYS CONTAINING SMALL QUANTITIES OF ALUMINUM AND IRON

Yu. V. Bobkov, I. A. Naskidashvili, V. V. Petrosyan, and Yu. N. Sokurskii

UDC 621.039.531

The effect of reactor irradiation at 60-80°C on the phase composition of binary uranium alloys containing 0.21 at. % iron or 0.88 at. % aluminum (together with 0.68 at. % silicon) was studied in [1] by reference to corresponding changes in the electrical resistance; it was found that under the influence of irradiation specific changes of phase composition occurred in these alloys. It was established at the same time that the substantial rise taking place in the electrical resistance of the alloys irradiated after preliminary aging in the α-phase was associated with the dissolution of finely-dispersed precipitates of the intermetallic compounds U_6Fe, UAl_2, and U_3Si and the accumulation of radiation defects. After the irradiation of alloys quenched from the γ-phase there was a fall in the electrical resistance due to the partial decomposition of the supersaturated solid solutions.

On irradiating the dilute uranium alloys, dissolution of the finely-dispersed intermetallic precipitates and decomposition of the solid solutions took place simultaneously, so that after prolonged irradiation a dynamic equilibrium was established for the phase composition, in which the proportion of alloying element passing into solid solution was equal to the proportion of this element precipitating from the solid solution.

The theory of the radiation-induced homogenization of alloys containing fissile material based on thermal-peak concepts [2] does not allow for the irradiation temperature, since this is usually far lower than that to which the material is heated within the peak. Hence the rate of radiation-induced homogenization should not depend on the irradiation temperature. This is confirmed by experiments on the irradiation of a uranium alloy containing 9 wt. % molybdenum at temperatures between −90 and +200°C [3].

At the same time, the rate of decomposition of the solid solutions, determined by the degree of supersaturation of the material with the radiation defects, should depend very considerably on the irradiation temperature, since it is the temperature which determines their mobility. Thus a reduction in the irradiation temperature should retard the decomposition of the supersaturated solid solutions, as a result of which the level of electrical resistance of the alloy in which the phase composition is in a state of "dynamic equilibrium" approaches the values characteristic of the homogenized state.

The aim of the present investigation (suggested by Konobeevskii) was to study the changes taking place in the phase composition of dilute binary uranium alloys irradiated at −150°C. The electrical-resistance method was used to estimate these changes.

Materials Studied and Experimental Method

We studied binary uranium alloys containing 0.21 at. % iron or 0.88 at. % aluminum in two states: quenched from 850°C, and aged at 350°C for 100 h after quenching. For comparison purposes we also studied pure uranium after quenching from the γ-phase and aging at 500°C for 24 h. The proportions of the main impurities in these materials are indicated in Table 1.

The electrical-resistance method employed for estimating the changes in phase composition taking place during irradiation was based on the linear dependence of the electrical resistance of binary uranium

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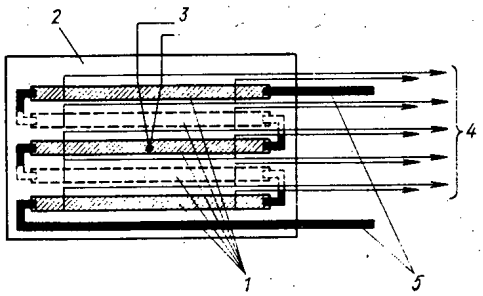

Fig. 1. Arrangement of the samples on the plate placed in the reactor channel: 1) samples; 2) plate; 3) thermocouple; 4) potential leads; 5) current leads.

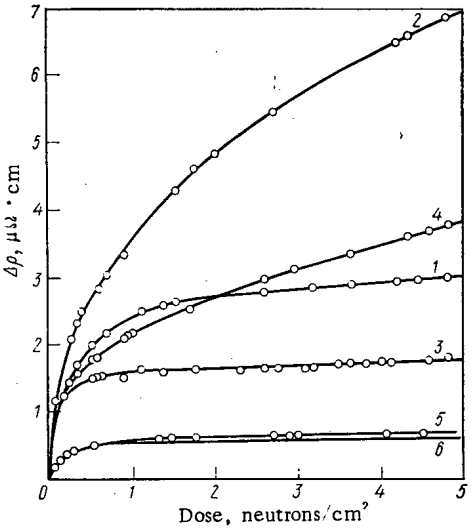

Fig. 2. Changes in the electrical resistance of uranium and its alloys on irradiation: 1, 2) quenched and aged samples of a U 0.88 at. % Al alloy; 3, 4) quenched and aged samples of a U 0.21 at. % Fe alloy; 5, 6) uranium, present results and those of [5], respectively.

TABLE 1. Chemical Composition of the Materials Studied

Material	Impurity content $\cdot 10^3$, wt. %					
	Al	C	Cu	Fe	Ni	Si
Uranium	<1	3	1.7	3	<1	<1
Alloy: U + 0.21 at.% Fe	10	60	1	50	0.8	5
Alloy: U + 0.88 at.% Al	100	60	1.4	6.3	1.2	7

alloys containing small proportions of iron and aluminum on the concentrations of these elements in the solid solution. The corresponding proportionality factors equal 20 and 18 $\mu\Omega \cdot$cm /at. % for aluminum and iron, respectively.

Detailed data as to the phase composition and microstructure of the alloys under consideration obtained by electron microscopy and electrical resistance measurements were published earlier [1]. The quenched alloys constituted supersaturated solid solutions stable with respect to thermal annealing up to ~300°C. After aging for 100 h at 350°C these solutions decompose into alloys with iron (70%) and aluminum (36%). Precipitates of a secondary phase (U_6Fe or UAl_2) are also formed, their mean size being 0.06-0.07 μ.

The samples were irradiated in the low-temperature channel of the nuclear reactor in the Institute of Physics, Academy of Sciences of the Georgian SSR [4]. The integrated thermal neutron flux was $5 \cdot 10^{17}$ neutrons/cm^2. Five samples 45 × 2.5 × 0.1 mm in size with current and potential leads soldered to them (Fig. 1) were irradiated at the same time. The electrical resistance of the samples was measured directly in the course of irradiation to an accuracy of ±0.01%, using a potentiometric method. The sample temperature during irradiation, monitored with a copper–constantan thermocouple fixed to the sample in the center of the plate, averaged –152°C. The temperature deviations from the mean value never exceeded ±3°C, and were eliminated in the determination of the electrical resistance by the introduction of corrections allowing for the temperature coefficients of the electrical resistance of the materials studied.

Results and Discussion

The changes in the electrical resistance of the alloys and the pure uranium are shown in relation to the irradiation dose in Fig. 2. The electrical resistance of pure uranium increases on irradiation as a result of the accumulation of defects and defect complexes in the crystal lattice. We see that a rapid growth in electrical resistance only occurs for small doses not exceeding ~$5 \cdot 10^{16}$ neutrons/cm^2. For larger doses saturation is practically reached, the maximum increment being ~0.65 $\mu\Omega \cdot$cm. These data agree with those of Quere, who studied pure uranium at –196°C [5].

A considerably greater increment in electrical resistance occurred on irradiating the alloys. The extent of this increment and its dependence on the dose are determined by the preliminary heat treatment of the alloy (Fig. 2). In contrast to irradiation at 60°C, which leads to a fall in the electrical resistance [1], the irradiation of quenched samples at –152°C leads to an increase. The different signs of the radiation-induced changes in electrical resistance indicate that a fall in irradiation temperature completely halts the decomposition of the solid solutions of aluminum and iron in α-uranium (which took place during irradiation at 60°C, accompanied by a considerable reduction in the electrical resistance).

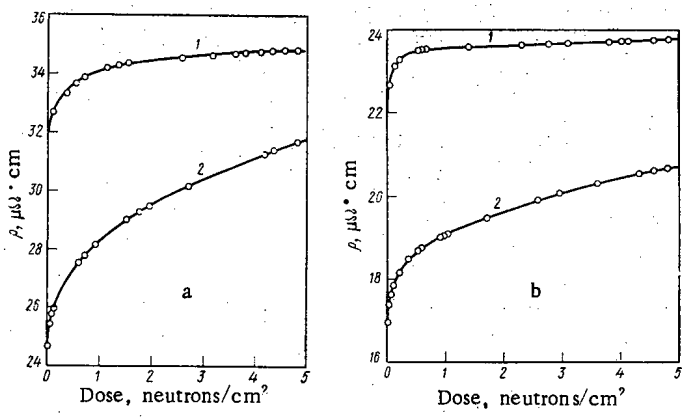

Fig. 3. Dependence of the electrical resistance of uranium alloys containing 0.88 at. % Al (a) and 0.21 at. % Fe (b) on the irradiation dose: 1) quenched; 2) aged samples.

A comparison between the corresponding curves in Fig. 2 shows that the dependence of the increment in the electrical resistance of the quenched alloys on the irradiation dose is analogous to the dependence observed in the case of pure uranium. It is reasonable to assume that, as in the case of pure uranium, this increment is associated with an accumulation of radiation defects, and its greater value indicates a higher concentration of these defects, due to their greater stability, which may itself be attributed to the presence of dissolved atoms of the alloying elements in the α-uranium lattice. Qualitatively the relationship between the increment and the concentration of the quenched solid solution is confirmed by the fact that the greatest increment in electrical resistance occurs for the quenched alloy containing 0.88 at. % Al, its value being twice as great as that of the alloy containing 0.21 at. % Fe, in which the concentration of the solid solution is four times less.

The electrical resistance of the aged alloys increases most sharply on irradiation. In contrast to the case of the quenched alloys, the electrical resistance now rises over the whole range of doses. At the maximum dose the increments in the electrical resistance of the aged samples are more than twice as great as those of the quenched samples, being 6.95 and 3.8 μΩ · cm for alloys with aluminum and iron, respectively. The greater increment in electrical resistance by comparison with the quenched alloys may be explained, as in the case of irradiation at 60°C [1], not only by the accumulation of radiation defects in the aged samples, but also by the dissolution of the finely-dispersed intermetallic compounds UAl_2 and U_6Fe, leading to an increase in the concentrations of the alloying elements in the solid solution in α-uranium and to a corresponding rise in electrical resistance. For a rough quantitative estimate of this effect we may consider that the concentration of the radiation defects and the value of the increment attributable to these do not depend on the heat treatment, i.e., they are the same for quenched and aged samples. On this assumption, the amount of the alloying additive passing into solid solution by virtue of the dissolution of the precipitates may be estimated from the difference between the increments in electrical resistance of the aged and quenched samples. For example, at the maximum dose of $5 \cdot 10^{17}$ neutrons/cm² the values of this difference for alloys with iron and aluminum amount to 2 and 3.9 μΩ · cm, respectively, which implies the passage of 0.2 at. % aluminum and 0.1 at. % iron into the solid solution. The corresponding quantities obtained on irradiating these materials at 60°C [1] amount to 0.15 at. % aluminum and 0.06 at. % iron. We may thus consider that a fall in irradiation temperature from 60 to −150°C does not reduce the rate of radiation-induced homogenization.

With increasing dose the electrical resistance of the aged samples approaches the level corresponding to the quenched alloys. Extrapolation of the curves in Fig. 3 shows that the electrical resistance of the aged alloys reaches the level of the quenched material for doses of $\sim 1 \cdot 10^{18}$ neutrons/cm². On this basis we may assume that for such doses the aged (heterogeneous) alloys pass into the homogeneous state.

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RADIOLYSIS OF SOLUTIONS OF TBP IN CONTACT WITH NITRIC ACID

II. PROCESSES OF OXIDATION AND NITRATION

E. V. Barelko and I. P. Solyanina

UDC 541.15

In [1] the principles of the formation of alkylphosphoric acids and a "polymer" from tri-n-butyl phosphate (TBP), which was irradiated in a solution of synthine in contact with HNO_3, were discussed; it was shown, in particular, that oxygen influences the radiolysis of TBP and the extraction of zirconium under these conditions.

In addition to studies in which the action of radiation on the diluent was simulated by experiments on its prolonged boiling with concentrated nitric acid [2-4], direct data on the radiation-chemical processes that occur in kerosene diluents in the presence of HNO_3 are available only in [5-8].

Huggard and Warner [5] investigated the radiolysis of TBP in a diluent of the grade "odorless kerosene." Irradiation was conducted in a stream of oxygen at a dose rate of 2-10 W/liter in the dose range 0-120 W·h/liter. Kinetic curves of the formation of compounds of the type of RCO^-, $RONO_2$, and RNO_2, as well as data on the extraction of zirconium nitrate from a solution of it (concentration $3 \cdot 10^{-3}$ M) in the presence of the salt uranyl nitrate, were obtained. By determining the degree of influence of model solutions of the primary and presumed secondary radiolysis products on the extraction of zirconium, the authors arrive at the conclusion that of the entire assortment of investigated compounds, the extraction of zirconium is influenced only by hydroxamic acids, the stability of the complexes of which with zirconium is rather high.

In [6, 7] the action of HNO_3 on the radiolysis of solutions of TBP in dodecane, mepazine, "odorless kerosene," and mixtures of paraffins in an atmosphere of air was investigated. Compounds containing the groups $>C=O$ and $-N\begin{matrix}O\\O\end{matrix}$ were identified among the radiolysis products. Compounds of the type of $RONO_2$ were not identified, and the absorption band 1640 cm^{-1} corresponding to them was assigned to the complex TBP–HNO_3. Data were obtained on the formation of products of oxidation and nitration, as well as on the extraction of zirconium from a 10^{-5} M solution of it as a function of the HNO_3 concentration. It was concluded that nitration of the system occurs as a result of the formation of the complex TBP–HNO_3.

In our investigation of the radiolysis of the system synthine–HNO_3, we determined the yields of compounds of the type of RNO_2, –R–NO, $RONO_2$, RONO, $RC\begin{matrix}O\\OH\end{matrix}$, $RC\begin{matrix}O\\\end{matrix}$ as a function of the dose, the oxygen content in the irradiated system, the HNO_3 concentration, and the presence of impurities in the diluent. The conditions of radiolysis promoting the passage of zirconium into the organic phase were also determined [8].

We suggested that a certain discrepancy in the composition of the radiolysis products in the presence of HNO_3 observed in the studies indicated above, primarily in the ratio between nitro compounds and compounds containing a carbonyl group, is associated with a difference in the concentration of dissolved oxygen in the corresponding experiments. The present work was devoted to an experimental verification of this hypothesis and a determination of the principles of the reactions of nitration and oxidation in the system TBP–synthine–HNO_3.

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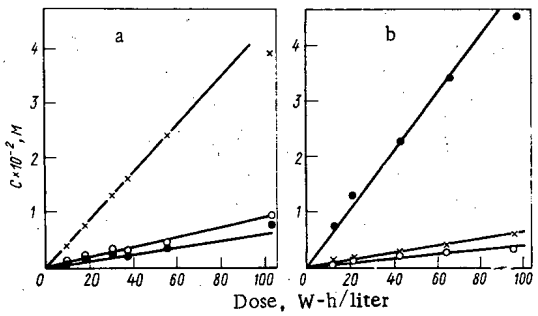

Fig. 1. Dependence of the concentration of products of nitration and oxidation on the dose: a) atmosphere of oxygen; b) vacuum; ×) $RC\overset{O}{\diagdown}$; ○) $RONO_2$; •) RNO_2 + RNO.

The method of irradiation was described in [1]. The irradiation dose rate was varied from 2 to 30 W/liter in the range of doses 0-100 W·h/liter. The experiments were conducted under vacuum, in air (without bubbling) and with continuous passage of oxygen through the irradiated system. The radiolysis products were analyzed by the method of IR spectroscopy according to a two-beam system – against an analogous, HNO_3-saturated but nonirradiated organic phase. Purified synthine, dodecane, and carbon tetrachloride were used as diluents. Together with TBP, dihexylphenylphosphonate (DHPP) was also used as an extraction reagent.

Figure 1 shows the curves of the accumulation of products of nitration and oxidation in the system 25% TBP –synthine–3MHNO_3 as a function of the dose (dose rates 20 W/liter), taken during radiolysis in an atmosphere of oxygen and under vacuum. From the data presented it follows that under conditions of a gas phase that does not change during the experiment, there is a proportionality of the concentration of compounds formed, containing the groups $\diagup C=O$, (–NO) + ($–NO_2$), $–ONO_2$, to the dose. In the case of irradiation of systems in which the penetration of oxygen into the organic solution is controlled by its diffusion (irradiation in an atmosphere of air), the nature of the dependence of the concentration of the products formed on the dose changes (Fig. 2).

Table 1 presents the values of the radiation-chemical yields of the products of nitration and oxidation, determined under conditions of a gas phase with constant composition, as well as the intervals of their variation with the dose under conditions controlled by the diffusion of air. The yields of the products calculated according to the data of [5, 6] are cited for comparison. The results obtained show that the composition of the radiolysis products actually depends substantially on the degree of saturation of the irradiated system with oxygen.

From the data presented in Table 1 it follows that when oxygen is entirely removed, the basic product of radiolysis of the system is compounds of the type of RNO–RNO_2; in an atmosphere of oxygen the yield of these compounds is sharply reduced, and there is an accumulation of carbonyl compounds. In systems irradiated in air without bubbling, as a result of the consumption of dissolved oxygen, the values of the yields of compounds of the type of RNO–RNO_2 and $RC\overset{O}{\diagdown}$ change, approaching the yields characteristic of oxygen-free systems. The values of G_{RONO_2} in this case are higher than in an atmosphere of oxygen and under vacuum.

From Table 1 and the data of [8] it follows that the presence of an extraction reagent in the system greatly influences the yield of nitration products and has little influence on the yield of oxygen-containing products. Figure 3 presents data on the influence of the dose rate on $G_{RNO-RNO_2}$, obtained under vacuum conditions. It is evident that within a broad range of dose rates (2-30 W/h at the dose 22 W/liter), $G_{RNO-RNO_2}$ is practically unchanged. We determined the dependence of the yield of nitro compounds on the concentration of nitric acid in the organic phase for the systems 25% TBP–dodecane and 25% TBP –synthine. In the first case, equal concentrations of HNO_3 were created in the organic phase before irradiation; in the second case a two-phase system, in which the HNO_3 concentration in the aqueous phase was varied, was irradiated.

A single curve of the dependence of $G_{RNO-RNO_2}$ on the concentration of HNO_3 in the organic phase was obtained, satisfactorily describing these two cases (Fig. 4a). The points obtained in the experiments in which the HNO_3 concentration in the organic phase was varied by varying the concentration of the extraction reagent extracting the acid in the system in contact with a solution of 3 M HNO_3, also lay on it. From the data cited it follows that the value of $G_{RNO-RNO_2}$ does not depend on the method of extraction of HNO_3 into the organic phase and is practically the same for synthine and dodecane.

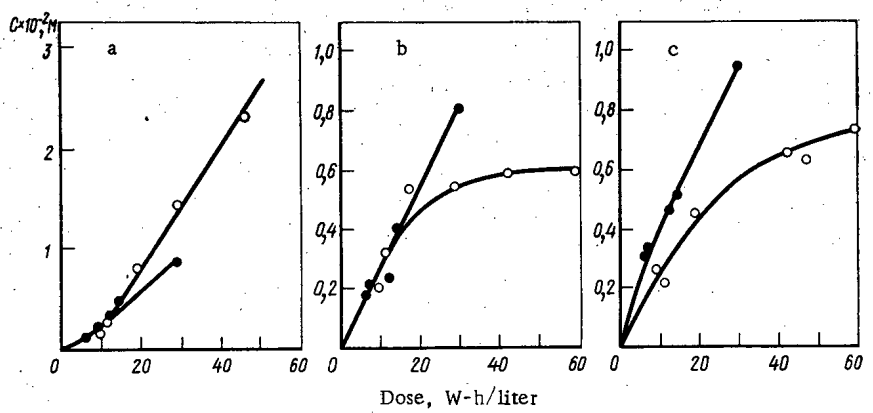

Fig. 2. Dependence of the concentration of products of nitration and oxidation on the dose in radiolysis in an atmosphere of air: a) RNO_2 $-RNO$; b) $RONO_2$; c) RCO^-; ○) dose rate 20 W/liter; ●) dose rate 5 W/liter.

TABLE 1. Radiation-Chemical Yield of Products of Nitration and Oxidation

Dose rate, W/liter	Dose range, W·h/liter	Atmosphere of radiolysis	G, molecules/100 eV			Source
			RNO+ +RNO_2	$RONO_2$	RC⟨O	
20	0—100	Bubbling of oxygen	0,2	0,3	1,2	This work
20	0—100	Vacuum	1,5	0,1	0,2	The same
5	0—30	Air without bubbling	0,4—0,96	0,76	1,2—0,75	» »
20	0—100	The same	0,4—1,6	0,76—0,2	0,7—0,2	» »
2—10	0—120	"Weak" current of air*	0,2	1,0	2,4	[5]
0,8	0—166	Air†	1,4	—	1,65	[6]

*Bubbling was characterized in this way by [6].
†Evidently without bubbling.

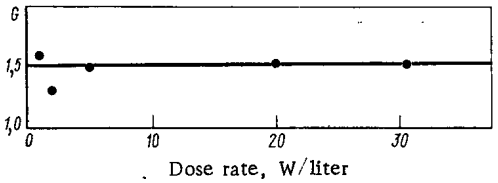

Fig. 3. Dependence of the yield of RNO $-RNO_2$ on the dose rate (vacuum).

Experimental data were also obtained, from which it is evident that the presence of an extraction reagent in the irradiated system influences the yields of nitration products not only as a result of the extraction of HNO_3 into the organic phase in the form of an extraction reagent $-HNO_3$ complex [6] but also as a result of the formation of intermediate or final products of radiolysis of the extraction reagent, capable of a nitration reaction. Thus, the replacement of TBP by DHPP, which extracts HNO_3 to practically the same degree as the first extraction reagent, in the case of irradiation under analogous conditions, leads to a decrease in $G_{RNO-RNO_2}$ within a broad range of HNO_3 concentrations (see Fig. 4b). However, it is possible that in this case the cause of the increase in stability of the system is the presence of a phenyl group, which protects not only the extraction reagent, but also the diluent from the reaction of radiation, according to an energy transfer mechanism [9].

The participation of TBP in radiation processes of nitration and oxidation is also evidenced by data on the radiolysis of its solutions in carbon tetrachloride – a diluent that does not undergo nitration. Figure 5 presents the IR spectra of CCl_4 and TBP–CCl_4, irradiated in contact with 3 M HNO_3 at a dose of 30 W·h/liter under vacuum. It is evident that the bands with absorption maxima 1680-1610 cm^{-1} and 1560 cm^{-1}, characteristic of products of the type of RONO and $RNO-RNO_2$, are detected only in the presence of TBP.

We attempted to compare the results obtained with the data published in [5, 6]. Such a comparison seems necessary, since in [6], the radiation chemical stability of solutions of TBP in various diluents, determined according to the yields of the nitration products, depends little on the nature of the diluent. Consequently, the criterion of stability is not the ratio of branched and normal paraffin hydrocarbons contained in the kerosene diluent, but primarily, as was indicated in [4, 8], the content of impurities of olefins and alcohols, from which the samples investigated in [6] were evidently freed (with the exception of mepazine).

From Table 1 it is evident that the experiments conducted in the indicated studies and this one differ in dose rate and in composition of the gas phase; therefore, their results might seem difficult to compare.

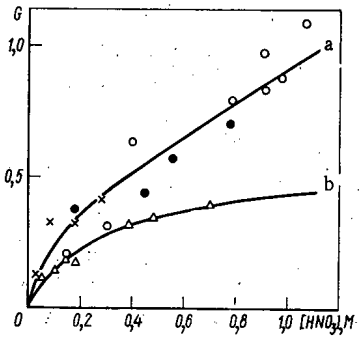

Fig. 4

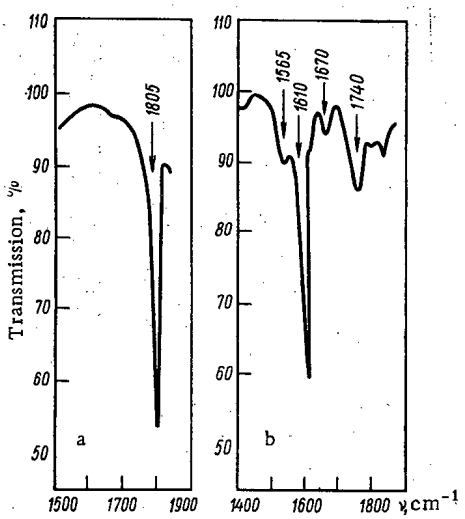

Fig. 5

Fig. 4. Dependence of the yield of nitro compounds on the HNO_3 concentration in the organic phase. a) TBP–diluent: ○) TBP–dodecane + HNO_3, irradiation of an anhydrous phase; ●) TBP–synthine + HNO_3, irradiation of a two-phase system; ×) TBP–synthine + 3 M HNO_3, irradiation with varying concentrations of the extraction reagent; b) DHPP–synthine.

Fig. 5. Infrared spectra of solutions of CCl_4 (a) and TBP–CCl_4 (b), irradiated in contact with 3 M HNO_3.

However, the data obtained show that the dose rate has no significant influence on the yields of the products under conditions when the oxygen content in the organic solution remains constant. In this case a proportionality of the concentration of the accumulated products to the dose is maintained within a broad range of doses, and, correspondingly, the ratio of the sum of the yields of the nitration products to the yield of the oxidation products:

$$\eta = \frac{G_{\mathrm{RNO-RNO_2}} + G_{\mathrm{RONO_2}}}{G_{-\mathrm{RCO}}}.$$

remains constant. In the case of irradiation in the absence of O_2, the value of $\eta \geq 8$, while in the case of passage of oxygen through the irradiated system it drops to 0.4. In the case of radiolysis under conditions of diffusion control of the delivery of oxygen to the solution, η has an intermediate value. For larger dose rates, as a result of the high rate of absorption of oxygen, its content in the system drops and at moderate doses the values of η reaches ~9, which coincides with the results of experiments conducted without oxygen. The results obtained permit us to conclude that the discrepancies between the ratios of the oxidation and nitration products in [5, 6] may be due to a different concentration of oxygen in the irradiated systems. In the experiments of [6] this shortcoming is obvious.

However, the data considered, obtained in the presence of an extraction reagent, do not give exhaustive answers to the questions pertaining to the mechanism of nitration. The fact that the role of the extraction reagent in such systems is reduced only to the extraction of HNO_3 also is not yet proven. In [8] we cited evidence with respect to the fact that the main source of nitro derivatives in the diluent is impurities of unsaturated compounds (especially alcohols) and that the influence of oxygen on radiolysis is associated with an inhibition of the radiolytic decomposition of NHO_3. It may also be that in this case the reactions of R + NO_2(NO) are also inhibited on account of competition with the reactions of R + O_2. All these considerations are also applicable to the present work, where the role of oxygen is obvious, while TBP, as is well known [10], is a source of butyl alcohol, formed in its radiolysis. This alcohol may also play the role of a nitrating impurity.

In conclusion, let us note the special character of the dependence of the yield of $RONO_2$ on the oxygen concentration in the system. In contrast to the dependence for carbonyl and nitro compounds, the yield of $RONO_2$ does not vary monotonically with the oxygen content in the system, but is a maximum for the middle oxygen concentration that we used, which corresponds to a gas phase consisting of air at the beginning of the experiment. The maximum value of G_{RONO_2} in our experiments was equal to 0.76, while in [6] it reached 1.

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ON THE MAGNITUDE OF THE MAGNETIC FIELD PRODUCED IN A LASER PLASMA

A. F. Nastoyashchii

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Spontaneous magnetic fields of fairly large amplitude have been observed in a number of experiments on laser heating of plasmas [1-3]. Thus, magnetic fields, $B \approx 1$ kG were measured on the boundary of a plasma in [2], and the fields increased rapidly upon approaching the center of the plasma. Several hypotheses about the origin of this magnetic field were proposed. One of the reasons for the field may be the laser radiation itself (for example, due to the inverse Faraday effect for a beam of circularly polarized waves [4]). Another reason may be an asymmetry in the laser plasma [2]. In the latter case, the rate of growth of the magnetic field is given by the formula [5, 2]

$$\frac{\partial \mathbf{B}}{\partial t} = \frac{c}{en}(\nabla T \times \nabla n) + \frac{c^2}{4\pi\sigma}\nabla^2 \mathbf{B}, \tag{1}$$

where n is the density; T is the temperature; and σ is the plasma conductivity. We shall stay within the framework of this model and attempt to find out whether the magnetic fields which arise spontaneously in a laser produced plasma can significantly influence the physical properties of the plasma. For example, in the widely discussed scheme by Teller for laser triggering of a thermonuclear reaction [6, 7], an important role is played by the plasma thermal conductivity which must ensure the necessary heat flux from the periphery of the compressed target to its dense core. Since the ordinary (no magnetic field) thermal conductivity already hardly provides the needed heat transfer [8], the presence of a magnetic field may make the situation much worse. Actually, if we substitute typical parameters of a laser produced plasma ($n \approx 10^{21}$ cm^{-3}, $T \approx 1$ keV, laser pulse duration $\Delta t \approx 10^{-9}$ sec) into Eq. (1), then we obtain for the ratio of the electron cyclotron frequency in the magnetic field to the collision frequency, $\omega\tau_e \approx 10^2$-$10^3$. In this case, Teller's scheme would in fact cease to work.

However the situation is more promising. In reality there is a saturation in the magnitude of $\omega\tau_e$. Since it is not possible to solve the exact problem of the magnitude of B, we limit ourselves to a qualitative consideration of the problem. In several typical examples we shall show that in each case it is possible to find a saturation mechanism which limits the value of B.

We assume the laser produced plasma borders on a material wall which absorbs heat so that a temperature (and density) gradient arises in a direction away from the wall and toward the plasma. In addition, there is a pressure gradient along the wall, so that $\partial B/\partial t > 0$.

After some time, when $\omega\tau_e$ becomes appreciable, a heat flux arises in the direction of the vector $(\nabla T \times B)$, which leads in turn to a rotation of the temperature gradient vector such that the directions of ∇T and ∇n rapidly become almost antiparallel and further growth in the field is impeded. The "restructuring" time for these gradients is

$$t \approx (nT/|\nabla n||\nabla T|)/2\chi_\wedge, \tag{2}$$

where $\chi_\wedge$ is the thermal conductivity of the plasma in the direction of $(B \times \nabla T)$ [9].

Substituting Eq. (2) in Eq. (1), we find

$$B \approx (cT/2e\chi_\wedge) \propto n/T^{3/2}, \tag{3}$$

from which it is apparent that the dimensionless ratio $\omega\tau_e$ remains near unity, i.e., $\omega\tau_e \approx 1$.

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The initial symmetry of the plasma may be destroyed due to instabilities. Some instabilities with excitation of "thermomagnetic waves" have been discussed previously in solid state plasma and astrophysical applications in papers by Gurevich and co-authors [5]. Temperature (and density) gradients are necessary for the development of these instabilities.

Furthermore, instabilities are possible which lead to a purely exponential growth in the magnetic field [10, 11]. It can be shown that for aperiodic instabilities to arise it is entirely enough that there be a density gradient alone.

In all of these cases the growth of the magnetic field may be limited by the "restructuring times" of both the temperature and density gradients. We shall now consider the situation in which the hydrodynamic plasma motion which arises due to the development of an instability (which is a stabilizing factor) limits the maximum attainable value of B. Substituting the characteristic restructuring time for the density gradient, $t \approx \Delta'/v_s$, where Δ' is the dimension of the perturbation and v_s is the speed of sound, into Eq. (1), we find

$$\omega\tau_e \approx \frac{l}{\Delta_T}\sqrt{\frac{M}{m}}\frac{\Delta'}{\Delta} \leqslant \frac{l}{\Delta_T}\sqrt{\frac{M}{m}}. \tag{4}$$

Here Δ and Δ_T are, respectively, the characteristic dimensions of changes in density and temperature; l is the mean free path of an electron in the plasma. Since in a Coulomb plasma Δ_T is about an order of magnitude larger than the mean free path, according to Eq. (4) $\omega\tau_e$ cannot be much larger than unity in a hydrogen plasma. A plasma of heavier elements requires special consideration.

We shall now discuss the magnetic pulsations excited by ion-acoustic oscillations.* The magnetic field must oscillate at the ion-acoustic frequency in this case. We assume for simplicity that the wave vector of the oscillations is at a right angle to the direction of the density (or temperature) gradient.

1. Let $\nabla n \neq 0$ and $\nabla T = 0$. Magnetic oscillations are possible in this case only in the presence of long wavelength oscillations with $k^{-1} \gg \sqrt{M/ml}$ since in the opposite case there would be no temperature fluctuations (isothermal sound). Putting $t \approx 1/kv_s$ in Eq. (1) we obtain the following upper estimate:

$$\omega\tau_e \approx \tilde{x}, \tag{5}$$

where $\tilde{x} \approx \tilde{n}/n$ is the relative amplitude of density fluctuations in the wave.

2. We now take $\nabla T \neq 0$ and $\nabla n = 0$. By analogy with the preceding, we find

$$\omega\tau_e = \sqrt{\frac{M}{m}}\frac{l}{\Delta_T}\tilde{x}. \tag{6}$$

Since the factor on the right-hand side of Eq. (6) cannot be much larger than unity for a hydrogen plasma and the amplitude of the oscillations, as a rule, is $\tilde{x} \ll 1$, it follows that $\omega\tau_e$ in both cases cannot be appreciably larger than unity.

Therefore, the maximum possible magnetic fields arising in a laser produced plasma are such that the dimensionless ratio $\omega\tau_e$ cannot significantly exceed unity and there is no catastrophe for Teller's scheme. Besides this, the production of magnetic fields in a laser compressed plasma may lead to certain difficulties. In fact, already for $\omega\tau_e \approx 1$ the thermal conductivity of the plasma is reduced by several times [9]. This means that the thermal conductivity is not able to provide for a rate of transfer of the incident energy from a neodymium laser greater than 10^{15} W/cm^2. If a CO_2 laser is used, where the density of the heated plasma is 100 times less, the situation is even worse. Thus, the spontaneous production of magnetic fields may lead to definite limitations in the ultimately attainable degree of compression of a DT droplet and, hence, to a requirement for an increased amount of radiant energy. It would be advantageous to investigate this question in detail.

The author thanks the participants of R. Z. Sagdeev's seminar for discussions.

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PARAMETERS OF THE NEUTRON RESONANCES OF ^{241}Am IN THE ENERGY RANGE FROM 8 TO 30 eV

T. S. Belanova, A. G. Kolesov,
V. M. Nikolaev, V. A. Safonov,
V. Ya. Gabeskiriya, V. A. Poruchikov,
S. M. Kalebin, R. I. Ivanov,
and O. M. Gudkov

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Measurements of neutron cross sections of ^{241}Am are of significant interest not only for the development of the theory of the nucleus but also for the solution of a series of important practical problems. These measurements are greatly complicated as a consequence of the high activity and low accessibility of americium. The experimental information – specifically the information on the total neutron cross sections of the ^{241}Am nucleus – is inadequate [1-4]. In the work described in the present paper the measurements of the energy dependence of the total neutron cross sections were carried out on a SM-2 reactor by the time-of-flight method. The neutron burst was formed by a selector having three synchronously rotating motors suspended in a magnetic field [5]. The neutron detector consisted of a bank of helium counters. The spectrometer resolution was 70 nsec/m.

Measurements and Results

Measurements of the total neutron cross section of ^{241}Am were performed using a powder consisting of stable dehydrated americium oxide having a known oxygen content (AmO_2). For this purpose the powder was calcined for 3 h in an oxygen atmosphere at a temperature 400°C. Then (99.9 ± 0.2) mg of powder was poured into an aluminum cassette having a wall thickness of 1 mm. The volume of powder was 0.8 × 8.0 × 0.42 mm. The fabricated target was remotely mounted with its "thin" or "thick" side mounted in the path of the neutron beam, which allowed measurement of the transmission at two samples that differed considerably in thickness: $0.63 \cdot 10^{21}$ and $0.33 \cdot 10^{22}$ atoms/cm^2. The sample contains 99.99% ^{241}Am; the ^{239}Pu and ^{237}Np impurities do not exceed 0.002%; the impurities of the other two transuranium elements are absent. The statistical accuracy of the measurements was held within the limits of 0.6 to 1%; the background varied within the limits of 1 to 3%. A 3% correction taking into account scattering by oxygen was introduced into the transmission data for a thick sample. The measured energy dependence of the total neutron cross section is displayed in Fig. 1.

Figure 2 shows the experimentally obtained dependence of the transmission T on neutron energy in the 8 to 30 eV interval. The high purity of the americium sample allows the statement to be made that all of the detected levels belong to the ^{241}Am nucleus. Up to an energy of 26.0 eV the resolution of the installation allowed calculation of the parameters of the neutron resonances by the shape method according to the one-level Breit–Wigner formula. The calculations were performed on a BÉSM-6 computer. The principal singularity of the computation program consisted in the fact that the resolution function of the installation was not approximated by an analytic expression but was calculated directly on the computer for each resonance with allowance for its energy and the experimental characteristics of the installation (including the angular divergence of the neutron beam). As a result of the calculation the values of neutron width $2g\Gamma_n$, the position of the E_0 level and the values of the total width Γ for a series of levels were determined (Table 1). The errors likewise took into account the errors associated with determination of the shape of the resolution function. The latter three resonances of Table 1 were computed solely by the method of invariants, the radiation width being taken to equal 40 mV under these conditions. Levels having energies of 26.50, 26.67, 27.52, and 27.65 eV were not processed in view of their inadequate resolution. The

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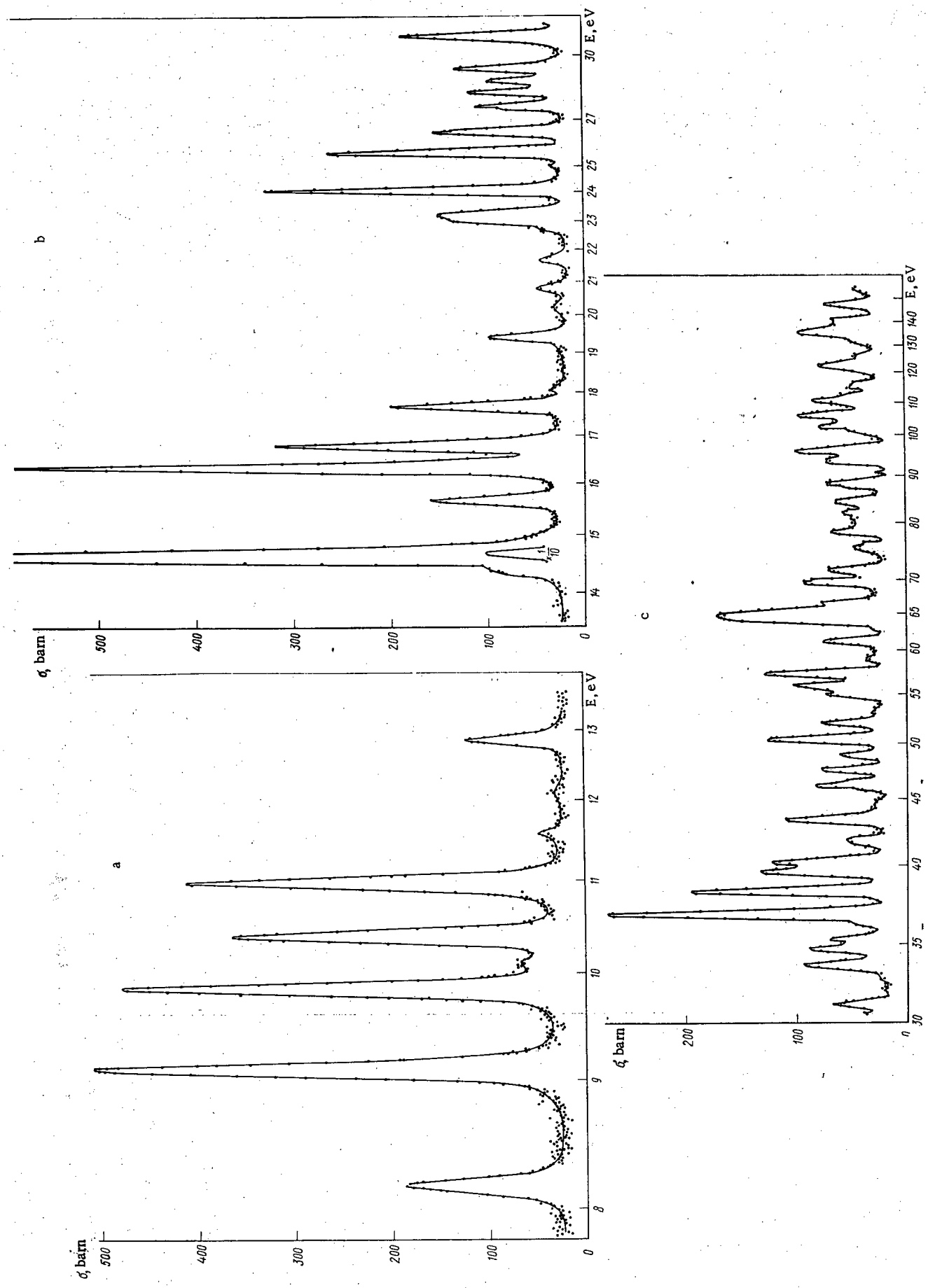

Fig. 1. Total neutron cross section: a) for 8 to 13 eV; b) 4 to 30 eV; c) 31 to 150 eV.

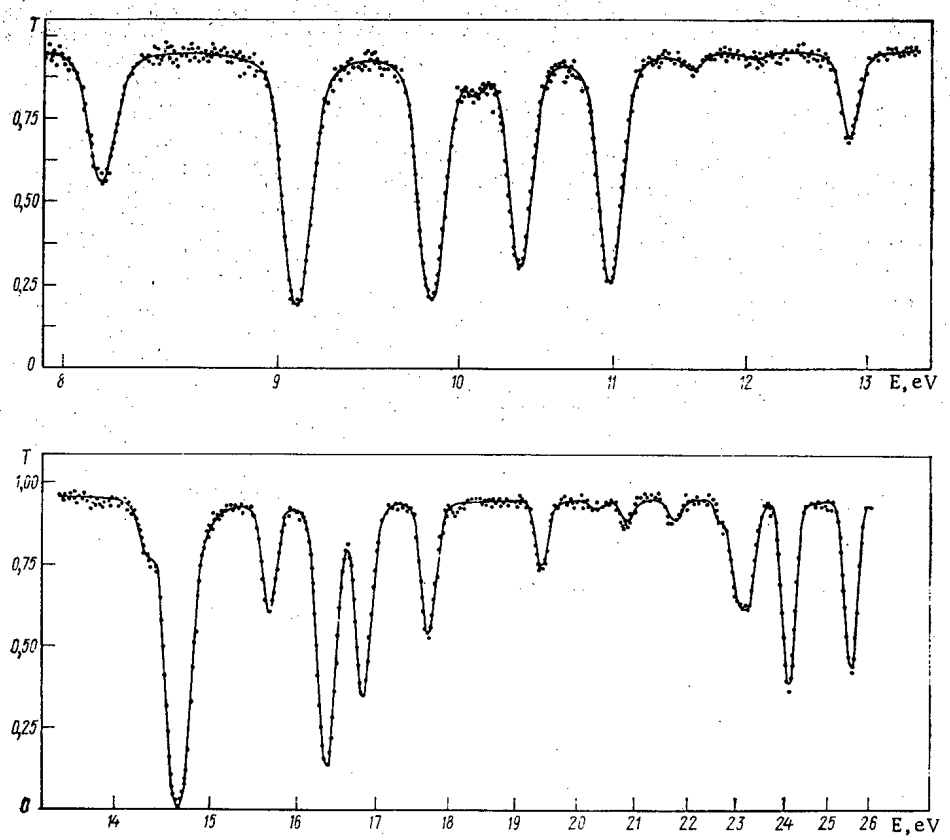

Fig. 2. The transmission of ^{241}Am: ———) theoretical calculation by the shape method; •) experimental data.

TABLE 1. Parameters of the Neutron Resonances of ^{241}Am

E_0, eV	Γ, MeV	$2g\Gamma_n$, MeV	σ_0, barn
8,17±0,02	42±5	0,096±0,004	360±40
9,11±0,02	48±3	0,358±0,006	1060±70
9,84±0,03	48±3	0,370±0,007	1030±70
10,11±0,03		0,025±0,004	80±40
10,39±0,03	45±4	0,294±0,007	820±70
10,99±0,04	52±4	0,382±0,008	880±70
11,58±0,05		0,018±0,003	50±20
12,06±0,06		0,007±0,003	20±10
12,86±0,06	44±5	0,116±0,009	260±70
14,32±0,06		0,066±0,012	150±50
14,66±0,07	44±5	2,30±0,13	4600±80
15,66±0,07	32±12	0,215±0,012	560±200
16,35±0,07	44±5	1,185±0,033	2100±500
16,81±0,07	31±8	0,575±0,020	1450±340
17,69±0,07	40±10	0,373±0,016	690±160
18,09±0,07*			
19,39±0,07	37±12	0,182±0,016	330±200
20,28±0,07		0,050±0,010	40±20
20,84±0,08		0,064±0,011	100±40
21,72±0,08		0,067±0,012	100±40
22,74±0,09		0,070±0,012	100±40
23,08±0,09		0,39±0,05	550±200
23,33±0,09		0,40±0,05	550±200
24,17±0,09		1,27±0,08	4100±2000
25,00±0,10*			
25,65±0,10		1,21±0,08	3400±2000
26,50±0,10†			
26,67±0,10†			
27,52±0,10†			
27,65±0,10†			
28,31±0,11		0,40	
28,82±0,12		0,35	
29,43±0,12		0,61	

* Doubtful resonance.
† Poorly resolved resonance.

TABLE 2. The Values of s_0 and $\overline{D}$ Obtained for ^{241}Am

$s_0 \times 10^{-4}$	$\overline{D}$, eV	Energy range, eV	Paper
0,85±0,25	0,7±0,2	0.3—10.3	[2]
1,10±0,20	—	0—40	[3]
0,97±0,12	0,67	0—50	[4]
0,75±0,23	0,70±0,10	8—26	Present paper

theoretical transmission curve displayed in Fig. 2 in the form of the solid line was calculated according to the parameters of the neutron resonances.

The problem of resonances having energies of 18.09 and 25.00 eV is for the time being difficult to solve uniquely. In order to prove their existence it is necessary to perform measurements with a thicker sample. The level having the energy 12.06 eV gives rise to a certain doubt. The new resonances which were found having energies of 10.11, 11.58, 14.32, and 20.28 eV are completely reliable. It should be noted that the present paper calculates the values of the resonance parameters for the first time for the 21.72 eV level detected in measurements of the fission cross section during an atomic explosion [6]. Simultaneously with finding the parameters of the resonances in the energy range from 8 to 26 eV by means of the computer, the magnitude of the potential scattering cross section σ_p for the ^{241}Am nucleus was varied. The best agreement with experiment was obtained for $\sigma_p = (12.8 \pm 1)$ barn.

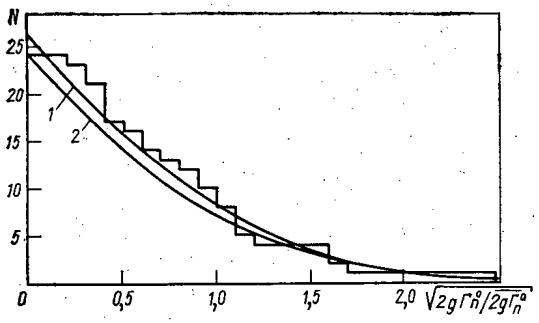

Fig. 3

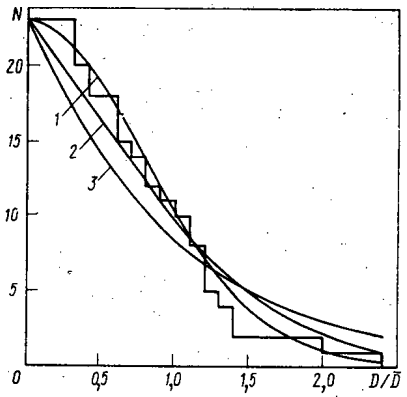

Fig. 4

Fig. 3. Integral distribution of the reduced neutron widths of ^{241}Am (N is number of resonances): 1) Porter–Thomas distribution for one degree of freedom, normalized to 26 resonances; 2) the same distribution normalized to 24 resonances; the histogram represents the experimental distribution.

Fig. 4. Integral distribution of the distances between the neutron levels of ^{241}Am (D is distance between resonances): 1) the Wigner distribution for one system of levels; 2) superposition of two Wigner distributions with equal average distances between the levels; 3) exponential distributions; histogram represents the experimental distribution.

Discussion of Results

The values obtained for the parameters of the resonances mainly agree with those published in [1–4]. Let us note the discrepancies.

1. For the 8.17 eV resonance the quantity $2g\Gamma_n$ (see Table 1) is approximately one order smaller than the value 0.80 ± 0.3 MeV given in [3], while for the neighboring 9.11 eV resonance the magnitudes of the neutron widths obtained in the present paper (0.358 ± 0.006 MeV) and in [3] (0.42 ± 0.06 MeV) coincide within the limits of the indicated errors. From the transmission curve (see Fig. 2) it is evident that the 8.17 eV resonance is considerably weaker than the 9.11 eV resonance, and therefore the magnitude of its neutron width must not exceed the value of the neutron width for the 9.11 eV resonance. It is of interest to note that the earlier [1] value for the 8.17 eV resonance was still greater and equaled 5 ± 2 MeV.
2. In [3] a fairly strong level having an energy of 8.28 eV is given ($2g\Gamma_n$ = 0.128 MeV). In the work described in the present paper we were not able to detect this level, although resonances having a smaller neutron width were registered in it.
3. The very strong 18.37 eV resonance, for which $2g\Gamma_n$ = 0.43 MeV was determined in [3], was likewise not detected. Judging from the magnitude of the neutron width, the 18.37 eV level must be no lower than the neighboring 17.69 and 19.39 eV resonances for which $2g\Gamma_n$ is correspondingly equal to 0.373 and 0.182 MeV (see Table 1).

The noted discrepancies are most probably associated with the fact that in [3] an insufficiently pure ^{241}Am sample was used and levels having energies of 8.28 and 18.37 eV were erroneously ascribed to this nucleus. In [3] the chemical composition of the target was not given, and therefore one may only assume that the indicated resonances belong to isotopes that accompany ^{241}Am.

The data of Table 1 in the energy range from 8 to 26 eV were used for statistical handling of the experimental results. Figure 3 displays the integral distribution of the reduced neutron widths. The experimental histogram is in satisfactory agreement with the Porter–Thomas distribution for one degree of freedom. The agreement improves if the theoretical distribution is plotted by adding any two levels in the considered energy range, which may serve as a certain indirect indication of the existence of resonances having the energies 18.09 and 25.00 eV. Figure 4 displays the experimental histogram of the integral distribution of the distances between levels. For comparison the integral theoretical distributions [7] are shown. The exponential distribution differs noticeably from the experimental results. This is not

in agreement with the assumption [8] that in odd nuclei the distribution of distances between levels follows an exponential law. For the time being it is difficult to decide which of the distributions (1 or 2; Fig. 4) describes the experiment better. It is not excluded that a somewhat better coincidence of the histogram with the curve 1, contrary to the expected coincidence with the curve 2, is random and can be explained by the slight resort to levels to construct the histogram. It is possible that the situation will change if the resonances lying below 8 eV are included. However, the remarks made above concerning [1-3] do not allow levels below 8 eV to be used without additional checking. In the present paper we have assumed that the energy range from 8 to 26 eV is sufficiently resolved. Starting from this condition for the considered energy range with allowance for the assumed two omitted levels, the following values of the average distance $\overline{D}$ between levels and the neutron force function s_0 have been obtained [9]: $\overline{D} = (0.70 \pm 0.10)$ eV; $s_0 = (0.75 \pm 0.23) \cdot 10^{-4}$.

The errors were calculated on the basis of the Porter–Thomas and Wigner distributions. The comparison of the obtained values with the data cited in other papers is displayed in Table 2.

The authors thank Yu. G. Abov and Yu. S. Zamyatnin for the interest manifested in the work and for their useful discussions, and likewise associates T. V. Denisova, S. I. Babich, S. N. Nikol'skii, G. A. Pelevin, N. G. Kocherygin, G. A. Timofeev, G. V. Kuznetsov, G. V. Rukolain, and V.S. Artamanov who helped at various stages of the work.

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RECOVERY OF NEUTRON SPECTRUM BY THE METHOD OF SIMULATED SPECTRA USING SET OF THRESHOLD DETECTORS

G. M. Obaturov and A. A. Tumanov

UDC 539.125.5.164

The article provides a computational and experimental verification of the method of simulated spectra. This method presupposes presentation of a real reactor spectrum in the form of a superposition of five simulated spectra* which include the Fermi spectrum, the perturbed Fermi spectrum, the perturbed and unperturbed fission spectrum, and a spectrum in the form of a Gaussian curve.

The recovery of the spectra is carried out on the basis of readings of two sets of threshold detectors consisting of the following elements:

1) ^{239}Pu, ^{237}Np, ^{232}Th, ^{238}U (all in boron absorber with 1 g/cm^2 ^{10}B wall thickness), and ^{239}Pu in a cadmium jacket;

2) the same set, but with ^{238}U in boron replaced by ^{31}P in cadmium.

The calculated recovery of 26 distinct reactor neutron spectra published in the literature is discussed. The criteria for accuracy in the recovery of the spectra are: the difference in the values of the recovered neutron flux and the real flux for flux over the entire energy range and for separate distinct energy intervals; the accuracy with which the shape of the real spectrum is retrieved; the difference in the kerma of the neutrons in the recovered spectrum and in the real spectrum.

The calculations demonstrated a high degree of reproducibility for most of the spectra under consideration (accuracy on the part of the recovered total flux and kerma within a few percentage points). Spectra in which the neutron distribution peaks fell within the range of energies below 0.8 MeV (iron-shielded reactor, BR-1, FLATTOP) were not recovered in satisfactory manner. In the case of those spectra, the error in the kerma determination amounted to 50%. The results of recovery based on the two sets of threshold detectors were found to be quite close.

The paper also presents examples illustrating recovery of reactor spectrum on the basis of experimental data from the set of detectors. The method of simulated spectra is intended primarily for

*A. Fasso et al., SESR/SESRCI N 70/52 AF/BS, Cadarache (1970).

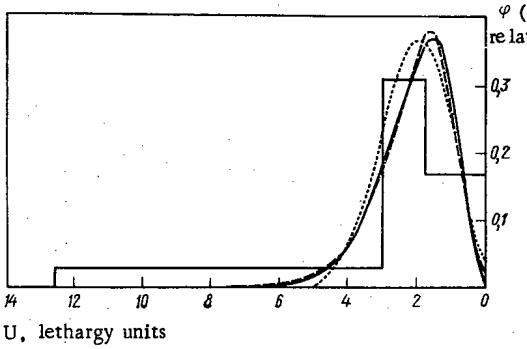

Fig. 1. Spectra of PLUNIT reactor ($U = \ln E_0/E$): ——) real spectrum; – – – and · · · ·) spectra recovered by the method of simulated spectra with the respective sets of threshold detectors employed.

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neutron dosimetry, and makes it possible to reproduce, in a simple and quite accurate manner, a broad class of reactor spectra. The illustration shows an example of a neutron spectrum restored by the proposed method. The bar diagram was constructed by the method of threshold detectors using the first set only.

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Abstract submitted August 30, 1974

RATING OF THE CAPABILITIES OF PHOTONEUTRON INSTALLATIONS FOR DETERMINING DEUTERIUM CONCENTRATION UTILIZING SNM COUNTERS

R. P. Meshcheryakov, S. V. Popov, and E. S. Solodovnikov

UDC 543.53

It is advisable to utilize the photoneutron method as the fastest in gravimetric determinations of the concentration of deuterium in water. However, the relatively low accuracy of the analyses (2-3%) reduces the effectiveness of the application of this method.

A photoneutron installation, allowing one to enhance the accuracy of the methodup to 0.5% andsimultaneously to increase its capacity several-fold, was constructed in the Scientific-Research Institute of Nuclear Physics at Tomsk Polytechnic Institute. The photoneutron counting rate during the analysis of samples with a volume (300 ml) usual for this method is 240 pulses/sec, a background of 1 pulse/sec, and a measurement time of 400 sec.

TABLE 1. Counting Rate and Maximum Activity of Source for the Detection Units of Photoneutron Installations as a Function of the Counter Type*

d, cm	N/A †			
	SNM-11	SNM-30	SNM-17	SNM-18
6	55/30	136/4,5	211/3	500/0,6
10	32/380	78/57	120/38	290/2
12	22/1000	55/150	86/100	202/20
14	12/2500	30/375	45/250	106/50

*The moderator is water with a lead shield of thickness d.
†The quantity (N) is the counting rate of the counters (pulses/sec) when the activity of the γ-source is 1 Ci; the denominator (A) is the activity of the source (Ci) corresponding to the maximum permissible dose rate of γ-radiation for the counters.

Further development of the method must be combined with a reduction in the volume of the sample and with an improvement in its accuracy. Therefore, an attempt to rate the capabilities of photoneutron installations utilizing standard SNM-type (SNM-11, SNM-17, SNM-18, SNM-30) slow neutron counters, a water moderator with a lead shield, and a graphite moderator was undertaken.

The experimental results and calculations conducted by statistical sampling techniques allowed one to evaluate the counting rate of various counters as a function of the activity of a γ-source (^{24}Na) for samples 10-15 ml in volume (see Table 1). From the table it is seen that there exists a real possibility of reducing the volume of the samples to 10 ml or lower with the accuracy of the analyses no worse than 0.5%.

Original article submitted February 25, 1974

MONTE CARLO CALCULATION OF THE DISTRIBUTION OF ELECTRON-BEAM ENERGY ABSORBED IN MATTER

A. M. Zlobin, E. N. Donskoi, and V. V. Khizhnyakov

UDC 539.121.17:518

In this paper, the spatial distributions of the energy absorbed dE/dm and the transmission coefficients for electrons with energy ε from 10 keV to 1 MeV in plates and double-layer systems are calculated by the Monte Carlo method. A procedure for separating the singular part of the total function from the angular distribution [1, 2], which corresponds to the separation of the electrons into unscattered and scattered groups, is utilized in the methodology developed. This allows one to calculate more properly the distribution function for multiply scattered electrons. Ionization losses are taken into consideration according to the relativistic theory of Bethe–Møller. The magnitude of the interval in the calculations was selected in the form

$$\Delta s = \frac{1}{25} R_G \varepsilon_0^{1,75}$$

(here $R_G = 4 \cdot 10^{-6}$ g/cm^2 is the so-called extrapolated range of Grün for $\varepsilon = 1$ keV; ε_0 is the initial energy of the electrons, keV).

The results of the calculations agree well with experiment [3-7] in the neighborhood of the energy ε considered, for which in a number of cases, the agreement is substantially better than those with available calculations [8].

In this paper, the dependence of $(dE/dm)_{max}$ on the thickness of a sample of aluminum, as well as the distinctive features of the distribution for dE/dm in double-layer systems are investigated. The theoretical curves for the energy distribution dE/dm are shown in Fig. 1. Calculation indicates a strong dependence of dE/dm on the thickness of the sample and the electrons scattered from the second layer, when the thickness of the first layer is less than or comparable to R_e.

It is shown that the maximum energy absorbed in a double-layer system when $Z_1 < Z_2$ (Z_1 and Z_2 are the atomic numbers of the elements of the first and second layers, respectively) can exceed the maximum energy absorbed in each substance of the system individually. When $Z_1 > Z_2$, the maximum energy absorbed in a double-layer system is found to be lower than the maxima in each substance of the system.

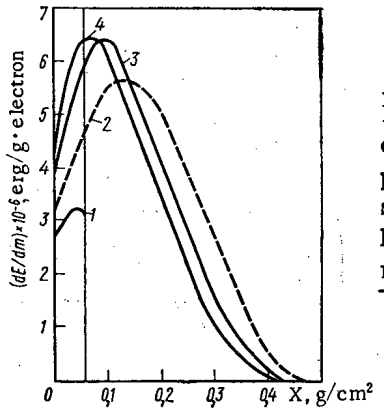

Fig. 1. Distribution of absorbed energy dE/dm ($\varepsilon = 1$ MeV): 1) in an aluminum plate of thickness $x = 2 \cdot 10^{-2}$ cm; 2) in semiinfinite aluminum; 3, 4) in double-layer systems of Al–Cu and Al–Ag, respectively.

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Original article submitted April 25, 1974

CONSIDERATION OF THE DECAY AND ACCUMULATION OF MIXTURES OF GENETICALLY RELATED SHORT-LIVED ISOTOPES IN γ-SPECTRAL ANALYSIS

Yu. A. Zaitsev

UDC 539.183.2

Certain methodological questions in γ-spectrum analysis are considered. It is shown that in the analysis of mixtures of genetically related short-lived isotopes, it is necessary to consider the dynamics of the mixtures. Correct relationships between the activity of the isotopes and the average counting rates are presented. For example, for two isotopes related by a line of decay, with the initial condition $Q_2(t_0) = 0$, the relationship

$$n_k = \alpha_1(\Delta t)\, a_{K1} Q_1(t) + \beta_2(t - t_0;\ \Delta t)\, a_{K2} Q_2(t),$$

will be correct, where n_k is the average counting rate in the k-th energy range; $Q_1(t)$ and $Q_2(t)$ are the activities of the parent and daughter isotopes at the beginning of the measurement t; t_0 is the beginning of the accumulation of the daughter isotope; Δt is the duration of the measurement; a_{Ki} are calibration coefficients;

$$\beta_2(t - t_0;\ \Delta t) = \frac{\alpha_1(\Delta t) - \alpha_2(\Delta t)\, e^{(\lambda_1 - \lambda_2)(t - t_0)}}{1 - e^{(\lambda_1 - \lambda_2)(t - t_0)}};$$

$$\alpha_i(\Delta t) = \frac{1 - e^{-\lambda_i \Delta t}}{\lambda_i \Delta t}$$

(here λ_i is the decay constant of the i-th isotope).

Taking into account the individual components of a daughter isotope, $Q_2'(t)$ results in the relationship

$$n_k = \left\{ \alpha_1(\Delta t)\, a_{K1} + \frac{P_2^1 \lambda_2}{\lambda_2 - \lambda_1} [\alpha_1(\Delta t) - \alpha_2(\Delta t)\, e^{(\lambda_1 - \lambda_2)(t - t_0)}]\, a_{K2} \right\} Q_1(t) + \alpha_2(\Delta t)\, a_{K2} Q_2'(t),$$

where P_2^1 is the yield of a nucleus of a daughter isotope (in a given case, a metastable state of an isomer is also understood to be a daughter isotope).

In the relationships presented, Δt is the actual, physical time of measurement. Present-day measurement apparatus, as a rule, operates at a rate of a "life" time Δt_L, corresponding to a given Δt. The difference in these values depends on the rate of utilization of the analyzer and can attain 50% or more, which in turn results in additional systematic errors.

Taking into consideration the dynamics of mixtures of short-lived isotopes is especially important for semiconducting spectrometry. This is associated with the long duration of the measurements and the analysis, as usual, of complex, chemically unseparated mixtures of radioactive isotopes. The distinctive features of the analysis discussed are important independently of the method of treatment of the spectrograms.

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APPLICATION OF ALBEDO EQUATIONS

V. V. Orlov

UDC 539.125.52:621.039.51.12

An extremely effective mathematical method, based on the invariance of the radiation transfer equations under a displacement of the coordinate system, has been proposed by Ambartsumyan [1] for solving astrophysical problems associated with the reflection of radiation from a medium. This method, which was generalized in [2, 3], allowed one to formulate immediately nonlinear equations for the probability of radiative reflection from a medium without solving the problem of the distribution of the radiation inside of the medium.

In [4-6] et al., an albedo method was applied to neutron transport problems occurring in reactor physics. However, until now this method has been fully (and in our opinion it is merited) omitted and scarcely used in theory and practical reactor calculations. Meanwhile, the albedo method, possessing not only mathematical elegance, but also purely practical merit, could serve as a basis for very effective approximations in the solution of a number of reactor problems.

For example, let us consider the problem of the reflection of neutrons from a cylindrical rod, the solution of which can be used in the formulation of the boundary conditions on the surface of the control rods of a reactor. We shall use the simplest approximations, calculating the neutron flux incident on a rod and emitted from it, and the angular distribution of the neutrons scattered isotropically, as well as the multigroup representation of the energy spectrum of the neutrons.

In order to derive the equation determining the albedo of a rod $\hat{\beta}(R)$,* let us add a thin layer of width Δ to a rod of radius R (Fig. 1).

Let us express $\hat{\beta}(R+\Delta)$ in terms of $\hat{\beta}(R)$, taking into account only single collisions of the neutrons in a layer Δ (then letting $\Delta \to 0$). Five possible cases are found: the neutrons traverse the layer Δ without collisions, are reflected from the cylinder of radius R and traverse the layer Δ in the opposite direction without collisions (1); the neutrons are scattered in the layer Δ and return (2); the neutrons are scattered in the layer Δ, fall on the inside cylinder and are reflected from it (3); the neutrons are reflected from the cylinder of radius R, are scattered in the layer Δ and are emitted outwards (4); the neutrons are reflected from the inside cylinder, are scattered in the layer Δ and are again reflected from the cylinder R (5).

The probability of the collision of a neutron in a thin layer of thickness Δ is $2\Delta\Sigma$. With isotropic scattering, $\Delta\hat{\Sigma}_s$ of the neutrons fly off in the forward direction and $\Delta\hat{\Sigma}_s$ in the opposite direction ($\hat{\Sigma}_s$ is the matrix of the differential neutron scattering cross sections $\|\Sigma_{ik}\|$). It is necessary to also take into account that the neutrons incident on the rod $R+\Delta$ can traverse the layer Δ without collisions, missing the inside cylinder (see Fig. 1, 6). The probability of this is Δ/R.

As a result, neglecting quantities of the order of Δ^2 and higher, we obtain

$$\beta_{ik}(R+\Delta) \approx \beta_{ik}(R)+\Delta\frac{d\beta_{ik}}{dR} \approx \left(1-2\Delta\Sigma_i-\frac{\Delta}{R}\delta_{ik}\right)$$

$$\times\beta_{ik}(1-2\Delta\Sigma_k)+\Delta\Sigma_{sik}+\frac{\Delta}{R}\delta_{ik}+\Delta\sum_l\Sigma_{sil}\beta_{lk}+\Delta\sum_l\beta_{il}\Sigma_{slk}+\Delta\sum_l\sum_m\beta_{il}\Sigma_{slm}\beta_{mk}$$

*In a group representation, the albedo $\hat{\beta}(R)$ is a matrix $\|\beta_{ik}(R)\|$, an element of which $\beta_{ik}(R)$ is the probability of emitting a neutron, which entered the rod with energy in group i, in group k.

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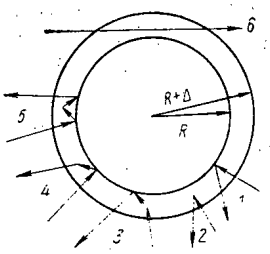

Fig. 1. Diagram of the passage of neutrons in a cylindrical rod.

or after cancellation

$$d\hat{\beta}/dR=[1+\hat{\beta}(R)]\,\hat{\Sigma}_s[1+\hat{\beta}(R)]-2\|(\Sigma_i+\Sigma_k)\,\beta_{ik}(R)\|+1/R\,[1-\hat{\beta}(R)]. \quad (1)$$

The initial condition for Eq. (1) is

$$\beta_{ik}(0)=\delta_{ik}. \quad (2)$$

Previously [7], using the one-group problem for a semiinfinite medium as an example, it was shown that Eq. (1) leads to nearly correct results even with strong absorption, when the diffusion approximation is inefficient. In the other limiting case $R \to 0$, Eqs. (1) and (2) result in the solution

$$\beta_{ik}=\begin{cases} 1-2R(\Sigma_i-\Sigma_{sii}) & \text{for } k=i; \\ 2R\Sigma_{sik} & \text{for } k\neq i, \end{cases}$$

corresponding to the correct one. This allows one to count on the high accuracy of the equation obtained in the general case. The practical utilization of Eq. (1) requires investigations into the stability of its numerical solution.

An equation for the albedo exterior to the cylinder of the medium can also be obtained by similar means:

$$-\frac{d\hat{\beta}}{dR}=[1+\hat{\beta}]\,\hat{\Sigma}_s[1+\hat{\beta}]-2\|(\Sigma_i+\Sigma_k)\,\beta_{ik}\|-\frac{\hat{\beta}}{R}, \quad (3)$$

as well as the corresponding equation in spherical geometry [in this case, the 1/R on the right-hand sides of Eqs. (1) and (3) is replaced by 2/R].

Equations (1) and (3) can also be applied to nonuniform media, when Σ and Σ_s are functions of the radius. For the calculation of multilayer rods, methods are utilized, which are based on the calculation of the transmission and reflection probabilities for the individual layers.

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DISTRIBUTION OF FAST NEUTRONS IN IRON – WATER SHIELDING

D. L. Broder, K. K. Popkov,
V. G. Taroev, I. N. Trofimov,
and S. A. Tsvetkova

UDC 539.125.52:621.039.538

An attempt is made in this paper to investigate the spatial and energy distribution of fast-neutron fluxes in homogeneous iron–water shielding. Such distributions were obtained by multigroup calculations on the M-220 computer using a special program for the Carlson S_n method in the P_6S_{12} approximation [1]. The calculations were made for one-dimensional spherical geometry and the shielding composition was an infinite homogeneous iron–water mixture in which the radiation source – the core, 30-cm in radius, of a water-cooled, water-moderated reactor – was located.

The group constants needed for the calculations were prepared from the data in [2–4] and the energy intervals for the groups were taken from [2]. Calculations were made for iron–water mixtures containing 0, 10, 20, . . . , 90, 100 vol. % of water at a density $\gamma = 1$ g/cm^3.

The energy distributions for fast-neutron fluxes in iron compared with the results of [5, 6] and also the fluxes in iron–water mixtures containing 20 and 70 vol. % of iron are shown in Figs. 1–3.

The dependence of the relaxation length on the volume concentration of iron in the mixture is shown in Fig. 4 for neutrons of various energies.

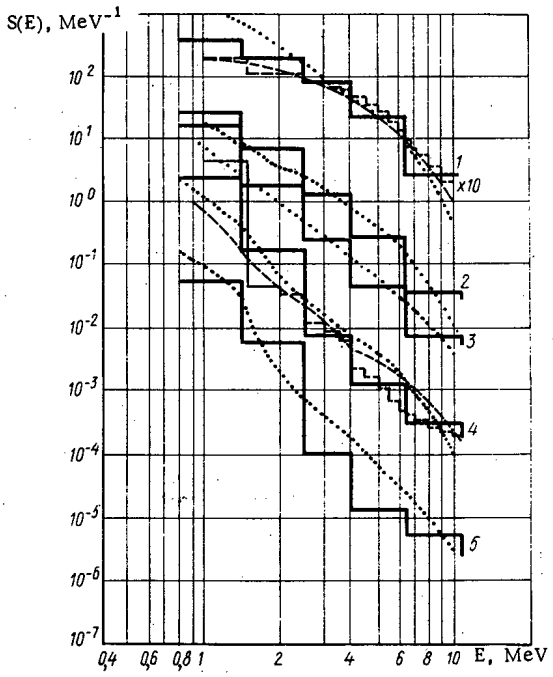

Fig. 1. Spatial and energy distributions of neutrons in iron. Histograms: ———) P_6S_{12} calculation; – – –) calculated with RASH program [6]; experiment [5]. Dotted line calculated by NYOBE program [6]; 1–5) iron thickness 0, 10, 20, 40, and 65 cm.

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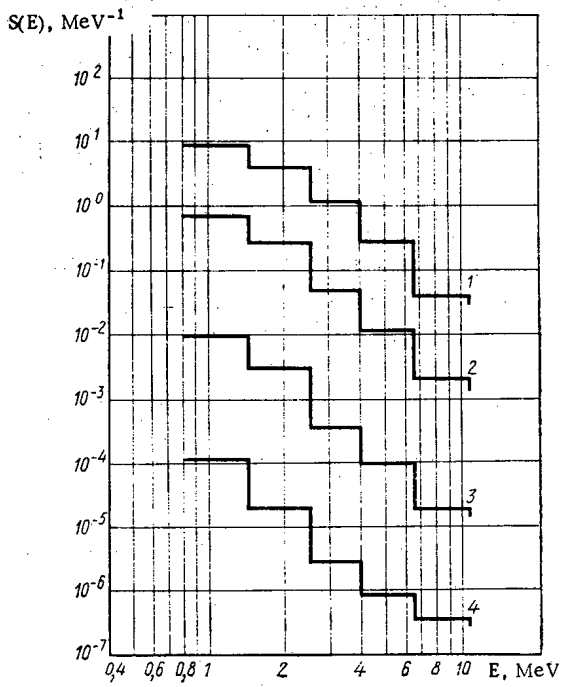

Fig. 2

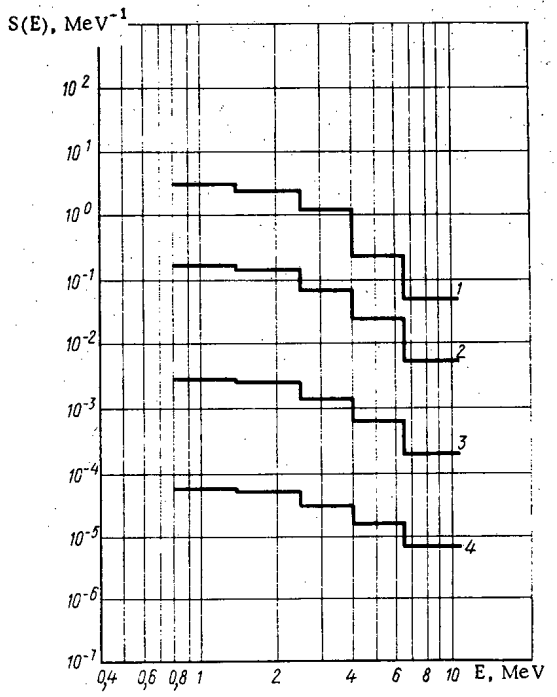

Fig. 3

Fig. 2. Energy distribution of neutrons in an iron–water mixture containing 70 vol. % of iron at various distances from the source: 1) 10; 2) 30; 3) 60; 4) 90 cm.

Fig. 3. Energy distribution of neutrons in an iron–water mixture containing 20 vol. % of iron at various distances from the source: 1) 10; 2) 30; 3) 60; 4) 90 cm.

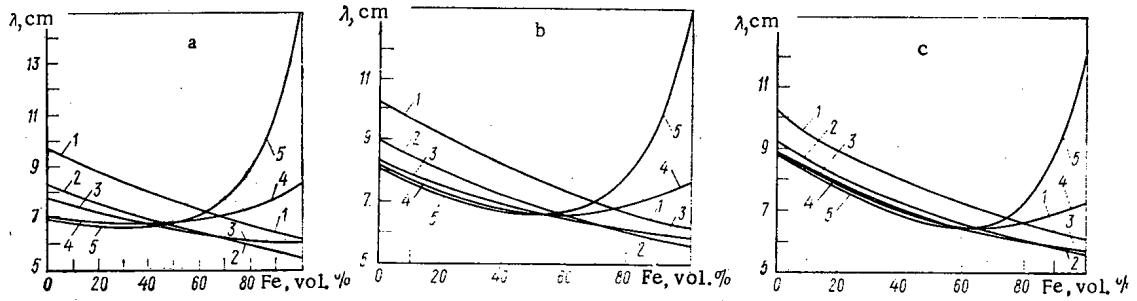

Fig. 4. Dependence of relaxation length for fast-neutron fluxes in iron–water shielding on the volume concentration of iron at source distances: a) 10-30; b) 30-60; c) 60-90 cm and for neutron energies (MeV): 1) 6.5-10.5; 2) 4.0-6.5; 3) 2.5-4.0; 4) 1.4-2.5; 5) 0.8-1.4.

TABLE 1. Relaxation Lengths for Neutron Fluxes (E > 1.4 MeV) in Iron–Water Homogeneous Shields at Various Source Distances

Iron content, vol. %	Distance, cm		
	10–30	30–60	60–90
0	7,7	8,5	9,1
10	7,45	8,05	8,55
20	7,2	7,6	8,05
30	7,0	7,3	7,6
40	6,9	7,0	7,2
50	6,8	6,8	6,8
60	6,8	6,65	6,5
70	6,85	6,65	6,45
80	7,0	6,75	6,55
90	7,3	7,0	6,9
100	7,8	7,5	7,3

Table 1 gives values of the relaxation length for a flux of neutrons with an energy E > 1.4 MeV in iron–water mixtures of varying composition; the values were obtained by an analysis of the results of the calculations. The table makes it clear that an insignificant reduction in relaxation length is observed as the source distance increases for iron–water mixtures containing more than 50 vol. % of iron. A similar effect was observed in an analysis of the penetration of fast neutrons from monoenergetic sources in heavy media [7].

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DOSE DISTRIBUTIONS IN PHANTOMS OUTSIDE 1-1000-GeV ACCELERATORS

N. V. Mokhov, E. L. Potemkin, and V. V. Frolov

UDC 621.039-78

We present here calculated results obtained from the FORTRAN program SHIPHA for the depth dose in phantoms produced by ionizing radiation produced in iron shielding of various thicknesses by primary protons with energies E_0 = 1-1000 GeV and unit flux density. In the SHIPHA program, calculation of the spectral and angular distributions of protons, neutrons, $\pi^{\pm}$-mesons, $K^{\pm}$-mesons, muons, and γ-rays outside the shield is performed in the subprogram HAMLET [1, 2] over the energy range 10^{-2}-10^{12} eV. Calculation of the dose distribution in a slab phantom 30-cm thick outside the shield is performed in the subprogram FANTOM [3].

The depth distribution for absorbed dose in the phantom is plotted in Fig. 1 for protons of energies 1, 10, 100, and 1000 GeV incident on an iron shield having a thickness H = 1500 g/cm^2. The distribution is exponential in nature because the dose is ~90% determined by neutrons with E ≤ 10 MeV. Calculations show that the addition of a concrete layer ~110 g/cm^2 thick to the iron shield reduces the dose by roughly a factor of five and leads to comparable contributions to the dose from low-energy neutrons and high-energy hadrons.

The energy dependence of the dose equivalent and the absorbed dose in a phantom outside an iron shield is shown in Fig. 2. There is a "critical thickness" of the shield at each energy such that the dose in the phantom for shield thicknesses less than the critical thickness is greater than without a shield.

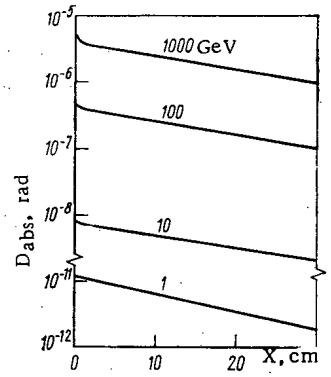

Fig. 1

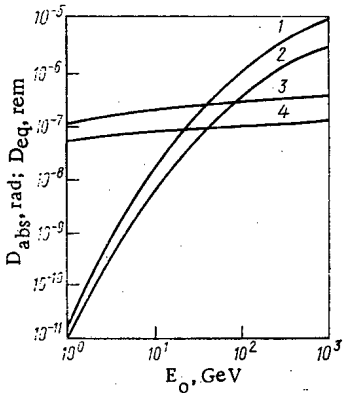

Fig. 2

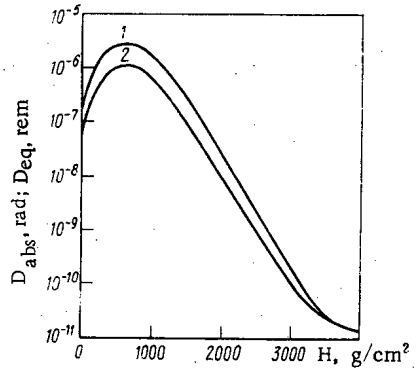

Fig. 3

Fig. 1. Depth distribution of absorbed dose in a phantom outside an iron shield for various primary proton energies.

Fig. 2. Dependence of dose equivalent (1) and absorbed dose (2) at a phantom depth x = 5 cm on the energy of protons incident on a shield of thickness H = 1500 g/cm^2; 3, 4) corresponding values without shielding.

Fig. 3. Dependence of dose equivalent (1) and absorbed dose (2) at a phantom depth x = 12 cm on thickness of an iron shield (E_0 = 50 GeV).

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Dependence of dose equivalent and absorbed dose on the thickness of a shield irradiated by protons with an energy $E_0 = 50$ GeV is shown in Fig. 3. Outside a shield of thickness $\gtrsim 3500$ g/cm^2, the dose is determined entirely by the muon component and the absorbed dose and dose equivalent are the same.

The program SHIPHA makes it possible to calculate dose distribution in phantoms of arbitrary shape outside heterogeneous shields of thicknesses up to 5000 g/cm^2. The program calculates the component contribution to absorbed dose and dose equivalent. The primary energies of broad, monodirectional beams of hadrons are $E_0 = 0.05$-1500 GeV. The reliability of the resultant data was established by comparison of the calculated results from the HAMLET and FANTOM programs with experimental data in the energy range 0.1-100 GeV. The disagreement did not exceed 10% for dose calculations in phantoms [3, 4] and 30-50% for calculation of the intranuclear cascade in the shield [1, 2]. The computing time required to obtain results with an accuracy of 5-10% is ~1-1.5 h on the BÉSM-6. The program is useful for the estimation of the radiation environment to be encountered by space vehicles and that around existing and planned accelerators.

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MEASUREMENT OF $\sigma_f(^{239}Pu)/\sigma_f(^{235}U)$ AND $\sigma_f(^{233}U)/\sigma_f(^{235}U)$ FISSION RATIOS AT 2 keV

V. G. Dukhsherstnov, V. L. Petrov, and V. M. Furmanov

UDC 539.1.083

Evaluations of the effect of uncertainties in nuclear data on the design parameters and economics of fast breeder reactors [1, 2] indicate the need for refinement of the energy dependence of the effective cross sections for basic reactor materials. This refers primarily to the fission and radiative capture cross sections in U and Pu isotopes, for which the measurements still fail to show sufficiently good agreement.

Useful information for the formulation of a recommended library of evaluated nuclear data can be obtained from measurements of the fission cross sections in U and Pu isotopes relative to the fission cross section of ^{235}U. In the neutron energy range 1-5 keV, however, there is an extremely limited set of experimental data from "direct" measurements of the $\sigma_f(^{239}Pu)/\sigma_f(^{235}U)$ [3] and $\sigma_f(^{233}U)/\sigma_f(^{235}U)$ [3, 4] ratios or from simultaneous measurements of $\sigma_f(^{239}Pu)$ and $\sigma_f(^{235}U)$ [5, 6]. Furthermore, the differences in the $\sigma_f(^{239}Pu)/\sigma_f(^{235}U)$ ratios in [3, 5, 6] are significantly greater than the errors cited by the authors.

This paper gives the results of "direct" measurements of the quantities $\sigma_f(^{239}Pu)/\sigma_f(^{235}U)$ and $\sigma_f(^{233}U)/\sigma_f(^{235}U)$ in the scandium neutron beam (2 keV) of the reactor at the Obninsk nuclear power station.

METHOD

Measurements of fission cross-section ratios were carried out in the "signal" spectrum of the scandium beam of the reactor at the Obninsk nuclear power station [7] by means of double fission chambers. Each chamber ($^{239}Pu/^{235}U$ and $^{233}U/^{235}U$) had five aluminum-foil electrodes ~0.1 mm thick on two of which a layer of fissile material was deposited on both sides in the form of a spot 14 × 9 mm in size. The total amount of fissile material in a single chamber was ~3 mg: 1 mg of ^{239}Pu dioxide (~99.8%) or of uranous–uranic oxides of ^{233}U (~99.9%) and 2 mg of uranous–uranic oxides of ^{235}U (~90%). The electrodes were mounted on plastic insulators with a spacing of 1.5 mm inside aluminum housings having a wall thickness of 1 mm and filled with argon to a pressure of ~1.5 atm. The chambers were used in the current-pulse mode [8]. Calibration measurements with the double fission chambers were made in the neutron beam from a crystal monochromator [9] yielding energies of 0.051-0.056 eV. The admixture of neutrons with energies 0.193-0.235 eV in the spectrum of the diffraction beam was estimated by the authors [9] to be 10% of the flux of neutrons in the main group with an average energy of 0.053 eV. The results

TABLE 1. Basic Experimental Characteristics

Parameter	Chamber $^{239}Pu/^{235}U$		Chamber $^{233}U/^{235}U$	
	monochromator beam	scandium beam	monochromator beam	scandium beam
Signal/background ratio	~ 33/48	~ 0,7/1,9	~ 60/53	~ 2,3/1,5
Signal counting rate, cts/sec	~ 38/17	~ 0,6/0,7	~ 30/13	~ 1,7/0,5
Ratio of chamber counts	2,245±0,011	0,791±0,012	2,355±0,028	3,460±0,052
Measured cross section ratio in "signal" spectrum of scandium beam	0,578±0,014		1,41±0,04	
Cross section ratio for energy range 1.5-2.3 keV (2-keV peak)	0,562±0,015		1,41±0,04	

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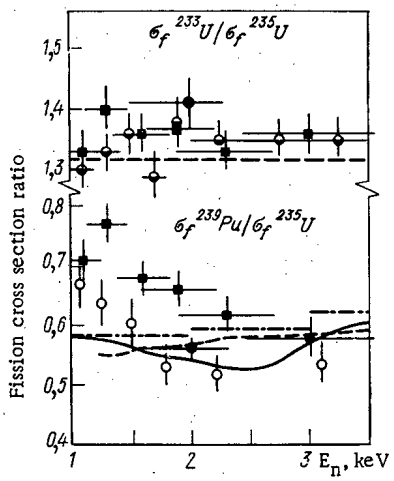

Fig. 1. Data for $\sigma_f(^{233}U)/\sigma_f(^{235}U)$ and $\sigma_f(^{239}Pu)/\sigma_f(^{235}U)$ in the neutron energy range 1-3.5 keV. ◒) [4]; ■) [3]; ○) [5]; ——) [17]; – · – · –) [18]; – – –) [16]; ●) present work.

in the neutron beam of the monochromator were normalized to values of the ratios $\sigma_f(^{239}Pu)/\sigma_f(^{235}U)$ and $\sigma_f(^{233}U)/\sigma_f(^{235}U)$ which were 1.65 ± 0.03 and 0.96 ± 0.02, respectively, and which were calculated from evaluated data [10, 11] and from experiment [12-14].

RESULTS

Table 1 gives the main results of the experiment in the neutron beam from the crystal monochromator and of the experiment in the scandium beam for a reactor thermal power of ~11 MW at the Obninsk nuclear power station. To obtain the fission cross-section ratios for the energy range 1.5-2.3 keV, a correction was introduced for the nonmonoenergetic nature of the spectrum of the "signal" scandium neutron beam in accordance with data of the BNAB system of constants [15]. The mean-square error given in Table 1 includes the statistical error of the measurements and the uncertainties in the calculated normalizing constants and corrections.

Figure 1 shows data for $\sigma_f(^{233}U)/\sigma_f(^{235}U)$ and $\sigma_f(^{239}Pu)/\sigma_f(^{235}U)$ in the neutron energy range 1-3.5 keV. It is clear that the present data for $\sigma_f(^{233}U)/\sigma_f(^{235}U)$ agrees within the limits of error with the results of spectrometric "direct" measurements with double fission chambers for the slowing-down time in lead [3] and from a high-altitude nuclear explosion [4]. The data obtained for $\sigma_f(^{239}Pu)/\sigma_f(^{235}U)$ agrees satisfactorily with experimental results [5] where $\sigma_f(^{239}Pu)$ and $\sigma_f(^{235}U)$ were measured "simultaneously" in a lead cube and also with previous evaluations [16, 17]. The values of the ratio $\sigma_f(^{239}Pu)/\sigma_f(^{235}U)$ calculated from the data in [5] and [6] for the spectrum of the 2-keV peak of the scandium beam are 0.556 and 0.713. The first result is in good agreement with the data in Table 1.

In conclusion, the authors thank V. S. Golovkin for providing the opportunity to carry out the calibration measurements in the beam of the crystal monochromator and Yu. A. Kazanskii and S. P. Belov for consideration of this work.

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FAST NEUTRON SPECTROMETRY OF (α, n) REACTIONS USING A DEUTERATED SCINTILLATOR

E. M. Burymov, S. P. Korsunova, and N. N. Spendiarov

UDC 539.125.5.164.07

A deuterated scintillator (octadeuteronaphthalene) was used in the work of Burymov et al. [1] for neutron spectrometry in an investigation of inelastic scattering. Neutron spectra from (α, n) reactions in the nuclei ^{9}Be, ^{10}B, and ^{11}B were measured by the same method in the present work.

The experiment was performed at the 120-cm cyclotron of the Research Institute for Nuclear Physics, Moscow State University with 23- and 25-MeV α-particles. Target thicknesses were 1-2 mg/cm^2. A crystal of octadeuteronaphthalene 30 mm in diameter and 20 mm high at 23 cm from a target was used in conjunction with an FÉU-29 photomultiplier for neutron spectrometry. The spectrometer was calibrated with 14.1-MeV neutrons.

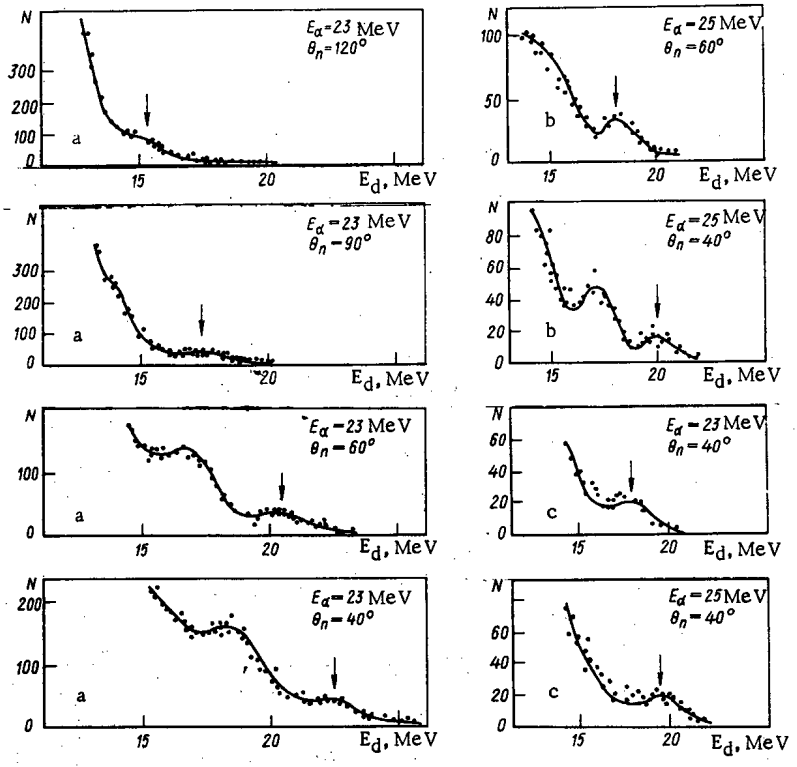

Fig. 1. Pulse-height spectra of scintillator recoil deuterons from elastic scattering of neutrons produced in the reactions: a) ^{9}Be(α, n)^{12}C; b) ^{10}B(α, n)^{13}N; c) ^{11}B(α, n)^{14}N [E_d) energy of recoil deuterons; N) number of counts per analyzer channel; arrows indicate the position of the peak in the recoil deuteron spectrum corresponding to the energy of emitted neutrons when the final nucleus remains in the ground state; $E_n \approx (9/8)E_d$].

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Neutron spectra are shown in Fig. 1 for several angles of emission. The time of measurement for each spectrum was ~10 min at a beam current of ~0.1 μA.

Since a single-crystal arrangement was used, great difficulties arose because of the γ-background. It was particularly great for signal pulses smaller than pulses corresponding to neutrons with energies of 18-20 MeV (this energy decreases somewhat as the angle increases). In those cases where the neutron energy exceeds this value, peaks are observed in the experimental spectra corresponding to the ground state in the final nucleus (see Fig. 1) and to the first excited state in particular cases. This is observed in all the reactions studied at angles less than 90°. In the $^{12}C(\alpha, n)^{15}O$ reaction, the maximum neutron energy is 13.5 MeV for E_α = 25 MeV and therefore peaks cannot be observed.

Thus we have demonstrated the possibility of using a single-crystal deuterated neutron spectrometer for studying (α, n) reactions in cases where the neutron energy is sufficiently high (greater than 18-20 MeV).

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INITIAL STATIC FOCUSING IN SMALL LINEAR TRAVELING-WAVE ACCELERATORS

A. D. Vlasov

UDC 621.384.64.01

In a traveling-wave linear proton accelerator, an effect occurs which is similar to static electric focusing in cyclotrons [1] and in linear accelerators with drift tubes [2, 3]. A momentum directed towards the axis acts on a particle upon entrance into the accelerating wave and a momentum oppositely directed upon leaving the wave. In a low-energy accelerator for acceleration of protons from 50-100 keV to several MeV, the focusing action of the initial momentum can be sufficient to allow one to get along without special focusing devices.

The longitudinal electric component of the accelerating wave has the form

$$E_z = E_M \cos\left(\omega \int \frac{dz}{v} - \omega t\right) = E_M \cos\varphi, \tag{1}$$

where z, φ, and v are the longitudinal coordinate, phase, and velocity of a particle; t is time; $\omega = 2\pi c/\lambda$ (here, λ is wavelength and c is the velocity of light).

We assume the wave amplitude E_M and the equilibrium phase φ_p are constant along the accelerator, and we confine our considerations to the equilibrium particle. The longitudinal motion of this particle occurs at a constant acceleration $eE_M \cos\varphi_p/m_0$ and is described by the equations

$$z = \frac{eE_M \cos\varphi_p}{2m_0} t^2 = \frac{W_0\beta^2}{2eE_M \cos\varphi_p}.$$

Here e, m_0, and W_0 are the change, mass, and rest energy of the particle; $\beta = v_p/c$; v_p is the equilibrium velocity. If, for example, $E_M = 3$ Mv/m, $\varphi_p = 30°$ and the protons are accelerated from 100 keV ($\beta_i = 0.0146$), $z_i = 38.5$ mm. For a final proton energy of 750 keV ($\beta_f = 0.0400$) and 5 MeV ($\beta_f = 0.1028$), the accelerator length, $z_f - z_i$, is 0.25 and 1.86 m, respectively. If $\lambda = 2$ m, the duration of particle acceleration, $c(t_f - t_i)/\lambda$ will be 4.6 and 16 cycles of the hf field. For $\lambda = 24$ m, we obtain only 0.4 and 1.3 cycles, respectively.

The number of phase oscillations is given by

$$n_\Phi = \sqrt{\frac{2W_0 \sin\varphi_p}{\pi e E_M \lambda \cos^2\varphi_p}} (\sqrt{\beta_f} - \sqrt{\beta_i}).$$

For the parameters given above and $\lambda = 2$ m, it becomes 0.5 and 1.62, and for $\lambda = 24$ m we obtain the values 0.2 and 0.5.

The transverse momentum acting upon a particle when it enters or leaves the field of the wave (Eq. (1)) is calculated by integration of the equation of transverse motion

$$m_0 \frac{d^2x}{dt^2} = W_0\beta^2 \frac{d^2x}{dz^2} = -\frac{ex}{2}\frac{\partial E_z}{\partial z}.$$

For the equilibrium particle, these momenta are

$$\Delta\left(\frac{dx}{dz}\right) = -\frac{ex}{2W_0\beta^2}\Delta E_z = \mp\frac{eE_M \cos\varphi_p}{2W_0\beta^2} x = \mp\frac{x}{4z},$$

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and their effect is given by the matrices

$$\begin{pmatrix} 1 & 0 \\ \xi_{i,f} & 1 \end{pmatrix}, \quad \xi_{i,f} = \mp \frac{1}{4z_{i,f}}.$$

The transverse motion of the equilibrium particle in the field (1) is described by

$$m_0 \frac{d^2x}{dt^2} = e(E_{1x} + E_{2x}) = \frac{\varepsilon \pi e E_M \sin \varphi_p}{\lambda} \frac{x}{\beta}, \tag{2}$$

where

$$E_{1x} = \frac{\pi E_M \sin \varphi_p}{\lambda} \frac{x}{\beta}, \quad E_{2x} = \frac{60\ \Omega \cdot I}{R^2} \frac{x}{\beta} -$$

are the transverse components of the accelerating wave and the intrinsic Coulomb field of the accelerated beam (I is beam current; R is beam radius) is

$$\varepsilon = \frac{E_{1x} + E_{2x}}{E_{1x}} = 1 + \frac{60\ \Omega\ \lambda I}{\pi E_M R^2 \sin \varphi_p}.$$

We obtain $\varepsilon \leq 1.1$ for $E_M = 3$ MV/m, $\varphi_p = 30°$, $\lambda = 24$ m, and $I/R^2 \leq 33$ mA/cm^2. The expression for E_{2x} corresponds to a cylindrical beam. The radius of the beam varies and the beam breaks up into bunches in proportion to acceleration. Both these factors lead to a variation in ε. For simplicity, we assume below $\varepsilon = \varepsilon_{max} = \text{const}$.

Expressing β through t, we bring Eq. (2) into the form

$$\frac{d^2x}{dt^2} = \frac{px}{4t}, \quad p = 2\varepsilon\omega \operatorname{tg} \varphi_p.$$

The solution of this equation and its derivative are given by

$$x = A\tau I_1(\tau) + B\tau K_1(\tau),$$
$$x' = \frac{dx}{d\tau}\frac{d\tau}{dt}\frac{dt}{dz} = \frac{\tau^2}{4z}[AI_0(\tau) - BK_0(\tau)],$$

where $\tau = \sqrt{pt}$. Setting $\tau = \tau_i = \sqrt{pt_i}$ in these equations, we determine the constants of integration A and B through x_0 and $x_0' = (dx/dz)_0$, and obtain a matrix equation connecting the instantaneous values x and x' with the initial values,

$$\begin{pmatrix} x \\ x' \end{pmatrix} = \begin{pmatrix} b_{11} & b_{12} \\ b_{21} & b_{22} \end{pmatrix} \begin{pmatrix} x_0 \\ x_0' \end{pmatrix}.$$

Here,

$$b_{11} = \tau[K_0(\tau_i) I_1(\tau) + I_0(\tau_i) K_1(\tau);$$
$$b_{12} = \frac{4z_i\tau}{\tau_i}[K_1(\tau_i) I_1(\tau) - I_1(\tau_i) K_1(\tau)];$$
$$b_{21} = \frac{\tau_i^2\beta_i}{\beta}[K_0(\tau_i) I_0(\tau) - I_0(\tau_i) K_0(\tau)];$$
$$b_{22} = \frac{\tau_i\beta_i}{\beta}[K_1(\tau_i) I_0(\tau) + I_1(\tau_i) K_0(\tau)].$$

The matrix determinant $b_{11}b_{22} - b_{12}b_{21} = \beta_i/\beta$ differs from unity and decreases like β^{-1}. The variation of x and x' over the length of the accelerator including the initial and final momenta is given by the matrix product

$$\begin{pmatrix} x_f \\ x_f' \end{pmatrix} = \begin{pmatrix} 1 & 0 \\ \xi_f & 1 \end{pmatrix} \begin{pmatrix} b_{11} & b_{12} \\ b_{21} & b_{22} \end{pmatrix} \begin{pmatrix} 1 & 0 \\ \xi_i & 1 \end{pmatrix} \begin{pmatrix} x_i \\ x_i' \end{pmatrix} = \begin{pmatrix} a_{11} & a_{12} \\ a_{21} & a_{22} \end{pmatrix} \begin{pmatrix} x_i \\ x_i' \end{pmatrix}.$$

Multiplying the matrices, we find

$$a_{11} = \tau_f \left\{ \left[K_0(\tau_i) - \frac{K_1(\tau_i)}{\tau_i} \right] I_1(\tau_f) + \left[I_0(\tau_i) + \frac{I_1(\tau_i)}{\tau_i} \right] K_1(\tau_f) \right\};$$
$$a_{12} = \frac{4z_i\tau_f}{\tau_i}[K_1(\tau_i) I_1(\tau_f) - I_1(\tau_i) K_1(\tau_f)];$$

$$a_{21}=\frac{\tau_i^2\beta_i}{\beta_f}\left\{\left[K_0(\tau_i)-\frac{K_1(\tau i)}{\tau_i}\right]\left[I_0(\tau_f)+\frac{I_1(\tau_f)}{\tau_f}\right]-\left[I_0(\tau_i)+\frac{I_1(\tau i)}{\tau_i}\right]\left[K_0(\tau_f)-\frac{K_1(\tau_f)}{\tau_f}\right]\right\};$$

$$a_{22}=\frac{\tau_i\beta_i}{\beta_f}\left\{K_1(\tau_i)\left[I_0(\tau_f)+\frac{I_1(\tau_f)}{\tau_f}\right]+I_1(\tau_i)\left[K_0(\tau_f)-\frac{K_1(\tau f)}{\tau_f}\right]\right\}.$$

The matrix elements a_{ij} depend on the four quantities τ_i, β_i, β_f, and z_i, with the last depending mainly on E_M and β_i since $\cos\varphi_p \approx 1$. Thus for given E_M, β_i, and β_f the matrix elements a_{ij} are determined by the values of τ_i or, alternatively, by the quantity

$$\frac{E_M\lambda}{\beta_i\varepsilon}\cdot\frac{\cos^2\varphi_p}{\sin\varphi_p}=\frac{4\pi W_0}{e\tau_i^2}=T_i. \tag{3}$$

The optimal choice of τ_i depends on the parameters of the injected beam and on the requirements imposed on the beam at the accelerator exit. In particular, one can select τ_i from the condition

$$K_0(\tau_i)-\frac{K_1(\tau_i)}{\tau_i}=0.$$

Then, neglecting terms containing $K_0(\tau_f)$ and $K_1(\tau_f)$ in the expressions for a_{12} and a_{22}, we obtain

$$a_{11}=\left[I_0(\tau_i)+\frac{I_1(\tau i)}{\tau_i}\right]\tau_f K_1(\tau_f);$$

$$a_{12}=\frac{4z_i}{\tau_i}K_1(\tau_i)\tau_f I_1(\tau_f);$$

$$a_{21}=-\frac{\tau_i^2\beta_i}{4z_i\beta_f}\left[I_0(\tau_i)+\frac{I_1(\tau i)}{\tau_i}\right]\left[K_0(\tau_f)-\frac{K_1(\tau f)}{\tau_f}\right];$$

$$a_{22}=\frac{\tau_i\beta_i}{\beta_f}K_1(\tau_i)\left[I_0(\tau_f)+\frac{I_1(\tau_f)}{\tau_f}\right].$$

The root of Eq. (4) is $\tau_i = 1.33$, to which $T_i = 6630$ MV corresponds. A wavelength $\lambda = 23.6$ m is required according to Eq. (3) when $E_M = 3$ MV/m, $\varepsilon = 1.1$, $\varphi_p = 30°$, and the initial proton energy is 100 keV. If the protons are accelerated to 750 keV and 5 MeV, we, respectively, obtain $\tau_f = 2.2$ and 3.5 and

$$a_{11}=0.50;\ \ 0.16,\ \ a_{12}=0.174;\ \ 0.925\ \text{m},$$
$$a_{21}=-0.36;\ \ -0.044\ \text{m}^{-1},\ \ a_{22}=0.61;\ \ 0.64.$$

These elements a_{ij} are not too large and therefore the transverse deflections and trajectory slopes of the particles at the accelerator exit will have acceptable values. If $\lambda = 2$ m, we obtain $\varphi_p = 3°14'$ for the same E_M, ε, and β_i. The values of $\tau_{i,f}$ and a_{ij} obtained in this case are not much different from those obtained above. Thus, although the choice $\tau_i = 1.33$ may differ from the optimal choice, the initial static focusing is still sufficiently effective.

Having selected the quantity τ_i, it is necessary to choose an optimal wavelength and equilibrium phase, which are related through Eq. (3). To do this, it is necessary to know the spreads in exit energy, deflection and trajectory slope of nonequilibrium particles corresponding to various λ and φ_p. The motion of nonequilibrium particles can only be calculated by numerical methods on a computer.

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DEPENDENCE OF THE FISSION-FRAGMENT SPUTTERING RATIO FOR THIN LAYERS OF A SUBSTANCE ON THE MEAN ENERGY OF THE FRAGMENTS

B. M. Aleksandrov, N. V. Babadzhanyants, I. A. Baranov, A. S. Krivokhatskii, L. M. Krizhanskii, and V. V. Obnorskii

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In [1], it was shown qualitatively that the sputtering ratio K for thin layers of ^{241}Am oxide is larger, the higher the mean energy of the fission fragments. We assume that the difference, obtained in [2-5], in the measurements on thin films of UO_2 ($K \approx 10^4$ atoms/fragment), on thick, coarse-grained baked disks of UO_2 ($K \approx 9$ atoms/fragment) and on metallic disks of plutonium and uranium ($K \approx 10^3$ atoms/fragment), respectively, can be partly associated with the difference in the energy spectra of the fission fragments, since the latter originated inside layers of different thickness and crossed the sputtered surface at different angles. The problem in this paper is the quantitative determination of the dependence of the sputtering ratio for thin layers of a substance on the mean energy of the fragments. Studies of a dependence of this kind allow one to ascertain those interactions of the fission fragments with the atoms of the substance (elastic or inelastic) which cause the sputtering, to verify the assumption expressed concerning the source of the spread in the experimental data on K in [2-5], and also have a definite practical value.

In this paper, we investigated the sputtering of fine-grained layers, prepared from ^{238}Pu by electrolysis of an aqueous solution and representing hydrated plutonium oxide, desiccated in air at room temperature. A thin layer of ^{252}Cf served as a source of the fragments. The plutonium layer undergoing sputtering and the californium layer were mounted parallel to each other, 7 mm apart, in a holder and placed in a special vacuum chamber. A rotating disk was placed between the layers. Ten thin nickel films, transparent to the fission fragments, and two thick foils, nontransparent to the fragments, were placed in openings along the rim of the disk. The fission fragments from the californium layer passed through the two protective nickel films, then the nickel-film collector in the disk, after which they fell almost perpendicularly on the plutonium layer, sputtering it. The sputtered atoms of plutonium were collected by the nickel collector-film located 2 mm from the plutonium layer. After 2 h of exposure, the disk was reversed (without any change in the high vacuum) and another collector with a different thickness of nickel film was placed between the layers of plutonium and californium, whereas the first collector-film was located above a surface-barrier, silicon α-counter. The α-spectrum of the collector was measured with the aid of an AI-256 multichannel amplitude analyzer with a precision-amplitude generator. The number of plutonium atoms collected by the collector was determined from the α-particle counting rate and the presence or absence of contamination from the ^{252}Cf was monitored in the spectrum. At the same time, a correction for the self-sputtering of the plutonium layer due to α-decay was taken into account. After the first series of measurements (sputtering in all of the 12 collectors and production of α-spectra with each one), two more series of measurements were conducted without impairment of the high vacuum, after which the layer being sputtered was replaced. In all, four plutonium layers with different sputtering ratios (500, 185, 130, and 70 atoms/fragment) for a maximum mean energy of the fragments ~87 MeV were subjected to sputtering. With a decrease in the mean energy to 23-27 MeV, the ratios for these layers were reduced to 100, 33, 40, and 20 atoms/fragment, respectively. The dependence of the sputtering ratio on the mean energy of the fragments for a layer with a maximum sputtering ratio of 500 atoms/fragment is shown in Fig. 1. The errors indicated represent the variances obtained as a result of three series of measurements. The curves for other plutonium layers have the same character; however, one should mention a certain

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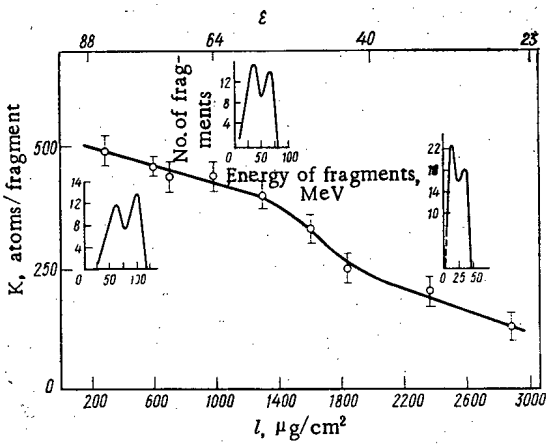

Fig. 1

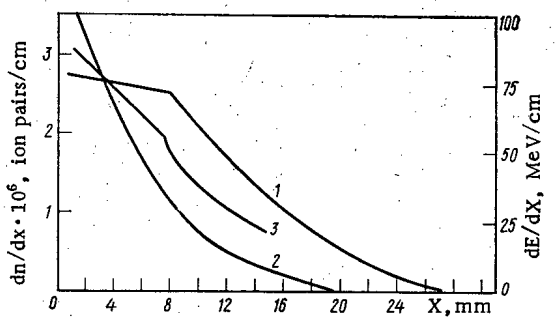

Fig. 2

Fig. 1. Dependence of the sputtering ratio of a thin layer of hydrated plutonium oxide on the mean energy of the fission fragments for ^{252}Cf (l is the thickness of the nickel films).

Fig. 2. Change in specific ionization dn/dx and specific energy losses dE/dx along a track x of ^{244}Cm fission fragments [6]: 1, 2) for light and heavy fragments, respectively; 3) averaged values of curves 1 and 2.

tendency towards a reduction in the slope of the curves for plutonium layers with a lower sputtering ratio. The film thicknesses range from 300-2800 $\mu g/cm^2$, and the energy spectra of the fragments after passing through these films were determined with a semiconductor spectrometer before the start of the experiment. Three energy spectra for the fission fragments, which characterize the energy distribution of the fragments after passage through 300, 1600, and 2800 $\mu g/cm^2$ nickel films, are given in Fig. 1. With a decrease in the mean energy of the fragments from ~87 to ~25 MeV, the sputtering ratio is reduced by a factor of 3.5-6 as a function of the quality of the layer undergoing sputtering. Thus, for fine-grained layers of a nonconducting substance, the sputtering ratio depends substantially on the energy of the fragments, if this energy >20 MeV. One can assume as completely valid that the difference in orders of magnitude in the sputtering ratios of thin, fine-grained films [2] and thick, metallic disks [4, 5] could be associated, completely or to a significant degree, with the difference in the mean energy of the fission fragments. On the other hand, the rate of change of the sputtering ratio with a change in the energy attained in this experiment can probably not explain the large difference in the sputtering ratios of the fine-grained layers and films [1, 2], and the coarse-grained disks [3]. This apparently says something about the different mechanisms for the sputtering of coarse-grained and fine-grained samples. From the data obtained, it also follows that the sputtering of coarse-grained, nonconducting layers by fission fragments with energies from ~20 up to ~100 MeV occurs mainly because of inelastic interactions of the fission fragments with the atoms of the substance. This is seen from the character of the curve in Fig. 1, which agrees qualitatively with the experimentally obtained curve for dE/dx as a function of x (Fig. 2, [6]). By investigating the question of the possible effect of the size of the grains on the value of the sputtering ratio, one can attempt to utilize the same reasons which were applied in [7, 8] for explaining the formation of tracks from the fission fragments in different substances. Thus, one can assume that reflection of phonon waves, low-energy electrons, and excitons, which arise during the passage of a fragment through a grain, occurs at the boundary of the grain. These phenomena must inhibit the dissipation of energy, evolved within the grain, beyond the limits of its boundary. At the same time, the amount of energy evolved in the grain by the fission fragments and the degree of dissipation of this energy beyond the extent of the grain will depend on its size. In fact, tracks from the displaced material were observed only in fine-grained films; the maximum sputtering ratio (10^4 atoms/fragment) was measured for fine-grained layers. Hardly noticeable tracks of a discontinuous character were observed in coarse-grained, insulating materials; the sputtering ratio for such materials proved to be a minimum (5-10 atoms/fragment). However, there is also a discrepancy. Thus, no tracks are observed in metals, although the sputtering ratio for metals proved to be significant (10^3 atoms/fragment). It is of interest to investigate the dependence of the sputtering ratio on the mean energy of the fission fragments in an experiment similar to the one described here for metals and a coarse-grained, nonconducting material.

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SILVER CHLORIDE TRACK DETECTORS

N. P. Kocherov, N. R. Novikova, and N. A. Perfilov

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Since 1960, work has been done in France and FRG on the recording of charged-particle tracks in flat single crystals of AgCl. The principle of operation of such detectors is based on the process of photolysis in silver halide crystals. When a charged particle passes through the crystal, a chain of defects is formed along its trajectory. If the crystals exposed to particles are thereafter subjected to photolysis under the action of actinic light ($\lambda \approx 410$ nm), metallic silver will be produced at the defects caused by the particles, and the resulting tracks will be visible under the microscope. The tracks found in crystals are very similar in appearance to tracks in emulsions.

In 1972, Schopper [1] established that the latent image of the particles in AgCl crystals is stable when the crystals are illuminated with yellow light at the instant of exposure, whereas when there is no illumination at the instant of particle action, the image regresses rapidly. This makes it possible to turn the sensitivity on and off whenever necessary for the experimenter.

In the present paper we report on our experience in the preparation of such detectors and on their properties.

The original AgCl salt was recrystallized by dissolving in ammonia and subsequently precipitating it by means of HCl. The resulting reagent was spectroscopically pure with respect to metals of valence 2 or higher.* The single crystals were grown by letting the solution flow between quartz and glass plates [2] and had dimensions of $15 \times 10 \times 0.15$ mm. In principle the area of the crystals can easily be increased by a factor of 2-2.5. Crystals grown from AgCl with no additives do not record even fission fragments. Crystals alloyed with 0.1% $CdCl_2$ by weight record the α-particles from the decay of ^{252}Cf but are insensitive to protons. Crystals alloyed with 0.5% $CdCl_2$ record protons up to an energy of 4 MeV.† The rate of regression of the latent image when the AgCl is exposed to particles without being illuminated depends on the specific energy losses of the recorded particles: the images persist for minutes in the case of protons, for hours in the case of α-particles, and for days in the case of ^{252}Cf fission fragments. If the crystal is illuminated with yellow light at the same time as it is exposed to particles, the latent image is stable for several months [1].

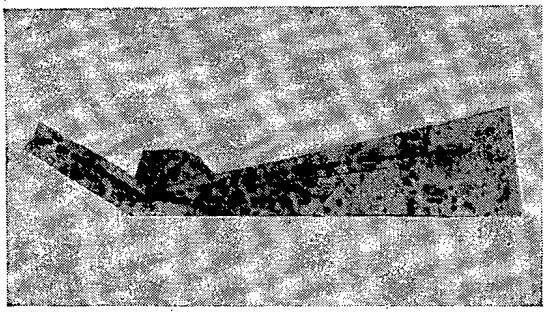

Fig. 1. Photomicrograph of tracks in an AgCl crystal irradiated with 660-MeV protons.

AgCl detectors are transparent crystals; they can be stored for long periods and used under conditions of illumination with long waves measuring 440 nm or more. An image appears when the crystal is illuminated with light having a wavelength of $\lambda \approx 410$ nm (for example, 5-8 min of exposure to direct sunlight filtered through ZhS-10 and PS-15 filters). Fixing the image is not necessary. In external form (see Fig. 1) the particle

*The recrystallization of the reagent was carried out by P. I. Chaikin and O. V. Kesarev at the Scientific-Research Institute of the Jewel Industry.

†Detectors capable of recording protons with energies up to 14 MeV are being produced today.

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tracks in the crystals exhibit almost no differences from tracks obtained with nuclear photographic emulsions [3].

Thus, the use of silver chloride detectors in the case of prolonged exposure (of the order of weeks or longer) makes it possible to turn the detector sensitivity on and off whenever necessary. The particles recorded in the crystal will then be those which passed during the time of exposure to yellow light plus the time of regression. In the case of short exposures (seconds) without illumination, the crystals can be developed after various intervals of time following the exposure, which makes it possible to obtain a series of detectors with different recording thresholds, since the regression time of the latent image depends on the specific energy losses of the recorded particles. It is possible to conduct biological experiments with objects mounted on the surface of the detector: development using light does not destroy the objects and makes it possible to observe them together with the tracks of the particles impinging on them.

Silver chloride detectors make it possible to study nuclear reactions on silver and chlorine without any background from reactions involving light nuclei, which interfere seriously with the use of nuclear emulsions; they can also serve as fast-neutron dosimeters, using the (n, p) and (n, α) reactions on silver and chlorine [4].

Since these detectors are simple to produce and have convenient properties, it may be supposed that they will come into wide use in the near future.

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CHANGE IN THE OPTICAL DENSITY OF PMMA UNDER THE ACTION OF DEUTERONS WITH ENERGIES 4-150 keV

S. P. Kapchigashev, V. P. Kovalev,
É. S. Barkhatov, and V. A. Sokolov

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The measurement of the difference in the optical density of polymethyl methacrylate (PMMA) before and after irradiation is a widespread method of determination of the absorbed energy of high-intensity ionizing radiation. However, the principles of the variation of the optical density (ΔS) as a function of the characteristics of the radiation (LET, type of particles, etc.) have thus far been insufficiently studied [1-4].

In this work we investigated the change in the optical density of PMMA under the action of deuterons with an energy of 4-150 keV. The method of irradiation was described earlier [5]. Plates of PMMA 1 mm thick were irradiated with deuterons of various energies and the change in the optical density observed on an USV-1 spectrophotometer at the wavelength 300 nm. A comparison of the dependence of the effectiveness of radiation chemical changes on the energy was conducted with an integral flux equal to $4.3 \cdot 10^{14}$ deuterons/cm^2. It was preliminarily shown that in this region of fluxes ΔS is a linear function of the dose.

The ranges of protons of the investigated energies are hundredths of a micron to units of microns, which is substantially less than the thickness of the PMMA plates. Therefore, the absorbed energy is proportional to the total energy of the particles E. The effectiveness of optical changes in η can be determined as $\eta = \Delta S/E$. The values of η for various deuteron energies are presented in Table 1.

From Table 1 it is evident that the effectiveness of the optical changes decreases substantially with increasing deuteron energy, which does not agree with the conclusions drawn in [3] with respect to low-energy particles. The increase in the effectiveness of radiation changes with decreasing dose of deuterons may be associated with a relative increase in the contribution of elastic nuclear collisions to the "exchange" of energy. Here it must be assumed that elastic processes are significantly (more than three times) more effective in radiation chemical disruptions of polymethyl methacrylate than processes of ionization and excitation. This phenomenon must be taken into consideration in the practical use of polymer materials in the dosimetry of nuclear radiation.

TABLE 1. Dependence of the Effectiveness of Optical Changes in PMMA on the Deuteron Energy

E, keV	η, cm^2/MeV	E, keV	η, cm^2/MeV
4	11±0,5	50	3,4±0,2
10	6,6±1,0	100	3,9±0,2
25	3,7±0,6	150	3,6

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PROVISIONAL INTERNATIONAL WORKTEAM

P. M. Tyukhtin

On February 23, 1973, authorized representatives of seven countries with membership in the Council for Economic Mutual Aid [COMECON] (viz. Bulgaria, Hungary, GDR, Poland, Rumania, USSR, Czechoslovakia) signed an Agreement on setting up a Provisional International Scientific-Research Workteam to carry out reactor physics research on a VVÉR type critical assembly.

The purpose of this research will be to devise and work out exact methods for design calculations relevant to VVÉR type reactors, and to elaborate straightforward and reliable techniques for handling variant calculations in the process of designing and operating reactors, and also to expedite research on perfecting methods in measurements and in the design of measuring equipment.

The experimental research facilities available to the staff of this provisional international workteam will be a VVÉR type critical assembly (3R-6 assembly) located at the Central Physics Research Institute of the Hungarian Academy of Sciences. This critical assembly was started up in November, 1972. That was followed by a period during which technical staff personnel were trained on the job, and in February, 1973, the scientific staff of the provisional international research team got started on the experimental research program.

A Scientific Council staffed by no more than three representatives from each country has been set up to provide scientific guidance for the activities of the research team, but each nation signatory to the Agreement has one vote. The Scientific Council schedules its sessions to gear into the work plan of the Council, but no more often than three times a year. These sessions are held in Budapest. Four sessions have been held to date: the first in February, 1972, the second in December, 1972, the third in June, 1973, and the fourth in March, 1974; a fifth session is scheduled for January, 1975.

The head of the research team is the director of the Central Physics Research Institute, L. Páal.

A detailed analysis of results achieved and of scientific topics being tackled is presented at meetings of specialists organized every eight months. To date two such conferences have been held, one in Czechoslovakia on the statics of VVÉR type reactors (October 31 through November 3, 1972), the second in Poland and dealing with reactor neutron statics (October 2-5, 1973). A third conference on VVÉR physics is planned for December 2-7, 1974, in GDR.

Three subteams have been set up to expedite the solution of some concrete problems. In the period intervening between sessions of the Scientific Council, these subteams and the research personnel working on the 3R-6 assembly are responsible for organizing their own work.

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COLLABORATION DAYBOOK

A conference of specialists of COMECON member-nations on standardization of radiation equipment and radiation protection equipment was held in Mamaia (Rumania) September 3-6, 1974.

Drafts of COMECON standards applicable to modular lead shielding units were discussed and agreed upon, as well as draft recommendations on standardization referable to shielded dry boxes (types, basic parameters and dimensions), shielded gloveboxes (types, basic parameters and dimensions); γ-therapeutic equipment (basic parameters and general technical specifications); complex mobile laboratories for nondestructive testing and quality control work (general technical requirements); systems and devices for alarm annunciation and flaw detection signalling in radiation flaw detection work.

Technical reports on recommendations referable to shielded cabinets or safes, packing sets for shipping radioactive materials safely, were heard and discussed. Draft recommendations on proper classification of laboratories for handling sources of ionizing radiations, and general technical requirements applicable, were discussed; a draft glossary of terms and definitions in the field of radiation protection work was discussed and agreed upon, as were a work plan on standardization of radiation equipment and radiation protection equipment for the year 1975, and a draft plan on collaboration over the 1976-1980 period.

* * *

The 14th session of the panel on nuclear electric power generating stations of the COMECON Permanent Commission on Electric Power was held September 17-21, 1974, at Herceg Novi (Yugoslavia). Urgent problems in the development of nuclear power in COMECON member-nations and in Yugoslavia came under discussion at the session. Materials on "Equipment type related engineering safety validations for installation and operation of nuclear power station equipment at the commissioning stage" and on "Standard types of operating report forms for nuclear power stations with VVÉR-440 power generating units" were approved. These documents enable those countries to go ahead with preparations on a high technical level to put nuclear power stations into service and to raise the safety levels of those power stations. Close attention was given to the development of power station equipment for nuclear power stations with VVÉR-1000 type reactors, in particular for expanding cooperation between COMECON member-nations in that area. A report entitled "Training of operating personnel for nuclear power stations," approved by the panel, is of considerable practical interest. The use of this material will be helpful in significantly improving the training of highly skilled cadres for service in COMECON member-nations. The 1975 work plan of the panel, envisaging research on such pressing topics as protection of the environment, nuclear power station safety, and improvements in nuclear power equipment, was approved.

* * *

The fourth session of the KNTS-RB (Scientific-technical coordinational council on radiation safety) was held in Leipzig (GDR), September 24-27, 1974. The council heard information on how the resolutions adopted at the XXVIIth Session of COMECON on environmental conservation were being carried out, reports and communications on work being done under the collaboration program, including recommendations aiding physical design calculations of biological shielding for power stations with pressurized-water reactors; results of a comparison of personnel monitors and dosimetric systems; suggestions aimed at further

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development of the topic "Determination of volume of monitoring operations in radioactive wastes burial areas"; requirements on radiation safety when sources of ionizing radiations are used in medicine, and on general radiation safety criteria when radioactive isotopes are used in industry, and so forth.

Proposals on the holding of a conference of specialists from COMECON member-nations, sometime in 1975 in Czechoslovakia, to deal with radiation safety problems in the context of operation of nuclear power stations and within the framework of a program of international intercalibration of whole-body γ-radiation spectrometers, were discussed at this session. A draft drawn up by the IAEA leadership, entitled "Estimates of collective population exposure dose," was discussed, and remarks were prepared for submission to the IAEA Secretariat, with special attention given to the most important measures to be taken by IAEA in 1975 in the area of radiation safety from the standpoint of the interests of KNTS-RB in this matter.

The council reached agreement on a tentative agenda for the fifth session of KNTS-RB, and a draft work plan covering the 1975-1976 period.

* * *

A conference of specialists on ways of developing techniques and equipment for cleanup of radioactive aerosols and gases from air-vented wastes was held September 30-October 4, 1974, in Leningrad.

Results of research work on methods of trapping radioactive iodine by trapping from the vapor phase, and methods for analyzing radioiodine content, were discussed. The use of adsorbers packed with activated charcoal and presenting not less than 1000 m^2/g surface area and 0.4 m bed height, with a linear gas stream flow velocity of 0.3 to 0.4 m/sec in the adsorber, is recommended as a suitable means of trapping iodine. It was pointed out that active charcoal cannot be used if nitrogen oxides, ozone, fluorine, or other powerful oxidizing agents are present in the off-gases, and the same applies to admixtures of materials of high molecular weight.

Specialists reported on research findings in studies of methods for determining the total radioactive iodine content in air and in off-gases vented from nuclear power stations, the iodine content by components, and the content of individual iodine compounds. The use of special cartridges (columns) filled with activated charcoal such as used in gas chromatography was recommended for periodic determinations in the absence of nitrogen oxides or ozone. Continued research on developing distinct methods for analysis of iodine and alkyl iodides was deemed advisable, as well as further work on methods for testing the filtering capacity of materials employed. Comparative tests on specimens of filtering materials, using methods generally applied in the participating countries, were proposed.

Attention was focused on how to remove radioactive noble gases and aerosols from off-gases vented from nuclear power stations, on how to design filtering devices, and also on how to work out a unified terminology referable to radioactive aerosols.

Reports were heard from specialists on the results of work in studying regular patterns in the dissemination of radioactive materials exhausted from the stacks of nuclear power stations. Concrete data on isotope make-up, on the concentrations and physicochemical characteristics of radioactive products, and also on models of the propagation of radioactive materials under conditions typical of short-term and continuous discharges were discussed, with attention given to special features of the terrain and environment, wind conditions and variation, and the influence of those factors on scattering and propagation of radioactive pollutants through the ground layer of the atmosphere.

Elaboration of a unified procedure, common to all COMECON member-nations, for the tolerance level of radioactive products in the neighborhood of industrial plants and power plants, was proposed.

Long-term trends in the field of deactivation of radioactive aerosols and gases were also discussed.

* * *

The seventh session of the KNTS on radioactive wastes and wastes deactivation was held in Leningrad, September 30 through October 4, 1974.

Proposals sumitted by the delegations of the various countries dealing with research in the field of deactivation of liquid wastes, solid wastes, and gaseous wastes, and deactivation of contaminated surfaces in rooms and on equipment, came under discussion. This research is to be conducted over the 1976-1980 period. These proposals were recommended as a basis in the preparation of a draft program for collaborative efforts in this area. The proposals envisage an extended range of efforts associated with processing and burial of radioactive wastes, including:

processing of liquid radioactive wastes of low and medium levels of radioactivity;

improvements in techniques and equipment for processing, immobilization, and burial of radioactive wastes of all levels;

improvements in methods and equipment for averting radioactive contamination and spills, and for deactivating rooms and equipment in nuclear power stations, in research centers, and in nuclear fuel processing and reprocessing plants;

investigations of techniques and equipment for cleanup of air discharges to get rid of radioactive aerosols and gases;

investigations of the Danube River basin and of the Baltic Sea for detection and assessment of pollution by radioactive materials.

The results of developments in the technology of disposal of low-level and medium-level solid and immobilized wastes by burial in salt mine caverns, and also problems in shipping such wastes from the point of formation to the burial site, were discussed (in reports covering developmental work in this area by the GDR delegation); also results of research on physicochemical processes involving interaction of radioactive wastes and materials of underground strata, and the development of a method for preparing wastes for subterranean burial; data on research on applications of electrodialysis to cleanup of low-level radioactive wastes (research and development work by the USSR delegation). Resolutions were adopted on the appropriate use of these developments in the particular countries, and on recommendations relevant to the more advanced research trends in these fields.

The council discussed a reference catalog of ion-exchange resins produced in COMECON member-nations and used in technological processes for cleanup and deactivation of liquid radioactive wastes. The catalog provides the characteristics of ion-exchange resins, defined according to unified evaluation procedures adopted in the various countries.

THE NINTH WORLD ENERGY CONFERENCE

Yu. I. Koryakin

The ninth world energy conference was held September 23-27, 1974, Detroit (USA). Participating in the deliberations were over 4300 delegates from 80 countries, and from virtually all the major international organizations concerned with power problems [COMECON (Council for Mutual Economic Aid), the European Economic Commission of UNO, the International Institute of Applied Systems Analysis, IAEA, the European Economic Commonwealth, etc.]. The Soviet delegation was headed up by the Minister of Power and Electrification P. S. Neporozhnii. This conference also functioned as an anniversary gathering: the year 1974 marked 50 years of the existence and activities of the World Power Organization. The US President G. Ford delivered an address inaugurating the conference, with the presentation of a World Comprehensive Energy Program entitled Interdependence Project.

It is worth pointing out that one special feature of the conference was its timing following immediately upon the heels of the 1973-1974 energy crisis which broke out in the western countries and which affected international relations and adversely affected the economic situation in a number of countries. The conference consequently went beyond the framework of a simple forum for discussion of scientific and applied problems in the field of industrial power, power resources, and power utilization. In the discussions, reports, and floor discussion following presentations, allusion was frequently made to problems in the area of philosophy, politics, sociology, psychology, international collaboration, and so on.

The slogan of the conference was "Economics and the environment in the light of future energy needs," itself a reflection of the topics around which 229 papers presented centered, as well as in character with the discussion unfolding and the classification of the reports. All of the reports were grouped under the following headings and panels:*

I Division. Population and Energy Resources.
- Panel 1. Population growth and population distribution (5).
- Panel 2. Power resources and power utilization (33).
- Panel 3. Distribution of resources (14).

II Division. The Environment and Energy Availability.
- Panel 1. Quality of the air (12).
- Panel 2. Quality of the water (12).
- Panel 3. Utilization of land resources (5).
- Panel 4. Noise level (4).
- Panel 5. Esthetics (2).
- Panel 6. How the environment influences energy availability (20).

III Division. Restoration of Energy Resources.
- Panel 1. Progress in the technology of energy resources recovery (20).
- Panel 2. Effect of resources recovery technology on the environment (8).

IV Division. Energy Transformation.
- Panel 1. Progress in energy conversion technology (22).
- Panel 2. Effect of energy conversion on the environment (24).

*The number of reports submitted and presented on a particular set of topics appears in parentheses.

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V Division. Energy Transport.
Panel 1. Progress in energy transport technology (18).
Panel 2. Effect of energy transport technology on the environment (3).

VI Division. Utilization of Energy.
Panel 1. Progress in energy utilization (21).
Panel 2. Effect of energy utilization on the environment (6).

There were 26 papers dealing specifically with nuclear power topics, and of these 23 dealt with reactor electric power problems and the remaining three with applications of reactor heat in desalination of seawater, a possible role for nuclear power in the utilization of ocean resources and thermonuclear energy. Most of the papers on nuclear power fall under the heading of "Energy conversion" in the breakdown above.

In practically all of the reports dealing with plans and forecasts for the development of power in the various countries, a decisive role was cut out for nuclear power. In the discussion of topics pertinent to nuclear power, attention was focused on macroscopic aspects of nuclear energy production: the scale of development and forecasting patterns, the fuel problem, and the fuel cycle in nuclear power systems. In contrast to preceding conferences of this series, here there was no place for descriptions of reactor types and nuclear power stations, reactor engineering or reactor technology. Attention was instead centered not so much on nuclear power stations per se as a means of generating electric power as on those factors and conditions governing the functioning of nuclear power stations (uranium resources and uranium mining, fabrication of fuel elements) or those factors and conditions accompanying the functioning of nuclear power plants (disposal of radioactive wastes by burial, nuclear power station safety topics, environmental effects of nuclear power stations). These topics also came under discussion in relation to the entire nuclear power grid, not just in relation to individual isolated nuclear power stations. In general, it should be stressed here that the systems approach applied in the discussion of energy problems typified this particular conference.

Reports dealing with nuclear power topics can be grouped arbitrarily under the following headings:

1. Scales and forecasts of nuclear power development.
2. The role of fast reactors and nuclear power structure in the future development of nuclear power, and alternative structures for nuclear power station systems.
3. Experience in the building and operation of nuclear power stations, new types of nuclear power stations and new projects and concepts.
4. Topics relating to the external fuel cycle in the nuclear power industry.
5. Nuclear power and the environment.

The role played by nuclear power in making energy available throughout the world in the future (figured or extrapolated to the year 2000 as a rule) was the subject of papers presented by national delegations and by international organizations. Various authors advanced predictions of the scale of nuclear power development throughout the world, with a breakdown by large-scale geographic zones: Europe, North America, the Pacific zone, Central and South America, Asia and the Far East and New Zealand, Africa, and the Middle East, countries with a centralized planned economy. Data on some individual countries were also presented.

It was emphasized that the considerable rise in worldwide prices for crude oil, from the area of 21.2-23.8 dollars/ton to ~133 dollars/ton, brought on by the 1973 energy crisis, greatly enhanced the economic competitiveness of nuclear power stations. This found expression in the economic feasibility of using more expensive uranium (22 dollars/kg U_3O_8), in an appreciable drop in the economically justifiable unit power output of power generating units of nuclear power stations, and consequently an expansion of the demand for nuclear power stations on the world market, and also in the greater amount of interest displayed in building major nuclear power generating stations. All of this, as might be expected, will bring about an intensified tempo of development of nuclear power, and will lead to nuclear power making a significant contribution to the generation of electric power around the year 2000. On the whole, the fraction accounted for by nuclear power stations in total electric power generating capacity by the year 2000 will be anywhere from 50 to 83%, depending on the authors making the estimates. The last percentage figure, corresponding to a total power output of 5.3 billion kW, is a limiting figure, and is apparently on the high side. The range of 2.7 to 3.0 billion kW is judged more realistic.

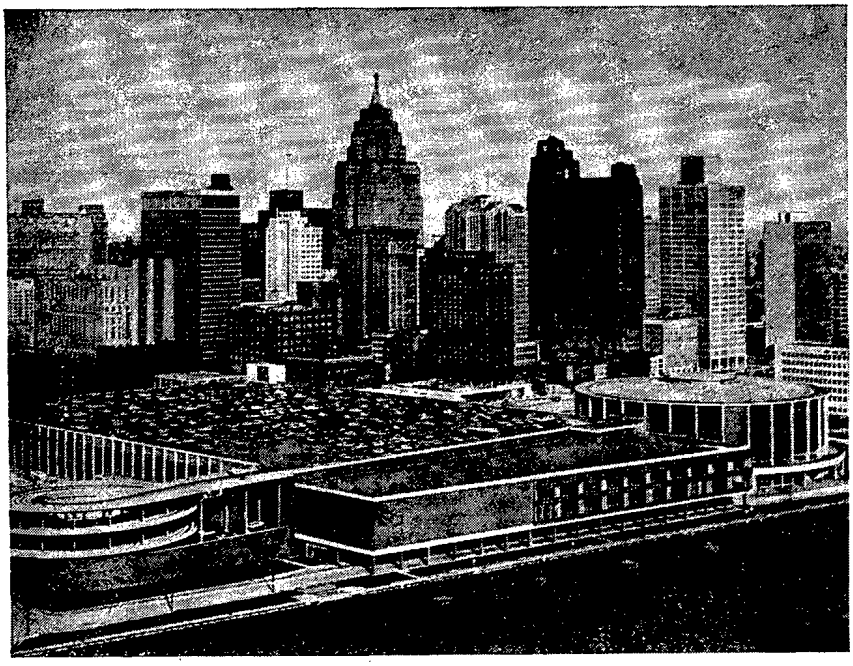

Fig. 1. Detroit. Cobo Hall (sports and exhibition complex housing the World Energy Conference) appears in foreground.

Almost all of the authors pointed out the role that fast reactors can play in the future economy of uranium resources. The development of fast reactors continues to be regarded as a general developmental trend in nuclear power. But for the time being the inadequate level of technical preparation for widespread commercial exploitation continues to lend a low profile to fast reactors in the forecast nuclear power structure (accounting for no more than 30% of the total) at least till the end of the century. In other words, until the end of the century nuclear power will remain predominantly "thermal," and no appreciable savings in uranium are expected from the use of fast breeders up to the year 2000 or thereabouts. Meanwhile, the cost of building such reactors is simply exorbitant at the present state of the art. For example, the building of the fast liquid-metal LMFBR reactor with its 350 MW(e) rating, at Oak Ridge (USA), is now estimated at a total of 1.736 billion dollars.

It was also emphasized in this context that the economic competitiveness of nuclear power stations with LWR type reactors holds its own even in the face of a doubling of uranium prices, and possibly even in the event that uranium prices increase three- or fourfold. There are on that account no special worries as to the long-term viability of LWR type thermal reactors, even looking at them outside of the question of fast breeders. But if effective fast breeders become a serious factor in the nuclear power picture within the next 10 to 20 yr, then LWR type reactors can be used to process plutonium for those reactors with uranium demand running into the millions of tons. This does not amount to a significant portion of the estimated worldwide uranium resources. It is felt that LWR type reactors can be converted to converter operation. For example, a reduction in burnup from 34,000 to 15,000 MWd/ton U in those reactors would mean a 50% increase in plutonium yield.

Reports submitted by French and Canadian specialists were an exception. The former felt that industrial utilization of fast breeders by the turn of the century would be held back by a shortage of plutonium and the rather long doubling time of the first breeders to appear on the scene (15 yr doubling time on oxide fuel). Nevertheless, a doubling time of 10 to 15 yr is judged adequate, and there is no need to switch to some other type of fuel (say, gaseous fuel).

The Canadian specialists are looking into CANDU type heavy-water converter-reactors as a possible alternative to fast breeders. Even though the nuclear power systems of other countries hold open a major role for fast breeders, these authors are of the opinion that the very considerable reserves of uranium and thorium existing in the Earth's land mass and oceans (respectively, $2.5 \cdot 10^{12}$ and $8 \cdot 10^{12}$ tons) make it possible, in principle, to satisfy any anticipated needs for power for hundreds of years through reliance on thermal reactors.

We should take note of the growing interest in HTR type high-temperature reactors, even though their place in the nuclear energy picture as of the year 1990 is not expected to be greater than 3%, and they are not expected to account for more than 7% by the year 2000.

We also note a significant increase in the specific costs of nuclear power stations. As an example, the average specific cost of nuclear power stations with LWR type reactors in the USA rose from 134 dollars/kW in 1967 to 397 dollars/kW in 1973. This increase reflects a greater realism in cost estimates, but also stems from several other factors: escalation of prices for equipment and materials, an increase in labor construction costs, inflation in general, and changes in the percentage of capital costs during construction and delays in construction amounting to several years' time, tightening of safety and reliability requirements imposed on nuclear power generating stations, greater complications and higher costs in licensing and authorization procedures, etc.

The construction time for nuclear power stations has been stretched to 8-9 yr in the USA. The assumption is that the cost of nuclear power stations will go on climbing in the future, principally on account of additional safety requirements. It is also assumed that the cost of nuclear power stations cannot be determined exactly, since a matter of 9 to 10 yr from the time the decision is taken to begin construction till the plant is ready for commercial operation. During that period of time the cost picture can undergo far-reaching changes, as can public opinion, positions taken by leading organizations and governments, labor costs, and labor productivity.

At the present time, the unit power output of LWR type reactors in the USA is limited to 1300 MW, even though the existing technology even now makes it possible to raise the power output level to 2000 MW. It is still unclear just what factors will act to limit further increases in unit power output (well above 2000 MW). It is assumed that savings in specific costs through raising the unit power level will be partially lost because of the rise in construction costs due to the factors alluded to.

According to the view entertained by USA specialists, no new reactor types of any kind will be required for a two-component nuclear power system (LWR + FBR), since this system meets the requirements imposed on nuclear power, and exhibits significant technical and economic potentials.

The new nuclear power station projects presented for discussion at the Ninth World Energy Conference are characterized by new construction methods (floating nuclear power stations and underground nuclear power stations). The former variant is important in the case of the USA. Mass construction of floating nuclear power stations on the continental shelf would be preferable in terms of the difficulties involved in selecting and acquiring construction sites, supplying cooling water, and long-term construction of ground-level nuclear power stations. Moreover, in the USA over half of the population lives in shoreline areas. The first such nuclear power station with two PWR reactors each of 1150 MW(e) ratings is sited three miles from the shoreline (at a depth of 12 to 22 m) near New Jersey, on an anchored platform behind an artificial protecting breakwater barrier. The capacity of the Jacksonville wharf for the construction of such nuclear power stations will be brought up to a level of four floating nuclear power stations per year by the beginning of the 1980s. There is some hope that expensive protecting breakwaters will not be needed eventually.

Underground construction of combined nuclear and fossil-fuel electric power generating stations was described in a report by Swedish specialists. The need and the economic prerequisites for nuclear generation of space heat calls for shortening the length of space heat pipelines, which is in fact possible when safety conditions are properly observed in underground construction of power stations.

Close attention was given, at the conference, to the fuel resources available to nuclear power and to satisfaction of nuclear fuel needs. Over the last 10 years substantial progress has been observed in uranium prospecting techniques. The average borehole drilling depth has been increased from 61 m in the early 1960s to 153 m in 1973, while drilling is being carried out at depths of 610 to 1200 m in a significant number of cases. At the same time, the uranium content in the occurrences brought to light decreased from 14.87 kg/m^3 (before the 1960s) to 4.46 kg/m^3 in more recent years, which in turn results in increased uranium mining and recovery costs. If further uranium finds are possible at existing costs of 2.2 dollars for each kilogram of U_3O_8, capital investments in uranium prospecting to ensure that uranium supplies will be forthcoming during the service life of 1000 MW(e) nuclear power stations with LWR reactors will run below 10 million dollars, which is not more than 2% of capital investments in nuclear power stations as such. Consequently, an inevitable rise in uranium prices can be managed with, and the ultimate effect on nuclear power costs will not be appreciable. Worldwide nuclear industry needs for uranium are estimated

at 5.2 million tons up to the year 2000, while potential uranium reserves recoverable at economically acceptable costs (up to 66 dollars/kg) are estimated at ~6.2 million tons. On the whole, the fuel problem confronting the nuclear power industry seemed fairly bright in the light of the conference discussion.

As for one other major link in the external fuel chain, regeneration of spent nuclear fuel, here we have to take note of various economic factors that are beginning to receive their share of attention and which are stimulating international integration in this area. A capacity of 5 to 6 tons/day (or 1500 to 1800 tons annually) is deemed economically optimum for a nuclear spent fuel reprocessing plant. Such a plant should be responsible for reprocessing fuel from nuclear power stations with LWR reactors of total output placed at 50 to 60 million kW(e). But this is an expensive and cumbersome process. That accounts for the joining of Great Britain, France, and FRG to cope with common "fuel recovery interests."

IAEA General Manager Z. Eklund, taking the floor at the conference, reported that international integration of the external fuel cycle will be the subject of a forthcoming IAEA-sponsored conference scheduled for 1977.

Radiation safety problems associated with nuclear power development are viewed as reliably resolved according to the conference proceedings. As for disposal of considerable quantities of radioactive wastes by underground burial, this confidence is based on over 20 yr of positive experience in wastes storage and on the remarkable progress achieved in recent years in wastes processing technology (immobilization of aqueous solutions, particularly).

Several measures were carried out simultaneously at the conference: plenary and panel sessions, seminars, lectures, round-table discussions, press conferences, technical excursions, and film showings were held. In line with the expressed wish on the part of a large number of delegates to take part in the floor discussions, time allotted to take the floor was held at a minimum, in 6-min slots or even 2-min slots. Even general reports presenting review papers on basic conference topics were accorded only 10- to 15-min time. A daily conference bulletin aided measurably in keeping participants abreast of the conference proceedings and offered helpful comments.

The fairly high degree of optimism with respect to the present status and future development of nuclear power was shared in common in all of the reports on nuclear power topics, and also in most of the floor discussion.

The next scheduled World Energy Conference (the 10th) will be held in 1977 in Istanbul (Turkey).

INTENSIFIED HEAT TRANSFER IN CHANNELS OF RBMK TYPE REACTORS

A. N. Ryabov and V. N. Filippov

A conference on intensification of heat transfer in the channels of RBMK type reactors was held in Moscow in early April, 1974, under the chairmanship of Academician N. A. Dollezhal', to provide a forum for discussion of the state of the art, results of experimental research, and pathways of future research.

The problem of how to increase the electrical power output of a power generating unit in a nuclear power station with RBMK type reactors, while keeping the dimensions and the number of process channels constant, and also keeping the same basic reactor equipment, came under discussion. An increase to 4500 kW in the thermal power output of the channels should not bring about burnout phenomena at elevated heat flux densities and high coolant steam content levels. Moreover, the temperature of the outer surface of the wall must be kept constant in the process channel, and the maximum temperature of the graphite moderator must be kept within permissible limits. Head losses over the circulation loop must be limited to 18.6 bar. Coolant parameters can be smoothed out over the transverse cross section of the fuel assembly in order to optimize the positioning of fuel elements in that cross section, which should in turn allow an increase in critical (burnout) heat flux density. Calculated and experimental studies on hydrodynamical optimization of fuel element positioning in the fuel assembly have shown that the critical output level of the fuel assembly can be increased by 5 to 10% simply by judiciously altering the arrangement of fuel elements in the assembly.

Increasing the inner diameter of the reactor channel from 80 to 82 mm by decreasing the channel wall thickness makes it possible to raise the channel output level by 5-7% while holding to permissible hydraulic drag values.

An increase in channel output to 4500 kW without any changes in the design of the channel and fuel elements, and while keeping the fuel loading constant, can be attained by using what are known as heat-transfer intensifiers in the core, so that the steam content at the exit from a maximally stressed channel can be increased by as much as 50 to 70%, and as much as 30 to 40% as averaged over the entire reactor. In this case the total flowrate of coolant through the reactor experiences an abrupt drop, leading to a reduction in electric power costs for the local needs of the plant itself.

On the basis of experimental data obtained in various scientific-research organizations throughout the country, thermal hydraulic characteristics of a channel with 80 and 82 mm inner diameter were calculated for power levels of 4200 and 4500 kW, in the case of fuel assemblies containing fuel elements 13.5 mm in diameter and various heat-transfer intensifiers. It was assumed in these calculations that the lower fuel assembly would have 10 regular spacing grids, while the upper fuel assembly would have 20 spacing grids with heat-transfer intensifiers. The local hydraulic drag ratio for spacing grids with intensifiers was assigned a value of 2.5, in contrast to the 0.5 value assigned in the case of regular grids.

Calculations showed that the maximum steam content is ~65% when the allowable loop hydraulic drag is 18.6 bar, the channel inner diameter is 80 mm, the channel output level is 4500 kW, and the maximum heat flux density is 1.16 MW/m^2. Preliminary results on thermal-hydraulic investigations pointed to the possibility of actually attaining such steam content levels.

Results of thermal-hydraulic investigations of channel models with different intensifiers, obtained within the recent period at different scientific-research organizations, came under discussion. All of these investigations were carried out according to a generally recognized procedure: the experimental

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sections were heated by low-voltage current, and the onset of burnout was attained by raising the electrical power smoothly until the temperature conditions of the rods were changed at constant pressure, constant flowrate, and constant enthalpy at the entrance.

Results of experimental burnout studies using rod assemblies with spacer grids and local intensifiers – vortex generators or swirl vanes located in the grids and spaced 125 mm apart over the entire height of the assembly – were reported out. The intensifiers in question consisted each of six pairs of segments of twisted variable-cross-section strips radiating outward at 60° angles.

The experiments were conducted on assemblies of seven cylindrical rod type fuel elements designed to simulate 12 mm diameter fuel rods with 2200 mm heated length and situated in a regular triangular grid array in a 45.5 mm diameter channel. The empirical data were obtained at pressures 68.5, 88.1, and 117.6 bar and mass velocities of 600, 800, 1000, and 1500 $kg/m^2 \cdot sec$.

Comparison of the empirical data obtained both with and without intensifiers demonstrated that the intensifiers account for as much as a 50% power output increment, and that hydraulic drag increases by a factor of 1.5. In addition, the use of intensifiers has the effect of substantially broadening the range of burnout-free performance of the assemblies while increasing the critical heat flux density level.

The effect brought about is accounted for by the existence of a rotational flow pattern on the part of the coolant as it flows through the vortex swirler and by separation of moisture from the core of the stream on the heat-generating surface of the rods. The twist in the coolant flow also contributes to intensified agitation or turbulization and to vigorous mixing between adjacent cells, which in turn leads to leveling out of the enthalpy in the transverse cross section of the assembly.

Results of an experimental investigation of burnout in rod assemblies with heat-transfer intensifiers made in the form of strips twisted in spirals and spaced 165 mm apart, running the entire length of the assembly, were reported out and discussed. The intensifiers filled the clearances between the fuel rods over the entire heated length.

These experiments were carried out on assemblies of 21 fuel elements each 13.5 mm in diameter and of 900 mm heated length. The rods were arranged on a square grid in a 90 mm channel. The experimental data were obtained at 68.6 bar pressure, 800 to 1000 $kg/m^2 \cdot sec$ mass velocities, and 0.58 to 1.16 MW/m^2 heat flux densities.

As the experimental data revealed, the critical steam content in this type of assembly is increased 1.5 to 2 times over that in an assembly with no intensifiers present. It is evident from the findings of this investigation that the hydraulic drag presented by the assembly increases by ~50% in this instance.

The conference also heard data on experiments designed to study the effect of the spacer grids functioning as turbulizers or vortex generators on the temperature conditions of an annular channel and a bundle consisting of four rods arranged on a square lattice array. The rod diameter in both experimental sections was 13.5 mm in this case, and the heated length was 3500 and 7000 mm. The experiments were conducted at pressures 68.5 to 127.3 bar and at mass velocities in the range from 1000 to 2500 $kg/m^2 \cdot sec$.

In the experiments using an annular channel, turbulization of the flow was achieved through reliance on transverse ribs of rectangular cross section, 1.0 or 2.0 mm in width. The rib spacing pitch was 375, 187.5, and 94 mm. It was found that the temperature conditions varied without burnout type discontinuities as usually observed in a channel with no intensifiers present. But the rate of wall temperature rise with increasing power level increased abruptly in response to this (by as much as 20 to 50 times). The greatest power output gain was 35-40% when ribs were spaced 187.5 mm apart. The gain achieved was smaller in experiments with four-rod assemblies.

One of the reports cited results of an experimental investigation on temperature patterns and burnout behavior for an experimental section comprising a thermal-hydraulic model simulating a channel with a full-size transverse cross section for the flow passages. The fuel assembly in question consisted of 18 heated tubes 15 mm in diameter and arranged in square and triangular lattices within a 79 mm diameter channel. The heated length of the bundle was 1100 mm. Both spacer intensifier grids contracting the effective cross section of the assembly by 65% because of the lobes on them that were bent back, and regular spacer grids set 175 and 350 mm apart, were employed in these experiments.

The temperature conditions brought about in the assemblies with the regular spacer grids were investigated at pressure 74 bar, mass velocities of 600, 1000, and 1500 $kg/m^2 \cdot sec$, and with unheated water

supplied at the entry to the section. The results of these experiments showed that qualitatively the same temperature distribution at the walls occurred at all of the mass velocities studied, viz.: the highest temperature was exhibited by surfaces located in the central cells, and the lowest by those in the peripheral cells, while those located in intermediate triangular cells featured a still lower temperature, and the lowest temperature of all was found in the intermediate square cells. Since the heat flux was distributed uniformly over the rods, that would mean that heating of the coolant was maximized in the central cells of this pattern.

The investigation of temperature conditions obtaining in assemblies with spacer intensifier-grids showed that the coolant was heated the most in the peripheral cells. Heating was pretty much the same in the central and intermediate square cells.

Burnout phenomena were studied at pressures 58.8 and 74 bar, and at mass velocities of 600, 1000, and 1500 $kg/m^2 \cdot sec$. The sites of onset of burnout, the rise of the wall temperature accompanying the onset of burnout, and fluctuations in wall temperature in the region beyond burnout, were all recorded in the experiments. In the assemblies with regular spacer grids, burnout generally occurred in the narrow clearances separating the central cells, and the level of critical heat flux densities was in accordance with calculations based on the V. S. Osmachkin formula. In assemblies with spacer intensifier-grids, burnout occurred in the peripheral cells in the narrow clearances intervening between the peripheral rods and the channel.

As the experiments revealed, the effect brought about by the use of spacer intensifier grids involved a smooth rise in the wall temperature of the rods in the region beyond burnout, while the critical heat flux density rose by 30-40% in the nucleate boiling region.

The findings of the hydraulics study revealed that hydraulic drag increases on the average by 2.9 times, when spacer intensifier grids with a pitch of 175 mm are installed, as compared to the situation in assemblies with regular grids.

The feasibility of using a "multistage" channel, i.e., a channel separated in height into a number of sections included in parallel as far as coolant flow is concerned, in a RBMK type reactor was reported. In a channel of this type, the critical heat flux densities would be slightly higher because of the low mass velocity, within the confines of each distinct stage, but hydraulic drag opposing the inflow and efflux of coolant would be greatly increased.

One of the reports dealt with an analytical treatment of burnout at high steam content levels. A procedure for solving the balance equation for liquid in the film was presented as an aid in calculating limiting steam content. Equations were derived for use in calculating the limiting liquid flowrate through the film and the wetting intensity. Theoretical data were found to be in adequate agreement with empirical data.

The conference noted the fact that the scientific-research work and experimental design work on intensified heat transfer in relation to the operating conditions of RBMK type reactors is making it possible to anticipate a 20-50% rise in the output level of the reactor technological (process) channel without any substantial changes in the reactor design as a whole, and primarily through improvements in the design of the fuel assemblies. It was recommended that steps be taken toward gaining practical acceptance, on an experimental level, of intensifying devices in the form of improved regular spacer grids, without any changes introduced into the design of the fuel elements as such

THE SECOND ALL-UNION CONFERENCE ON THE METROLOGY OF NEUTRON RADIATION AT REACTOR AND ACCELERATOR FACILITIES

R. D. Vasil'ev

The Second All-Union conference on the metrology of neutron radiation at reactor and accelerator facilities was held in Moscow on October 14-17, 1974. This conference was organized and sponsored by the VNIIFTRI (All-Union scientific-research institute for physics and engineering and radio-engineering measurements) on the initiative of the State Committee on Standards attached to the Council of Ministers of the USSR, and of the State Committee on the Peaceful Uses of Atomic Energy of the USSR. Participating in the conference were specialists assigned from over 80-odd organizations under the aegis of various ministries, departments, and the Academy of Sciences.

The purpose of this gathering was to draw up a balance sheet of metrological research over the three-year period elapsed between the first and second conferences. A collection of papers, in two volumes, entitled "Metrology of neutron radiation at reactor and accelerator installations," published to coincide with the inauguration of the conference, was distributed among the participants.

Review papers on topics pertaining to the metrology of neutron measurements at nuclear physics installations and facilities were presented at a plenary session. Achievements of the past few years were discussed and new problems on the horizon were formulated. The structure of metrological servicing of measurements work at nuclear physics installations was discussed, as well as standard specimens and physical standards for sources (fields) of neutrons, standard physical specimens of materials, counters and instruments, nuclear physics constants, standard procedures to be followed in taking measurements, calibration work, and certification, intercomparisons at different reactor installations, problems involving how to go about determining errors and design of experiments, terminology in the field of neutron radiation. It was pointed out that the beginnings of the organization of a system of metrological servicing of measurements at nuclear physics installations which would be unique for the various administrative departments have been laid with the devising of a checkout pattern for equipment used in measuring the characteristics of neutron fields at given installations, a state-sponsored special physical standard of the unit of neutron flux density using an accelerator as the basic metrological tool (at VNIIFTRI), the first standard specimens of neutron sources based on nuclear reactors [at the I. V. Kurchatov Institute of Atomic Energy (IAÉ) and at other organizations as well], and also by the production of a variety of standard specimens equipment including inparticular sets of standard AKN specimens (neutron activation sets) and NDS specimens (neutron tracking detectors). This system encompasses measurements of the characteristics of neutron fields over the entire range of reactor neutron energies at flux densities to 10^{14} neutrons/cm^2·sec.

There were 43 reports presented at a panel session called to discuss "Equipment for measuring the characteristics of neutron fields, and standard physical specimens; calibration and certification." These 43 reports can be grouped according to the following basic trends covered in them: standard sources based on nuclear reactors and other equipment for calibration work, continuous-acting instruments and detectors (gas-discharge counters, calorimetric detectors, multisphere spectrometers, neutron identifiers responding to pulse shape, etc.), detectors consisting of fissionable specimens and specimens susceptible to activation in boron and cadmium absorbers, activation specimens for analysis of materials, detectors and targets composed of fissionable isotopes, and fission fragment recording devices. Separate aspects of

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measurements of induced activity, and problems involving isotope analysis of fissionable materials, were also discussed at the panel.

A panel on "Measurements of the characteristics of neutron fields at reactor and accelerator installations" drew 23 reports. Some of these dealt with measurements of thermal neutrons and intermediate-spectrum neutrons, the remainder with measurements of fast neutrons. Investigations carried out at reactors accounted for almost the totality of the reports presented, only two of them delving into problems encountered in work with accelerators. Applications of individual types of continuous-acting detectors were among the topics covered, as well as various methods of recovering the spectrum of fast neutrons, and results of experimental separation of fast neutron spectra. A report was given on attempts at evaluating errors in the results of spectrum restitution. Attention of researchers working with reactors and accelerators continued to focus on indirect and combined measurements, as the modes of measurements providing the most complete volume of reliable information on the characteristics of neutron fields. The number of experimental physicists making use of standard procedures in measurements work is on the increase. Attention was drawn to the way closer attention is being given to more precise estimates of errors. For example, values on the low side are being reported less and less often because of the painstaking analysis of sources of systematic error.

Six reports were presented at the panel session on "Intercomparisons at reactors." Interest in this problem has been steadily on the rise. For example, in recent years several organizations have been participating in six national intercomparison programs and one international intercomparison program, encompassing indirect measurements and combined measurements of spectral and dosimetric characteristics. Of the most remarkable national intercomparisons reported on, we must mention intercomparisons of results of measurements on a static reactor that were carried out during 1974. The organizations, using equipment for measurements over the entire range of reactor neutron energies, took part in this intercomparison project. The intercomparison program included preliminary comparisons of methods for restituting a test spectrum. The national and international intercomparisons made it possible to discern the most highly improved procedures and equipment for taking measurements, and to recommend promising detectors and instruments for practical applications.

The panel on "Nuclear data for neutron measurements, and metrological aspects of nuclear data research" heard contributions by the leaders of the Nuclear Data Center, V. N. Manokhin [FÉI (Power Physics Institute, Obninsk], and Center for Data on the Structure of the Atomic Nucleus and on Nuclear Reactions, F. E. Chukreev [I. V. Kurchatov IAÉ (Atomic Energy Institute), Moscow]. These reports gave accounts of the trends evident in the work of their data centers in relation to the needs of experimental physicists for data used in neutron measurements. They acknowledged the importance of the activities undertaken by the Nuclear Data Commission of GKAÉ (State Committee for the Peaceful Uses of Atomic Energy), by coordination teams on measurements of nuclear physics constants, and nuclear data needs of reactor facilities and in shielding and protection work, by topic-oriented teams on various types of neutron constants, standards, and reference quantities. Reports were also presented on measurements of the fission neutron spectrum of ^{252}Cf, on approximation of cross sections of threshold reactions, on metrological analysis of experimental determinations of particular constants.

The panel on "Determination of errors and design of experiments" discussed problems concerning calculation of errors in indirect measurements, and approximation of calibration dependences. Methods for estimating the information content of various experiments involving measurements of cross sections were discussed, in addition to problems involving design of experiments and neutron data estimates for reactors.

A general discussion was held at the concluding plenary session. The efforts of the various organizations to solve various problems in metrology in the field of neutron radiation were approved. It became clear from the discussion on the floor and from the reports presented that metrological studies are being carried out more and more often by several organizations jointly. Coordination of these efforts is being aided by exchange of views and opinions such as that afforded by these All-Union conferences.

The next such conference is scheduled for sometime in 1977. Organization of this next conference is also entrusted to VNIIFTRI.

MIFI TWENTY-SIXTH SCIENTIFIC CONFERENCE HELD

V. V. Frolov and V. A. Grigor'ev

The twenty-sixth scientific conference of instructors, students and staff personnel of MIFI (Moscow Engineering and Physics Institute), devoted to the 250th anniversary of the Academy of Sciences of the USSR, was held in June, 1974. A total of 480 reports and communications were presented at 26 panel sessions.

The panel on experimental nuclear physics listened with keen interest to reports submitted on experiments dealing with searches for the W-boson at the IFVÉ (Institute of High-Energy Physics) accelerator located at Serpukhov. It was shown that a W-boson having a muon decay mode $\geqslant 10\%$ does not exit with a mass in the interval from 5 to 8 GeV, at the level $\sim 10^{-35}$ cm^2 of the production cross section of a W-boson in p–N collisions. These experiments also yielded data on interaction cross sections of particles at energies of 70 GeV in the case of large transverse momenta (1.8 to 2.8 GeV/c). A new μ-meson generation mechanism was detected (the intensity of these μ-mesons amounts to $\sim 2.5 \cdot 10^{-5}$ the intensity of pions in the range of transverse momenta investigated).

F. M. Sergeev gave an account of experimental research on pion–nuclear reactions using the method of heavy-liquid bubble chambers. From among the physical results reported, we should mention the measurement of angular scattering of pions of both signs on carbon nuclei in the 5-20 MeV energy range. This experiment made it possible to establish the fact that the pion–nuclear interaction in the p-state exhibits a repulsion nature.

A report by P. S. Baranov et al., delivered at the panel on experimental techniques in nuclear physics, cited new experimental data on the time resolution of large-size scintillation counters when Soviet-make FÉU-85 and FÉU-87 photomultipliers were made with quantity-manufactured plastic polystyrene base scintillators. The time resolution of counters of dimensions $400 \times 100 \times 20$ mm was $\lesssim 1$ nsec when relativistic particles were recorded.

The panel on theoretical nuclear physics responded with greatest interest to a report by Academician A. B. Migdal presenting recent results he obtained on the theory of the π-condensation in nuclei, which is related to the production of π-meson pairs in the nucleus.

The plasma physics panel heard reports by O. A. Vinogradova, S. K. Dimitrov, A. N. Igritskii, V. M. Smirnov, D. A. Panov, V. A. Leitan, and A. S. Luts'ko, discussing engineering problems of thermonuclear fusion reactors. Results pertaining to the further development and improvements of an ion energy recuperation system proposed on an earlier occasion were discussed at this panel. The ions in question escape from an open type reactor trap. On the basis of experiments centered around the LIN-5 research facility, an estimate of the efficiency of ion energy recuperators in the injector circuit of Tokamak machines was arrived at: the expected efficiency of real direct conversion systems is 0.85 to 0.90.

D. A. Knyazev et al. evaluated the outlook for acceptance of various physical and chemical methods of isotope separation, in a report presented before the panel on physics of separation processes.

A lively discussion was touched off at the heat-transfer physics section by a presentation by V. A. Andreev et al. on investigations of heat transfer in the cryogenic temperature range. We should also mention a fundamental contribution by L. S. Kokorev et al. laying the basis for physical representations on the nature of burnout phenomena on heat-generating surfaces.

Translated from Atomnaya Énergiya, Vol. 38, No. 1, p. 59, January, 1975.

The panel on applied nuclear physics responded with deep interest to reports on applications of activation radiography techniques in nuclear γ-resonance work as part of a rounded integrated study of diffusion and redistribution processes affecting impurity and alloying elements in the corrosion of zirconium and zirconium alloys. The same methods were used in obtaining data on the parameters and nature of oxygen diffusion, on the distribution of alloying elements in oxide type zirconium films, and in oxide films of zirconium alloys.

The panel on electrophysical equipment heard results obtained in the development of new models of linear electron accelerators for applied purposes. The first model was designed for a pulsed current of accelerated electrons of 1 A, and is capable of producing 4 to 10 MeV energy. The second model operates at a high frequency (~10 GHz), so that a successful compact design was feasible. This model has been exhibited at the Exposition of Achievements of the National Economy (VDNKh).

The panel on solid state physics sustained a lively discussion stimulated by a report presented by T. I. Kozin et al., outlining a method for recording long-lived plasma with the aid of a stabilized laser. This method favors the use of noncontacting techniques in investigating low-density plasma with densities down to $\sim 10^{12}$ cm^{-2}, and is distinguished by its comparative simplicity.

The concluding plenary session heard a report entitled "High-energy physics and the Academy of Sciences of the USSR," delivered by Corresponding Member of the Academy of Sciences of the USSR A. M. Baldin. The history of the design and development of accelerators in the Soviet Union was presented in his brilliant and content-packed presentation, which also took up the role of scientists belonging to the Academy of Sciences of the USSR, and particularly the role played by FIAN (Lebedev Institute of Physics of the Academy of Sciences of the USSR) in establishing and developing Soviet accelerator facilities and equipment.

THE FOURTH ALL-UNION PLASMA PHYSICS SCHOOL

V. A. Papadichev

The Fourth All-Union plasma physics school was held in Novosibirsk, July 29 through August 7, 1974. Participating in this session of the school, an institution which has become traditional in its field and which has been scheduled once every two years, were 200-odd Soviet and foreign scientists. The 1974 school agenda included studies of high-power beams of relativistic electrons and applications of beams of relativistic electrons in research on controlled thermonuclear fusion, and also plasma confinement systems and heating systems for high-density plasma. One of the problems tackled by the school, as stated in the introductory remarks by the chairman of the Organizing Committee, G. I. Budker, was analyzing the present status of research in those areas. The lecturers included leading scientists of the USSR, USA, and Italy. Seminars were also staged as forums for detailed and informal discussion of a variety of problems: high-density plasma, collective acceleration of ions as beams of relativistic electrons traverse a gas, techniques and equipment in the generation of ultrahigh-power electron beams, coupling of high-power electron beams and plasma. The floor discussions at the lectures, seminars, and in the corridors, contributed to a more complete exchange of ideas and information and to the establishing of close contacts between the scientists of different countries and laboratories.

A report by M. S. Rabinovich [FIAN (Lebedev Inst. Physics)] contained a review of the status of thermonuclear research efforts throughout the world. The present stage is characterized by a transition to the construction of large machines and installations (one being scheduled for commissioning in Europe in 1977, another for the early 1980s in the USA) at which work on basic scientific problems will compete with efforts to solve technological and engineering problems in the design of a successful thermonuclear fusion reactor.

Problems pertaining to the production of powerful beams of relativistic electrons for achieving pulsed thermonuclear reactions found reflection in a report presented by M. V. Babykin [I. V. Kurchatov IAÉ (Institute of Atomic Energy)]. Relevant calculations indicate that a beam of duration ~1 nsec and energy 10^7 J (power level ~10^{16} W) would be required for the purpose. Reliance on a heavy shell around the D–T mixture would necessarily lead to relaxing the restrictions on the beam (10 nsec, $3 \cdot 10^6$ J, $3 \cdot 10^{14}$ W). Conversion to disk-shaped and conically tapered shaping circuits with peripheral switching and to positioning of a cold-emission gun at the center would be feasible in efforts to generate such beams. In order to arrive at the required high-power growth rate (> $3 \cdot 10^{22}$ W/sec), switching inductance would have to be kept down to 2-3 nH, which would in turn call for the use of 100 to 200 discharge switches working in parallel. Magnetic insulation would do for the transport of the pulse to the gun. Experiments on multichannel switching devices are to be continued at IAÉ, using the new Angara-1 facility (2.5 MeV, 400 kA).

Similar problems were approached in a report delivered by A. Kolb (Maxwell Corp. Laboratory, USA). Beams with a high power growth rate can also be used to excite high power UV gas lasers. The principal problems confronting designers of plasma machines are, in Kolb's view, how to develop a compact storage device, pulse switching (appropriate discharge switches), and generation of the beam per se through the use of a field-emission diode. Several types of multichannel discharge switches functioning in water, oil, or compressed gas environments have been developed, with voltage to 5 MV and power switched in the range up to 10^{12} W. Trigger pulse height is generally 20-25% of the voltage switched. How to decrease the inductance of the discharge switch is the very first problem to be tackled, although the diode itself actually accounts for anywhere from 20 to 30% of the total inductance. The diode wave impedance is closely described by the Child–Langmuir law when proper attention is given to the displacement of the plasma interface in the diode, i.e., to changes in the nonconducting clearance in the diode. The

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duration of the voltage pulse rise is 5 to 10 nsec in the new medium-energy (1-2 MeV) high-current electron accelerators developed by the Maxwell Corp.

Some interesting experiments on focusing of high-current electron beams were described in a report by J. Witkowitsky (Naval Research Laboratory, USA). In these experiments, a hollow cylindrical beam is admitted into a drift chamber with a side wall made of insulating material and with a central conductor made in the form of wire running between the anode and the second end face of the chamber, to short back current. The beam was focused on the wire, and later on (in the case of a thin wire) the beam was not immediately destroyed but allowed to follow the wire, even if it was a bent wire. A yield of 10^9 to 10^{11} neutrons/pulse resulted when the wire was coated with deuterated polyethylene. The beam current was ~300 kA in these experiments, the electron energy 200 to 400 keV, and the beam duration 50 nsec.

A report by G. A. Mesyats (Institute of Atmospheric Optics, Siberian Branch, Academy of Sciences of the USSR) dealt with the initial stage in the shaping of a plasma cathode and the current flowing through the field-emission diode. The decisive factors here are, in the author's opinion, explosive emission and the propagation of the cathode flare (plasma and cathode).

The results of work done on devising pulsed high-current electron accelerators at the Nuclear Physics Institute, Siberian Branch, Academy of Sciences of the USSR (IYaF SO AN SSSR) were reported by V. M. Logunov. The shaping circuits were shaped in these machines with the aid of a Tesla pulse transformer; the usual circuitry with discharge switches in the line and a field-emission diode were employed later. At the present time, accelerators with a single 250 keV, 250 kA, 40 nsec and 800 keV, 110 kA pulse shaping circuit are being used. An accelerator with a double pulse-shaping circuit in water (1 MeV, 230 kA, 60 nsec) is being built, and a 1 MeV, 300-400 kA accelerator is being planned. All of the accelerators are being used or are being designed for experiments on beam–plasma coupling in a variety of configurations (multiple-mirror configurations included). Some interesting experiments using mockups to simulate a rise in the electrical conductivity of the water with the aid of a pressure rise (a threefold rise of 120 atm) and screening of microcorona points on the electrode surfaces as an electrolyte solution is forced through the electrode surfaces into the electrode sheath region were also reported on. In the case where both electrodes are screened, the electrical conductivity of the clearance rose by a factor of 2 to 2.5. But we should draw attention to some engineering difficulties impeding the materialization of the last technique, when electrode areas in existing accelerators are appreciable.

A basic research trend in evidence at Sandy laboratory (USA) centers on the use of high-power electron beams in heating and compression of a high-density plasma formed from D–T mixture grains. A report by J. Jonas entitled "Nuclear fusion in microparticles of a D–T mixture" cited research findings on beam focusing in a diode. Calculations showed current densities up to 10^7 A/cm^2 at 300 kA current and a 2.5 mm anode-to-cathode gap (2.5 MV); densities of around $5 \cdot 10^6$ A/cm^2 were obtained on the Hydra machine (1 MV). A holographic study was made of a plasma formed when a beam from an angular cathode impinged on a hemispherical anode in the diode. It was demonstrated that the plasma was distributed uniformly and symmetrically about the anode, which corresponds to symmetrical compaction of the plasma in this geometry. An experiment being planned will hopefully shed light on irradiation of a D–T mixture particle in the form of a spherule in two opposite directions (i.e., two cathodes), as a way of achieving uniform compression. The large 3 MeV, 10^{12} W EBFA accelerator, with ~20 nsec pulse duration, is being built especially for this experiment. This machine includes six pulse-shaping circuits in the form of 27-m disks commutated by specially designed multispark discharge gaps with overvoltage features, and two diodes positioned at the center of the system. Beam generation on the EBFA accelerator is scheduled for the spring of 1975.

A seminar dealing with equipment and techniques for generating ultrahigh-power electron beams covered accelerator parts and subassemblies. Some of the reports cited data on current distribution in the channels of multispark discharge gaps and some of the time characteristics of such devices. There was a report on experiments using foil type switches in work with inductive storage devices, on the development of a high-current long-pulse electron accelerator at NIIÉFA based on an inductive storage device, etc.

New results from an investigation of high-current ion beams were cited in a paper by R. Sudan (Cornell University, USA). A three-electrode gun (anode in the center) was employed to produce 2.5 kA proton current of 50 nsec duration and 300 keV energy. It is expected that 50 kA current will be generated on a larger (5 MV) accelerator. These beams will be employed in plasma heating, and injection into an Astron machine to bring about an E-layer (as reported by H. Fleischmann), and in the formation of dense ion rings in investigations of shock-waves propagating through solids, etc.

The widest variety of aspects of the problem of beam–plasma interaction came under discussion: traversal of plasma by the beam giving rise to instabilities, mechanisms of effective energy transfer from beam to plasma, acceleration of ions in the system plasma–beam–magnetic field, beam–plasma interactions in the presence of an rf field, etc. It should be pointed out that the experimental research conducted in this area is of a highly complex nature requiring development of special custom-engineered equipment and measurements procedures, and that considerable experimental material obtained to date still awaits clearcut and definitive theoretical analysis for the most part.

A seminar on acceleration of ions when high-current electron beams are passed through a gas was held over and above the school program, on the initiative of A. A. Kolomenskii (FIAN). The seminar discussions brought to light a trend toward increasing complexity in models of acceleration (two acceleration stages, treatment of effects of ion motion, etc.), since none of the models proposed earlier has been found to account satisfactorily for all of the available experimental data.

Some basic research findings obtained by Novosibirsk physicists in plasma studies were reported out. This work is being carried out generally in a variety of nontraditional approaches: experiments on plasma confinement in multiple mirror configurations, a rotating-plasma trap, high-intensity beams of monatomic hydrogen, and theoretical work.

Reviews of theoretical concepts on beam–plasma coupling were provided by prominent specialists (A. A. Rukhadze, N. Rostoker, L. I. Rudakov), and rested to a significant extent on original work done by those authors. In addition to deepening analytical research, closer attention is being given to simulation with the aid of high-capacity analog computers; the fruitfulness of this approach was demonstrated in studies of strong Langmuir turbulence,

On the whole, the work of the school demonstrated that high-power electron beams and ion beams are being used ever more widely in research on plasma physics and in other areas of science, and also in the solution of various applied problems. Generation of high-power electron beams and ion beams, the study of special features of the propagation of such beams through various media, are at this point of independent interest as a branch of science in its own right. The school provided fresh confirmation of the feasibility of such a rounded discussion on one or two of the most urgent and timely problems selected from among the many problems confronting plasma physics research.

FAMILIARIZATION TRIP BY SOVIET SPECIALISTS TO SWEDEN

A. D. Amaev

A team of Soviet specialists headed up by Academician A. P. Aleksandrov visited Sweden September 3-14, 1974. The purpose of this trip was to become better acquainted with the work being done in the field of nuclear science and nuclear engineering and industrial plants producing reactor equipment. The delegation visited the State Hydroelectric Power Authority, and also visited various nuclear power plants now in the startup stage or under construction, at Oskarshamn and Ringhals; a plant engaged in the production of heavy reactor equipment (reactor pressure vessels, steam generating units, etc.), the Uddkomb firm, the ASEA-atom design and production center, where our specialists became familiarized with a new fuel element and fuel assembly manufacturing plant for deliveries to water-cooled water-moderated boiling water reactors and heat-transfer experimental test stands; hot laboratories, the R-2 test reactor, and test stands in operation at the AB Atomenergi research center in Studsvik, the well-known Sandvik concern and its tube manufacturing plant which puts out jackets for fuel elements, as well as steam generators and reactor loops, and metallographic and metal physics laboratories.

On the request of the delegation, a meeting was organized, above and beyond the previously agreed-upon program, with a group of specialists to discuss the nuclear power station safety programs being implemented at the former Marviken nuclear power plant.

On September 12, A. P. Aleksandrov spoke at the Royal Academy of Engineering Sciences before an audience of prominent specialists in nuclear science and nuclear engineering, and heads of Swedish firms, on "A rational fuel cycle in the nuclear power industry," eliciting keen interest and numerous inquiries from the floor. This occasion also prompted a discussion in a smaller group which led to an exchange of views with Professor G. Hambreus, General Director of the Royal Academy of Engineering Sciences, with A. P. Aleksandrov delivering a brief review of problems providing a basis for possible future development of mutually profitable liaisons between the Soviet Union and Sweden. The close attention given to the Soviet delegation is accounted for by the significant contribution made by the Soviet Union in the foundation and further development of nuclear science and nuclear industry, as well as the traditionally benevolent relations prevailing between our two countries.

Sweden, as a highly developed nation, has practically everything it needs (in terms of natural resources, its own developed and tested boiling-water reactor projects involving BWR reactors, the Westinghouse Corp. PWR reactors, scientific-research facilities with highly trained engineering and scientific personnel, a high industrial potential with up-to-date equipment) for carrying out its plans for introduction of nuclear power capacity expected to reach a total level of 15,800 MW(e) by the year 1990. But Sweden is experiencing certain difficulties stemming from the resistance being put up by firms not connected with the Nuclear Industrial Group, as well as oppositional public opinion in the country on the possible harmful effects of nuclear power stations on the natural environment. These factors have been shaping scientific and engineering policy in the planning of the activities of research centers, and also in the use of the former Marviken nuclear power station for organizing full-scale highly expensive special projects within the framework of the nuclear power station safety program (verifying core cooling performance in response to scramming brought on by loop leaks or failure, service-life and endurance tests on specific pieces of equipment and subassemblies, etc.). Experiments carried out at Marviken within the framework of the nuclear safety program involve participation and financing by several other countries as well. The scientific findings resulting are the property of those countries, and can be made available to still other countries only through mutual agreement and financial compensation of costs incurred in the program.

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Taking into consideration the unstable petroleum pricing situation and the constant rise in petroleum costs, Sweden is now engaged in cooperation with other countries in a costly safety program aimed at expediting successful fulfilment of nuclear power station construction plans. The Soviet Union is also interested in establishing contacts with those participating in this program.

Highly sophisticated manufacture of fuel elements and fuel assemblies for BWR reactors, including fabrication of sintered fuel pellets, has been organized at the plant run by the ASEA-atom firm. A production line for the fabrication of fuel elements with helium pressurized to 30 atm under the canning jacket is being planned for PWR reactor equipment. This plant will not only meet the needs of Sweden's own industry, but will also be able to deliver fuel elements as an export item. The firm plans to expand fuel element manufacture. Point-of-entry monitoring of tubes for Zircaloy-2 jackets of Sandvik fuel elements will be carried out using plant equipment manufactured by Swedish industry or imported from other countries. Scrap, put at 0.2%, will be examined to update information in order to eliminate causes of scrap and improve the reliability of the fuel elements at the same time. Fuel elements designed by ASEA-atom offer some design advantages over fuel elements produced by our industry.

Operation of BWR fuel elements in the first power generating unit of the Oskarshamn nuclear power station has turned up leakage in only two out of ~30,000 fuel elements in service after 12,000 MWd/ton U burnup. Even though the burnup attained is about half of the level planned for, this result must be considered highly encouraging.

Fairly extensive research has been pursued both at the Studsvik research center and at the Sandvik plant, and this includes reactor investigations involving studies of the effect of heat treatment and machining on the mechanical properties of reactor materials and studies of the structural stability of zircaloy tubing and fuel-element cans.

It is worth pointing out that the Sandvik firm is placing great hopes on utilization of the steel grade 12R72HV (15% Cr, 15% Ni, 1.8% Mn, 1.2% Mo, 0.4% Ti, 0.006% B) which the firm itself developed, for the fabrication of cans of fuel elements in fast reactors. Extensive studies have been carried out on creep behavior, long-term strength in the 550-750°C temperature range over a 50,000-h time span, a set of in-pile tests demonstrating the superiority of this grade of steel over the American grade 347 steel devised for similar applications. The positive effect brought about by additions of titanium and boron on high-temperature long-term mechanical properties is now evident.

Further investigations on materials for the cans of fuel elements designed for thermal reactors are being carried out from the standpoint of nuclear power plant safety. A lot of work is being done in the field of reactor materials studies along the lines of joint programs involving several participating countries (Belgium, Norway, Britain, FRG, France, and Italy).

A relatively small-size plant for the production of heavy reactor equipment, operated by the Uddkomb AB firm, is fabricating as many as seven reactor pressure vessels per year to cover Sweden's needs of one or two pressure vessels per year with the remainder available for export. This plant is operated mainly as a source of goods for export (to FRG, France, other Scandinavian countries), and is equipped with highly productive and very sophisticated outsize equipment for all types of machining and welding operations and inspection of welded joints. The plant is engaged in a program of cooperation with other foreign firms which serve as vendors for forgings and partially finished fuel-element jackets and shell type products. The plant handles such operations as machining, hardfacing, welding, inspection, and inspection of BWR and PWR reactor pressure vessels and steam generators for 1500 MW(e) nuclear power stations. Reactor pressure vessels are being fabricated from American steel grades A553-B (for BWR reactors) and A508-B (for PWR reactors). Special attention is being given to inspection techniques, and costs for inspection work total 10%. Uddkomb AB intends to expand its plant capacity to keep up with long-term orders already in and in view of the steady interest shown in its products by clients.

Here we may point out that the method currently in use at the Uddkomb plant of applying hardfacing by means of wire electrodes is inferior to the strip hardfacing method now in use in the Soviet Union. The Uddkomb firm is now interested in placing orders for reactor pressure vessels manufactured in the Soviet Union.

In 1967, Sweden, Denmark, Norway, and Finland pooled their efforts in an attempt to design a reinforced-concrete pressure vessel for use in a water-cooled water-moderated boiling-water (BWR type) reactor, and in 1973 Britain and France joined in financing this program (without any direct participation otherwise). The "Scandinavian program," as it is known, embraces the solution of problems associated

with the strength, design of the vessel cover sealing, thermal insulation, monitoring and safety of such reinforced-concrete pressure vessels in reactor design. The design of a suitable reinforced-concrete pressure vessel will be completed in 1975, after which steps can be taken to proceed to installation of a reinforced-concrete structure in a 900 MW boiling-water reactor facility.

The heads of the ASEA-atom and AB Atomenergi firms displayed intense interest in the work being done in the USSR on the development of a boiling-water reactor with a reinforced-concrete pressure vessel.

NEW BOOKS FROM ATOMIZDAT

(FOURTH QUARTER, 1974)

N. G. Gusev, P. M. Rubtsov, V. V. Kovalenko, and V. M. Kolobashkin

Radiation Characteristics of Fission Products; a Reference Handbook*

Reviewed by A. A. Moiseev

This reference manual cites quantitative data on the radiation characteristics of fission products formed in nuclear reactors and in instantaneous fission events. These data include independent yields of fission products of ^{233}U, ^{235}U, ^{238}U, ^{239}Pu, ^{232}Th bombarded by thermal neutrons, fast neutrons ($E_n = 14$ MeV), and fission spectrum neutrons; total and differential activities and γ-ray equivalents of fission products and of mixtures of fission products; energy release by γ-photons per unit dose rate; effective dose and energy spectral composition of γ-ray emission by a mixture of fission products. Data on the buildup of ^{134}Cs in nuclear reactors are also tabulated.

The integrated radiation characteristics of a mixture of krypton and xenon after these escape from fuel elements will be of great interest to designers and engineering physicists engaged in calculating doses from a cloud of inert radioactive gases.

Data are published for the first time on the radiation characteristics of a mixture of fission products from a reactor operated with shutdowns (shutdowns for partial refueling and for preventive maintenance) and also from reactors operated in a critical-assembly mode.

The empirical formulas cited in the appendices and the required coefficients for determining total activity [Q(T, t), Ci/kW] and energy release by γ-photons [S(T, t), MeV/sec·kW] of a mixture of nuclear reactor fission products as a function of the reactor campaign T and exposure time t will also be of great practical interest. These relationships are analogs of the familiar Way–Wigner formula for a mixture of instantaneous-fission products.

We can make a special point of the originality of virtually all of the material in the handbook. There is no doubt that this will be a desk-top reference book for designers and production experts at nuclear power stations, spent fuel reprocessing and recovery plants, for physicists and engineers concerned with protection and shielding against ionizing radiations and radiation safety, and for students and instructors in physics and engineering colleges.

We can express some regret that the handbook has not space for data on the differential specific activities and γ-equivalents of radionuclides with a very short half-life. Data are cited for differential characteristics only in the case of four campaigns (the most important ones, to be sure), and on total characteristics we have appropriate values for nine reactor campaigns. This presents certain difficulties when the task is to find solutions for some radiation safety problems.

Among the shortcomings of this reference manual, we can mention the fact that the authors resort to the term "isotope" (or "radioisotope") instead of "nuclide" (or "radionuclide"), even though the former

*Atomizdat, Moscow (1974), 224 pp.

Translated from Atomnaya Énergiya, Vol. 38, No. 1, pp. 63-64, January, 1975.

term can be correctly used only in those cases where we are referring to atoms of the same chemical element. If we have in mind atoms of different chemical elements, then it would be more correct to hold to the term "nuclide."

* * *

Yu. V. Kuznetsov, V. N. Shchebetkovskii, and A. G. Trusov

Fundamentals of Cleanup of Radioactive Contaminants from Water (edited by V. M. Vdovenko)*

Reviewed by D. S. Gol'dshtein

The appearance of this book in print is an important event for all those involved to some extent with removal of radioactive contaminants, and particularly with cleanup of contaminated water and effluents. The first edition appeared back in 1968 with the title Fundamentals of Water Deactivation.

The authors have done a good deal of work on their monograph, and have perused new publications in the field as well as their own research findings. The first and fourth chapters of the book are those that have been enlarged the most.

Extensive factual material is presented in 168 tables of data and 56 diagrams and illustrations. The book contains an impressive bibliography (with 986 references).

An interesting feature of this monograph is its approach to the problem of water cleanup to get rid of radioactive contaminants from the vantage point of concepts relating to the physicochemical state of the radioactive isotopes present in aqueous media, and the behavior of those isotopes in the sorption systems under study. The authors convincingly demonstrate the serious value of information on the modes of existence of radioactive isotopes in aqueous media for solution of practical water cleanup problems. The first chapter reserves considerable space for a thorough airing of this information. The first chapter also covers possible sources of radioactive isotopes in natural bodies of water, and processes involving interaction between radioactive isotopes and inorganic or organic components of natural waters. The chapter ends with a prognosis, the first of its kind in the literature, of the probable physicochemical state of radioactive isotopes in surface waters on land, in the seas and oceans.

The second chapter centers on the basic techniques of water treatment and water management, and their effectiveness when used to remove radioactive contaminants from water. In addition to a description of traditional methods at a contemporary level of sophistication, the text presents concepts relating to the physicochemical essentials of processes involving coprecipitation of trace amounts of radioactive elements with hydroxy-hydrate collectors. The chapter ends with information cited on new water cleanup processes for use at water conduit stations in our country and in other countries.

The third chapter offers a concise but content-packed description of other-than-basic methods in water treatment and water preparation: soda-lime softening and ion exchange, phosphate coagulation, electrodialysis, distillation, flotation, biological cleanup, and so on. It is of the utmost importance that the authors cover not only the better known methods, but also methods which have only begun to be accepted in practice (back osmosis, electroflotation, electrocoagulation, desalination of brines by freezing out, solvent extraction techniques, etc.), and the promise each of these methods appears to hold for practical applications, and the cost feasibility of the methods, are estimated.

The fourth, and most capacious, chapter cites extensive information on the use of sorbents of the most widely varied classes in cleanup flowsheets for removal of radioactive contaminants from water and effluents. Here we have to take note of the successful (in our view) arrangement of factual material in this chapter: the description starts off with presentation of basic information on the structure of each particular sorbent, its ion-exchange capacity, and the reasons governing the range of practical application of the sorbent, and ends with arrays of factual data on the effectiveness of using the sorbents in particular flowsheets.

*Atomizdat, Moscow (1974).

One very obvious advantage of this monograph is the conciseness of presentation of the enormous wealth of factual material that has been accumulating up to the present writing. We should also point out that all of the material is systematized and reviewed in a successful manner, and that each chapter ends with a critical resume of the situation by the authors. The monograph can be approached as a valuable scientific text and reference manual for specialists concerned with cleanup of water to remove every conceivable type of (radioactive) contaminants. The authors' own data, obtained in the course of many years of research on processes designed to remove radioactive contaminants from water, will be of unquestionable interest to readers and users of the reference.

The value of the book is also evident in the close attention given to current problems pertaining to protection of the natural environment from radioactive contaminants emanating from whatever sources.

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