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ARTICLES

ICEBREAKER "ARCTIC" — A NEW ACHIEVEMENT OF SOVIET NUCLEAR SHIPBUILDING

F. M. Mitenkov, B. G. Pologikh, A. K. Sledzyuk, and N. S. Khlopkin

UDC 621.039.578:621.12

The nuclear icebreakers built in the Soviet Union represent a successful experiment in the use of nuclear power plants for ships. Nuclear power offers the greatest advantages when a ship must remain at sea for long periods of time without returning to port. Now that nuclear power for ships has demonstrated its reliability, the question of the extent of its use in the marine fleet is determined only by economics.

Many years of experience in the operation of the nuclear ship "Lenin" showed that the construction of icebreakers is economically feasible particularly with commercially available power plants and a common shore base for maintenance. This is even truer for the more powerful icebreakers needed for prolonging the Arctic navigation period and for increasing the rate of ship passage under heavy ice conditions. A marked improvement in icebreaking capability is only achieved by a significant rise in the power of icebreakers.

The new nuclear icebreaker "Arctic" was laid down on July 3, 1971 at the Baltic plant in Leningrad. Its turboelectric propulsion plant generates 75,000 hp and its displacement is about 21,000 tons. The icebreaker went into the water on December 26, 1972, and construction was finished and sea trials completed in 1974. In the design and construction of the icebreaker, the requirements of the Convention on the Preservation of Human Life at Sea of 1960 relating to nuclear ships were taken into account. The nuclear ship was constructed under the supervision, and in accordance with the rules, of the Registry of the USSR. These measures in combination with the experience acquired in the building and operation of the icebreaker "Lenin" ensured high reliability and safety both on board the "Arctic" and in the environment.

A steam-producing plant is the source of power on the icebreaker. The steam produced by it is delivered to two main turbines which drive ac generators. The current is delivered to three electric motors for driving the propellers after passing through rectifiers. Each of the two nuclear units of the steam-producing plant include a pressurized-water reactor, four steam generators, four main circulating groups, volume compensators, chillers, auxiliary pumps, and filters. The reactors of the nuclear ship "Arctic" are similar to the reactors on the nuclear ship "Lenin" [1]. They are located in separate gas-tight spaces, the so-called reactor compartments, which are behind a reliable biological shield. Above the reactor compartments there are equipment rooms containing equipment not requiring constant maintenance. When necessary, entrance to these rooms is possible through hermetically sealed doors.

The primary loop of the steam-generating plant, which contains radioactive coolant, is completely welded. Hermetic seals around the drives for control units, circulating pumps, and coolant cleanup systems make it possible to reduce to a minimum the escape of liquid and gaseous radioactive products. A certain amount of them appears only when sampling the water in the primary loop, which is done to check the state of the coolant. In this case, the water samples are placed in hermetically sealed montejus in the biological shield beneath the reactors; they are stored there until the icebreaker returns from a voyage. There is no need for frequent sampling of the primary loop, because the chemical regime of the primary loop is self-maintaining and does not require interference by the crew during navigation. When accidental leaks appear because of equipment failure (for example, leaks in the steam generators), they are rapidly detected and the defective equipment is cut off by a double shut-off device. From experience in the operation of the icebreaker "Lenin," the volume of water from the primary loop which is stored is small and does not exceed 1-2 m^3 per trip.

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The ventilating system which supplies the necessary removal of air in hazardous spaces operates on an open cycle with discharge of the exhaust air at the highest portion of the ship — the mainmast. The discharged air is carefully monitored for the presence of radioactive gases and aerosols by means of dosimetric instruments. The ventilating system is converted to a closed cycle of operation in the case of leakage of radioactive materials from the primary loops and an increase in air activity to an assigned control value. In this situation, all air passes through filters and conditioners.

A thoroughly checked system of radiation monitoring is used on the icebreaker "Arctic" [2]. It handles a large group of problems. The state of the coolant and the integrity of fuel-element cladding are evaluated by means of a spectrometric detector adjusted for the reference isotopes $^{88}Kr + ^{88}Rb$, a gamma-ray detector, and also a delayed-neutron detector. Several β detectors monitor the integrity of various equipment, of each steam generator, and of all steam generators together; several γ detectors monitor the integrity of auxiliary heat exchangers. Beta detectors installed in the ventilation ducts monitor the discharge of gases and aerosols. Discharge waters are monitored with a γ-ray scintillation spectrometer. The radiation environment in a number of spaces in the central compartment is evaluated with γ and neutron detectors. Portable and fixed β detectors monitor contamination of body and clothing and surface contamination in the spaces. The information obtained by the radiation monitoring detectors is centrally handled and displayed on a panel. Analysis of this information makes it possible to predict the radiation environment and to take timely measures for re-establishment of normal conditions and for protection of personnel. Among such measures are the disconnection of portions of the power plant, conversion of ventilation to the closed cycle of operation, the introduction of filters for purification, the use of measures for personnel protection, the reduction of personnel residence time in spaces with increased background, and decontamination.

All personnel doing maintenance on steam-generating equipment wear personnel dosimeters for monitoring external radiation; in addition, equipment is provided for "lifetime" determination of the content of radionuclides in the body.

The icebreaker is reliably provided with electric power from two independent power stations located in separate compartments; the power from each can satisfy all demands under normal and accident conditions. The forward power station has two 2000-kW turbogenerators and the aft power station has three. In addition, there is a reserve diesel generator located in the aft power-station compartment. Its power is sufficient to cool the steam-generating equipment and to provide for other needs of the ship if both power stations should suffer a breakdown. The reserve diesel generator is automatically cut in whenever any of the power stations cut out. Two emergency diesel generators are also provided which are automatically started up when any of the reactors are scrammed. The emergency diesel generators can provide electrical power for the most important requirements of the ship including the means for cooling equipment with complete cutoff of both power stations.

The power plant of the icebreaker is completely automated. This makes it possible to reduce the number of maintenance personnel, and to decrease operating costs while simultaneously improving the characteristics of the equipment, particularly in the transition and emergency modes. Automatic and remote control is provided for the steam-generating plants, power stations, electromechanical equipment, and the electric motors driving the propellers together with interconnections between these systems.

A central control computer was installed on the icebreaker for the collection and analysis of information about the operation of technical equipment. The most important parameters of the reactor equipment, the main and auxiliary turbogenerators, the electric motors for propulsion, and general ship systems are monitored continuously and the remaining parameters monitored cyclically with the production of warning and emergency lights and audible signals when necessary. In accordance with a set program, the computer automatically records the most important parameters and displays their values on a printer. All parameters which exceed standard limits are also recorded automatically. The machine follows the logic of illuminated mnemonic diagrams which make it possible for the operator to monitor the operation of equipment and to issue commands. The need for stability is ensured by a structure which provides the formation and transmission of information over two independent systems.

In 1974, individual types of equipment were first tested on test stands and in the ship, and then combined tests were made of the equipment as a whole. The combined tests were made on test stands at the Baltic plant in Leningrad. Sea trials were held in the Baltic Sea and ice tests in the Kara Sea.

At the time of the tests, parameters characterizing the state of the equipment and various equipment operating modes were measured with the regular monitoring system with computer recording of data and also with special measuring equipment — a multichannel magnetograph which made it possible to reproduce repeatedly the data obtained immediately after the tests and in the necessary time scale and to compare actual data with calculated data operationally. This made it possible to judge whether the parameters studied went beyond the limits established in the plans. The tests of the steam-generating equipment and of the ship were performed successfully and were completed in a short time.

Design characteristics were confirmed in the course of the tests and convincing data were obtained which pointed to satisfactory functioning of the equipment under normal conditions and simulated emergency conditions. Equipment ensuring safety of the installation was checked particularly carefully. The tests confirmed its high reliability. The biological shield also met all requirements. The steam-generating equipment provided the required output of steam at a given flow and pressure and was sufficiently flexible for maneuvering capability. The various modes of operation of the steam-generating equipment were thoroughly investigated both when separated from the steam turbine equipment and with interconnected controls. With interconnected controls, the reactor power automatically followed the power required by all ship mechanisms and determined by the flow of feed water through the steam generators for the maintenance of steam pressure in the main steam piping.

These tests were also completed successfully and in a short time. The Marine Ministry of the USSR placed the ship into operation. Our icebreaker fleet was supplemented with a new nuclear ship, the most powerful of the civilian ships in the world. Putting it into operation expanded the ability of our country to conquer the northern areas and the Arctic regions.

LITERATURE CITED

1. F. M. Mitenkov et al., "New nuclear plant of the icebreaker Lenin," IV Geneva Conference, 1971, Paper 722.
2. V. S. Zhernov et al., "System for radiation monitoring of nuclear power plants with water cooling," IV Geneva Conference, 1971, Paper 687.

RADIATION SAFETY IN THE OPERATION OF NUCLEAR POWER STATIONS

A. I. Burnazyan

UDC 621.039.58

During the entire operating lifetime of a nuclear power station, questions involving the provision of radiation safety for workers, for the population living near the site, and for the environment are subjects for close attention and thorough systematic monitoring by the USSR Ministry of Health.

Scientific studies and systematic health-protection observations by members of State Health Protection Groups at the first atomic power stations made it possible to justify the need for carrying out a number of planning, organizational, health-engineering, and other measures for the prevention and elimination of the harmful effects of radiation on the health of workers and the population in the surrounding area. The results of this work was subsequently the basis for the "Health Rules for the Design of Nuclear Power Stations" that had to be satisfied by all organizations, institutions, and establishments participating in the design, construction, or operation of nuclear power stations.

In accordance with the rules mentioned, the production areas of a nuclear power station are divided into a free zone, in which the possibility of effects from ionizing radiation on personnel is eliminated, and a control zone, where effects on personnel from radiation-harmful factors are possible (external α, β, γ, and neutron radiation, contamination of air by radioactive gases and aerosols, contamination of surfaces and equipment by radioactive materials).

Personnel enter the control zone through health passages and are dressed in special work clothing which eliminates the possibility of contamination of regular clothing by radioactive materials. In the free zone, personnel work in their own clothing or use ordinary work clothes without a complete change.

The results of careful dosimetric and radiometric monitoring and also of systematic scientific studies demonstrate that the radiation environment in the free zones of all operating nuclear power stations is completely satisfactory and practically no different from the working areas of other non-nuclear industrial plants.

In the initial operating period of the first units of the Novovoronezh (NNPS) and Beloyarsk (BNPS) nuclear power stations, increased radiation levels in comparison with the calculated levels were noted in particular portions of the control zone which were connected with adjustment of equipment and development of technology. Measures for the shielding, rebuilding, and improvement of individual units, assemblies, and system were intensified. In particular, the installation for special water purification and the system for storage of radioactive gases of the first unit of NNPS were reconstructed and improved, following which working conditions for personnel were considerably upgraded. The radiation environment at the first unit of NNPS was always poorer than that at subsequent units. This resulted from the increased volume of construction and repair work performed on equipment of the primary loop and was connected with individual structural deficiencies of the main circulating pumps (MCP), of the drives for the control and protection systems (CPS), of the main shutoff valves (MSV), and of the upper unit of the equipment. The

TABLE 1. Annual Radiation Doses for Personnel at Novovoronezh and Kolsk Nuclear Power Stations

Station	Year	Fraction of controlled personnel receiving dose, rem/yr			
		0–1,0	1,1–2,5	2,6–5	> 5
Novovoronezh	1972	76	14	10	—
	1973	79	13,7	7,3	—
	1974	72	18	10	—
Kolsk	1974	99,9	0,1	—	—

Deputy Minister of Health, USSR. Translated from Atomnaya Énergiya, Vol. 39, No. 3, pp. 167-172, September, 1975. Original article submitted March 17, 1975.

greatest radiation dose was recorded when performing repair work on equipment and piping of the primary loop; however, the maximum annual dose did not exceed permissible values.

In the initial operating period of the BNPS, failure of the integrity of fuel-element cladding and the entrance of fission activity into the graphite pile of the reactor was observed, which hindered maintenance work on the CPS boxes located in subreactor spaces of the compartments for disconnect equipment. The γ-ray dose rate in these spaces exceeded permissible levels, which necessitated limitation on stay time for repair personnel.

Because of the buildup of radioactive corrosion products on internal surfaces of equipment and piping, the γ-ray dose rates in the turbine hall gradually increased as well as those in the neighborhood of disconnect equipment and other units. Decontamination performed on the first unit of BNPS by a specially developed technique considerably reduced the γ-ray dose rate. It was established that it was necessary to wash and decontaminate systematically (approximately once a year) the internal surfaces of piping and pipe fittings in order to reduce, and prevent, irradiation of workers at the first and second units of BNPS.

On the whole, the radiation environment and working conditions at nuclear power stations are satisfactory. One of the criteria for the radiation environment is the type and level of radiation of personnel at nuclear power stations (see Table 1).

The dose distribution shown for MNPS, where repair personnel were the same for all four units, resulted mainly from repair work on the first units. The radiation doses for personnel working on the third and fourth units of this station were approximately the same as those at the Kolsk nuclear power station.

Operational results for the third and fourth units at NNPS and at the first and second units of the Kolsk nuclear power station demonstrate the considerably greater reliability of equipment and technical systems of the commercial VVER-440 units. These units are being designed and built for a number of nuclear power stations in the USSR and abroad (in East Germany, Hungary, Czechoslovakia. Bulgaria, Rumania, and Finland) and give no cause for comment with respect to radiation safety. The radiation environment in the periodically maintained, and even unmaintained, spaces of the control zone in these units is satisfactory and the radiation levels are considerably below permissible values in the majority of cases. The use of improved technical equipment with longer periods between repairs markedly reduces the amount of repair work in the control zone, which significantly improves working conditions for repair personnel.

Along with the improvement in the construction of the basic technical equipment and the increase in its operational reliability, high-quality fuel elements mainly determine the radiation environment at units with VVER-440 reactors. The reliability of fuel-element cladding integrity during an operating period practically eliminates the entrance of fission products into the coolant of the primary loop and the nature of the radiation in the production areas of the control zone of these units will result mainly from induced activity from impurities in loop water and from corrosion products, which are rapidly decaying isotopes with short half-lives having no harmful effect on personnel doing repair work. At the present time, with normal operation of a nuclear power station having VVER-440 reactors, working conditions for workers in the control zone completely meet the requirements of existing health regulations and radiation-safety standards.

The results of periodic compulsory annual medical examinations of people working at nuclear power stations demonstrate the absence of not only occupational diseases but even of fuctional deviations from the norm. Not a single case of occupational disease of workers from the effects of radiation has appeared in ten years of operation of high-power nuclear power stations. The health of people working in control zones does not differ from the health of workers not exposed to radiation effects.

Despite the comparatively favorable situation at all operating nuclear power stations in the world, reports have often appeared in the foreign press in recent times on the need for revision of the circulation piping in nuclear power stations. This is because in the USA cracks were observed in large pipes of stainless steel 6 mm and more in thickness at three pressurized-water reactors (like the VVER). The development of such cracks and their propagation through the entire thickness of the pipe wall can lead to a radiation accident. In such a case, all necessary measures are provided at nuclear power stations in our country that would localize escaping radioactive material within the volume of hermetically sealed spaces and would not permit an increased discharge outside the limits of the nuclear power station. Since this kind of accident is extremely undesirable, it is necessary to develop and introduce at all nuclear power stations methods for monitoring equipment and piping, standards for the scheduling of such monitoring, and protective measures which would prevent the possible creation and propagation of cracks and other defects that could lead to a radiation accident.

The development of nuclear power basically depends on a solution of the problem of protecting the environment against contamination by radioactive materials and the problem of ensuring the radiation safety of the population in the surrounding area. These problems acquire special importance because of the resolution of the Central Committee of the Communist Party of the Soviet Union and of the Council of Ministers of the USSR "On Intensification of the Preservation of Nature and the Improvement of the Use of Natural Resources."

To ensure radiation safety for the environment and population, the following requirements are put forward in existing health regulations for planning nuclear power stations:

satisfaction of the basic health rules for the location of a nuclear power station as a commercial unit;

effective decontamination of radioactive wastes and their disposal in the environment in accordance with existing health requirements;

monitoring of the conditions for decontamination and disposal of radioactive wastes and of their effect on the state of the environment.

Selection of a site for the location of a nuclear power station on a health basis is of exceptional importance in the creation of safe operating conditions. A nuclear power station, as far as possible, should be located in an area of low population with good natural ventilation of the area toward the leeward side with respect to populated areas. The selection of a site in a basin or in an area with slight air movement and with a large number of calms is not recommended. The hydrogeologic conditions of the area should eliminate the possibility of radioactive materials entering ground water. The highest level of standing ground water must be not less than 1.5 m below the outline of a tracer field from the proposed structures produced by radioactive methods.

The location of a nuclear power station must admit the possibility of arranging a health-protection zone around the station, the size of which is established separately in each specific case in conjunction with units of the State Health Protection Group depending on local climatic, meteorological, and topographic conditions; the type, construction, and power of the reactor; the calculated amount of radioactive discharge into the atmosphere and its surface concentration, etc. In the health-protection zone, one forbids continuous residence by individuals, the location of health establishments and food-processing plants; but one allows the construction of subsidiary and auxiliary structures and buildings for the nuclear power station, the growth of agricultural products, pasturage of cattle, etc. subject to compulsory radiometric monitoring of this area and of the products produced in it.

In the operation of a nuclear power station, various radioactive wastes are formed which are the main source for the entrance of radioactive materials into the environment. Therefore, effective decontamination of radioactive wastes and their disposal into the environment in accordance with the appropriate health requirements are definitive conditions for the provision of a normal radiation environment in the area of a nuclear power station. In the absence of special measures, radioactive wastes can be potential sources of radioactive contamination of the air, soil, plants, open sources of water supply, ground waters, etc.

In the decontamination and disposal of radioactive wastes, one should not allow gross discharge of radioactive materials into the atmosphere above maximum permissible levels which are regulated by the health rules for designing a nuclear power station, should forbid the discharge of waste waters, except for drain water, into storage lagoons even when the content of radioactive materials in them is not above established standards for water, and should eliminate the possibility of contamination of the nearby areas and ground waters during transportation and storage of wastes.

We briefly consider the measures which are provided in planning and employed during the operation of a nuclear power station in order to satisfy the specified requirements. Air contaminated by radioactive materials, depending on the degree and nature of the contamination, is directed into an independent special ventilation system and undergoes purification before discharge into the atmosphere. Filters of fine-fiber FPP cloth (with a polyvinylchloride base) with a trapping factor of 99.9% are used for the removal of aerosols. Removal of iodine is accomplished with charcoal filters having a removal factor of 99.9% and higher. If necessary, the efficiency for removal of radioactive materials from the air is increased by the inclusion of additional stages in the flow using the same filters. Radioactive noble gases are removed by natural decay through storage in gas holders the operating efficiencies of which depend on the time the gases remain in them, i.e., on the volume of the gas holders. The air is filtered with aerosol filters in order to remove aerosols formed by decay of the gases.

At new nuclear power stations, there are in addition to the gas holders for decontamination of gases the so-called devices for suppression of activity (DSA), which are based on the adsorption of gases by activated charcoal with subsequent radioactive decay. The operating efficiency of a DSA depends on the amount of loading and the temperature of the medium. The rate of gas adsorption by charcoal increases as the temperature decreases. As shown by many years of experience in the operation of nuclear power stations, the existing gas cleanup systems consisting of aerosol and charcoal filters and gas holders satisfy the health requirements for the decontamination and removal of gaseous and aerosol wastes from a nuclear power station; the actual discharges from operating nuclear power stations are evidence of this. In ten years of nuclear power station operation, not a single case has been recorded where established standards for permissible discharges of radioactive materials into the atmosphere have been exceeded. The actual rates were tens and hundreds of times below the maximum permissible values. Ventilation exhausts are discharged through stacks 100-150 m high for better dissipation in the atmosphere. In the area of operating nuclear power stations, the atmospheric content of radioactive materials is practically at the level of the natural background resulting from global fallout.

All radioactive waste water at a nuclear power station is collected in a separate sewer system and is purified. Evaporation and ion exchange are the main methods used for purification and provide a reduction in activity of 2-4 orders of magnitude depending on the original amount. The concentration of radioactive materials in waste waters is below the mean annual permissible concentration for water. When necessary, purification efficiency can be increased by repeated purification in the same equipment. After purification, radioactive waste water is mainly used repeatedly in the return water supply for loop makeup and for washing and decontaminating rooms and equipment. However, waste waters from special laundries, health passages, and laboratories cannot always be used over and over. These drain waters with an activity not exceeding permissible values are discharged into industrial waste lines or directly into a lagoon. Such discharge is not in conflict with existing health legislation.

The soil method for decontamination of drain waters, which was successfully used at NNPS, deserves serious consideration. All drain waste waters are discharged into a special filtration field, in the surface layer of which the main portion of the radioactive materials are held. In the drainage waters from the filtration field, the content of radioactive materials does not exceed natural levels, which is evidence of the reliability of this method of purification. The surface layer of soil (thickness 1 cm) in such fields is taken up annually and removed for storage with other radioactive wastes from the station. This method can be used generally since it does not require large ground areas because of the insignificant amount of debalance water and completely eliminates discharge of contaminated waste water into a lagoon.

Under favorable hydrogeologic conditions underground storage of liquid radioactive wastes is used; this has been done at the Ul'yanovsk nuclear power station in Dimitrovgrad for more than eight years. Since 1966, the main portion of the liquid wastes has been discharged into deep underground strata which are not used for agricultural purposes and which are reliably isolated from higher water-bearing strata. During the entire period of operation of an underground storage test field, there was observed no harmful effect of the discharged waste waters on the health conditions of higher strata used for agricultural needs, including food. In our opinion, great attention should be given to this method for the storage of radioactive liquid wastes when favorable hydrogeologic conditions are present, of course.

A possible means for the entrance of radioactive materials into open water supplies, besides drain waters, is the technical water supply system intended for cooling turbine condensers and other equipment. However, the scheme for technical water supply was developed with a design that completely eliminated the possibility of radioactive materials entering the cooling water. At a nuclear power station with VVER reactors, this is achieved by means of two hermetically isolated closed loops. The technical water supply system forms a third loop, the radioactive contamination of which is impossible under the conditions specified. The possibility of radioactive contamination of process water is also prevented by an increase in the pressure of process water to a value higher than the pressure in the turbine condensers.

At a nuclear power station, therefore, only drain water with an activity not exceeding permissible values can be discharged into open water supplies. Such a discharge cannot give rise to radioactive contamination of the water in open water supplies to levels above the natural background characteristic of the given locality as is confirmed by many years of monitoring in the areas around operating nuclear power stations. Because of the use of boron regulation in reactors of the VVER type, tritium is detected in the coolant of the primary loop and the tritium content may become significant in time. Because of this, it is necessary to prohibit the discharge into water supplies of waste waters formed as the result of water leaks in the primary loop.

In the clean-up of waste water contaminated by radioactive materials, various slurries, tank residues from evaporation, and waste ion-exchange materials are formed. These wastes have activities of the order of 10^{-4} to 10^{-1} Ci/kg and are stored in special vaults. The construction of the vaults eliminates the entrance of radioactive materials into ground waters. To accomplish this, tanks are made of stainless steel either in the form of a "can in a can" type or as a single tank having a pan for the collection of possible leaks and subsequent transfer to reserve tanks.

The plans for new nuclear power stations provide for consolidation of tank wastes from evaporators by bituminization with storage of the bituminous waste in vaults of simple and inexpensive construction. This practice will show how promising this method is for extensive commercial application.

Depending on levels of specific activity, solid wastes are sent for burial either in special ferroconcrete tanks or in ordinary earth trenches. Wastes of high specific activity are placed in the tanks and wastes of low specific activity in the trenches. When hydrogeologic conditions are unfavorable, the earth trenches are hydraulically insulated in order to prevent leaching of radioactive materials and their transport from the waste to the ground water. Disposal of solid wastes of low specific activity presents great difficulty because of the tremendous amount of waste. The introduction of compacting should play a large role in the solution of this problem.

Careful radiometric monitoring at decontamination and disposal of radioactive wastes was set up at nuclear power stations. Continuous monitoring of discharges of radioactive gases into the atmosphere is supplemented by spectrometric or radiochemical determinations of the most radiotoxic isotopes (^{90}Sr, ^{89}Sr, ^{131}I, ^{137}Cs). Air samples for these studies are taken from the central ventilating stack before discharge into the atmosphere. Monitoring of debalanced water is accomplished in intermediate tanks, discharge from which is permitted only in accordance with the results of the measurements. In the common sewer, monitoring of the composition of waste water is performed before discharge into the lagoon. Possible leaks from storage vaults for liquid wastes of high specific activity are discovered by means of instruments installed in the vault sumps and by investigation of ground water from test wells located in the storage area. Reserve tanks are provided in the case of possible loss of tank integrity.

Special external dosimetry services at nuclear power stations and units of the State Health Protection Groups of the USSR Ministry of Health are studying the effect of station operation on the state of the environment and are monitoring the radiation environment around the site of nuclear power stations.

An area out to a radius of 10-12 km from the nuclear power station is under observation. For comparison of data obtained in the area of obsevation, a control (background) point is established which is to the windward of the nuclear power station and outside the influence of its discharges. In addition, before a nuclear power station goes into operation, the natural radiation background, which results mainly from global fallout, is recorded for comparison with actual data obtained during operation of the nuclear power station and for an objective evaluation of the effect of nuclear power station operation on the change in radiation environment in the surrounding area.

The area under observation is divided into several radial bands. Monitoring of the radiation environment is carried out with samples collected from various elements of the environment — air, soil, plants, local agricultural products, snow, water from open water supplies, hydrobionts, etc. — at specially selected points. At the same time, the γ-ray dose rate at the locality is noted.

The results of systematic monitoring indicate that for the entire period of operation of existing nuclear power stations there has been no change in the radiation environment around station sites as compared with conditions in the period before startup. This is evidence that the present set of health and technical measures ensure safe operation of a nuclear power station both for the staff and people in the surrounding area.

It is necessary to carry out further studies which would facilitate an increase in the reliability of nuclear power station operation because of an increase in the specific power of station units, the tendency toward location of nuclear power stations in the immediate vicinity of power consumers and in areas of high population density, the use of nuclear reactors for central heating and hot-water supply in large cities, and other needs of the national economy.

In this plan, measures have been developed for localization of various accidents, including the maximum permissible accident. At a nuclear power station, there are specific plans for protection of staff and population in case an accident occurs. These anti-accident measures are directed toward reduction of accident hazards to a minimum and toward elimination of the consequences of an accident.

COMPARISON OF RADIATION HAZARD FROM FISSION AND FUSION REACTORS

Yu. V. Sivintsev

UDC 621.039.58

An attempt was made in the paper by Hafele and Starr* to compare the advantages and disadvantages of two types of nuclear devices for the production of electrical power: fast reactors based on the fission of heavy nuclei (FIR) and thermonuclear reactors based on the fusion of light nuclei (FUR).

In a comparison of FIR and FUR, a decisive factor is the estimate of the radiation hazard for staff and population and the potential effect on the environment. However, the approach and results of the analysis described in the paper cited call for several critical remarks. This is all the more necessary because a reprint of the Hafele and Starr paper is being prepared as an official report of the International Institute of Applied Systems Analysis (IASA).

This paper discusses the text of the third version of the report distributed by IASA. The remarks relate formally only to Sec. III in which the questions of radiation hazard and environmental protection are discussed.

Statement of Purpose and Area of Investigation. The problem of comparison of the radiation hazards of FIR and FUR is formulated correctly in the report. However, the authors confine themselves to an analysis of only a single aspect — exposure of the population under accident conditions and as the result of disposal of wastes. These important questions do not encompass the entire problem. Therefore, one should refine the formulation of the purpose and expand the area of investigation for such a comparison. Obviously, the problem in one of the corresponding sections of the new text of the report is more advantageously formulated in the following manner: "Quantitative evaluation of the degree of radiation hazard from FIR and FUR (of identical thermal power)."

In accordance with the basic concepts developed by the International Commission on Radiation Protection (ICRP) [1] and the analogous national commission (NCRP) [2], the radiation hazard for three population groups — occupational workers (A), individuals in the general population (B), and the general popula- (C) — is subject to evaluation for such a comparison. In the appropriate calculations, the dose from external and internal radiation should be evaluated.

*W. Hafele and C. Starr, J. Brit. Nucl. Energy Soc., 13, 131 (1974).

TABLE 1. Long-Lived Radioactive Isotopes (Half-Lives > 10 yr) in FUR and FIR [3]

Reactor	Radioactive	$T_{1/2}$, yr	Production of radioactive isotopes		MPC in water, μCi/liter	Hazard factor, km^3/MW (t)
			Ci/MW (t)·yr	Ci/MW(t) in 10^3 yr		
FUR	^{93m}Nb	19,6	8 800	173 000	$4 \cdot 10^{-4}$	0,4
	^{94}Nb	$2,9 \cdot 10^4$	2,9	2 900	$3 \cdot 10^{-6}$	1,0
	V (containing Nb)	—	—	—	—	$1,4 \cdot 10^{-3}$—10^{-4}
FIR	^{90}Sr	40,4	670	27 000	$3 \cdot 10^{-7}$	90
	^{137}Cs	43,4	930	40 000	$2 \cdot 10^{-5}$	2,0
	^{99}Tc	$3 \cdot 10^5$	0,12	120	$2 \cdot 10^{-4}$	$6 \cdot 10^{-4}$
	^{238}Pu	128	0,9	120	$5 \cdot 10^{-6}$	$2,5 \cdot 10^{-3}$

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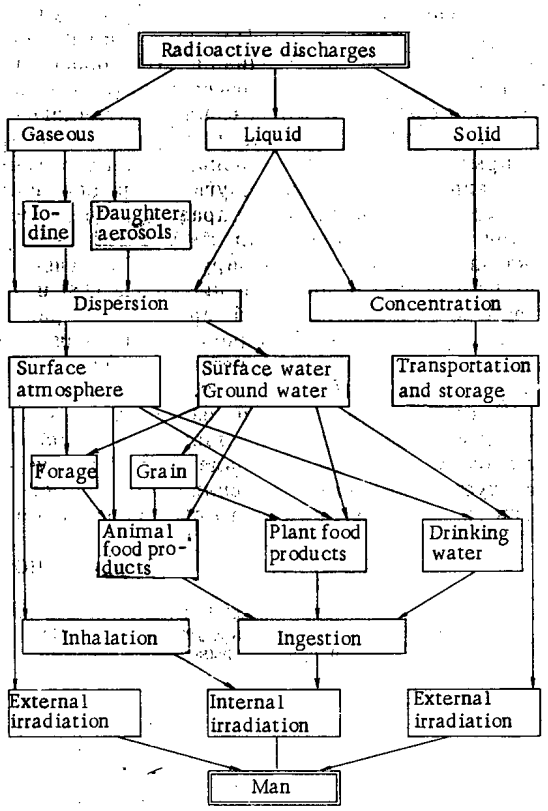

Fig. 1. Paths for the effects of radioactive discharges on the human body.

In Russian scientific literature, the term "radiology" has a narrower meaning (the science of the use of ionizing radiations in medicine) than in Anglo-American publications. Therefore, it is preferable to use the term radiation hazard instead of radiological hazard in the English text of the report.

Remarks within the Bounds of the Traditional Approach. Evaluation of the relative radiation hazard was carried out by the authors in terms of the traditional approach: an activity A (curies) built up in the reactor or discharged into the environment (the air, for example) is related to the value of the maximum permissible concentration (MPC, Ci/liter). The result is a volume of air needed to dilute the analyzed radioactive material to the MPC, which is taken as a quantitative criterion of radiation hazard.

In comparing FIR and FUR with respect to the amount of long-lived radioactive isotopes contained in the reactor after an extended operating period, the authors of the paper being discussed did not consider ^{137}Cs and ^{99}Tc in Table 6. We note that such estimates were made in [3], which is cited as reference 7 in the paper. According to this data, the radiation hazard of ^{137}Cs is twice as great as that of ^{94}Nb, and ^{99}Tc is comparable to vanadium containing an admixture of niobium with respect to this parameter (Table 1).

We also direct attention to the inconsistency of the data for ^{90}Sr given in Table 1 [3] and in Table 6 of the subject paper. According to the data in [3], the relative radiation hazard $^{90}Sr/(^{93}Nb+^{94}Nb)=90/1.4=50$; from the data in the second column of Table 6, where a mixture presenting a greater radiation hazard — all fission products (FP) — is being considered, $\Sigma FP/(^{93}Nb+^{94}Nb)=10^4/1.7\cdot10^4\approx0.6$. This difference (almost two orders of magnitude) demands a detailed explanation.

In the same Table 6, the analysis of the radiation hazard from wastes is not carried out to the point where values of A/MPC are obtained. In the third row of Table 6, one should take into account the abundance of radionuclide components in the fission fragment sum.

The FIR/FUR comparison was made for only a part of the volatile gaseous materials (^{131}I and ^{3}H). In normal operation of the FIR and under accident conditions, however, one of the main sources of radiation hazard is the radioactive isotopes of the noble gases (argon, krypton, xenon) and their daughter aerosol products (rubidium, strontium, barium, cesium, etc.). The former are predominantly responsible for external β and γ irradiation of the body and the latter (together with ^{3}H, ^{14}C, and ^{85}Kr) for internal irradiation [4]. These isotopes should be included in the appropriate calculations.

In making the comparative evaluation, liquid radioactive wastes of low and medium specific activity, which are constantly discharged from an FIR, were also not taken into account. The authors of the report, without taking these wastes into consideration, started from the assumption that the criterion of 1 mrem/yr would be met in any case. However, the conversion from the evaluation of A/MPC values to the calculation of dose requires a study of the so-called "critical" path of a radioactive isotope in the environment including factors of dispersion and concentration. A classical example is the food chain air — grass — cow — milk — human thyroid gland for the fission product ^{131}I. When all the links on this critical path are considered, the concentration factor of ^{131}I is estimated to be 700 [5, 6].

TABLE 2. Relative Radiation Hazard of Radioactive Wastes from FIR and FUR

Radioactive element	Yield of radioactive isotopes, Ci/MW (t) in 10^8 sec	MPBB, μCi	Relative hazard
^{94}Nb	35	90	$4\cdot10^5$
^{239}Pu	35	0,04	$9\cdot10^8$

The bioaccumulation factor for certain radioactive materials in the bodies of fish amounts to 10^3 (cesium,

cobalt, zinc, etc.). Just as great a concentration was recorded in experiments on roe. These experiments showed that radionuclides related to the group of rare-earth elements as well as zirconium, niobium, ruthenium, and other elements are concentrated predominantly in the covering of the roe. Strontium, barium, and cesium are mainly concentrated in the contents of the roe [7, 8].

Unfortunately, the values for permissible concentrations recommended by ICRP and NCRP [2] were established without consideration of food chains and bioaccumulation factors. In addition, they were accepted on the basis of recommendations from national committees and may contradict one another. Thus, the content of radioactive materials (25 pCi/liter in fresh water) which is acceptable in West Germany exceeds the standard established in Hungary [9].

From what has been said, it is preferable from our point of view to use for an estimate of the relative radiation hazard RH the ratio

$$RH = A/10^{-6} q,$$

where A has its previous meaning and q is the maximum permissible content of a radioactive isotope in the body (MPBB in the terminology of ICRP), μCi. The ratio given is a dimensionless quantity which characterizes (in arbitrary units of radiation hazard) the number of people who may be exposed to permissible doses if the considered activity A is dispersed into the environment. Applying this approach to some of the data in the aforementioned Table 6, we obtain the values given in Table 2. A comparison of relative hazards gives the value $^{239}Pu/^{91}Nb \approx 2 \cdot 10^{-3}$.

Need for Expansion of Systems Approach. In the analysis of radiation hazard from specific nuclear reactors, either FIR or FUR, for people in category B, one should evaluate the effect of gaseous, liquid, and solid radioactive materials including possible paths of migration in the environment and all types of irradiation of the body. The scheme proposed in [10] (see Fig. 1) can be used as a basis for this. The report [11] can serve as an example.

It is convenient to accomplish a quantitative evaluation by calculation of the integral radiation effect (in units of man-rem/yr). This makes it possible not only to compare FIR and FUR but also to make a comparison with the natural radiation background. A technique for such calculations has been proposed [12]. It is extremely desirable to evaluate the genetically and leucogenically significant doses separately (GSD and LSD).

New complexities have been observed for this problem recently. Thus, it was noted at a meeting of IAEA experts in Budapest (September 1973) that the discharge waters from a nuclear reactor with a thermal power of 237 MW at Gundremingen which were emptied into the upper reaches of the Danube River contained 18 isotopes (up to 98% ^{3}H, $^{89, 90}Sr$, $^{134, 137}Cs$, ^{131}I, ^{140}Ba-^{140}La, $^{58, 60}Co$, and others), but the radiation dose produced by them in the population of the adjacent area did not exceed 1 mrem/yr. This is 1/170 of the corresponding maximum dose. By 1977, however, the total thermal power of the nuclear power stations discharging their waste waters into the Danube will increase to 12,800 MW, i.e., by a factor of 50 [9], which can bring the irradiation dose to the population close to a dangerous level. In addition, considerable miscalculation has occurred in precisely this area of radiation protection in the recent past. Thus, the amount of liquid radioactive wastes produced during the operation of the Shipping Port nuclear power station was underestimated by a factor of 15 [13]. In the design of a radiation-safety system for the first nuclear icebreaker, a similar underestimate by a factor of 10 was accepted [14]. Therefore, it is advisable to supplement the material in the report with a new section containing quantitative estimates of the radiation hazard from electrical power systems using FIR or FUR in a given geographical area (for example, Western Europe or the Mississippi River basin) as was done recently [10, 15].

The radiation hazards of FIR or FUR under normal operation and in accident situations should be compared for several designs for specific reactors. As examples of FIR, it is convenient to consider the BR-350, "Phoenix," and PFR; as examples of FUR, designs carried out to a high degree of engineering development (for example, the RTPR [16] and others).

In the subject matter of such a comparison, it is extremely desirable to include questions concerning the minimization of the radiation hazard from individual assemblies in FIR and FUR as was done [17] for FUR blankets of niobium, vanadium, and aluminum.

A comparison of the radiation hazards of FIR and FUR can only be complete when the global aspects of the effect of peaceful power based on fast and thermal reactors and on fusion reactors are subjected to analysis. It is then necessary to consider the entire technological chain (from ore treatment and enrichment,

refinement and preparation of fuel, to burial of radioactive wastes) and to take into account the effect of other harmful factors accompanying the use of ionizing radiation and radioactive materials (radium, radon, and daughter aerosols in uranium mines; silica and lead dusts in ore-processing plants; chemically corrosive media at radiochemical plants, etc.). Along with this, it is extremely desirable to evaluate the radiation hazard to the biosphere from FIR and FUR, considering it as a total system with the human population a subsystem. Such studies might reveal subtle, and as yet unobserved, mutual effects between radiation and other natural or artificial factors. Typical examples are the known dependence of the harmful effects of ionizing radiation on the partial pressure of oxygen [18] and on the ambient temperature [19], the previously published extraordinarily high values for the relative biological efficiency of ^{3}H included in DNA [20], etc.

Perhaps the last of the studies we have mentioned cannot possibly be included in the new text of the report. In that case, one should exert every effort to organize such a multifactor systems analysis under the aegis of the IASA. It seems to us that these studies particularly fit the spirit and approach of the IASA. What has been said still does not furnish a basis for considering FIR and FUR equivalent from the viewpoint of radiation safety. The conclusion that a significant difference between them is in favor of the fusion reactor rather comes to mind.

LITERATURE CITED

1. Recommendations of the International Commission of Radiological Protection (ICRP Publ. 9), Oxford (1966).
2. Radiation Safety Standards (NRB-69) [in Russian], Atomizdat, Moscow (1972).
3. D. Steiner and A. Fraas, Nucl. Safety, 13, 353 (1972).
4. Yu. V. Sivintsev, Radiation Safety at Nuclear Reactors [in Russian], Atomizdat, Moscow (1967).
5. T. Burnett, Health Phys., 18, 73 (1970).
6. L. A. Il'in, I. A. Likhtarev, and Yu. D. Konstantinov, Radioactive Iodine in Radiation Safety [in Russian], Atomizdat, Moscow (1972).
7. N. I. Mashneva, S. Ya. Sukal'skaya, and A. I. Tikhonova, Radiobiologiya, 12, 156 (1972).
8. V. P. Shvedov, G. G. Polikarpov, and I. A. Sokolov, in: Radioecology of Aquatic Organisms [in Russian], No. 2, Zinatie, Riga (1973), p. 263.
9. G. V. Shishkin et al., At. Énerg., 36, 154 (1974).
10. J. Kastner, Trans. Amer. Nucl. Soc., 17, 535 (1973).
11. P. McGrath, KFK-1992 (1974).
12. J. Pensko, Nukleonika, 19, 47 (1974).
13. C. Abrams et al., Chem. Engng. Progr. 58, 70 (1962).
14. Yu. Sivintsev and A. Stefanovitch, Bull. of the Permanent Intern. Association Congress, 2, No. 8, 5 (1963).
15. Nucl. Safety, 15, 56 (1974).
16. T. Coultas et al., in: Proc. IAEA Conf. on Rusion Reactor Design Problems, Culham, 29 Jan.-15 Feb., 1974, p. 151.
17. J. Powell et al., ibid, p. 343.
18. Primary Radiobiological Processes [in Russian], Atomizdat, Moscow (1973).
19. V. Ya. Aleksandrov, Cells, Macromolecules, and Temperature [in Russian], Nauka, Leningrad (1975).
20. Yu. I. Moskalev, in: Tritium Oxide [in Russian], Atomizdat, Moscow (1968), p. 366.

BEHAVIOR OF ^{90}Sr AND ^{137}Cs IN THE WATERS OF THE BALTIC SEA

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At the present time, a program for the broad development of nuclear power is being realized in countries bordering on the Baltic. A number of nuclear power stations are being built directly on the shores of the Baltic Sea. Others are located on rivers or lakes draining into the Baltic. The fraction of radioactive products from the operation of a nuclear power station which is discharged into aqueous systems is decreasing in proportion to the improvement in reactor construction and in means for technical safety. However, a rise in the total power of the nuclear installations being put into operation is occurring even more rapidly. Therefore, increasing attention is being paid to the behavior of radioactive impurities in the Baltic Sea and in the rivers and lakes in its basin.

Products of global radioactive fallout presently in the waters are being studied in order to develop ideas about the behavior of certain radioactive contaminants. Despite the high sensitivity of the methods used for the determination of radionuclides in water [1, 2] and the possibility of using other sensitive methods as well [3], one can actually observe in our day only ^{90}Sr and ^{137}Cs, and perhaps ^{54}Mn and $^{239+240}Pu$, in the waters of the Baltic Sea and its basin.

Information about the behavior of ^{90}Sr and ^{137}Cs can also give some ideas about ^{89}Sr, ^{134}Cs, and other isotopes of the alkali-earth and alkali elements which may enter the water system of the Baltic as components of discharges and wastes.

Studies of the radioactive contamination of Baltic waters are being carried on by scientific institutions in various countries [4-8]. A review of the results for 1959-1967 was carried out by the National Academy of Sciences (USA) [9]. New publications have appeared in recent years, including [10, 11], where attempts were made to generalize published materials available up to 1974. The present paper also includes the latest data. Table 1 gives the results of our determinations of ^{90}Sr and ^{137}Cs in the surface waters of the Baltic Sea in 1972 and 1973. Averaging with a 68% confidence coefficient gives the following values: 150 ±25 and 120 ±25 dis/min in 1972, and 130 ±20 and 120 ±20 dis/min in the 1973 for ^{90}Sr and ^{137}Cs respectively.

TABLE 1. Average ^{90}Sr and ^{137}Cs Concentrations in Surface Waters of the Baltic Sea

Area	1972			1973		
	concentration, dis/min ·100 liters		$^{137}Cs/^{90}Sr$	concentration, dis/min ·100 liters		$^{137}Cs/^{90}Sr$
	^{90}Sr	^{137}Cs		^{90}Sr	^{137}Cs	
NE Baltic including the western portion of the Gulf of Finland	190±30 [5]	110±20 [5]	0,6±0,2	145±20 [4]	107±15 [4]	0,7±0,15
E Baltic	180±25 [8]	130±20 [8]	0,7±0,2	130±20 [4]	115±15 [4]	0,9±0,2
W Baltic	150±20 [14]	120±25 [13]	0,8±0,2	140±25 [5]	120±20 [4]	0,9±0,15
S Baltic	120±15 [5]	110±20 [5]	0,9±0,2	120±20 [5]	120±20 [5]	1,0±0,2
Area bordering on the Danish straits	110±15 [3]	110±20 [3]	1,0±0,25	120±25 [4]	130±20 [4]	1,05±0,25
Danish straits	130 [8]	—	—	170±30 [1]	130±25 [1]	0,7±0,25

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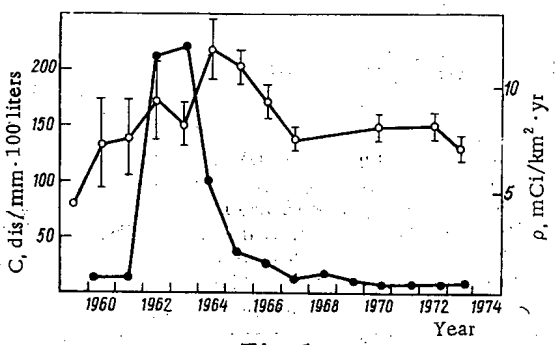

Fig. 1

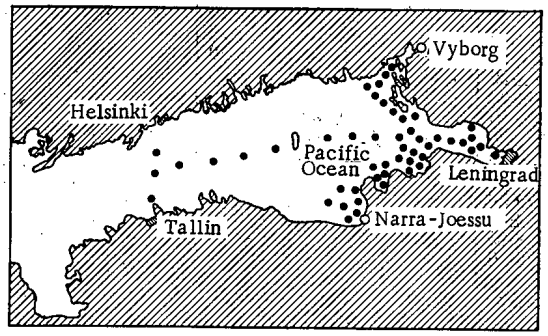

Fig. 2

Fig. 1. Variation of ^{90}Sr content in the Baltic Sea and of fallout on its surface in 1959-1973; ○) concentration C; ●) fallout density ρ.

Fig. 2. Location of water sampling points in the Gulf of Finland.

The ratio between the average values of the concentrations, $^{137}Cs/^{90}Sr$, was 0.8 ± 0.15 in 1972 and 0.9 ± 0.15 in 1973.

A comparison of the average results of the observations for the entire Baltic Sea in 1972 and 1973 shows that they are practically identical. Consideration of the average values for individual areas gives evidence for some decrease in 1973 within the limits of the experimental error in the concentration of ^{90}Sr. This is not observed for ^{137}Cs within the limits of the accuracy of the determinations. Along with this, it was noted that the ^{90}Sr content was highest in the northeastern portion of the sea and in the Gulf of Finland and lowest in the area of the straits, especially for the 1972 data. This is explained by the inflow of water from the North Sea with lower concentrations of radionuclides and by the great separation from the straits of precisely the northeastern portion of the sea with consequent lowered possibility for exchange of water from this area. In the shallow and almost fresh-water northeastern portion of the sea, and particularly in the Gulf of Finland, the drainage of rivers carrying ^{90}Sr from a large area of dry land plays a decisive role in the balance of materials from global radioactive fallout.

The results of the observations in 1972 and 1973, and the inclusion of data previously published [7-11], make it possible to extend the estimate of the time changes in the average values of ^{90}Sr and ^{137}Cs concentrations (Fig. 1). Figure 1 indicates that the waters of the Baltic are an "inertial" system with respect to ^{90}Sr contamination. This is understandable if one keeps in mind the shallowness and enclosed nature of the Baltic Sea. Apparently, the ^{90}Sr carried down by the rivers, and possibly also deposited on the surface of the sea, is in solution in ionic form. Because of the shallowness, all of the water is exchanged with the consequence that ^{90}Sr lost from the surface layer to deeper waters is compensated for. The ^{90}Sr settles out on the bottom extremely slowly. It is known that in some stagnant shallow areas the ratio of ^{137}Cs to

TABLE 2. Some Determinations of the Ratio of ^{137}Cs to ^{90}Sr in the Waters of Seas and Oceans

Place and time of sampling	No. of samples	Range of values	Mean arithmetic ratio	Mean-square deviation of a single result	Error of a single experimental result ± 20%
Atlantic Ocean	172	0 6—4,4	1,8±0,04	±0,5	±0,36
surface water	159	0,6—4,4	1,8±0,04	±0,5	±0,36
1963—1964	78	0,7—3,7	1,7±0,07	±0,6	±0,34
1969—1970	81	0,6—4,4	1,9±0,07	±0,6	±0,38
to 700 m depth (1969—1970)	13	0,7—2,6	1,6±0,14	±0,5	±0,32
Pacific ocean	83	0,4—2,0	1,4±0,07	±0,4	±0,28
surface water (1966—1970)	53	0,8—2,0	1,5±0,04	±0,3	±0,3
to 700 m depth (1966—1967)	15	0,4—2,0	1,4±0,12	±0,5	±0,28
from 700 m to 7000 m (1966—1970)	15	0,6—1,8	1,3±0,1	±0,4	±0,26
Baltic Sea (1970)	63	0,4—1,4	0,9±0,05	±0,2	±0,24
Totals and averages	318	0,4—4,4	1,7±0,05	±0,5	±0,34

TABLE 3. Determination of ^{90}Sr and ^{137}Cs in Surface Waters of the Gulf of Finland

Sampling area	Sampling data			Concentration, dis/min · 100 liters						$^{137}Cs/^{90}Sr$		
				^{90}Sr			^{137}Cs					
	1971	1972	1973	1971	1972	1973	1971	1972	1973	1971	1972	1973
Neva Bay	15.VII	15.IX	5.IX	200±40	440±60	240±30	50±10	20	40±15	0,3	—	0,2
	15.VII	15.IX	5.IX	160±30	420±60	220±30	50±10	20	40±15	0,3	—	0,2
	15.VII	15.IX	5.IX	220±30	410±60	250±40	30±10	20	20	0,1	—	—
	15.VII	15.IX	5.IX	190±30	410±60	210±30	50±10	20	35±15	0,3	—	0,2
	15.VII	17.IX	5.IX	200±30	350±50	240±30	30±10	20	30±10	0,2	—	0,1
	16.VII	17.IX		—	320±10	—	30±10	20	—	—	—	—
	16.VII	17.IX		—	380±50	—	30±10	30±15	—	—	0,08	—
	16.VII	18.IX		320±50	550±80	—	30±10	70±20	—	0,1	0,15	—
	16.VII			—	—	—	30±10	—	—	—	—	—
	16.VII			120±20	—	—	50±10	—	—	0,4	—	—
	Area average:			210±20	410±60	230±30	40±10	—	—	0,2	0,1	0,2
Koporo-skaya Bay	30.VII	12.X	29.VII	340±50	240±40	160±30	60±10	130±20	80±20	0,2	0,6	0,5
	30.VII	12.X	29.VII	260±40	280±40	170±30	80±10	130±20	80±20	0,3	0,5	0,5
	30.VII	12.X	29.VII	220±30	230±30	170±30	100±20	170±30	90±20	0,5	0,7	0,5
	30.VII	12.X	29.VII	320±50	100±20	160±30	100±20	70±20	110±20	0,3	0,7	0,7
	30.VII	12.X	29.VII	340±50	100±20	180±30	110±20	70±20	120±20	0,3	0,7	0,7
	30.VII	12.X	29.VII	330±50	120±20	190±30	120±20	80±20	100±20	0,4	0,7	0,5
			29.VII	—	—	160±30	—	—	100±20	—	—	0,6
			29.VII	—	—	200±30	—	—	70±20	—	—	0,4
			29.VII	—	—	180±30	—	—	60±20	—	—	0,3
			29.VII	—	—	180±30	—	—	70±20	—	—	0,4
	Area average:			300±45	180±30	180±30	100±20	110±20	90±20	0,3	0,7	0,5
Gulf of Finland (central part)	30.VII	18.IX	8.IX	230±40	300±50	180±30	70±10	90±20	120±20	0,3	0,3	0,6
	30.VII	18.IX	8.IX	250±40	180±30	180±30	70±10	80±20	100±20	0,3	0,4	0,6
	30.VII	18.IX	8.IX	240±40	170±30	190±30	60±10	80±20	90±20	0,3	0,5	0,5
	30.VII	18.IX	8.IX	400±60	180±30	150±30	70±10	80±20	100±20	0,2	0,4	0,7
	30.VII	18.IX	8.IX	230±40	210±30	120±20	90±10	90±20	90±20	0,4	0,4	0,8
	30.VII	18.IX	8.IX	280±40	260±40	170±30	70±10	100±20	90±20	0,3	0,4	0,5
	30.VII	18.IX	8.IX	490±70	350±50	170±30	100±20	100±20	100±20	0,2	0,3	0,6
	30.VII	19.IX	11.IX	330±50	190±30	150±30	80±10	110±20	100±20	0,2	0,6	0,7
		19.IX	11.IX	—	140±20	180±30	—	80±20	90±20	—	0,5	0,5
		20.IX	11.IX	—	220±30	160±30	—	150±30	80±20	—	0,7	0,5
		20.IX	15.IX	—	320±40	150±30	—	120±20	90±20	—	0,4	0,6
		20.IX	15.IX	—	240±30	170±30	—	80±20	110±20	—	0,3	0,6
		3.X	15.IX	—	150±20	140±20	—	170±30	120±20	—	1,2	0,9
		3.X	15.IX	—	190±30	140±20	—	230±40	110±20	—	1,2	0,8
		3.X	15.IX	—	300±50	110±20	—	100±20	110±20	—	0,3	1,0
		3.X	15.IX	—	270±30	140±20	—	110±20	120±20	—	0,4	0,9
		3.X	15.IX	—	200±30	130±20	—	120±20	120±20	—	0,6	0,9
		3.X	20.IX	—	220±30	140±20	—	130±20	190±30	—	0,6	1,4
			20.IX	—	—	170±30	—	—	90±20	—	—	0,5
			20.IX	—	—	140±20	—	—	80±20	—	—	0,6
			20.IX	—	—	130±20	—	—	80±20	—	—	0,7
			20.IX	—	—	150±30	—	—	60±20	—	—	0,6
			20.IX	—	—	150±20	—	—	80±20	—	—	0,4
			20.IX	—	—	160±30	—	—	80±20	—	—	0,5
	Area average:			310±50	230±30	150±30	80±10	110±20	100±20	0,3	0,5	0,7
Narvskyi Gulf	14.VIII	3.X	24.VII	630±90	180±30	190±30	80±10	230±40	100±20	0,1	1,3	0,5
	14.VIII	3.X	24.VII	330±50	290±40	200±30	90±10	140±20	80±20	0,3	0,5	0,4
	14.VIII	3.X	24.VII	270±40	280±40	200±30	70±10	170±30	70±20	0,3	0,6	0,4
	14.VIII	3.X	26.VII	290±40	140±20	160±30	100±20	220±30	110±20	0,3	1,6	0,7
		11.X	26.VII	—	130±20	150±30	—	120±20	100±20	—	0,9	0,7
	Area average:			380±60	200±30	180±30	90±10	180±30	90±20	0,3	1,0	0,5
Vyborg Gulf	24.VII	20.IX	20.IX	280±40	260±40	170±40	40±10	110±20	80±20	0,1	0,4	0,5
	24.VII	20.IX	20.IX	380±50	300±40	180±30	40±10	100±20	90±20	0,1	0,4	0,5
	24.VII	20.IX	20.IX	—	270±40	180±30	40±10	130±20	80±20	—	0,5	0,4

TABLE 3 (continued)

Sampling area	Sampling data			Concentration, dis/min · liters						$^{137}Cs/^{90}Sr$		
				^{90}Sr			^{137}Cs					
	1971	1972	1973	1971	1972	1973	1971	1972	1973	1971	1972	1973
Viborg gulf	24.VII	20.IX	23.IX	440±60	130±20	180±20	30±10	110±20	80±20	0,1	0,8	0,5
	24.VII	20.IX	23.IX	280±40	310±40	170±30	50±10	130±20	60±20	0,2	0,4	0,4
	24.VII	20.IX	23.IX	250±40	250±40	220±30	70±10	120±20	80±20	0,3	0,5	0,4
	24.VII	20.IX	23.IX	220±40	180±30	190±30	80±10	100±20	90±20	0,4	0,6	0,5
	24.VII	20.IX	23.IX	270±40	200±30	210±30	60±10	90±20	60±20	0,2	0,5	0,3
	31.VII	20.IX	23.IX	310±50	240±40	190±30	70±10	70±20	40±15	0,2	0,3	0,2
	31.VII	21.IX	24.IX	380±60	210±30	190±30	80±10	70±20	40±15	0,2	0,3	0,2
	31.VII	21.IX		290±50	240±40	—	70±10	150±30	—	0,2	0,6	—
	31.VII	21.IX		320±50	210±30	—	80±10	60±20	—	0,3	0,3	—
	31.VII			310±50	—	—	80±10	—	—	0,3	—	—
	31.VII			210±30	—	—	80±10	—	—	0,4	—	—
	31.VII			320±50	—	—	80±10	—	—	0,3	—	—
	31.VII			400±60	—	—	80±10	—	—	0,2	—	—
	Area average:			310±50	230±40	190±30	60±10	—	70±20	0,2	0,5	0,4
Luzhskaya Bay	—	—	27.VII	—	—	200±30	—	—	90±20	—	—	0,5
	—	—	27.VII	—	—	160±30	—	—	120±20	—	—	0,7
	—	—	27.VII	—	—	170±30	—	—	100±20	—	—	0,6
	Area average:					180±30			100±20			0,6

^{90}Sr in the bottom deposits is 4 and more while it is 1.6 on the average in fallout [12]. Consequently, ^{90}Sr settles out into the bottom deposits much more slowly than ^{137}Cs. The chance for ^{90}Sr removal from the water mass of the Baltic is low because of this. In years of low river runoff, as was the case in 1972 and 1973, there is also little loss of ^{90}Sr through exchange of water with the North Sea. In some years, water from the Baltic sea does not even reach the North Sea. Evaporation of water obviously does not remove ^{90}Sr. As a result (see Fig. 1), the process of ^{90}Sr removal from the water of the Baltic is proceeding slowly after the intense inflow in 1964-1965. The large lag of the maximum ^{90}Sr concentration with respect to the maximum of the fallout density is evidence of the important role of drainage with river waters. This process lags behind fallout and determines the position of the maximum concentration. But for this effect, the ^{90}Sr concentration would follow the variation in the density of radioactive fallout with considerably less shift in phase. The reasons given for the buildup of ^{90}Sr in the waters of the Baltic Sea lead to the fact that the ^{90}Sr concentration in this sea is 5-6 times greater than that in the open ocean at the same latitudes [10]. Information for 1969-1970 [11] shows that if one takes the ^{90}Sr concentration in the open Atlantic Ocean in the surface waters of the middle northern latitudes as unity, one then has the values 1.5, 3.2, 4, and 6 respectively for the Mediterranean, North, Black, and Baltic Seas.

The distribution of ^{137}Cs in the waters of the Baltic Sea is markedly different from the distribution of ^{90}Sr. The range of ^{137}Cs concentrations is somewhat narrower. A great difference in the various regions of the Baltic is not seen, as was noted for ^{90}Sr in 1972 (see Table 1). Minimum and maximum values were found at comparatively nearby points [11]. Adsorption processes have a great effect on the behavior of ^{137}Cs and, consequently, the conditions which are built up in one or another part of the basin. River waters are stripped of ^{137}Cs. This means that to the degree the content of the sea water is determined by river drainage, the poorer in ^{137}Cs its water is. In addition, adsorption processes directly in the sea facilitate the transfer of ^{137}Cs to bottom deposits.

TABLE 4. Averaged ^{90}Sr and ^{137}Cs Concentrations in Waters of the Gulf of Finland.

Year	Concentration, dis/min · 100 liters		$^{137}Cs/^{90}Sr$
	^{90}Sr	^{137}Cs	
1971	280±40	70±10	0,3±0,1
1972	250±40	120±20	0,5±0,15
1973	180±30	80±10	0,4±0,1

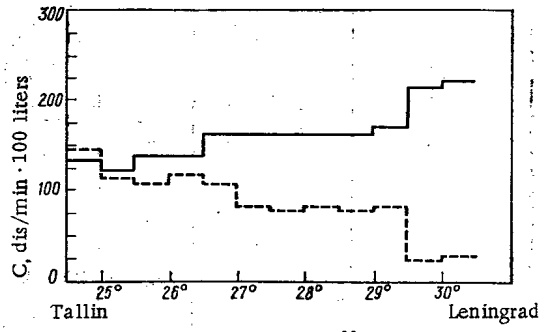

Fig. 3. Concentration of ^{90}Sr (——) and ^{137}Cs (-----) in waters of the Gulf of Finland in 1973, dis/min•100 liters.

A difference in the behavior of ^{90}Sr and ^{137}Cs in the waters of the Baltic Sea is also observed in an analysis of the concentration ratio $^{137}Cs/^{90}Sr$. Data from [13] is reproduced in Table 2, but later results are given for the Baltic Sea. This data indicates that the Baltic Sea differs markedly from the Atlantic and Pacific Oceans with respect to the value of this ratio. This fact emphasizes the slowness of ^{90}Sr removal and the ^{137}Cs deficit in river waters.

Of all the waters of the Baltic Sea, the Soviet Union has particular interest in the Gulf of Finland which is mainly fed by river drainage from the northwestern part of our country and from a considerable portion of the territory of Finland. In recent years, the V. G. Khlopin Radium Institute has been making systematic observations of the content of long-lived radionuclides in the waters of the Gulf of Finland. Samples are collected in the waters of the gulf itself and in separate parts of it: Neva Bay, Koporskaya Bay, Narvskyi Gulf, Vyborg Gulf, Luzhskaya Bay (Fig. 2, Table 3). The average concentrations of ^{90}Sr and ^{137}Cs for the water of the entire gulf are given in Table 4. The data in Tables 3 and 4 show that the ^{90}Sr content in the Gulf of Finland tends to decrease. There appears to be no change in the ^{137}Cs level during the period of observation.

A reduction in the ^{90}Sr concentration from 1971 to 1973 was also observed in separate regions of the Gulf of Finland. An increase in the ^{90}Sr content was noted in 1972 in the Neva Bay area. It coincided with a sharp decrease in depth because of high evaporation and a lack of precipitation as the result of an unusually hot and dry year for Leningrad and the entire northwestern USSR. Under such conditions the concentration of all forms of contamination, including radioactive contamination, can increase. The isotope ^{137}Cs, which has a greater capability for absorption by silts and bottom depositis than ^{90}Sr, may be strongly carried to the bottom by suspended matter resulting in the creation of a deficit of ^{137}Cs together with a simultaneous buildup of ^{90}Sr.

The $^{137}Cs/^{90}Sr$ ratio (see Table 3) in areas of the eastern portion of the Gulf of Finland is lower than it is in the western portion, which is clearly confirmed by Fig. 3 showing the ^{90}Sr and ^{137}Cs content in the surface waters of various parts of the Gulf of Finland running from west to east. The water content of the eastern areas of the Gulf of Finland, which are most distant from the water of the Baltic Sea itself, is determined mainly by river waters. Hence, we have as a consequence the lower value of the $^{137}Cs/^{90}Sr$ ratio in the eastern portion of the Gulf of Finland. In the western portion, the ^{137}Cs concentration is practically equal to the concentration of ^{90}Sr. This equalization occurs because of the exchange of water in the Gulf of Finland with water from the open Baltic Sea.

The change in ^{90}Sr and ^{137}Cs content in waters of the Gulf of Finland noted in 1972-1973 is explained by global radioactive fallout. The difficulty of water exchange with the Baltic Sea proper and then with the ocean prevents a rapid reduction in the concentration of radionuclides. In addition, the drop in radionuclides by water exchange is reversed because of continuing global fallout and also because of river drainages which leach radionuclides from the surface of the ground.

LITERATURE CITED

1. L. M. Ivanova, Radiokhimiya, 9, 622 (1967).
2. L. M. Ivanova, I. L. Gedeonov, and A. G. Trusov, in: Proceedings of SEV Conference on Studies in the Area of Decontamination of Liquid, Solid, and Gaseous Radioactive Wastes and Decontamination of Contaminated Surfaces [in Russian], Kolobzheg, Oct. 2-7, 1972.
3. Reference methods for marine radioactivity studies, Techn. Rep. Series, IAEA, Vienna (1970), No. 118.
4. C. Gyllander, in: Proc. IAEA Symp. on Disposal of Radioactive Wastes into Seas, Oceans, and Surface Waters, Vienna, May 16-20, 1966, p. 207.
5. A. Voipio, in: Proc. of the First National Coastal Shallow Water Research Conference, NSE-ONR, 1962, p. 785.
6. A. Voipio and O. Paakola, ICES Hydrographical Committee, 1964, No. 147.
7. A. Salo and A. Voipio, in: Radioecological Concentration Processes, Pergamon Press, New York (1966), p. 837.
8. Environmental Radioactivity in Denmark in 1972, Riso, Rep. No. 291 (1972).
9. Radioactivity in the marine environment. Prepared by the panel on radioactivity in the marine environment of the Committee on Oceanography, National Research Council, Washington (1971).
10. L. M. Ivanova et al., At. Énerg., 33, 835 (1972).
11. N. A. Rakov, At. Énerg., 37, 373 (1974).

12. J. Harley et al., Fission Yield and Fission Product Decay, HASL-300, Procedures Manual, New York (1972), p. F-02-01.
13. A. G. Trusov, L. M. Ivanova, and L. I. Gedeonov, At. Énerg., 34, 391 (1973).

^{90}Sr, ^{137}Cs, AND TRITIUM IN THE BALTIC SEA IN 1972

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The expanding construction of nuclear industrial plants and nuclear power stations on the shores of the Baltic Sea [1] is creating a real possibility for the introduction of radioactive wastes into the sea water and bottom deposits. Because of this, a detailed examination of the sea was made to determine the content of long-lived radioactive isotopes in its water. This makes it possible to evaluate the background "global" level of radioactive contamination of the sea before startup and operation of nuclear industrial plants and nuclear power stations and to make future judgements of their contaminating effects. The problem is even more pressing because the Baltic Sea has limited water exchange with the ocean and is comparatively shallow.

The first data on ^{90}Sr content in the waters of the Baltic Sea were published by Finnish workers [2]. The ^{90}Sr concentrations observed by them during the period 1959-1963 varied over a wide range (0.05-1.8 pCi/liter). Subsequently, the variation in ^{90}Sr and ^{137}Cs accumulation in the Baltic Sea was determined for the period 1960-1967 [3]. L. I. Gedeonov and associates, using the data of the Finnish workers and their own measurements, derived the dynamics of ^{90}Sr concentration variation in the Baltic Sea from 1960-1970 [4].

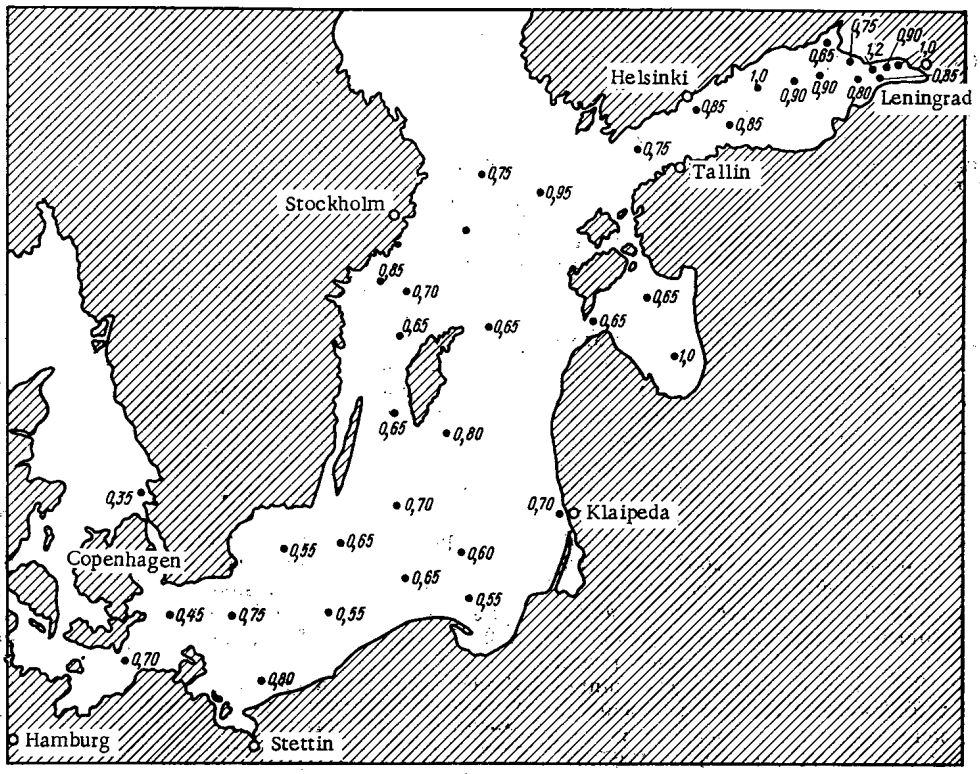

Fig. 1. Distribution of ^{90}Sr concentrations in surface waters of the Baltic Sea (pCi/liter).

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TABLE 1. ^{90}Sr and Tritium Concentrations in Surface Waters of Various Areas of the Baltic Sea (June 1972)

Area	Isotope	No. of meas.	Concentration†
Gulf of Finland	^{90}Sr	13	0,85±0,13 ‡
E Baltic	^{90}Sr	15	0,72±0,12
	^{3}H	5	70±23
W Baltic	^{90}Sr	9	0,60±0,13
	^{3}H	6	47±22
Av. for entire sea	^{90}Sr	37	0,73±0,14
	^{3}H	13*	62±23

*Including two measurements in the Gulf of Finland.
†pCi/liter for ^{90}Sr, TU for ^{3}H.
‡Root-mean-square deviation of a single measurement from the average value is given.

TABLE 2. Ratio of ^{137}Cs and ^{90}Sr in Waters of the Baltic Sea

Area	Depth, m	Rel. concn.
E part of Gulf of Finland	0	0,5
W part of Gulf of Finland	0	0,8
Central Baltic	0	1,0
	50—150	1,0
W Baltic	0	1,7
	50—150	1,7

We made a study of the radioactive contamination of the Baltic Sea in June 1972 on the research ship "Okeanograf." Bottom deposits were sampled, 10-liter samples of sea water were collected, a primary concentration of the ^{90}Sr and ^{137}Cs in the samples was performed, and 1-liter samples of sea water were collected for tritium analysis. Sea-water samples from depths less than 30 m were taken from surface layers and were taken from surface, bottom, and intermediate layers for depths greater than 30 m. Samples were analyzed for ^{90}Sr by the method given in [5, 6]. To increase the yield of ^{90}Sr, 4 g of calcium were added to a sample before primary concentration. The error in the determination of ^{90}Sr concentration was no more than 20%. ^{137}Cs was precipitated from the same sea-water samples with a carrier of the stable isotope in the form of the ferrocyanide. The concentration of ^{137}Cs was measured with a low-background gamma spectrometer. Several precipitates were combined into a single sample in order to increase the accuracy of the measurements.

The total error in the determination of ^{137}Cs was no more than 25%. The tritium concentration in samples collected was measured with a Nuclear Chicago liquid scintillation spectrometer after preliminary enrichment by electrolysis; the enrichment factor was 130-265.

A total of 37 samples of surface water from the Baltic Sea was analyzed for ^{90}Sr content; 17 samples were analyzed for ^{137}Cs content and 13 samples for tritium. The sampling points were distributed approximately uniformly over the entire water area of the sea. Only in the Gulf of Bothnia were no samples collected.

The results are shown in Tables 1 and 2. Data for ^{90}Sr are shown in Fig. 1. The tables and figure show that the concentrations of ^{90}Sr, ^{137}Cs, and tritium in the surface waters of the open ocean are considerably lower than those in the surface waters of the Baltic Sea. Thus, the concentration of ^{90}Sr in the North Atlantic in 1967-1970 was $0{,}13 \cdot 10^{-12}$ Ci/liter [4, 7]. On the other hand, in the northern part of the Pacific Ocean, the concentration of ^{137}Cs was $0{,}1 \cdot 10^{-12}$ Ci/liter [8]. The tritium content in the North Atlantic in 1967 was at a level of approximately 20 TU [9, 10], and in the northern portion of the Pacific Ocean in 1968 was 24 TU [11]. Thus the contamination of the Baltic Sea by long-lived isotopes is several times greater than the contamination of the open ocean. This is in agreement with [4] and is a consequence of the fact that the Baltic Sea is a closed, comparatively shallow body of water in which intense mixing of radioactive fallout from the atmosphere in a large layer of water does not occur as it does in the open ocean. The data we obtained 1972 for the average concentration of ^{90}Sr in the surface layer of the Baltic Sea is in good agreement with data referring to 1970 [4].

Tables 1 and 2 also show that the concentrations of these isotopes in the waters of the sea decrease somewhat from east to west. Furthermore, the ratio of ^{137}Cs concentration to ^{90}Sr concentration increases. Both these facts are in agreement with [4]. The reason for the observed behavior may be the effect of exchange with the open ocean which, of course, is greater in the western portions of the sea. On the other hand, the concentrations of global tritium and strontium in river water is higher than in the sea [12, 13]. The role of river drainage is rather large in the eastern part of the Baltic Sea. Thus the Neva alone produces 1/5 of the total river drainage into the Baltic Sea [4]. There are data [14] which purport to show that cesium is preferentially absorbed by bottom deposits in river systems in comparison with ^{90}Sr.

Therefore, the ratio of ^{137}Cs to ^{90}Sr for river water is somewhat lower than for radioactive fallout from the atmosphere. According to [14], the ratio of ^{137}Cs concentration to ^{90}Sr concentration at the mouth of the Neva is considerably less than one.

Analysis of bottom deposits from the Baltic Sea showed that the level of contamination is small (0.2-0.5 mCi/km^2 for ^{90}Sr and 7-12 mCi/km^2 for ^{137}Cs).

A series of measurements was made of the vertical distribution of the concentrations of these isotopes as a function of depth. The tritium concentration in a layer 30-50 m deep had an average value of 41 ±11 TU (based on five measurements). The vertical distribution of ^{90}Sr concentrations showed that these concentrations were somewhat higher, as a rule, in the surface layers of the water than in the deeper layers. However, at a depth of 50 m and in bottom waters in shallow areas of the sea with a depth less than 50 m, the ^{90}Sr concentrations were approximately the same and had an average value of $0.63 \cdot 10^{-12}$ Ci/liter.

The information obtained about the average concentrations of ^{90}Sr in separate water masses of the Baltic Sea was used to estimate the accumulation of this isotope in the water. Following [3], the following volumes were assumed for the water masses of the Baltic Sea: Gulf of Finland — above the discontinuity layer, 900 km^3, below the discontinuity layer, 200 km^3; the open sea — above the discontinuity layer, 10,000 km^3, below the discontinuity layer, 3500 km^3; Gulf of Bothnia — 6700 km^3; Gulf of Riga — 400 km^3.

Since we have no data for the Gulf of Bothnia, the ^{90}Sr concentrations in the waters of this gulf are assumed to be equal to the average concentration in the open portion of the sea, i.e., to 0.7 pCi/liter. Assuming that the average ^{90}Sr concentration obtained for the surface waters of the sea corresponds to the concentrations above the discontinuity layer and that in the deep waters corresponds to the concentrations below the discontinuity layer, we calculated the total accumulation of this isotope in the sea. It turned out to be ~15,000 Ci. This value is 5600 Ci less than that calculated in 1967 [3]. One should keep in mind, however, that according to [3], the efflux of ^{90}Sr into the North Sea is 800-1000 Ci/yr (afflux, 350 Ci/yr) and the actual rate is ~400 Ci/yr. Keeping this in mind and remembering the significant reduction in the level of global fallout of radioactive materials in the last five years, one can assume that our estimate of ^{90}Sr accumulation is in agreement with the previous estimate [3].

The resultant value of 15,000 Ci for a sea area of 365,000 km^2 corresponds to 40 mCi/km^2, is in practical agreement with the accumulation of ^{90}Sr on the area adjacent to the shores of the Baltic Sea [4], and indicates that ^{90}Sr contamination of the water results mainly from global radioactive fallout from the atmosphere.

It is more difficult to estimate the accumulations of ^{137}Cs and tritium because little data have been obtained for these isotopes, particularly their vertical distribution in the water of the sea. However, crude estimates can be made with respect to average concentrations. The average ^{137}Cs concentration in the surface layer of the Baltic Sea in 1972 was $0.9 \cdot 10^{-12}$ Ci/liter and the average $^{137}Cs/^{90}Sr$ ratio was 1.2. Hence, the ^{137}Cs accumulation in the Baltic Sea can be estimated to be 18,000 Ci. The tritium accumulation can be obtained in a similar manner (3.9 MCi).

LITERATURE CITED

1. C. Gyllander, in: Proc. Symp. on Disposal of Radioactive Wastes into Seas, Oceans, and Surface Waters, IAEA, Vienna (1966), p. 425.
2. O. Paakola and A. Voipio, Nature, 205, 274 (1965).
3. A. Voipio and A. Salo, Nordic Hydrology, 11, 57 (1971).
4. L. M. Ivanova et al., At. Énerg., 33, 835 (1972).
5. S. Vakulovsky and A. Silantjev, Nucl. Instrum. and Methods, 68, 42 (1969).
6. V. B. Chumichev, Trudy IEM, No. 25, 161 (1972).
7. Radioactivity in the Marine Environment, National Academy of Sciences, Washington (1971).
8. O. S. Zudin et al., At. Énerg., 32, 347 (1972).
9. G. Rooth and H. Ostlund, Deep-Sea Res., 19, 481 (1972).
10. W. Schell, G. Sauzay, and B. Rayne, IAEA Symp. on the Physical Behavior of Radioactive Contamination in the Atmosphere (Vienna, November 12-16, 1973), Rep. SM-181/34.
11. W. Roether, K. Munnich, and J. Ostlung, J. Geophys. Res., 75, 7672 (1970).
12. G. A. Sereda and Ts. I. Bobovnikova, in: Radioactive Isotopes in the Atmosphere and Their Use in Meteorology [in Russian], Atomizdat, Moscow (1965), p. 361.

13. B. Bennett, HASL-268 (1973).
14. A. C. Trusov, Author's Abstract of Dissertation, Moscow (1974).

NONSTATIONARY TRANSPORT OF THERMAL NEUTRONS IN A MODERATOR CONTAINING A LARGE SPHERICAL CAVITY

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A theoretical study was made [1, 2] of the transport of thermal neutrons in a spherical moderator containing a large spherical cavity at the center of which a pulsed isotropic source was located. It was shown that in a sufficiently long period of time after the neutron pulse, the law for reduction of neutron flux in the moderator and on its surface should be close to exponential with a damping constant $-\lambda_0$ which is uniquely related to the albedo averaged over the angles of neutron incidence and emergence. One should therefore expect that experiments of this kind would make it possible to measure the albedo of thermal neutrons with rather high accuracy.

However, the calculations in [1, 2] were based on certain assumptions which are not well satisfied in experiments published to the present time (for example, see [3,4]); a constant-cross-section approximation was used and the anistropy of neutron scatter in the moderator was neglected in [1, 2]. In addition, it was assumed that the average interval between time of neutron entrance into the moderator and the time of its return into the cavity (i.e., the average neutron residence time in the moderator) was small in comparison with the average neutron flight time across the cavity. As shown in [1 2], this assumption is satisfied if

$$R \gg \frac{3}{2} L_0 \quad \text{for} \quad H \gtrsim L_0 \tag{1}$$

and

$$R \gg H \qquad \text{for } H \ll L_0,$$

where R is the radius of the cavity, L_0 is the neutron diffusion length in the moderator, and H is the thickness of the moderator layer.

In this paper, therefore, nonstationary transport of thermal neutrons is considered in a spherical moderator containing a spherical cavity for an arbitrary ratio R/L_0 except for the conditions

$$R \gg \langle \Sigma_{tr}^{-1} \rangle_T; \quad L_0 \gg \langle \Sigma_{tr}^{-1} \rangle_T,$$

TABLE 1. Comparison of Experimental and Theoretical Results

R, cm	H, cm	ρ (B), mg/cm^3	$-\lambda_0$, sec^{-1}	$k_0(\lambda_0)$ (18)	$k_0(\lambda_0)$ (12)	$\langle D \rangle_T$, cm^2/sec	L_0, cm
13,26	7,5	0,00	2560±10	0,8265±0,0007	0,8249±0,0006	35493±80	3,997±0,017
13,4	4,0	0,00	2970±15	0,7985±0,0010	0,8057±0,0006	35519±80	4,427±0,027
10,4	7,0	0,00	3270±15	0,8263±0,0008	0,8305±0,0008	35537±80	4,848±0,036
10,4	10,36	0,00	3200±15	0,8299±0,0008	0,8360±0,0008	35533±80	4,739±0,033
13,26	7,5	0,95	4200±120	0,7249±0,0071	0,7130±0,0019	35051±81	1,930±0,014
10,4	10,36	1,046	5640±140	0,7117±0,0065	0,7104±0,0020	35085±81	1,989±0,017
13,26	7,5	2,37	4600±160	0,7013±0,0094	0,6084±0,0016	34261±81	1,242±0,006
15,76	5,0	2,6	4750±150	0,6420±0,0100	0,6018±0,0046	34139±81	1,189±0,005
13,4	4,0	2,6	5750±180	0,6329±0,0090	0,6029±0,0018	34200±82	1,214±0,005
10,4	7,0	2,6	6690±200	0,6642±0,0094	0,6016±0,0020	34257±82	1,240±0,007

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where Σ_{tr} is the macroscopic transport cross section of the moderator and the notation $\langle \ldots \rangle_T$ denotes averaging over a Maxwellian neutron spectrum corresponding to a moderator temperature T. In this case, the final results are presented in the form of expressions containing quantities directly measurable in pulsed experiments with finite blocks of moderator: $D_0 = \langle V/3\Sigma_{tr} \rangle_T$ is the diffusion coefficient and $\alpha = \langle v\Sigma_a \rangle_T$ is the probability of neutron absorption in the moderator per unit time (Σ_a is the macroscopic absorption cross section).

Derivation of General Formulas. Let $N^+(\mu, v, t)$ and $N^-(\mu, v, t)$ be the neutron fluxes at the surface of the moderator at an angle $\arccos \mu$ to the normal having a velocity v at the time t respectively escaping from the cavity into the moderator and vice versa. For a pulsed isotropic monoenergetic source (with a velocity v_s) of unit intensity located at the center of the sphere, these fluxes are associated through the relation

$$N^+(\mu, v, t) = N^-[\mu, v, t-(2R\mu/v)] + (1/4\pi R^2)\,\delta(1-\mu)\,\delta(v-v_s)\,\delta[t-(R/v_s)]. \quad (2)$$

When the condition $L_0 \gg \langle \Sigma_{tr}^{-1} \rangle_T$ is satisfied, neutron transport in the moderator can be described in the diffusion approximation. In addition, one can roughly assume the neutron spectrum in the moderator is Maxwellian, M(v). Then the total neutron density n(r, t) in the moderator at a distance r from the center of the sphere satisfies the equation

$$[\partial n(r, t)/\partial t] = \langle D \rangle_T \Delta n(r, t) - \alpha n(r, t) \quad (3)$$

for the boundary conditions

$$n(R+H+z_0, t) = 0 \quad (4)$$

(z_0 is the extrapolation length) at the external boundary, and

$$N^-(\mu, v, t) = \frac{M(v)}{2}\left[\mu v n(R, t) + 3\mu^2 D(v) \frac{\partial n(r, t)}{\partial r}\Big|_{r=R}\right]; \quad (5)$$

$$\int_0^\infty dv \int_0^1 d\mu N^+(\mu, v, t) = \frac{\langle v \rangle_T}{4} n(R, t) - \frac{\langle D \rangle_T}{2} \frac{\partial n(r, t)}{\partial r}\Big|_{r=R} \quad (6)$$

at the internal boundary.

Applying the Laplace transform with respect to t to Eqs. (2)-(6) and denoting the Laplacian forms of the functions by a tilde, we obtain:

$$\tilde{N}^+(\mu, v, \lambda) = e^{-\frac{2R\mu}{v}\lambda} \tilde{N}^-(\mu, v, \lambda) + \frac{1}{4\pi R^2}\,\delta(1-\mu)\,\delta(v-v_s)\,e^{-\frac{R}{v_s}\lambda}; \quad (2a)$$

$$\lambda \tilde{n}(r, \lambda) = \langle D \rangle_T \Delta \tilde{n}(r, \lambda) - \alpha \tilde{n}(r, \lambda); \quad (3a)$$

$$\tilde{n}(R+H+z_0, \lambda) = 0; \quad (4a)$$

$$\tilde{N}^-(\mu, v, \lambda) = \frac{M(v)}{2}\left[\mu v \tilde{n}(R, \lambda) + 3\mu^2 D(v) \frac{\partial \tilde{n}(r, \lambda)}{\partial r}\Big|_{r=R}\right]; \quad (5a)$$

$$\int_0^\infty dv \int_0^1 d\mu \tilde{N}^+(\mu, v, \lambda) = \frac{\langle v \rangle_T}{4} \tilde{n}(R, \lambda) - \frac{\langle D \rangle_T}{2} \frac{\partial \tilde{n}(r, \lambda)}{\partial r}\Big|_{r=R}. \quad (6a)$$

The solution of Eq. (3a) under condition (4a) takes the form

$$\tilde{n}(r, \lambda) = \frac{A}{r}\left(e^{-\frac{r}{L}} - e^{\frac{r}{L}} e^{-2\frac{R+H+z_0}{L}}\right), \quad (7)$$

$$L = \sqrt{\langle D \rangle_T/(\alpha+\lambda)} \quad (8)$$

(here A is a constant which generally depends on λ). Hence, we have in particular

$$\frac{\partial \tilde{n}(r, \lambda)}{\partial r}\Big|_{r=R} = -\tilde{n}(R, \lambda)\left(\frac{1}{R} + \frac{1}{L}\operatorname{cth}\frac{H+z_0}{L}\right). \quad (9)$$

Substituting Eqs. (2a), (5a), and (9) into Eq. (6a), we obtain

$$\tilde{n}(R, \lambda) = \frac{1}{\pi R^2} e^{-\frac{R}{v_s}\lambda}\left\{\langle v \rangle_T - \left\langle v \int_0^1 2\mu\, d\mu e^{-\frac{2R\lambda}{v}\mu} \right\rangle_T + \right.$$

$$+2\left(\frac{1}{R}+\frac{1}{L}\operatorname{cth}\frac{H+z_0}{L}\right)\left(\langle D\rangle_T+\left\langle D\int_0^1 3\mu^2\,d\mu e^{-\frac{2R\lambda}{v}\mu}\right\rangle_T\right)\Bigg\}^{-1}. \tag{10}$$

One can similarly obtain an expression for $\tilde{N}^{\pm}(\mu, v, \lambda)$ and determine $n(r,t)$ and $N^{\pm}(\mu, v, t)$ by means of the inverse Laplace transform. In particular, we obtain for the total neutron flux emerging from the moderator into the cavity.

$$N^{-}(t)\equiv\int_0^1 d\mu\int_0^\infty dv N^{-}(\mu, v, t)=\frac{1}{8\pi^2 i R^2}\int_{\sigma-i\infty}^{\sigma+i\infty} e^{\lambda\left(t-\frac{R}{v_s}\right)}\frac{d\lambda}{\frac{1}{k_0(\lambda)}-f\left(\frac{2R\lambda}{v_T}\right)}; \tag{11}$$

$$k_0(\lambda)=\frac{1-2\left(\frac{1}{R}+\frac{1}{L}\operatorname{cth}\frac{H+z_0}{L}\right)\frac{\langle D\rangle_T}{\langle v\rangle_T}}{1+2\left(\frac{1}{R}+\frac{1}{L}\operatorname{cth}\frac{H+z_0}{L}\right)\frac{\langle D\rangle_T}{\langle v\rangle_T}}; \tag{12}$$

$$f(x)=\frac{\left\langle v\int_0^1 2\mu\,d\mu e^{-\frac{v_T}{v}\mu x}\right\rangle_T-2\left(\frac{1}{R}+\frac{1}{L}\operatorname{cth}\frac{H+z_0}{L}\right)\left\langle D\int_0^1 3\mu^2\,d\mu e^{-\frac{v_T}{v}\mu x}\right\rangle_T}{\langle v_T\rangle-2\left(\frac{1}{R}+\frac{1}{L}\operatorname{cth}\frac{H+z_0}{L}\right)\langle D_T\rangle}, \tag{13}$$

where $v_T=\sqrt{2kT/m}=\sqrt{\pi/2}\,\langle v\rangle_T$; m is the neutron mass.

Since for small t (of the order of 2-3 times the mean neutron flight time across the cavity), the neutron distribution essentially depends on the initial conditions (which in this case differ greatly from those realized experimentally), the region of large t is of practical interest where only the normalization of the total flux depends on the initial conditions. In this region, the integral (11) is mainly determined by the behavior of the integrand for relatively small values of $|\lambda|$, namely for $|2R\lambda/v_T|\equiv|x|\leqslant 1$.

We first consider the behavior of the quantity f in this region. We represent Eq. (13) in the form

$$f(x)=\frac{1}{\langle v\rangle_T}\left\langle v\int_0^1 2\mu\,d\mu e^{-\frac{v_T}{v}\mu x}\right\rangle_T\frac{1-\varepsilon(x)}{1-\varepsilon(0)}; \tag{13a}$$

$$\varepsilon(x)=2\left(\frac{1}{R}+\frac{1}{L}\operatorname{cth}\frac{H+z_0}{L}\right)\frac{1}{\langle v\rangle_T}\left\langle D\int_0^1 3\mu^2\,d\mu e^{-\frac{v_T}{v}\mu x}\right\rangle_T. \tag{14}$$

It is necessary to know the dependence of D on v for an exact calculation of $\varepsilon(x)$. However, the difference between the factor $(1-\varepsilon(x))/(1-\varepsilon(0))$ and unity can be neglected in first approximation when $|x|\leqslant 1$.* Accordingly, we obtain

$$f(x)\approx\frac{1}{\langle v\rangle_T}\left\langle v\int_0^1 2\mu\,d\mu e^{-\frac{v_T}{v}\mu x}\right\rangle_T\equiv 4\int_0^\infty du e^{-u^2}u^3\int_0^1 d\mu\mu e^{-\frac{\mu}{u}x}. \tag{15}$$

This integral was investigated previously [1, 2] and a series expansion obtained for it which is valid for $x<0$. In particular, one can use the approximate expression

$$f(x)\approx 1-\frac{\sqrt{\pi}}{3}x+\frac{1}{4}x^2-\frac{\sqrt{\pi}}{15}x^3+\frac{1}{14}\left(\frac{9}{4}-\frac{3}{2}C-\ln x\right)x^4;\ (C=0.5772\ldots) \tag{16}$$

when $|x|\leqslant 1$.

As is easily verified, the quantity $k_0(0)$ is the albedo averaged over the angles of the incident and emerging neutrons in the diffusion approximation for a stationary source and $k_0(\lambda)$ is the same quantity in which the macroscopic absorption cross section Σ_a is replaced by $\Sigma_a+\lambda/v$. As should be expected, Eq. (12) agrees with the previously obtained expression [1, 2] for the average albedo, $k_0(0)$, in the approximation of constant cross sections and isotropic scatter for neutrons with $L/R\to 0$ and $\lambda/v\Sigma_a\to 0$.

If one replaces $k_0(\lambda)$ by $k_0(0)$ in the integral (11) and uses Eq. (15) for f(x), the total flux of neutrons emerging from the moderator into the cavity, which was determined in [1, 2], will be described accurately except for a normalization factor (connected with a different selection of source intensity). In the general

*For example, if $D\sim v$, $\frac{1-\varepsilon(x)}{1-\varepsilon(0)}=\frac{[f(x)-f(0)]^2}{16}+O(x^3)$.

case, this integral describes the neutron flux for arbitrary energy dependence, scattering anisotropy and L/R ratio as long as $R \gg \langle \Sigma_{tr}^{-1} \rangle_T$ and $L \gg \langle \Sigma_{tr}^{-1} \rangle_T$.

We consider the properties of the function $k_0(\lambda)$ for small λ. We first note that in first approximation one can neglect the dependence of $\langle D \rangle_T$ on Σ_a, and consequently on λ also, and assume

$$\langle D \rangle_T \approx \left\langle \frac{v}{3\Sigma_{tr}} \right\rangle \equiv D_0. \tag{17}$$

A more exact expression for $\langle D \rangle_T$ can be obtained by using the following chain of approximate relations:

$$\begin{aligned} \langle D \rangle_T &\equiv \left\langle \frac{v}{3(\Sigma_{tr}+\Sigma_a+\lambda/v)} \right\rangle_T \\ &\approx \left\langle \frac{v}{3\Sigma_{tr}} \left(1 - \frac{v\Sigma_a+\lambda}{v\Sigma_{tr}}\right) \right\rangle_T \approx \left\langle \frac{v}{3\Sigma_{tr}} \right\rangle_T \\ &-(\alpha+\lambda) \left\langle \frac{1}{3\Sigma_{tr}^2} \right\rangle_T \approx D_0 \left[1 - \frac{3D_0}{\langle v \rangle_T^2} (\alpha+\lambda)\right]. \end{aligned} \tag{17a}$$

If Σ_{tr} does not depend on velocity, this expression accurately described the coefficient for $(\alpha+\lambda)$ and the error in this coefficient is 10% at most when $\Sigma_{tr} \sim 1/v$. Since the term $3D_0/\langle v \rangle_T^2 (\alpha+\lambda)$ itself is small in comparison with unity * in the overwhelming majority of cases, Eq. (17a) is a sufficiently good approximation. Thus, Eqs. (8), (12), and (17a) allow one to express k_0 satisfactorily as a function of λ through the experimentally determined parameters D_0, α, and $\langle v \rangle_T$.

When $|\lambda| < \alpha$ the function k_0 is analytic (varying rather weakly). Therefore, consideration of the dependence of k_0 on λ when $|\lambda| < \alpha$ practically leaves unchanged the functional structure of the integrand in Eq. (11). It was shown earlier [1, 2] that the main contribution to the integral in Eq. (11) may be made by two maxima of the integrand, one located at $\lambda = -4/t$ and the second at $\lambda = \lambda_0$, where λ_0 is a root of the equation †

$$\frac{1}{k_0(\lambda)} = \mathrm{Re}\left[f\left(\frac{2R\lambda}{v_T}\right)\right]. \tag{18}$$

The first maximum exists only for very large t ($t \gg 4/|\lambda_0|$). Since small λ correspond to such t, one can replace $k_0(\lambda)$ by $k_0(0)$ in calculating the contribution $\Delta N^-(t)$ from this maximum in the integral (11), i.e., the results obtained previously [1, 2] are completely applicable. Accordingly, we have‡

$$\Delta N^-(t) = \frac{16}{3} \frac{k_0^2(0) R^2 |\lambda_0|^5}{[1-k_0(0)]^2 v_T^4 z^5} \left[1 - \left(1 + z + \frac{z^2}{2!} + \frac{z^3}{3!} + \frac{z^4}{4!} + \frac{z^5}{4!}\right) e^{-z}\right], \tag{19}$$

$$z = |\lambda_0| (t - R/v_s).$$

The second maximum is very narrow and high. It makes a contribution to the total flux which is given by

$$\begin{aligned} N_e^-(t) &= \frac{v_T k_0(\lambda_0)}{2R^3 \varphi(\lambda_0)} e^{\lambda_0 \left(t - \frac{R}{v_s}\right)}, \\ \varphi(\lambda_0) &= \frac{d\,\mathrm{Re}\,[f(x)]}{dx}\Big|_{x=-\frac{2R\lambda_0}{v_T}}. \end{aligned} \tag{20}$$

The total flux is

$$N^-(t) = N_e^-(t) + \Delta N^-(t). \tag{21}$$

When $t \lesssim 4/|\lambda_0|$, the second term in this equation is small, i.e., the flux decreases in time approximately exponentially with a damping constant $-\lambda_0$. For large t, deviation from the exponential law appears as the result of an increase in the role of the second term, and the total flux $N^-(t) \sim t^{-5}$ for $t \to \infty$.

Thus, consideration of the finiteness of neutron migration time in the moderator and of the radius of the cavity does not change the picture of the evolution of total neutron flux from the moderator into the

* For example, in the experiments of Antonov et al. [4], it does not exceed 5% as is seen from the data given in the seventh column of the table.

† This root is approximately equal to the root of smallest absolute value for the equation $1/k_0(\lambda) = 2\int_0^1 d\mu\mu \exp[-2R\mu\lambda/v_T]$ corresponding to the one-velocity approximation [1, 2].

‡ In this expression, terms proportional to e^{-z} were obtained on the basis of crude approximations and are an estimate.

cavity (and correspondingly of $N^{\pm}(\mu, v, t)$ also) and merely amounts to a small change in the damping constant of the exponential component of the flux resulting from the difference between $k_0(\lambda)$ and $k_0(0)$.

Comparison with Experimental and Discussion of Results. The greatest interest is in the experimental investigation of that region of values of t where $N^-(t)$ (and also $N^{\pm}(\mu, v, t)$ and $n(r, t)$] decreases almost exponentially since measurement of $-\lambda_0$ makes it possible to determine $k_0(\lambda_0)$ [from Eq. (15) or Eqs. (16) and (18)], i.e., the average albedo for various rates of neutron flux damping and, in particular, the albedo in the stationary case [extrapolation of the data to the point $\lambda_0 = 0$ or conversion by means of Eqs. (8), (12), and (17a)]. This region was investigated in experiments [4] with pure water and water with varying boron content $\rho(B)$. The experimental results are shown in the first four columns of Table 1. Values of $k_0(\lambda_0)$ calculated from Eqs. (16) and (18) are given in the fifth column. The errors shown there (determined from the expression $\Delta k_0 = |\partial k_0(\lambda_0)/\partial \lambda_0| \Delta \lambda_0$) illustrate the accuracy of the method and the sensitivity of the dependence of k_0 on λ_0. Calculated results for $k_0(\lambda_0)$ in the diffusion approximation, for $\langle D \rangle_T$, and for L respectively obtained from Eqs. (12), (17a) and (8) are in the sixth, seventh, and eighth columns. The following data were used in the calculations: $D_0 = 35630 \pm 80$ cm^2/sec; $\alpha(H_2O) = 4782 \pm 15$ sec^{-1} [5]; $\sigma_\alpha(B) = 758 \pm 4$ b at $v = 2200$ m/sec [6]; $v_T = \sqrt{\pi/2}(v)_T = 2202$ m/sec ($T = 21°C$); $z_0 = 2.13 D_0/\langle v \rangle_T = 0.305$ cm.

The errors given result only from the errors specified for these parameters. It is clear that the agreement of experimental and theoretical values for $k_0(\lambda_0)$ (fifth and sixth columns) is very good for water not highly poisoned by boron ($\rho(B) \leqslant 1$ mg/cm^3). The results given in the first two rows of the table are of particular interest. A marked increase in λ_0 (by 16%) is observed experimentally for a reduction in H from 7.5 to 4 cm and practically constant R. It corresponds to a marked change in $k_0(\lambda_0)$* which is difficult to explain if one does not take into account the significant increase of $L(\lambda_0)$ in comparison with the diffusion length for pure water in the stationary case (2.74 cm), as is clear from the last column of the table. If this agreement is not accidental, the results obtained demonstrate the validity of the diffusion approximation in the albedo calculation for a nonstationary source [i.e., Eq. (12)] and a weakly absorbing moderator; the possibility of albedo measurement in this case with an error of a fraction of a percent in experiments such as [4]; the possibility of determining in such experiments parameters such as D_0 and α with an error of a few percent at most. Since the actual possibilities for an increase in accuracy were far from exhausted in the experiment [4], there is reason to hope that this method can prove to be sufficiently effective.

As boron concentration is increased, the agreement between the values of $k_0(-\lambda_0)$ determined from experiment [from Eq. (18)] and in the diffusion approximation becomes somewhat worse. A possible reason for this is the inaccuracy of the diffusion approximation both with respect to the transport of neutrons in the moderator [Eq. (3)] and in the description of neutron fluxes from the moderator into the cavity and vice versa [Eqs. (5) and (6)]. Some error may also be introduced by the approximate equation (15) and by the assumption about a Maxwellian neutron spectrum in the moderator, i.e., the neglect of diffusion cooling. Finally, one cannot exclude the fact that experimental data for λ_0 at high boron concentrations is unreliable. Besides purely experimental difficulties, this can be associated with the uncertainty in the separation of the exponentially damped component from the total neutron flux, since the time period in which the nonexponential damping component is small decreases as the absorption increases. The existence of a nonexponentially damping component appears as an effective reduction of $|\lambda_0|$, i.e., leads to an overestimate of $k_0(\lambda_0)$ which is larger for larger t. Thus the question of albedo measurement in experiments such as [4] requires additional experimental and theoretical study for highly absorbing moderators.

LITERATURE CITED

1. K. D. Ilieva and M. V. Kazarnovskii, Kratkie Soobsh. po Fizike, No. 3, 19 (1973); At. Énerg., 35, 346 (1973).
2. K. D. Ilieva, Cand. Diss. FIAN, Moscow (1973).
3. G. Grosshug and G. Ronberg, Nucl. Energy, 25, 361 (1971).
4. A. V. Antonov et al., Kratkie Soobsh. po Fiziki, No. 11, 34 (1973).
5. E. Arai and M. Muchle, Nukleonik, 7, 416 (1965).
6. K. Beckurts and K. Wirtz, Neutron Physics, Springer-Verlag (1964).

*The quantity $1 - k_0(\lambda_0)$, which characterizes the fraction of "absorbed" neutrons, increases by 11% in this case.

EFFECT OF THE SHAPE OF THE PHASE-TRANSFORMATION FRONT ON THE TEXTURE OF BETA-QUENCHED URANIUM

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It is a well-known fact [1-4] that the texture found in quenched uranium does not often represent the remaining traces of the original, incompletely destroyed working texture, but rather arises in the course of quenching. The extent and character of the texture are in fact related in a very specific manner to the conditions of the quenching process (the shape of the phase interface, the stresses arising in the $\beta \rightarrow \alpha$ transition, and so on). The question as to the nature of the quenching texture and the effect of stresses on texture formation was studied in considerable detail earlier [5, 6]. It was found that during the martensitic transformation the rate at which the α uranium crystals formed and grew was determined by the interaction between the motive and restrictive forces (the former including the difference between the free energies of the phases, and the latter principally including the elastic deformation of the matrix and crystal nucleus), while the quenching texture was determined by the interaction between stresses of the second kind (arising during the generation and growth of the α crystals) and stresses of the first kind already existing in the quenched sample. Depending on the orientation of the nucleus, the external forces deforming the β matrix may either reduce or increase the degree of lattice distortion created by the formation and growth of the martensitic nucleus, and hence either favor or oppose the generation and growth of plates of the new phase. With this in mind the earlier authors studied the influence of tensile stresses (applied to a uranium sample in the course of quenching) on the quenching texture; for high cooling rates this led to the appearance of [010] texture in the direction of the applied forces. Taking account of Lomer's theoretical calculation [7] regarding the mode of deformation of the uranium lattice during the $\beta \rightarrow \alpha$ transformation, the deformations of the β matrix were then calculated in the direction of the axes of the nucleating α uranium crystals for the three cases of nucleus orientation in the matrix described in the literature; it was found that in the case of two of these orientations the generation of an α crystal required the extension of the β matrix in the [010] direction of the nucleating crystal. It was thus established in these papers that the main reason for the appearance of quenching texture lay in the orienting action of the stress field on the generation and growth of the crystals.

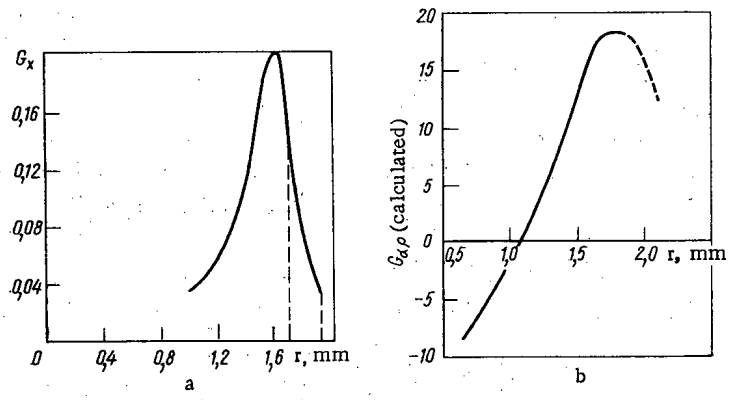

Fig. 1. Distribution of the growth indices G_x (a) and $G_{\alpha\rho}$ (b) over the cross section of a β-quenched uranium rod.

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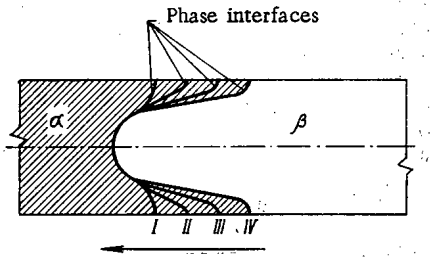

Fig. 2

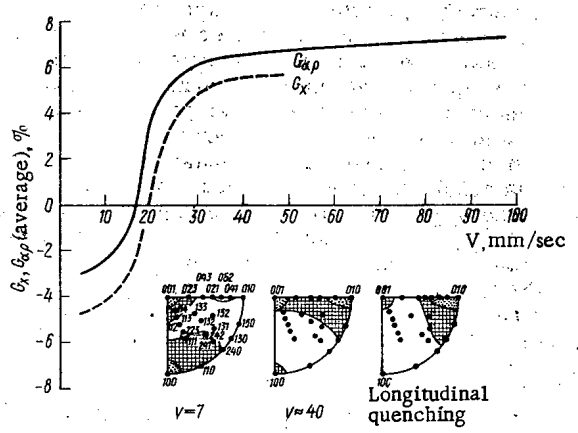

Fig. 3

Fig. 2. Lines of phase separation in uranium rods on quenching at different velocities in the cooling systems. I), II), III), IV) Rods moving at 14, 21, 28, and 41 mm/sec, respectively (longitudinal section in the diagram). (The arrow shows the direction of rod motion.)

Fig. 3. Texture and growth indices G_x (----); $G_{\alpha\rho}$ (———) β-quenched uranium rods as functions of v, their velocity in the cooling system.

On quenching uranium-based objects, considerable structural and thermal stresses arise in the latter, according to the particular cooling conditions employed. The magnitude and character of the stresses change as the phase interface penetrates into the interior of the sample, and this should lead to a change in the intensity of the quenching texture.

In this paper we shall analyze the effect of the orientation of the phase interface (i.e., the front or leading edge of the phase transformation) on the distribution of quenching texture over the sample cross section.

Experimental Material and Method. As original material we used uranium rods of reactor purity 6.5-7 mm in diameter prepared by the extrusion of the α phase in the high-temperature region. The diameter of the rods was then reduced to 4 mm by drawing at 200-250°C. Quenching from the β phase was effected by the progressive motion of the rods through the inductor of a high-frequency generator and a water spray. The temperature to which the rods were heated was 740°C and the holding period 7 sec. The velocity of the rod was varied from 7 to 100 mm/sec in different experiments.

We studied the texture by x-ray structural analysis [8] and also by the method of Stobo and Pawelski [9], including measurement of the specific electrical resistance and the linear thermal-expansion coefficient of the samples, and calculation of the growth index $G_{\alpha\rho}$, expressing the proportion of texture in percents.

Results and Discussion. Let us first consider the distribution of quenching texture in a rod quenched by cooling over the whole surface (the so-called longitudinal form of cooling).

It was found earlier [3, 10] that in this case there was a nonuniform distribution of texture over the cross section of the quenched uranium (Fig. 1a). Subsequent analysis of the texture distribution (using layer-by-layer etching of the samples) and a determination of the degree of texture by calculating this for each layer (each layer being 0.1-0.2 mm thick) enabled us to refine the true picture of the quench-texture distribution over the cross section of the rod (see Fig. 1b). It follows from Fig. 1 that the value of the growth indices, while tending to zero at the actual surface, first increases sharply on moving toward the center and then diminishes; at a certain distance from the axis of the rod it changes sign, which indicates a change in the type of texture.

These results may be explained on the basis of the foregoing consideration regarding the character of the $\beta \rightarrow \alpha$ transformation in uranium, allowing for the effects of developing stresses on this process. During the $\beta \rightarrow \alpha$ transformation a complex field of structural and thermal stresses due to the presence of phases with different specific volumes and also to the existence of a temperature gradient is established over the cross section of the uranium rod. An analysis of the interactions of these stresses shows that their sign and level both change as the phase boundary passes from the surface to the center of the sample.

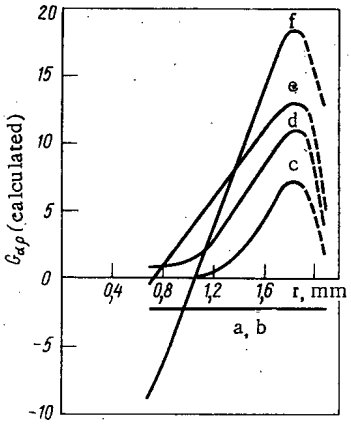

Fig. 4

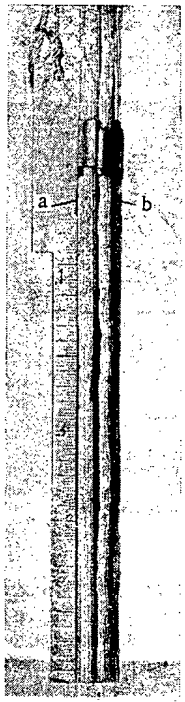

Fig. 5

Fig. 4. Distribution of texture over the cross section of β-quenched uranium rods for various velocities through the cooling system a, b) 7 and 14; c) 21; d) 28; e) 41; f) 100 mm/sec.

Fig. 5. Distortion of a rod-type fuel element after irradiation, caused by the loss of axial symmetry in the distribution of quenching texture over the core cross section.

In layers close to the surface, longitudinal tensile stresses are chiefly operative at the instant of the $\beta \rightarrow \alpha$ transformation, while the sample also experiences a radial compression. This ultimately determines the appearance of a [010] texture in the surface layers of the metal in the direction of rod axis. The reduction in the growth indices at the actual surface, at which the stress concentration is greatest, may be explained by the special conditions governing the cooling of the surface layer (namely, its greater cooling rate; the consequent high degree of supercooling of the β phase may lead to a reduction in the mobility of these structural defects which facilitate the transformation and the development of the [010] quenching texture). As the phase boundary moves toward the center of the sample, tensile radial stresses are created owing to the fact that the outer layers of the metal have already cooled while the inner layers are still undergoing a cooling process. This causes a change in the orientation of the [010] direction: The direction sets along the radius of the rod and the proportion of this type of texture increases as the center of the sample is approached. Along the axis of the rod a texture of the [100] type then develops and rapidly intensifies (Fig. 1b).

Thus, the experimentally established distribution of texture may be explained quite satisfactorily. The chief reason for the formation of the quenching texture in the present case is the stress field due to the manner of cooling the rod.

In the case under consideration, the conditions of longitudinal quenching are quite easily realized for a velocity of over 40 mm/sec in the cooling system. For lower velocities the phase interface is a cone of irregular shape (Fig. 2). The dependence of the quench texture on the velocity in the cooling system is shown in Fig. 3. We see from this figure that the type of texture and the textural intensity largely depend on the velocity of the rod through the water spray. The distribution of the texture over the cross section also changes (Fig. 4).

As already indicated, at velocities exceeding 40 mm/sec the shape of the phase interface is almost cylindrical. Hence, the quenching-texture and stress distributions over the cross section coincide with

those already considered for the case of longitudinal quenching. On reducing the pulling velocity the height of the irregular cone formed by the phase interface diminishes, and this leads to a corresponding change in the direction of the forces acting at the phase interface; the angle between the direction of the tensile forces and the axis of the uranium core increases with diminishing velocity. The effect of the compressive radial stresses also diminishes. All this leads to a reduction in the amount of [010] texture along the axis of the rod and the degree of textural nonuniformity over the cross section. At the lowest velocity of 7 mm/sec, a faint texture of the [100] type is created along the rod axis, this being distributed uniformly over the cross section.

It follows from the foregoing considerations that uranium rods of reactor purity quenched from the β phase hardly ever possess a quasi-isotropic structure, since in this case the phase transformations always take place in the presence of a sharply expressed phase interface and consequent high structural and thermoelastic stresses. By choosing the quenching technology in a suitable manner, in particular by selecting the configuration of the phase interface in a specific fashion, we may ensure that the quenching texture should be only weakly expressed, and most of all compensated over the cross section of the core. However, under ordinary working conditions, in which different parts of the fuel-element cross sections are subject to very different external effects (temperature, neutron density, etc.), this circumstance may lead to undesirable consequences. Thus, on irradiating rod-shaped fuel elements containing cores having a texture similar to that of Fig. 1 b (considerable in magnitude but compensated over the cross section) serious radiation-induced lengthening of the fuel elements was encountered ($G_i = 20$) for a surface temperature of about 350°C; this was associated with the existence of [010] texture in the surface layers.

This may be explained by the fact that the central layers of the core then had a temperature of about 400°C, and were thus less subject (by comparison with the outer layers) to radiation-induced growth; they did not, therefore, have the appropriate compensating action on the general change in the shape of the core. In an analogous way the nonuniform radiation-induced growth of the peripheral and central layers in the fuel-element cores of the Calder Hall reactors [11] led to a serious distortion of the ends of the core (a meniscus being created), which in turn caused the fuel elements to go out of action owing to the damage incurred by the can.

If the quenching texture is uniformly distributed over the cross section of the core (Fig. 4, curves a and b), the [010] texture is formed at an angle of ~45° to the core axis, and is distributed in the shape of a fan. If the symmetry of disposition of the [010] texture is preserved all along the rod and compensation is also ensured, the shape and size of the fuel elements remain perfectly stable under irradiation (Fig. 5a). If the compensation of the texture is disrupted, the irradiation of the cores of rod-like fuel elements leads to a serious distortion, appearing as indentations and projections (Fig. 5b).

Thus, the texture of a quenched uranium rod of reactor purity depends very substantially on the configuration of the phase interface on quenching. By modifying the shape of the phase interface we may alter the type of quenching texture, the extent of its development, and the character of its distribution over the cross section of the rod.

These laws agree closely with the earlier concept of the orienting influence of stresses on the generation and growth of α uranium crystals during the phase transition.

The structure of quenched uranium is usually far from quasi-isotropic. The radiation-induced deformation of the fuel-element cores may be suppressed by the creation of texture which is compensated over the whole cross section of the core.

LITERATURE CITED

1. I. Gittus et al., in: Proc. IAEA Symp. Radiation Damage in Solids and Reactor Materials, Venice (May 1962), p. 11. Preprint SM-25/83, Vienna (1962).
2. B. Butcher, in: Transactions of the Spring Symp. of the Inst. of Metals on Uranium and Graphite [Russian translation], London (March 1962), Paper No. 1.
3. V. E. Ivanov et al., At. Énerg., 16, No. 4, 325 (1964).
4. I. V. Batenin and B. V. Sharov, At. Énerg., 16, No. 4, 372 (1964).
5. V. F. Zelenskii, V. S. Krasnorutskii, and V. V. Kunchenko, At. Énerg., 27, No. 5, 411 (1969).
6. V. F. Zelenskii and V. S. Krasnorutskii, At. Énerg., 31, No. 5, 449 (1971).
7. W. Lomer, Inst. Metals, London, Monogr. and Rep., Ser. 18 (1955), p. 243.
8. E. Sturcken and W. McDonnell, J. Nucl. Mater., 7, No. 1, 85 (1962).

9. Y. Stobo and B. Pawelski, J. Nucl. Energy, 16, Part a/b (1962).
10. V. F. Zelenskii et al., At. Energ., 21, No. 3, 192 (1966).
11. I. Gittus, Brit. Nucl. Energy Soc., 3, No. 2, 106 (1964).

PENETRATION OF NIOBIUM BY D^+ IONS AND OUTGASSING OF DEUTERIUM

A. A. Pisarev and V. G. Tel'kovskii

UDC 533.924;539.12.17

In constructing a thermonuclear reactor, problems will arise in connection with the interaction between the plasma and the vacuum wall [1, 2]. Part of this interaction involves penetration of structural materials by hydrogen isotope ions. This process must be taken into account in dealing with the problems of plasma cooling by particles released from the walls, production of thermonuclear fuel, and changes in the physical and mechanical properties of metals impregnated with gases. The energy of the plasma ions is on the order of tens of keV. Niobium is a possible material for the vacuum wall of a thermonuclear reactor.

The absorption of hydrogen isotope ions with energies of tens of keV in various metals was studied in [3-9]. However, many questions were left unanswered. In this paper we present the results of a study of the capture and outgassing of deuterium ions with energies of 5-35 keV in niobium at temperatures of 290-1500°K.

Experimental Apparatus and Technique. The experiments on absorption of D^+ ions were carried out on a mass monochromator [10] designed for production of ions resolved in mass and energy with a coefficient of about 200. The base vacuum of the system is $1 \cdot 10^{-6}$ torr. A bombardment target of dimensions $70 \times 10 \times 0.1$ mm can be water cooled and ohmically heated. The target material is NVCh brand niobium of 99.99% purity. The temperature is monitored by a thermocouple. The partial pressure of deuterium is measured by a type IPDO-1 mass spectrometer.

The experiment was carried out as follows. The measurement chamber was pumped to a base pressure of about 10^{-6} torr, and the target was then annealed at a temperature of about 1500°K. Then the target was cooled to room temperature and subjected to bombardment by an ion beam having a density of about 50 $\mu A/cm^2$ uniformly distributed over an area of about 0.05 cm^2. In the experiments on irradiation of samples at high temperature the heat was turned on at the same time the beam was directed onto the target. The amount of bombarding particles was determined from the beam current with a correction for secondary ion and electron emission. After irradiation of the sample the remaining gas was pumped out. The target was heated until all the gas which had penetrated it was removed. As this was done the temperature increased continuously due to a linear increase in time of the heating voltage. The gas yield was examined in the so-called dynamic regime, in which the rate of outgassing from the sample (j) is proportional to the amount by which the partial pressure of the gas under study exceeds its base pressure during the heating process and the number of imbedded particles is proportional to the area under the curve j(t), where t is the time from the onset of heating. Before each new cycle of penetration-outgassing the sample was held at a temperature considerably higher than that at which outgassing ceased when the sample was baked out following ion bombardment.

Results and Discussion. These experiments used atomic (D_1^+) and molecular (D_2^+) ions. It is found that the outgassing curves obtained by heating after irradiation with molecular and atomic ions are the same if the energy of the molecular ions is twice that of the atomic ions and the dose of D_2^+ is half that of D_1^+, that is, if the energy per deuteron and the number of bombarding dueterons are the same. An analogous result was obtained in experiments with H_1^+ and H_2^+ ions bombarding molybdenum [11].

Following irradiation of the sample, as its temperature is increased, the outgassing rate j increases up to a maximum and then falls to zero. On analyzing the j(T) curves, one can see that they have

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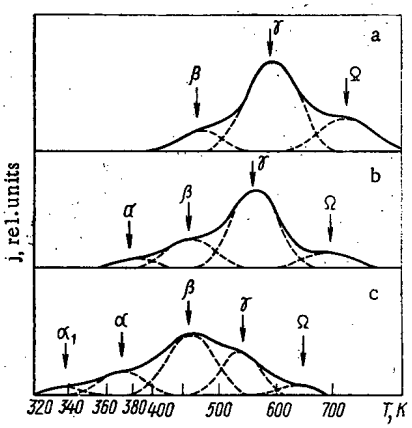

Fig. 1

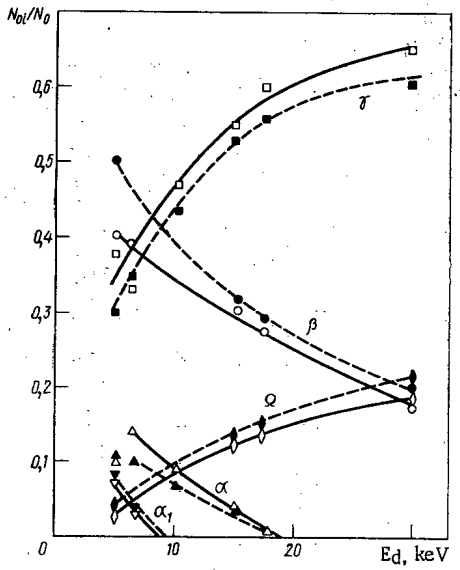

Fig. 2

Fig. 1. Dependence of the deuterium outgassing rate on the baking temperature following irradiation of niobium with (a) D_1^+ ions of energy 30 keV and (b, c) D_2^+ with energies of 35 and 10 keV. The dashed curves show the resolution of the outgassing curves into separate peaks.

Fig. 2. Dependence of the fraction of particles released in the individual peaks on the energy (per deuteron) of the bombarding ions: the continuous and dashed curves correspond to irradiation doses of $2 \cdot 10^{17}$ and $1 \cdot 10^{16}$ deuterons/cm^2, respectively.

a fine structure; that is, they consists of several peaks. Figure 1 shows the functions j(T) for various energies of the bombarding deuterons and shows the resolution of these curves into separate peaks. The irradiation dose was $3 \cdot 10^{17}$ deuterons/cm^2 and the heating rate, about 3-7°K/sec. It is clear from the figure that during the heating period outgassing of deuterium begins at 320-400°K and ends at 700-850°K, depending on the irradiation conditions. The beginning and the end of outgassing are defined by the lowest and highest temperature peaks, respectively. The maximum outgassing rate in the peaks is reached at T_{max} = 340, 370, 470, 540-610, and 650-730°K for the α_1-, α-, β-, γ- and Ω peaks, respectively. The temperature at which the maximum occurs for the last two peaks increases with the deuteron energy.

The fine structure in the j(T) curves means that the deuterium outgassing following D^+ ion bombardment is determined by several processes, each of which corresponds to one of the peaks in Fig. 1. In this case the relative amount of gas given off in the i-th peak (N_{0i}/N_0) depends strongly on the energy of the deuterons and comparatively weakly on the dose over the range E_d = 5-35 keV and $N_0 = 1 \cdot 10^{16} - 6 \cdot 10^{17}$ deuterons/cm^2. The total amount of gas captured in the target at a temperature of 300°K and released when it is heated (N_0) is constant for all energies and increases linearly with the irradiation dose. The latter result agrees with the data of [3, 7]. Figure 2 shows the dependence of the fraction of particles released in the individual peaks on the ion energy per deuteron (E_d). It follows from the figure that the fraction in the high temperature peaks increases with energy, while the amount of gas in the low temperature peaks decreases with increasing E_d.

The form of the function $T_{max}(E_d)$ for the γ- and Ω peaks indicates (according to [12]) diffusive release of deuterium in these peaks. No change in T_{max} with E_d is observed in the α and β peaks. This is evidence that the deuterium yield in the low temperature part of j(T) is described by the desorption equation and not by the diffusion equation [13, 14]. However, the appearance of the α_1 through β peaks cannot be a consequence of desorption of deuterium from the surface since the amount of gas in these peaks is very large and the yield in them is determined by a first order equation, while desorption of a molecular gas obeys a second order equation.

The nature of the fine structure of the outgassing spectra may be explained by considering the possible states of deuterium in the volume of the niobium. The principal states are solid interstitial solutions and deuteride compounds of various compositions [15-17]. Displacement solutions are also possible due

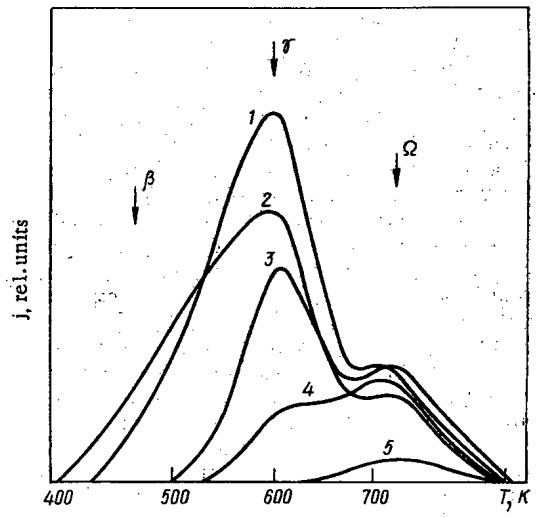

Fig. 3

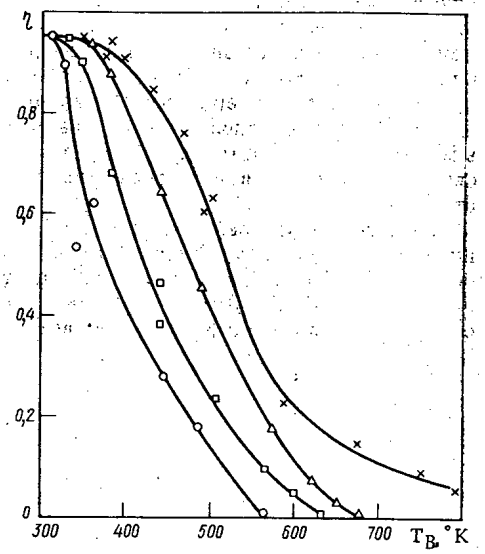

Fig. 4

Fig. 3. Deuterium outgassing curves obtained during heating of the target following bombardment of niobium at various temperatures with D_1^+ ions: 1) 290; 2) 400; 3) 500; 4) 550; 5) 620°K.

Fig. 4. Dependence of the capture coefficient, η, for D^+ ions on the target temperature for various energies of the bombarding ions (irradiation dose $2 \cdot 10^{17}$ deuterons /cm^2): ○, □) D_2^+ ions of energies 10 and 35 keV; △) D_1^+ ions at 30 keV; ×) data of [7].

to ion bombardment [8, 9]. It is known that deuterides decompose at a lower temperature [17]. Thus, the low temperature part of j(T) corresponds to a compound Nb_xD_y with a higher concentration of deuterium. Measurements of the amount of gas in the individual peaks showed that the β peak does not always appear except at some critical dose, when $y/x \approx 0.1$. According to [17], for these amounts of hydrogen the hydride phase begins to precipitate. Thus, we may assume that the γ and Ω peaks are due to separating out from a solution, and the β, α, and α_1 peaks, to separation from deuteride phases with increasingly larger values of y/x. The relations $T_{max\Omega} > T_{max\gamma}$ and $N_{0\gamma} > N_{0\Omega}$ indicate that the γ and Ω peaks are due to release of gas from interstitial and displacement solutions, respectively. This is so since deuterons are more effectively in radiative vacancies than at interstitial locations. This distribution of the γ, Ω, β, α and α_1 peaks is in accordance with the above noted dependence of T_{max} on E_d in the separate peaks and with the relation $N_{0\gamma} < N_{0\beta}$ which is realized for large N_0. The increase in $N_{0\gamma}(E_d)$ and the simultaneous decrease in $N_{0\beta}(E_d)$ is naturally explained by widening of the zone in which the ions are slowed down, thus making it possible to store more deuterium in the solid solution. In this case the deuteride β, α, and α_1 peaks occur for larger N_d. The increase in E_d, and, therefore, in the width of the gas accumulation layer also leads to a reduction in the concentration of deuterium and to an increase in the temperature stability of the Nb—H alloy. It is just this which should explain the difference among the j(T) curves of Fig. 1.

The experiments with heated targets are of practical interest. Figure 3 shows the outgassing spectra obtained upon heating a target which has been irradiated at various target temperatures with D_1^+ ions (energy 30 keV, dose $2 \cdot 10^{17}$ deuterons/cm^2). It is clear from the figure that as the target temperature (while the target is bombarded) is increased, the low temperature peaks in the outgassing spectrum whose T_{max} is near T_0 decrease and disappear. This means that even during irradiation part of the gas is released from the target due to thermally activated decomposition of hydrides and diffusion. Thus, for irradiation at 500°K the β peak disappears, and at about 600°K the γ peak disappears leaving only the Ω peak. The complete absence of imbedded particles, characterized by the disappearance of the last, Ω peak, begins at about 660°K.

To characterize the retention of ions in the target after bombardment we use the capture coefficient η which equals the fraction of bombarding particles captured in the sample during the entire irradiation time. Since the portion of atoms absorbed in niobium by various means depends on the ion energy, we might expect that at high irradiation temperatures the fraction of particles remaining in the target will also be a function of the energy E_d. Figure 4 shows the dependence of the deuteron capture coefficient on the target temperature during bombardment for various energies per deuteron. It is clear from the

figure that for $T_B > 320°K$, η increases with E_d, which is a consequence of an increased fraction of high temperature peaks and a reduced fraction of low temperature peaks at large energies (cf. Fig. 2). As the deuteron energy increases, there is an increase in both the temperature at which $\eta(T_B)$ begins to decrease and the temperature at which all the imbedded deuterium is not retained in the metal but completely released to the vacuum. In Fig. 4 we also show for comparison the measurements of $\eta(T_B)$ given in [7]. The qualitative variation in $\eta(T_B)$ given there agrees with our results. Some difference should be noted in the high temperature region: the tail of the curve of [7] extends to 1000°K, while we did not observe capture of deuterium ions at 700-1000°K. As shown in [18] this may be due to the presence of hydrogen active impurities in the sample used in [7]. It should be noted that at low temperatures $\eta(E_d)$ is independent of E_d, as also noted in [7].

Therefore, we may make the following conclusions:

1. The characteristics of deuterium ion capture are independent of the type of ion (molecular or atomic) provided the doses of bombarding deuterons and energies per deuterons are equal.

2. Capture and outgassing of deuterium are determined by several processes which are characterized by different activation energies. The principal states of deuterium imbedded in the metal are interstitial and displacement solutions as well as deuterides.

3. The ratio of the number of particles captured in the metal by one or another means depends strongly on the ion energy and comparatively weakly on the dose.

4. The capture coefficient for D^+ ions decreases with the temperature of the target during irradiation. The retention of deuterium begins to slacken at about 340-380°K. At temperatures of 550-680 to 1500°K the capture coefficient is close to zero and the imbedded deuterium is nearly completely released to the vacuum.

5. The increase in the capture coefficient for D^+ ions with the ion temperature at high target temperatures has been observed and explained. It has been shown that at room temperature it is independent of the energy, which is in agreement with previous data.

LITERATURE CITED

1. M. Kaminsky, IEEE Trans. Nucl. Sci., 18, 208 (1971).
2. I. Draley et al., Intersoc. Energy Convers. Eng. Conf., Boston, 1971, New York (1971), p. 1065.
3. G. McCracken and I. Maple, Brit. J. Appl. Phys., 18, 919 (1967).
4. E. S. Borovik, N. P. Katrich, and G. T. Nikolaev, At. Énerg., 18, 91 (1965).
5. E. S. Borovik, N. P. Katrich, and G. T. Nikolaev, At. Énerg., 21, 339 (1966).
6. N. P. Katrich, At. Énerg., 26, 307 (1969).
7. G. McCracken, D. Jefferies, and P. Goldsmith, in: Proc. 5th Intern. Vacuum Congress, Vol. 5/1 (1968), p. 149.
8. K. Erents and G. McCracken, Brit. J. Appl. Phys., 2, 1397 (1969).
9. K. Erents and G. McCracken, Radiation Effects, 3, 123 (1970).
10. V. A. Kurnaev, V. M. Sotnikov, and V. G. Tel'kovskii, in: Plasma Physics [in Russian], Moscow, Atomizdat (1967), p. 65.
11. A. A. Pisarev, V. M. Sotnikov, and V. G. Tel'kovskii, in: II-nd All-Union Symposium on the Interaction of Atomic Particles with Solids [in Russian], Moscow, Izd. ONTI IAE (1970), p. 240.
12. L. B. Bergambekov et al., At. Énerg., 31, 625 (1971).
13. G. Carter, Vacuum, 12, No. 5, 245 (1962).
14. P. Redhead, Vacuum, 12, No. 8, 203 (1962).
15. M. Mackay, Hydrogen Compounds of Metallic Elements, Halsted Press (1966).
16. N. A. Galaktionova, Hydrogen in Metals [in Russian], Moscow, Metallurgiya (1967).
17. A. J. Goldschmidt, Interstitial Alloys, Plenum Press (1968).
18. A. A. Pisarev and S. K. Fedyaev, in: The Interaction of Atomic Particles with Solids [in Russian], Pt. 2, Kiev, Naukova Dumka (1974), p. 101.

DEPENDENCE OF $\bar{\nu}$ ON NEUTRON ENERGIES UP TO 5 MeV FOR ^{233}U, ^{238}U, AND ^{239}Pu

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UDC 539.183.74

Of the nuclear physics constants, the average yield of prompt neutrons per fission event $\bar{\nu}$ makes the greatest demands on the precision of measurements in reactor technology. The development of fast reactors requires knowing the energy dependence $\bar{\nu}(E_n)$, while the low-energy neutron region $E_n \lesssim 5$ MeV, which makes the principal contribution to the number of fissions, is of greatest interest. Absolute measurements of $\bar{\nu}$ are extremely complex; hence, the energy dependence of this constant is investigated in a comparative experiment in which the ratio $\bar{\nu}(E_n)/\bar{\nu}_0$ is determined directly, where $\bar{\nu}(E_n)$ represents the isotope being investigated and $\bar{\nu}_0$ represents the standard. The value of $\bar{\nu}_0$ for the spontaneous fission of ^{252}Cf, the precision of which has reached ~0.3% at the present time, serves as the generally accepted standard in such measurements. But the relative measurements of $\bar{\nu}(E_n)$ for monenergetic neutrons also present specific difficulties, the chief of which is the necessity for cancelling the background associated with the primary neutron beam. In order to achieve this, the coincidences between the pulses from the fission fragments and the neutrons are recorded in the appropriate detectors. Random coincidences and the difficulties in using considerable quantities of fissionable materials in the detectors of the fission events severely limit the statistics of the useful radings. Detailed measurements of the energy dependence $\bar{\nu}(E_n)$ consists of a laborious multiday experiment; therefore, it is natural to seek those systematic solutions which would allow one to measure $\bar{\nu}$ for several isotopes simultaneously. Such a possibility was realized in [1].

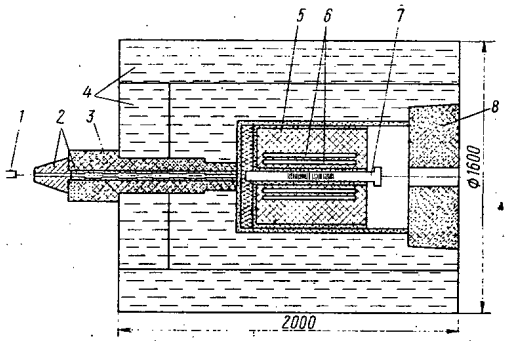

Fig. 1

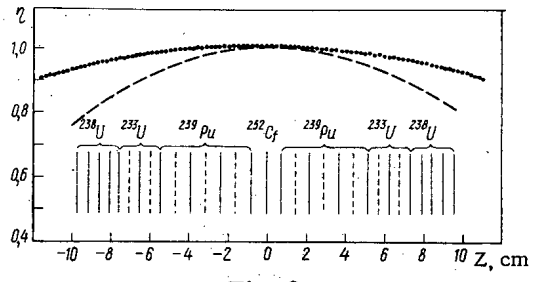

Fig. 2

Fig. 1. Diagram of experiment: 1) neutron target; 2) brass collimator; 3) borax–paraffin collimator; 4) water shield; 5) polyethylene block of the neutron detector; 6) ^3He neutron counter; 7) fission chamber; 8) rear shielding plug (borated polyethylene).

Fig. 2. Dependence of the sensitivity η on the displacement of the source along the axis of the neutron detector Z: •) the detector in this paper; -----) the detectors in [2-4]. The arrangement of the fissionable targets in the fission chamber is below.

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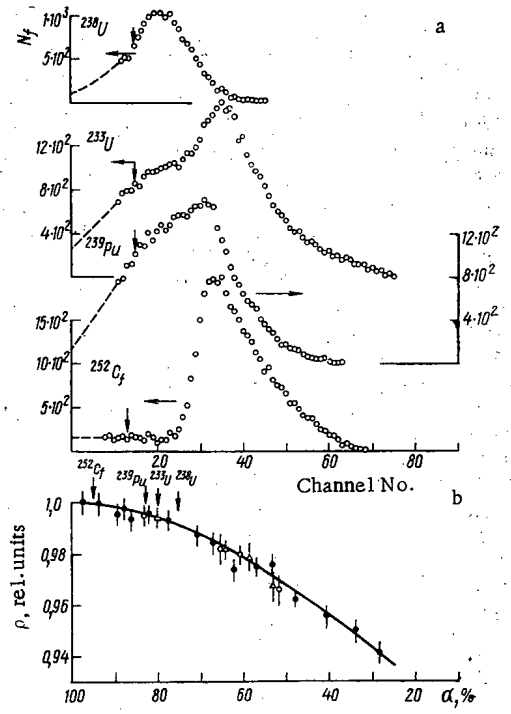

Fig. 3

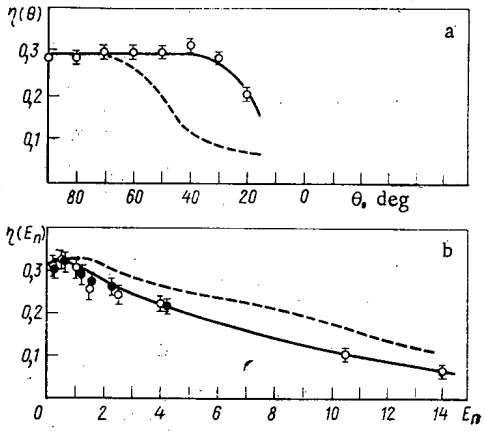

Fig. 4

Fig. 3. Distribution of the amplitudes of pulses from fragments in different sections of the fission chamber (a); ○) experiment; ---) extrapolated curves. Dependence of the experimental ratio ρ on the fraction of recorded fragments α (b): ●, ○, and Δ) experimental values of ρ for ^{252}Cf, ^{233}U, and ^{239}Pu, respectively. Discrimination levels (a) and fractions of fragments recorded in the operating of the fission detector (b) are indicated by vertical arrows.

Fig. 4. Dependence of the efficiency of the neutron detector η on the angle ϑ (a) and the neutron energy E_n (b): ○) calculated by the Monte Carlo method; ●) result of the experiment; and ---) response curve of the detector in [2-4].

In this work, relative measurements of $\bar{\nu}(E_n)$ were simultaneously conducted for ^{233}U, ^{238}U, and ^{239}Pu. The choice of nuclei to be investigated is determined by their practical value. Both ^{239}Pu and ^{238}U are basic fissionable materials in present-day breeder reactors, and the interest in ^{233}U is explained by its role in the thorium cycle. Measurements of $\bar{\nu}$ for ^{233}U and ^{239}Pu have been made previously for neutron energies up to 1.5 MeV [2, 3]. Therefore, in this experiment attention was mainly concentrated on the region E_n higher than the ^{238}U fission threshold.

Method of Measurement. In order to measure $\bar{\nu}$, we utilized the technique developed previously in [2-4] of recording the coincidences between the pulses of fissions in an ionization chamber and in a neutron detector surrounding it. The neutron detector represents a composition of 21 ^{3}He-counters in a polyethylene block, which are arranged coaxially in two rows around the channel for positioning the fission detector. The detectors are arranged inside a shield on the same axis as the collimator which forms the beam of bombarding neutrons (Fig. 1).

The recording of the neutrons, both the effective ones emitted during fission and the background ones (from the primary beam), scattered and passing through the shield, occurred during a period T equal to the gate width of the coincidence circuit, which is controlled by the pulses from the detector of the fission fragments. The choice of $T = 2\tau = 100$ μsec is determined by the lifetime of the neutrons ($\tau = 50$ μsec) in the detector. Calculation of the background neutrons (random coincidences) is conducted by simultaneous counting of the pulses in the same interval T, but delayed relative to the control pulse by a time $T_d = 250$ μsec.

The quantity $\rho = N_c/N_f = \bar{\nu}\eta$ for the isotopes under investigation and the standard $\rho_0 = \bar{\nu}_0\eta$ can be found from the number of recorded fission events N_f and actual coincidences N_c. The unknown quantity $\bar{\nu}/\bar{\nu}_0$

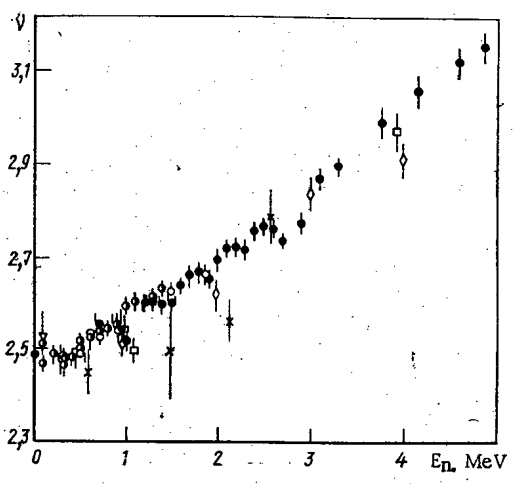

Fig. 5

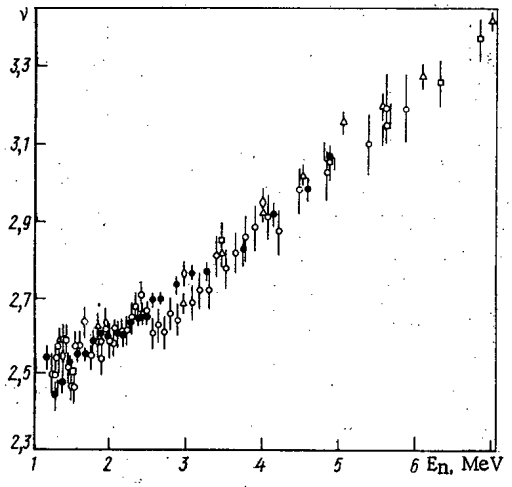

Fig. 6

Fig. 5. Dependence of $\bar{\nu}$ on the neutron energy E_n for ^{233}U: ▽) [9]; □) [10]; ◇) [11]; ○) [12]; ◑) [2]; ×) [13]; and ●) present experiment.

Fig. 6. Dependence of $\bar{\nu}$ on the neutron energy E_n for ^{238}U: ○) [14]; △) [1]; ◇) [11]; □) [15]; and ●) present experiment.

TABLE 1. Experimental Results

Cycle No.	E_n, MeV	^{233}U			^{238}U			^{239}Pu		
		$\rho(E_n)/\rho_0$ *	$\bar{\nu}(E_n)/\bar{\nu}_0$	$\bar{\nu}(E_n)$	$\rho(E_n)/\rho_0$ *	$\bar{\nu}(E_0)/\bar{\nu}_0$	$\bar{\nu}(E_n)$	$\rho(E_n)/\rho_0$ *	$\bar{\nu}(E_n)/\bar{\nu}_0$	$\bar{\nu}(E_n)$
118	0,00	0,6579±0,0021	0,6627±0,0034	2,489±0,013	—	—	—	0,7691±0,0019	0,7680±0,0036	2,884±0,014
11	0,700±0,055	0,6734±0,0036	0,6806±0,0091	2,556±0,034	—	—	—	0,7894±0,0090	0,7905±0,0096	2,969±0,036
35	0,900±0,059	0,6728±0,0044	0,6799±0,0053	2,553±0,020	—	—	—	0,7880±0,0050	0,7890±0,0059	2,963±0,022
17	1,000±0,064	0,6648±0,0053	0,6711±0,0060	2,520±0,023	—	—	—	0,7896±0,0073	0,7907±0,0080	2,970±0,030
20	1,200±0,060	0,6842±0,0049	0,6927±0,0057	2,602±0,021	0,6533±0,0084	0,6776±0,0096	2,545±0,036	0,7985±0,0062	0,8004±0,0071	3,006±0,024
41	1,300±0,056	0,6850±0,0045	0,0634±0,0053	2,604±0,020	0,6299±0,0086	0,6524±0,0096	2,450±0,036	0,8088±0,0039	0,8116±0,0051	3,048±0,019
49	1,400±0,061	0,6836±0,0044	0,6919±0,0052	2,599±0,020	0,6373±0,0071	0,6606±0,0083	2,481±0,031	0,8127±0,0036	0,8159±0,0048	3,065±0,018
30	1,500±0,059	0,6840±0,0038	0,6923±0,0048	2,600±0,018	0,6506±0,0031	0,6745±0,0043	2,533±0,016	0,8111±0,0046	0,8142±0,0056	3,058±0,021
52	1,600±0,060	0,6923±0,0035	0,7015±0,0046	2,635±0,017	0,6569±0,0026	0,6808±0,0048	2,557±0,018	0,8178±0,0041	0,8215±0,0053	3,085±0,020
38	1,700±0,057	0,6991±0,0048	0,7090±0,0056	2,663±0,021	0,6565±0,0023	0,6804±0,0043	2,555±0,016	0,8269±0,0040	0,8315±0,0052	3,123±0,020
20	1,800±0,060	0,7006±0,0059	0,7107±0,0066	2,669±0,025	0,6653±0,0043	0,6899±0,0056	2,591±0,021	0,8372±0,0064	0,8428±0,0073	0,165±0,028
52	1,900±0,054	0,6978±0,0035	0,7076±0,0045	2,658±0,017	0,6697±0,0036	0,6948±0,0051	2,610±0,019	0,8325±0,0042	0,8376±0,0053	3,146±0,020
20	2,000±0,053	0,7069±0,0057	0,7177±0,0064	2,696±0,024	0,6675±0,0051	0,6924±0,0065	2,601±0,024	0,8381±0,0066	0,8438±0,0075	3,169±0,028
30	2,100±0,053	0,7126±0,0039	0,7240±0,0049	2,719±0,018	0,6733±0,0039	0,6988±0,0053	2,625±0,020	0,8370±0,0045	0,8425±0,0057	3,165±0,021
20	2,200±0,055	0,7135±0,0051	0,7250±0,0059	2,723±0,022	0,6687±0,0042	0,6938±0,0056	2,606±0,021	0,8393±0,056	0,8451±0,0066	3,174±0,025
28	2,300±0,050	0,7120±0,0052	0,7233±0,0060	2,717±0,023	0,6768±0,0033	0,7025±0,0051	2,639±0,019	0,8428±0,0054	0,8489±0,0065	3,188±0,024
19	2,400±0,051	0,7218±0,0051	0,7342±0,0059	2,757±0,022	0,6799±0,0032	0,7059±0,0048	2,651±0,018	0,8383±0,0052	0,8439±0,0062	3,170±0,023
30	2,500±0,048	0,7237±0,0034	0,7363±0,0045	2,765±0,017	0,6801±0,0043	0,7060±0,0059	2,652±0,022	0,8539±0,0041	0,8610±0,0054	3,234±0,020
45	2,600±0,046	0,7223±0,0049	0,7348±0,0058	2,760±0,022	0,6909±0,0034	0,7178±0,0054	2,696±0,019	0,8546±0,0055	0,8618±0,0066	3,237±0,025
34	2,700±0,047	0,7166±0,0041	0,7284±0,0051	2,736±0,019	0,6916±0,0031	0,7187±0,0048	2,699±0,018	0,8703±0,0044	0,8790±0,0057	3,302±0,021
24	2,900±0,059	0,7249±0,0061	0,7383±0,0069	2,773±0,026	0,7004±0,0038	0,7289±0,0053	2,738±0,020	0,8711±0,0057	0,8808±0,0068	3,308±0,025
18	3,100±0,057	0,7476±0,0058	0,7637±0,0066	2,868±0,025	0,7074±0,0025	0,7365±0,0045	2,766±0,017	0,8793±0,0058	0,8899±0,0068	3,342±0,026
20	3,300±0,055	0,7540±0,0041	0,7708±0,0052	2,895±0,020	0,7091±0,0043	0,7387±0,0059	2,774±0,022	0,8762±0,0059	0,8865±0,0069	3,330±0,026
36	3,780±0,25	0,7726±0,0044	0,7958±0,0088	2,989±0,033	0,7209±0,0046	0,7530±0,0070	2,828±0,026	0,8954±0,0049	0,9137±0,0100	3,432±0,038
30	4,17±0,20	0,7875±0,0041	0,8131±0,0091	3,054±0,043	0,7430±0,0043	0,7776±0,0068	2,921±0,026	0,9110±0,0049	0,9314±0,0100	3,498±0,038
28	4,61±0,16	0,8020±0,0041	0,8290±0,0091	3,114±0,034	0,7572±0,0049	0,7944±0,0073	2,984±0,029	0,9416±0,0051	0,9643±0,0110	3,622±0,040
24	4,89±0,14	0,8097±0,0046	0,8376±0,0093	3,146±0,035	0,7745±0,0051	0,8154±0,0099	3,063±0,037	0,9462±0,0052	0,9694±0,0110	3,641±0,040

*Experimental values

TABLE 2. Structure of the Correction Factor k and the Experimental Error for $E_n = 3.0$ MeV

Example of data analysis	^{233}U	^{238}U	^{239}Pu
Experimental result			
Effects taking into consideration the correction	0.7069 ± 0.0057	0.6675 ± 0.0051	0.8381 ± 0.0066
Length of fission source along axis of detector	1.0259 ± 0.0025	1.0495 ± 0.0030	1.0074 ± 0.0020
Difference between diameters of layers of isotopes under investigation and standard	0.9971 ± 0.0010	0.9971 ± 0.0010	0.9971 ± 0.0010
Pile-up of neutron pulses from a single fission event	0.9935 ± 0.0010	0.9922 ± 0.0012	0.9963 ± 0.0007
Dependence of efficiency of detector on neutron energies	0.9911 ± 0.0025	0.9884 ± 0.0035	0.9970 ± 0.0022
Angular anisotropy of response of neutron detector	1.0009 ± 0.0002	1.0018 ± 0.0004	1.0007 ± 0.0002
Admixture of slow neutrons	1.0028 ± 0.0009	1.0000 ± 0.0003	1.0025 ± 0.0008
Discrimination of fraction of pulses from fission fragments	1.0042 ± 0.0014	1.0090 ± 0.0030	1.0058 ± 0.0020
Total correction	1.0153 ± 0.0042	1.0373 ± 0.0055	1.0068 ± 0.0062
Final result	0.7177 ± 0.0064	0.6924 ± 0.0065	0.8438 ± 0.0075

differs from the directly measurable quantity ρ/ρ_0 by the ratio of the respective sensitivities in recording the fission neutrons, which is close to unity.

Changes associated with the characteristics of the problem being resolved were introduced into the construction of the detectors and recording apparatus in contrast with those in [2-4]. The length of the assembly of layers of fissionable material in the ionization chamber was increased significantly (up to ~20 cm). Separated into a series of sections [six for ^{239}Pu, four for ^{233}U, and one each for ^{238}U and ^{252}Cf (Fig. 2)], it had a corresponding number of independent outlets for subsequent amplification, discrimination, and shaping of the pulses. The standard pulses produced by different sections of the chamber, composed of identical layers, entered a mixer and then into a coincidence circuit as control signals. The coincidence circuit had four channels (conforming to the number of isotopes), each of which contained circuits for counting the total number of coincidences and coincidences having a random character.

The overall amount of isotopes utilized in the experiment was ~80, ~100, and ~600 mg for ^{239}Pu, ^{233}U, and ^{238}U, respectively. Sections of fissionable layers were arranged in this order, left and right of a target of ^{252}Cf with an intensity of 10 fissions/sec (see Fig. 2). Compactness of the overall assembly of the chamber was achieved as a result of some worsening of the counting-rate curve compared with previous experiments [2-4]. Distribution of the amplitudes of the pulses from fission fragments in different sections of the chamber are shown in Fig. 3a; the fractions of fragments recorded for ^{252}Cf, ^{233}U, ^{239}Pu, and ^{238}U are 95, 83, 80, and 75%. The materials utilized had a satisfactory isotopic frequency in order not to introduce corrections due to the fission of contaminant nuclei. Discrimination of a part of the fission events resulted in a reduction in the measurable ratio, and this fact was taken into consideration during the processing of the experimental results. The pertinent characteristics are shown in Fig. 3b.

Increasing the length of the source of fission neutrons required an increase in the sensitive volume of the neutron detector. In order to do this, ^3He-counters longer than in [2-4] were utilized (length of the operating part ~45 cm as compared to ~25 cm). The dependences of the efficiency in the recording of the neutrons $\eta(Z)$ on the displacement in the location of the source along the Z axis in the case of the detectors utilized in this experiment and the ones in [2-4] are shown in the upper part of Fig. 2. Comparing them demonstrates the scale of the reduction in one of the principal corrections to the disparity in the neutron recording efficiencies with changes in ρ and ρ_0, which is achievable due to the increase in the length of the ^3He-counters. This fact also influenced favorably a decrease in the angular dependence of the efficiency of the neutron detector $\eta(\vartheta)$ (Fig. 4a), which has a fixed value in view of the small angular correlations between the fission neutrons and the incident neutrons [4].

An increase in the responsive region of the neutron detector and the amount of scattering material in the fission chamber could in addition involve an extremely undesirable deterioration in the relationship between the true and the random coincidences. Due to reinforcement of the shield, disposition of the counters in the region of maximum sensitivity of the detector to the fission neutrons, and the reduction in the number of counters in the less effective second row, one succeeded in retaining the relationship at the previous level of interest to us. However, the dependence of the recording efficiency on the neutron

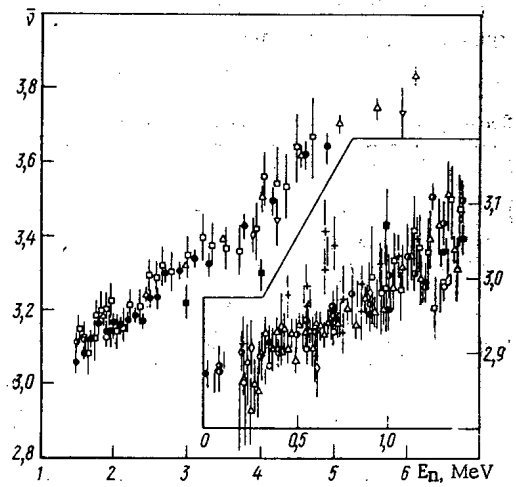

Fig. 7. Dependence of $\bar{\nu}$ on the neutron energy E_n for ^{239}Pu: △) [1, 16]; ◇) [10]; ○) [12]; □) [17]; ■) [11]; ▽) [18]; +) [19]; ◑) [3], and ●) present experiment. The E_n region up to 1.5 MeV is shown on a larger scale.

TABLE 3. Value of the Coefficients Which Specify the Equation of the Line Segments $\bar{\nu}(E_n) = (d\bar{\nu}/dE_n)E_n + \bar{\nu}(0)$ by Means of a Broken Line

Isotope	Energy range, MeV	$d\bar{\nu}/dE_n$	$\bar{\nu}(0)$
^{233}U	0—0,5	0,021±0,058	2,475±0,019
	0,5—2,5	0,120±0,005	2,450±0,007
	2,5—5,0	0,172±0,013	2,311±0,041
^{238}U	1,14—2,5	0,123±0,013	2,360±0,024
	2,5—6,0	0,178±0,006	2,203±0,021
^{239}Pu	0—1,0	0,117±0,008	2,869±0,005
	1,0—3,0	0,155±0,007	2,853±0,012
	3,0—5,0	0,200±0,012	2,689±0,050

energies $\eta(Z)$ became somewhat sharper (see Fig. 4b). The characteristics which were calculated by the Monte Carlo method are given in Fig. 4 [5]. The energy dependence of the efficiency $\eta(E)$ was also investigated experimentally using the radioactive neutron sources: $^{7}Li + ^{239}Pu$; $^{7}Li + ^{238}Pu$; $^{19}F + ^{239}Pu$; $^{19}F + ^{238}Pu$; $^{10}B + ^{239}Pu$, and $^{9}Be + ^{239}Pu$ (the average energies $E_n \approx 0.25$, 0.55. 1.20, 1.55, 2.3, and 4.2 MeV, respectively).

Since in this experiment one had to measure $\bar{\nu}$ over a broader range of energies E_n than in [2, 3], we obtained a higher value for the distortion in the spectrum of the incident neutrons because of an admixture of neutrons undergoing multiple scattering and moderation in the shield and the collimator. In order to reduce these effects, an elbow-shaped collimator [6] was utilized and the shield of the central channel was reinforced ($Cd + B_4C$).

The fraction of fissions by the moderated neutrons was estimated by two methods; by comparing the number of fissions of ^{233}U, ^{239}Pu, and ^{238}U in the beam and in the shadow of the collimator and by comparing the relative number of fissions of these isotopes inside and outside of the shield (in different ion ducts of the accelerator). Measurements indicated that this background is ~3-5% for nonthreshold isotopes and is taken into account by means of an appropriate correction. Measurements of the neutron spectra directly in the central channel with the aid of a scintillation counter also exhibited a small broadening in the monoenergetic neutron line that was taken into account in calculating E_n.

Let us mention yet another apparatus effect related to the shield correction. This is the pile-up of pulses in the neutron detector and the counters connected to it. In this experiment, ^{3}He-counters with an addition of ~2% CO_2 for reducing the collection time for electrons were utilized. The resolving time of the neutron detector was 2.5 μsec as compared to 3.7 μsec in [4], which resulted in a reduction in the correction to the pulse pile-up by a factor of 1.5.

The experiment was performed in the electrostatic and cascade generators of an FEI with the utilization of the T(p, n) and D(d, n) reactions and solid targets of TTi and DTi on a molybdenum backing. The conditions for measurements in the region $E_n > 3.5$ MeV occurring with the help of a D(d, n) reaction were worse: a higher correction to the contribution from slow neutrons, a breakdown in the monochromaticity of the neutrons due to the "packing" of deuterons from the primary beam in the target, contamination of neutrons from the C(d, n) reaction in the accumulating carbon, etc. In view of this, the error in the measurement over the range of high energies indicated was increased.

The results of the experiment are presented in Table 1, where the average energy of the primary neutrons E_n, the number of cycles of measurements N (roughly $1.5 \cdot 10^3$ true coincidences per cycle), the experimental and uncorrected ratios ρ/ρ_0 and $\bar{\nu}/\bar{\nu}_0 = k \cdot \rho/\rho_0$, and the absolute value of $\bar{\nu}(E_n)$ on the assumption $\bar{\nu}_0 = 3.756$ are given for each isotope [7]. The structure of the correction factor and the experimental

error are considered in Table 2 as illustrated by the measurements for $E_n = 2.0$ MeV. The ratio ρ/ρ_0 and its error, calculated from the variance of the measurements in separate cycles, are cited in the first row. In determining the error in the correction factor and the final result for $\bar{\nu}/\bar{\nu}_0$, the errors in the individual components were added quadratically. This assumption is to a certain extent justified by the fact that either the partial correction factors were measured experimentally or the calculations were based on experimental values. The values of the parameters Θ utilized which determine the average energy of the fission neutrons for ^{233}U, ^{239}Pu, ^{238}U, and ^{252}Cf are 1.33 ± 0.03; 1.37 ± 0.03 (thermal neutrons), 1.33 ± 0.04 (for $E_n = 2.0$ MeV) and 1.42 ± 0.02 MeV, respectively [8]. The results of this experiment are compared in Figs. 5-7 with the data in other experiments having a precision not worse than 2.5%.

Discussion. Of the isotopes investigated ^{233}U was studied least of all (see Fig. 5). Detailed direct measurements of $\bar{\nu}(E_n)$ were only completed in [2, 12] over a relatively narrow range of energies $E_n < 2.0$ MeV. Agreement of the data in Fig. 5 is satisfactory.

Experimental information concerning $\bar{\nu}$ for ^{238}U is richer than in the preceding case (see Fig. 6). Above the fission threshold ($E_n \approx 1.5$ MeV) the points of various authors lie extremely close together, with the exception of [14], the data of which fall outside of the 2.5-3 MeV range, forming a "resonance" structure. Measurements by us were sufficiently detailed to conclude that this effect was not verified in the present experiment. It is also supported by indirect data [20], obtained during a study of the distribution of the mass and kinetic energy of the fragments and an analysis of the energy balance.

The ^{239}Pu isotope has been most thoroughly investigated (see Fig. 7). An analysis of the errors was conducted in [3], which showed that systematic disparities between independent experiments are not significant in the data set for $\bar{\nu}$ in the case of ^{239}Pu up to 1.5 MeV. At the same time, the data in [19], which differ noticeably from the main bulk of points over the 0.5-0.7 MeV range, were excluded from consideration. The scatter of the points also increases over the 1.2-1.5 MeV range; however, in this case, in contrast to the previous one, it is difficult to give preference to any group of data. That this fact possesses a physical basis is not excluded; notably, that it is related to the structure considered in [3]. In the neutron energy range above 1.5 MeV, there are less data, but they are in good agreement. Nevertheless, it is worth paying attention to the dominant data in [11], whose results for ^{233}U also fall below the main bulk of points (see Fig. 5).

The results of the data which we studied as described by the broken line are given in Table 3. The data analysis was conducted using the method of least squares for the energy ranges E_n indicated. It was verified that if one excluded those points obviously falling outside of the aggregate of points the hypothesis concerning the linear dependence of $\bar{\nu}(E_n)$, common to the entire 0-5 MeV range, is rejected. A similar result can also be obtained for the data set for $\bar{\nu}(E_n)$ in the case of ^{235}U, although the extensive information in this case allows one also to construct a more detailed representation of the nature of the departure from a linear dependence [4]. Summing up, one can conclude that the breakdown in the linear dependence of $\bar{\nu}(E_n)$ is a systematically observable phenomenon, which one should take into account when solving applied problems, in particular when compiling multigroup constants in reactor design.

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LITERATURE CITED

1. M. Soleihac, J. Frehaut, and J. Gaurian J. Nucl. Energy, 23, 257 (1969).
2. B. Nurpeisov et al., At. Énerg., 34, No. 6, 491 (1973).
3. K. E. Volodin et al., At. Énerg., 33, No. 5, 901 (1972).
4. L. I. Prokhorova et al., At. Énerg., 30, No. 3, 250 (1971).
5. I. E. Bocharova, L. I. Prokhorova, and G. N. Smirenkin, in: Problems in Atomic Science and Technology [in Russian], Nuclear Constants Series, No. 17, Izd. TsNIIatominform (1974), p. 7.
6. A. Moat, D. Mather, and M. McTaggart, Reactor Sci. and Tech., 16, 102 (1961); D. Mather, P. Fieldhause, and A. Moat, Phys. Rev., 133, 1403 (1964).
7. G. Hanna et al., Atomic Energy Rev., 7, 3 (1969).
8. Proceeding of the IAEA Meeting on Prompt Fission Neutron Spectra, Vienna, 25-27 Aug., 1971.
9. B. Diven et al., Phys. Rev., 101, 1012 (1956).
10. J. Hopkins and B. Diven, Nucl. Phys., 48, 433 (1963).
11. D. Mather, P. Fieldhause and A. Moat, Nucl. Phys., 66, 149 (1965).

12. R. Walsh and J. Boldeman, J. Nucl. Energy, 25, 321 (1971).
13. D. Colvin and M. Sowerdy, Atomic Energy Rev., 10, 668 (1972).
14. M. V. Savin et al., At. Énerg., 32, No. 5, 408 (1972).
15. I. Asplund-Nilsson, H. Conde, and H. Starfelt, Nucl. Sci. and Eng., 20, 527 (1964).
16. M. Soleilhac et al., in: Proceedings of the IAEA Symposium on Nuclear Data for Reactors — 1970, Vol. II, Helsinki, June 15-19, 1970, p. 145.
17. M. V. Savin, Yu. A. Khokhlov, and Yu. S. Zamyatnin, ibid, p. 157.
18. H. Conde, J. Hansen, and M. Holmberg, J. Nucl. Energy, 22, 53 (1968).
19. D. Mather et al., Atomic Energy Rev., 10, 701 (1972).
20. V. G. Vorob'eva et al., in: Bulletin of the Nuclear Data Information Center [in Russian], Nuclear Constants Series, No. 15, Atomizdat, Moscow (1974), p. 3.

DETERMINATION OF CROSS SECTIONS FOR INELASTIC PROCESSES AND THE FORMATION OF 4.43-MeV γ QUANTA DURING THE INTERACTION OF CARBON WITH THE NEUTRONS FROM A $T(d,n)^4$ He REACTION

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UDC 539.171.4.017

A transmission method was utilized to measure the inelastic, interaction cross section σ_x taking into consideration multiple collisions [1]. The target of a low-voltage generator was placed at the center of spherical samples of graphite of various thicknesses t = 1-7 cm. The distance between the target and the detector (a stilbene spectrometer) was ~45 cm.

The spectrograms of the pulses $dN/dV = f(V)$ [V is the scale of the analyzer], obtained with samples of different thicknesses, were analyzed. The energy independence of the ratio $K = (dN/dV)_t/(dN/dV)_0$ in the 13.3-14.8 MeV region indicates that K = T (K is the transmission coefficient). A mean value of σ_x equal to (0.518 ± 0.014) b was obtained for four different samples 5.5 — 1.0 cm in thickness. The results of the present work and [2-6] indicate that the most reliable value is $\sigma_x = (0.531 \pm 0.010)$ b.

The curve $(dN/dV)_0 = f_0(V)$, obtained in the absence of a sample, is the result of the interaction of neutrons with the crystal of stilbene. Subtracting its contribution from similar curves obtained with the samples present, we find an integrated effect due to the interaction of the neutrons with the carbon. The middle of the linear region of the decay of the maximum energy Compton scattering peak in the difference curve corresponds to 4.43-MeV γ quanta to within 0.3%, while a second rise corresponds to an energy of 3.92 MeV = 4.43-0.51 MeV. Normalizing these curves beforehand to the shape of the $E_\gamma = 4.43$-MeV line for a stilbene spectrometer, taken from [7], we found the area S under them for each sample. After correcting for the presence of holes in the samples and the loss of 4.43-MeV γ quanta from the beam while passing through a graphite layer of thickness t, we found that the ratio $S/t = \alpha$ does not depend on t (0.2%) (here and below, the error is indicated in parentheses). Since $S = \Phi_{\gamma_0} \cdot \Omega \cdot \varepsilon_\gamma$ and $\Phi_{\gamma_0} = \Phi_0 (1 - e^{-\sigma_a \cdot N' t}) \approx \approx \Phi_0 \sigma_a N' t$, then $\sigma_a = \alpha/\Phi_0 \varepsilon_\gamma \cdot \Omega \cdot N'$ is the cross section for the production of quanta with $E_\gamma = 4.43$ MeV during the interaction of neutrons with carbon. Here, N' is the concentration of carbon nuclei, Ω is the solid angle (0.8%), Φ_0 is the neutron flux from the $T(d,n)^4$ He reaction (0.6%), and ε_γ is the efficiency of detecting the γ quanta using a crystal of stilbene. The latter quantity was found by normalizing the theoretical curve $\varepsilon_\gamma = f(E)$ to the experimental curve obtained for the 0.662-1.33 MeV energy range. For a detection threshold of 0.423 MeV, the sensitivity of the stilbene crystal to 4.43-MeV γ quanta proved to be 0.125 ± 0.004.

Thus, taking into consideration the constituent errors, $\sigma_a = (0.261 \pm 0.012)$ b. This value of the cross section agrees with the result in [8] (0.249 ± 0.028) b if one considers it to be equivalent to the excitation cross section of the 4.43-MeV level in the ^{12}C nucleus.

LITERATURE CITED

1. J. Marion and J. Fowler (editors), Fast Neutron Physics, Vols. 1 and 2, Interscience, New York (1960, 1963).
2. E. Graves and R. Davis, Phys. Rev., 97, 1205 (1955).
3. H. Taylor, O. Lonjo, and T. Bonner, Phys. Rev., 100, 174 (1955).
4. N. N. Flerov and V. M. Talyzin, At. Energ., 4, No. 4, 155 (1956).

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5. A. Bratenahl, J. Peterson, and J. Stocring, Phys. Rev., 110, 927 (1958).
6. M. MacGregor, W. Ball, and R. Booth, Phys. Rev., 108, 726 (1957).
7. A. I. Abramov, Yu. I. Kazanskii, E. S. Matusevich, Fundamentals of Experimental Methods in Nuclear Physics [in Russian], Atomizdat, Moscow (1970).
8. J. Benveniste, Phys. Rev., 119 448 (1960).

HEAT TRANSFER SURGE IN A TUBE WITH A HYPERBOLIC DISTRIBUTION OF HEAT FLUX ALONG ITS LENGTH

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UDC 536.248.2

In this paper we present the results of an experimental study of heat-transfer surge in tubes with a hyperbolic distribution of heat flux along the tube length (outer diameter d = 10 mm; heated tube length equal to 900 mm and 1500 mm; heat flux nonuniformity $q_{max}/q_{min} = 3.64$; $q_{max}/\bar{q} = 1.42$) with pressures of 140, 180, and 200 atm and mass velocities from 500 to 4000 kg/m^2 · sec.

The critical heat fluxes (at the heat-surge location) were found to be practically identical on tube sections having uniform and hyperbolic lengthwise heat flux distributions in the range of variation of the parameters and geometric dimensions studied.

The location of maximum heat generation has an influence on the magnitude of the critical power. If the maximum heat generation occurs at the exit end of a tube section, then, under identical conditions, the critical power may be 40 to 50% less than the N_{cr}^{un} for a uniformly heated tube of the same length. When the heat generation maximum takes place at the entrance to a tube section the critical power can be 5 to 10% more than N_{cr}^{un}.

We present graphs showing the variation with time of the temperature of a steam-generating surface when a heat-transfer surge occurs on tube sections with a nonuniform heat flux distribution.

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THE PROBLEM OF THE BUILDUP OF FISSION PRODUCTS IN FUEL ELEMENTS OF THE WATER-COOLED — WATER-MODERATED POWER REACTOR TYPE

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UDC 621.039.577:621.039.566

The paper is devoted to the determination of the burnup of fuel elements from power reactors of the VVER (water-cooled — water-moderated power reactors) type with low enrichment fuel by measuring the intensity of the γ lines of fission products. Experiments have been conducted on the buildup of certain fission products as a function of the fuel burnup for casettes with 3% initial uranium enrichment in the second unit of the Novovoronezh nuclear power station, with an average computed burnup of 7.96 kg/ton, by means of a gamma spectrometer with a germanium detector. Twenty-five fuel elements were investigated in the casette and each fuel element was measured at 10 points uniformly distributed over the height. In the γ spectra of the fuel, the intensities of the ^{106}Ru, ^{134}Cs, and ^{137}Cs lines were measured. The overall fuel burnup was determined experimentally by the buildup of ^{137}Cs.

It was observed that the spread of the individual experimental points of the buildup of these fission products corresponding to an identical burnup, but taken from different points of the casette, significantly exceeds the possible measurement errors. This can be explained by the fact that the neutron spectrum has a significant effect on the fission product buildup.

The dependence of the fission product buildup on the spectral parameters of the neutron field is found by a numerical method: by the effective neutron temperature, the ratio of the epithermal and thermal neutron fluxes and the coefficient of resonance absorption of ^{238}U. On the other hand, it is shown theoretically that the ratio of the epithermal and thermal neutron fluxes in the casette being studied can vary significantly within the bounds of the casette. Comparison of the calculated and experimental data leads to the conclusion that the change of spectrum of the neutron field even within the limits of a single casette over its height, and from fuel element to fuel element, considerably affects the magnitude of the fission product buildup, in the case of identical fuel burnup.

Obviously, the concentration of fission products depends on many conditions, the spectra formed at a given point of the active zone, for example on the ratio of the fuel burnup in the space being considered and in the surrounding fuel elements, the presence in the vicinity of the point being investigated of an absorbing element or a water cavity, the effect of adjacent fuel assemblies and the boundaries of the active zone, etc. Thus, when determining the characteristics of spent fuel according to the buildup of fission products, it is necessary to take into account the effect of the neutron spectrum in every specific case. The dependence found for the concentrations of isotopes on the spectral parameters reveals the feasibility in principle of determining the latter by measuring the quantity or the ratio of the quantities of different fission products in irradiated fuel.

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HYDRODYNAMICS OF FISSIONABLE MATERIALS.

III. STRUCTURE OF SHOCK WAVES AND DISCONTINUITIES

V. M. Novikov

UDC 621.039.542.51

An investigation of the dispersion of acoustic oscillations and the nature of the nonlinear solutions of simple-wave type, carried out in [1, 2], showed that for a fissionable liquid (gas) situated in neutron fluxes $\geqslant 10^{15}$–10^{16} cm^{-2} · sec^{-1}, there is a reduction in the time of growth of nonlinear processes and the formation of discontinuities in the wave. The study of the subsequent evolution of the wave leads to problems dealing with the features of the propagation of discontinuities and shock waves in fissionable materials. The latter problem arises in the investigation of pulsed nuclear power devices, in which a critical state is achieved by introducing into the core of the reactor a shock wave of the compression of the gas-phase fuel [3].

In the present paper a study is made of the spatially one-dimensional problem dealing with shock waves and discontinuities in fissionable materials situated in constant homogeneous neutron fluxes. An analysis of the solutions obtained allows us to draw the following conclusions. The energy release connected with the fission process leads to the formation in the shock wave of a relaxation zone — the shock-wave front. In this zone the gas parameters vary from values corresponding to the Hugoniot shock adiabat to values that lie on the curve of the equilibrium states. For neutron fluxes $\sim 10^{16}$ cm^2 · sec, the size of the front is ~10 cm. The shock waves produced by a moving piston are stable if the velocity of the piston v_p is greater than some critical value (Fig. 1). Far from the stability boundary the density near the front varies according to an exponential law. Near the stability boundary the front consists of two characteristic regions: in one of them the density varies almost linearly, in the second it varies considerably less than linearly over the dimensions: The variation in density has a sharply expressed exponential character (Fig. 2). The discontinuities that appear in a periodically simple wave owing to nonlinear effects for subsequent evolution can be converted into a wave with discontinuities of a steady finite amplitude, whereas in the hydrodynamics of an ideal liquid, the amplitudes of these discontinuities are damped down to zero.

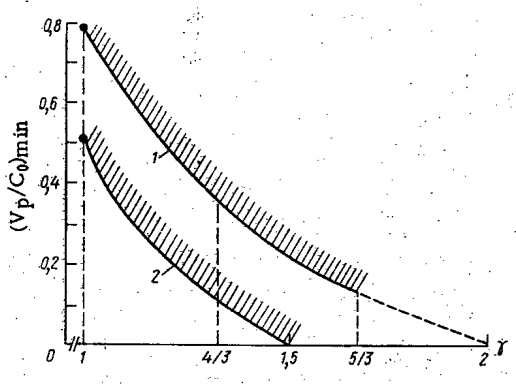

Fig. 1

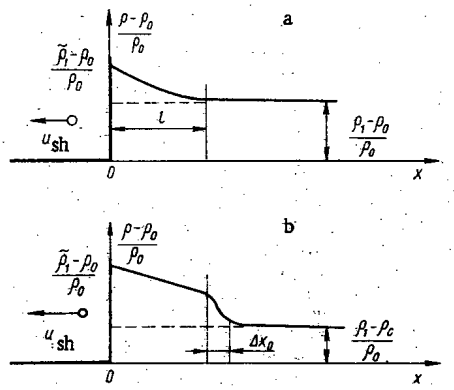

Fig. 3

Fig. 1. Boundary of stability region of shock wave: 1, 2) for a sink of heat of fission according to Newton's law and by means of the radiant heat conduction from an optically thick layer, respectively. The stability region is denoted by the hatching.

Fig. 2. Structure of shock-wave front: a) for values $M - M^{(m)} \geq M^{(m)}$; b) for values $0 < M - M^{(m)} \ll M^{(m)}$, where M is the Mach number; ρ is the density of the material.

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LITERATURE CITED

1. V. M. Novikov, At. Énerg., 30, No. 5, 446 (1971).
2. V. M. Novikov, At. Énerg., 38, No. 4, 246 (1975).
3. W. Podney et al., Physics. of Fluids, Suppl., 1, 168 (1969).

ADAPTIVE ALGORITHM FOR STABILIZATION OF THE POWER DISTRIBUTION OF A NUCLEAR REACTOR

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UDC 621.039.562:621.039.514

In [1], a system of coupled regulators was synthesized for controlling the power distribution in a nuclear reactor when all the coefficients in the equations of dynamics are known and are time-independent.

In the present study we consider questions of the stabilization of a three-dimensional power distribution of a nuclear reactor for incomplete information on the properties of the object. To solve such a problem we assume that the reactor parameters are coupled by differential equations of spatial kinetics within the framework of the one-group diffusion theory with account of delayed neutrons and power feedback. The given radial distribution of power over the reactor channels, which is called the control region, is divided into a finite number of controlled zones. In this case control and regulation of the power in each zone is ensured, and the controlling action ΔK_s varies discretely in time, which is connected with the use of a computer. It is assumed that the coefficients in the equations describing the reactor are partly unknown.

For the synthesis of the adaptive algorithm of stabilization we quantify the data of the equations according to the spatial coordinates and the time, and then, on the basis of the method of [2, 3], we give the target conditions of the stabilization in the form of the inequalities

$$\| Z_{s+1} \| < \varepsilon, \ \varepsilon > 0,$$

where $Z_{S+1} = \delta\Phi_{S+1} + h_{S+1}$; $\delta\Phi_{S+1}$ is the vector of the deviation of the power; h_{S+1} is the interference vector. On the controlling action we superimpose a constraint of the form $\| \Delta K_S \| \leq \delta$, $\delta > 0$.

The efficiency of the synthesized adaptation algorithm was examined by modeling on a computer. As a result we obtained a finite convergence for the algorithm, and hence, stabilization of the power distribution in the nuclear reactor under conditions of a priori uncertainty.

LITERATURE CITED

1. P. T. Potapenko, At. Énerg., 27, No. 3, 189 (1969).
2. V. A. Yakubovich, Dokl. Akad. Nauk SSSR, 189, No. 3, 495 (1969).
3. G. D. Penev and V. A. Yakubovich, Dokl. Akad. Nauk SSSR, 198, No. 4, 787 (1971).

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CALCULATION OF THE YIELD OF PHOTONUCLEAR REACTIONS

M. G. Davydov and V. A. Shcherbachenko

UDC 543.53

The increased utilization of activation techniques in analytical work requires the optimization of the conditions in analyses. Optimization techniques can be employed only to a limited extent when methods of gamma activation are developed, because it is difficult to calculate the yield of photonuclear reactions. Calculations of the yield of photonuclear reactions were treated in [1-3], but coarse approximations for the bremsstrahlung spectrum were used in [1, 2] and the calculations were made with a complicated, cumbersome method in [3]. The present paper derives a relatively simple formula for the calculation of the yields of photonuclear reactions; the thickness of the emitter of the accelerator and the geometry of irradiation are taken into account in the formula.

The integral cross section of the bremsstrahlung [4] was introduced to describe the energy dependence of the bremsstrahlung. Both the thickness of the emitter and the geometry of irradiation were brought into account by using a formula which described the angular dependence of the bremsstrahlung intensity (formula obtained in [5]). For the purpose of checking the applicability of the proposed calculation method, the yield curves of $^{12}C(\gamma, n)^{11}C$, $^{16}O(\gamma, n)^{15}O$, and $^{63}Cu(\gamma,n)^{62}Cu$ were measured. The curves were recorded in measurements made of the induced activity with a scintillation gamma spectrometer. The agreement between calculations and experiments was good.

LITERATURE CITED

1. A. M. Yakobson, in: Trudy Instituta Metallurgii im. A. A. Baikova, No. 8, Izd-vo Akad. Nauk SSSR, Moscow (1961), p. 178.
2. M. G. Davydov and V. A. Shcherbachenko, At. Énerg., 27, No. 3, 205 (1969).
3. G. Lutz, Anal. Chem., 41, 424 (1969).
4. L. Schiff, Phys. Rev., 83 252 (1951).
5. L. Lanzl and A. Hanson, Phys. Rev., 83, 959 (1951).

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INCREASE OF THE RADIATION STABILITY OF AN ORGANIC COOLANT BY MEANS OF ANTIRADS

M. F. Romantsev, D. Ya. Muchinskii, T. Ya. Livertovskaya, N. M. Astratova, V. A. Gorbach, and G. G. Kakabekov

UDC 541.14/15

A diphenyl mixture 23.5 wt.% of diphenyl and 76.5 wt.% of diphenyl oxide is the most widely used coolant [1]. Despite the relatively high radiation stability of the components of the mixture, radiation — chemical processes take place by the action of ionizing radiations which lead to the formation of various products, among which the greatest part consists of gaseous and polymer products.

It is well known that the radiation stability of certain organic coolants can be increased by means of antiradiation additives. In particular, the introduction into the mixture of terphenyls, benzopyrene, and certain other compounds retard the formation of gaseous products and polymers [2].

In this present paper the effect is investigated of certain polyphenyls on the radiation stability of the diphenyl mixture. The diphenyl mixture is prepared from diphenyl and diphenyl oxide subjected to prior distillation. Polyphenyls — pyrene and phenanthrene of "chemically pure" grade — were not additionally purified. Samples were irradiated in a ^{60}Co gamma source at a dose intensity of 42 rad/sec in temperature-controlled ampoules of Kh18N19T stainless steel. Before irradiation the ampoules were evacuated, filled with argon (0.5 atm), and sealed.

Analysis of the gaseous products was carried out chromatographically on LKhM-8M and KhL-69 instruments. High-boiling polymer products were determined by the molecular distillation method. The products of radiolysis, with a boiling point below that of the components of the diphenyl mixture, were determined by the method of chromatographic distillation.

TABLE 1. Release of Gaseous Products during Radiolysis of Diphenyl Mixture,* cm^3/ml

System	Dose, Mrad	Temperature, °C				
		150	260	330	370	385
Diphenyl mixture	50	0,024	0,048	0,074	0,103	0,117
	100	0,050	0,081	0,124	0,187	0,250
Diphenyl mixture + 1.5 wt. % phenanthrene	50	0,020	0,046	0,072	0,113	0,126
	100	0,041	0,081	0,100	0,159	0,229
Diphenyl mixture + 2 wt.% pyrene	50	0,017	0,038	0,078	0,101	0,109
	100	0,036	0,068	0,117	0,165	0,223

*Relative error of determination 5%.

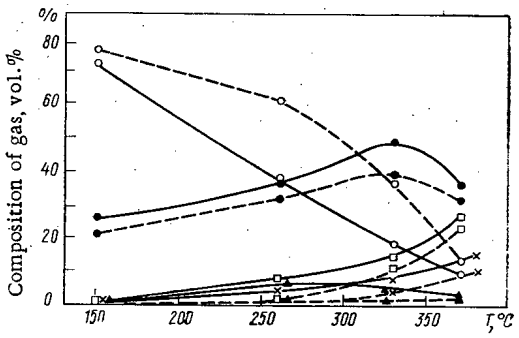

Fig. 1. Dependence of the composition of the gaseous radiolysis products of diphenyl mixture on the temperature (dose, 100 Mrad): ——) original diphenyl mixture; ---) in the presence of pyrene; ○) H_2; ●) CO; □) CH_4; ×) C_2H_4; ▲) $C_2H_2 + CO_2$.

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TABLE 2. Composition of Radiolysis Products Released from the Liquid Phase of Diphenyl Mixture, wt.%

Fractions	Temperature, °C				
	150	260	330	370	385
	Diphenyl mixture				
Low-boiling*	0,243	0,332	0,593	0,962	—
Intermediate†	0,073	0,113	0,180	0,256	—
High-boiling‡	0,90	1,25	1,48	2,86	—
	Diphenyl mixture + 2% pyrene				
Low-boiling*	0,160	0,273	0,491	0,783	1,443
Intermediate†	0,077	0,109	0,161	0,211	0,326
High-boiling‡	0,76	1,12	1,45	1,90	4,10
	Diphenyl mixture + 1.5% phenanthrene				
Low-boiling*	0,186	0,306	0,525	0,759	1,425
Intermediate†	0,127	0,120	0,169	0,218	0,387

*Up to 240°C; relative error of determination 5-7%.
†280-320°C.
‡Above 320°C.

Table 1 shows data on the buildup of gaseous products during radiolysis of diphenyl mixture. It can be seen that the introduction of pyrene and phenanthrene into the diphenyl mixture retards the formation of gas. However, if at 150° the amount of gas is reduced by 20-30%, then at high temperatures a reduction of only 10% occurs. It should be mentioned that the release of gas during irradiation of diphenyl mixture increases sharply over the temperature range 330-370°C, which is due to the development of thermal radiation cracking reactions. Similar rates of gas release are achieved in the presence of additives at temperatures of 10-20°C higher than for the original diphenyl mixture.

Figure 1 shows the composition of the gaseous radiolysis products from diphenyl mixture. It can be seen that in the region of low temperatures hydrogen is formed with the greatest yield. The buildup of CO — a product of the breakdown of diphenyl oxide — is accelerated with increase of temperature, and simultaneously the yield of hydrogen is reduced.

Hydrocarbon gases are formed with a low yield; however, with increase of temperature their content in the total mixture is increased, which indicates an acceleration of processes associated with the rupture of aromatic rings. In the presence of pyrene and phenanthrene, the relative content of hydrogen in the gas increases somewhat, but the content of carbon monoxide is reduced and therefore it may be assumed that the shielding action of the additives is directed to a considerable degree at the diphenyl oxide.

Table 2 shows data on the content of radiolysis products released from the liquid phase of diphenyl mixture. In the presence of pyrene and phenanthrene the yields of the products are found to be lower than for the original mixture. The content of low-boiling products is reduced by 25-35% in the region of low temperatures; at high temperatures it is reduced by approximately 20%.

In the presence of pyrene the formation of the products of the intermediate fraction and the high-boiling products is slowed down also. The results shown in Table 2 were obtained at a dose of 100 Mrad. Similar relationships were observed also at higher dosages. The breakdown of the diphenyl mixture in the case of a dose of 100 Mrad did not exceed 1.5 and 4% at 150° and 370°C, respectively.

Phenanthrene obviously possesses a somewhat low shielding action. It should be noted also that phenanthrene is easily sublimed and therefore, frequently it may get into the intermediate fraction and remain in the high-boiling fraction. As a consequence of this, the determination of the high-boiling products and of the intermediate fraction is associated with considerable errors.

Thus, the results have shown that the radiation stability of diphenyl mixture can be increased by means of antiradiation additives which, in particular are polyphenyl hydrocarbons.

LITERATURE CITED

1. A. V. Chechetkin, High-Temperature Coolants [in Russian], Énergiya, Moscow (1971), p. 94.
2. The Action of Radiation on Organic Materials [in Russian], Atomizdat, Moscow (1965), p. 277.

SPECTRA OF RESONANCE NEUTRONS IN HOMOGENEOUS MEDIA

A. A. Luk'yanov and A. P. Platonov

UDC 539.125.5.162.3

In investigating the slowing down, absorption, and diffusion of neutrons in media whose cross sections show a resonance structure it is necessary to know the detailed energy dependence of the neutron flux $\Phi(E)$ in the resonance region [1] since the collision density $\Psi(E) = \Phi(E)\sigma(E)$. At the present time calculations of macroscopic characteristics for extended uniform media generally use the very simple forms

$$\Psi(E) \approx \frac{\text{const}}{\bar{\xi}E}; \quad \Phi(E) \sim \frac{1}{\bar{\xi}E\sigma(E)}, \tag{1}$$

assuming that absorption in individual energy groups is small and that resonance widths are small in comparison with the average energy loss in elastic scattering on nuclei of the medium. These forms do not take account of so-called asymptotic effects, arising from the fact that the resonances are intermediate, and the total decrease of flux within a group as a result of absorption [1, 2].

The influence of these effects on the group characteristics in the resonance regions was investigated by calculating numerically the spectra of resonance neutrons in homogeneous media containing ^{238}U mixed with moderators of various atomic weights. The algorithm for the numerical solution of the corresponding slowing down equation and the calculational program are given in [3, 4]. The resonance cross sections were reproduced from the resonance parameters using the URAN program [5]. Figure 1 shows the function

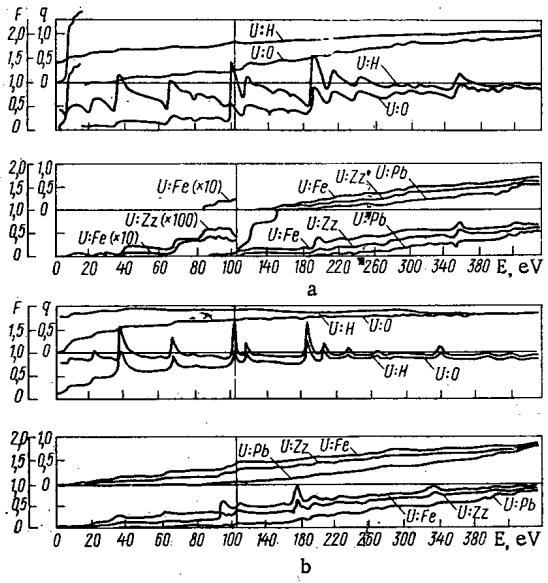

Fig. 1. Spectra of collision density and resonance escape probability in homogeneous mixtures containing ^{238}U: U and H, U and O, U and Fe, U and Zr, and U and Pb at T = 300°K for σ_m = a) 10; b) 100 b.

$$F(E) = \bar{\xi}\sigma(E)\Phi(E)E, \tag{2}$$

representing the spectrum of the neutron slowing down density attributed to the Fermi spectrum ($\sim 1/\xi E$). The function F(E) is normalized to unity at E = 500 eV. The considerable oscillations of the F(E) spectrum are correlated with the positions of the intermediate resonances in ^{238}U and are due to nonasymptotic effects in the slowing down of neutrons in multicomponent media [1]. Figure 1 also shows the energy spectrum of the slowing down density

$$q(E) = 1 - \int_E^{500} \frac{\sigma_a(E')}{\sigma(E')} \Psi(E') \frac{dE'}{E'}, \tag{3}$$

characterizing the total decrease in the number of neutrons per unit volume of the medium during slowing down from 500 eV to E (resonance escape probability). The calculations show a difference between the Fermi spectrum (1) and the actual spectrum in two-component media containing ^{238}U nuclei mixed with various moderators for two values of σ_m, the slowing down cross section per ^{238}U nucleus in the medium. For light moderators (H, O) this difference is due mainly to nonasymptotic oscillations close to

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the intermediate resonances, but for heavy moderators the effect of a sharp decrease of the slowing down density for energies below ~150 eV is more important.

The "exact" spectra obtained for resonance neutrons in homogeneous two-component media were used to calculate the ^{238}U group cross sections σ_m (group constants) as functions of the concentration, the atomic number, and the temperature of the moderator [4]. Comparison with calculations which assume a constant collision density within a group (1) shows the considerable influence of the effects appearing in the exact calculations on the values of the group cross sections in groups containing intermediate resonance in relatively concentrated media ($1\,b \lesssim \sigma_m \lesssim 100\,b$). The differences in the results are as much as 20-25% in some cases [4].

LITERATURE CITED

1. A. A. Luk'yanov, The Slowing Down and Absorption of Resonance Neutrons [in Russian], Atomizdat, Moscow (1974).
2. A. P. Platonov and A. A. Luk'yanov, At. Énerg., 35, 264 (1973).
3. A. P. Platonov, Zh. Vychisl. Matem. i Matem. Fiz., 12, 1325 (1972).
4. A. P. Platonov and A. A. Luk'yanov, in: Nuclear Constants [in Russian], No. 12, Atomizdat, Moscow (1973), p. 98.
5. L. P. Abagyan, M. N. Nikolaev, and L. B. Petrova, in: Bulletin of the Nuclear Data Information Center [in Russian], No. 3, Atomizdat, Moscow (1966), p. 418.

CRITERION FOR ESTIMATING THE POSSIBILITY OF USING RADIOACTIVE PRODUCTS TO MONITOR FUEL BURNUP

V. K. Shashurin

UDC 621.039.516

The method of gamma scanning is widely used to determine burnup in fuel elements. One of the requirements restricting the possibilities of the method, arising from the nonstationary character of the actual irradiation conditions and the dynamics of fission product accumulation, is the necessity of choosing as a monitor a fission product with a sufficiently long half-life [1].

On the other hand the shorter the half-life of the fission product, the greater its specific activity, and this simplifies the quantitative determination of its content by gamma spectrometry. The use of short-lived fission products as monitors permits the determination of burnup in less exposed fuel elements since the gamma radiation from longer-lived fission products can be identified only after the radioactive decay of an appreciable fraction of the nuclei of the short-lived isotopes.

A further advantage of the use of shorter-lived fission products is the rapid rate of change of their content during fuel irradiation, except in the region of maximum content, in comparison with the rate of accumulation of long-lived fission products.

Thus, in each specific case it is important to predict which identifiable fission products of different half-lives give equally reliable information for given irradiation conditions and can be used to determine fuel burnup with the same reliability.

We characterize the dependence of the number of fission product nuclei on the irradiation time by the so-called center of gravity of the distribution [2] defined as

$$S=\frac{\sum_i t_i}{N(T)}, \tag{1}$$

where t_i are the lifetimes of the nuclei existing at the instant irradiation ceases, and T is the fuel irradiation time.

The average lifetime of the nuclei existing at the instant irradiation ceases can be calculated from the standard definition

$$\tau=\frac{\int_0^T t\,dN(t)}{\int_0^T dN(t)}, \tag{2}$$

where N(t) is the number of fission product nuclei at time t.

Since the probabilities of the formation of individual nuclei in the irradiation process are statistically independent, we have for the relation under consideration

$$S=\tau. \tag{3}$$

It is obvious that the closer the center of gravity of the relation for the number of monitor nuclei as a function of irradiation time is to the average irradiation period, the better it reflects the irradiation history of the fuel. Thus, for a stable unburned fission product the condition

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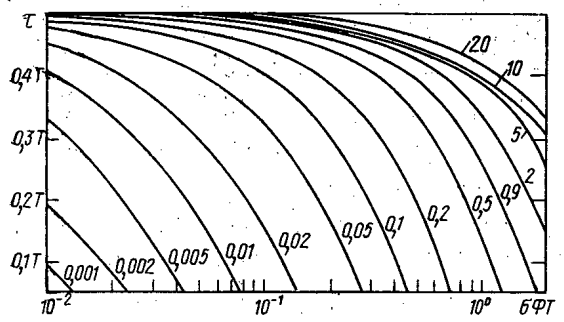

Fig. 1

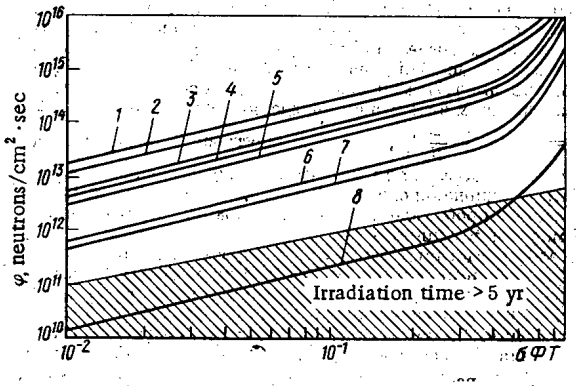

Fig. 2

Fig. 1. Dependence of τ on the integrated neutron flux in units of $\sigma\Phi T$ for various ratios of the rates of formation and radioactive decay of fission product nuclei (the parameter $\sigma\Phi/\lambda$).

Fig. 2. Lines of equal τ corresponding to 0.9 of the maximum τ for possible fuel burnup monitors; 1) ^{131}I; 2) ^{140}Ba; 3) ^{141}Ce; 4) ^{103}Ru; 5) ^{95}Zr; 6) ^{144}Ce; 7) ^{106}Ru; 8) ^{137}Cs.

$$\tau = 0.5T$$

must be satisfied if we neglect fuel burnup during its irradiation. The burnup of fuel, as the burnup and radioactive decay of nuclei generating a fission product, decreases the rate of accumulation of its nuclei, i.e., decreases τ. Thus, the value of τ can serve as a criterion for estimating the possibility of using a particular fission product with a finite half-life to monitor fuel burnup for given irradiation conditions.

The accumulation of a fission product in the irradiation process can be described by the expression

$$N(t) = q\sigma^f \Phi n_0 \frac{e^{-\sigma\Phi t} - e^{-(\lambda+\sigma'\Phi)t}}{(\lambda+\sigma'\Phi)-\sigma\Phi}, \quad (4)$$

where n_0 is the initial number of nuclei of the fissionable isotope, σ and σ^f are the absorption and fission cross sections of the fissionable isotope, Φ is the neutron flux density, q is the fission product yield, λ is the radioactive decay constant of the fission product, and σ' is the cross section for the burnup of the fission product nuclei.

Then using (2) and (3) the center of gravity of the relation described by Eq. (4) for a radioactive fission product with a negligibly small burnup cross section, which corresponds to the actual situation, can be obtained in the form

$$S = \tau = T\left[1 - \frac{\frac{1}{\sigma\Phi T}(1-e^{-\sigma\Phi T}) - \frac{1}{\lambda T}(1-e^{-\lambda T})}{e^{-\sigma\Phi T} - e^{-\lambda T}}\right].$$

According to this expression, the value of τ for a given fuel burnup depends only on the ratio of the rates of formation and decay of the fission product nuclei, i.e., on the ratio $\sigma\Phi/\lambda$.

Figure 1 shows the calculated dependence of τ on the integrated flux of thermal neutrons in a range corresponding to 1 to more than 90% ^{235}U burnup for various values of $\sigma\Phi/\lambda$. The quantity $\sigma\Phi T$, which is proportional to the integrated neutron flux but more convenient for burnup calculations, is plotted along the axis of abscissas.

The larger the value of $\sigma\Phi/\lambda$ the smaller the change in τ for an increase in fuel burnup. The fact that τ remains constant with increasing fuel burnup in a certain range of integrated fluxes indicates that a monitor whose half-life corresponds to a specified value of $\sigma\Phi/\lambda$ can be used in that range of fuel burnup with practically the same reliability. The less the change in τ corresponds to a given range of change of burnup (integrated neutron flux) the better the given fission product serves as a burnup monitor. Then specifying the maximum admissible change of τ it is possible to estimate the possibility of using one or another fission product as a fuel burnup monitor for given irradiation conditions, i.e., flux density and integrated neutron flux.

Figure 2 shows lines of equal τ differing from the maximum by 10% for the longest-lived gamma emitters among the fission products: ^{137}Cs (30 yr*), ^{106}Ru (368 days), ^{144}Ce (284 days), ^{95}Zr (65.5 days),

* The half-lives are given in parentheses.

^{103}Ru (39.5 days), ^{141}Ce (32.5 days), ^{140}Ba (12.8 days), and ^{131}I (8.05 days) as a function of the flux density and the integrated flux of thermal neutrons in units of $\sigma\Phi T$. The data on the half-lives of the isotopes are taken from [3]. The shaded area corresponds to an irradiation time of more than 5 yr, which is rarely encountered in practice.

From the character of the change in τ in the coordinate system chosen one can draw the following general conclusions:

1) The higher the rate of accumulation of fission product nuclei, determined by the neutron flux density, the larger the integrated fluxes for which it can be used as a fuel burnup monitor.

2) Only ^{137}Cs can be used as a monitor of burnup for practically all conditions of the irradiation of fuel which can be realized at the present time.

3) For $\sigma\Phi T$ larger than 0.4-0.5 all fission products are significantly poorer as monitors. This is probably related to the fact that after achieving its maximum value the rate of decrease of the density of fission product nuclei exceeds the rate of accumulation for small fuel burnup.

The graphs of Fig. 2 permit a quantitative estimate in each specific case of the possible use of one fission product or another as a fuel burnup monitor for the interpretation of data obtained in the gamma scanning of fuel elements.

LITERATURE CITED

1. R. Edwards, Nucl. Appl., 4, 245 (1968).
2. V. I. Gol'danskii et al., Counting Statistics in Recording Nuclear Particles [in Russian], Fizmatgiz, Moscow (1959).
3. C. Lederer et al., Tables of Isotopes, 6th Edition, Wiley, New York (1967).

ANGULAR DISTRIBUTIONS OF ELECTRONS REFLECTED FROM A METAL SURFACE FOR OBLIQUE INCIDENCE

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UDC 537.533.7

A number of practical problems of radiation technology related to shielding, the shaping of electron distributions, etc. require the reflection coefficients and the spectral and angular distributions of electrons reflected from various metal targets. A theoretical description of the differential characteristics of reflected electrons runs into substantial difficulties. Satisfactory results are obtained only under a number of restrictions [1-3].

We have measured the angular distributions of 15-25 MeV electrons reflected from metal targets of various thicknesses for oblique incidence of the primary beam.

The measurements were performed at the LUE-25 linear accelerator [4] (Fig. 1). The distance from the exit window to the target is 15 cm. Metal targets 100 mm in diameter of various thicknesses were made of aluminum, copper, cadmium, and lead. The thicknesses varied from 0.46 to 13.6 g/cm^2, and the angle of incidence from 30 to 70°. To prevent the detector from seeing the electrons emerging from the lateral surface, a 12-mm-thick and 200-mm outside diameter lead shielding ring was placed around the target.

The detector was a 0.25-cm^3 thimble chamber having aluminum walls 0.5 mm thick and was placed 15 cm from the target. The chamber and target were rotated by remote control. The sensitivity was determined by measuring the dependence of the chamber reading on the electron energy for a given incident flux. At energies of 10-25 MeV a thin 0.08-mm-thick tungsten target was placed along the path of the electron beam and the number of electrons incident on the chamber was calculated from the Moliere distribution. In order to go over to the region of lower energies a thick tungsten target in which the average energy of the transmitted electrons was determined from the expression $E_{av} = E_0(1 - R/R_e)$ was placed along the path of the beam and the number of electrons incident on the chamber was determined from the diffusion, proportional to $\cos^2\theta$. Here R_e is the extrapolated range of electrons in tungsten, and R is the target thickness in units of R_e.

Studies showed that the sensitivity of the chamber in the energy range from 1 to 25 MeV does not vary by more than 12%. The error in measuring the current is no more than 5%. All the measurements were performed in air in the plane of incidence. The diameter of the electron beam at exit from the accelerator did not exceed 10 mm. The background angular distribution of electrons was measured without the target and then with the target. The background varied from 2% for large values of θ to 50% for small values of θ.

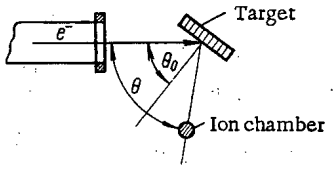

Fig. 1. Schematic diagram of experiment.

Figure 2 shows the dependence of the angular distributions of reflected electrons for E = 25 MeV on the thickness of lead and copper targets for an angle of incidence $\theta_0 = 50°$. For thick targets the position of the maximum in the angular distribution corresponds approximately to an angle of reflection equal to the angle of incidence. As the target thickness is decreased the maximum is displaced toward larger angles. Saturation begins when the thickness is approximately $R_e/2$ and the angular distribution of reflected electrons remains constant. Similar results are obtained for other angles of incidence.

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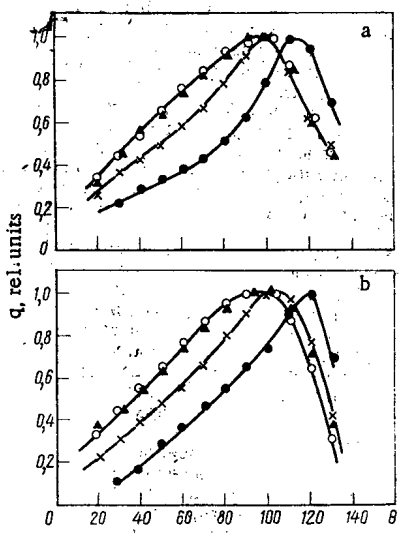

Fig. 2

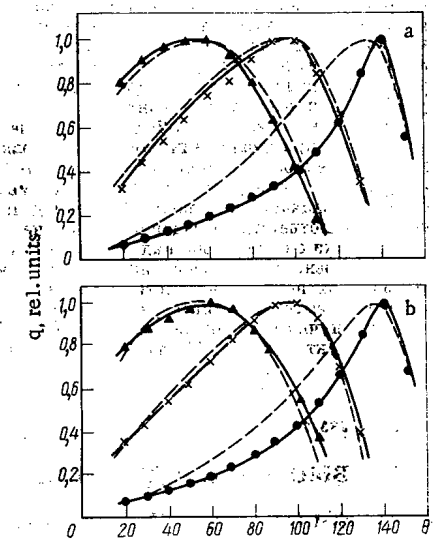

Fig. 3

Fig. 2. Angular distributions of reflected electrons q as a function of target thickness in g/cm^2: a) lead •) 0.57, ×) 1.13, ○) 3.14, △) 13.6; b) copper •) 0.53, ×) 2.14, ○) 4.45, ▲) 13.35.

Fig. 3. Angular distributions of reflected electrons as a function of the angle of incidence for semiinfinite targets of a) lead and b) copper. For lead t = 13.6 and for copper 10.57 g/cm^2: ▲, ×, •) $\theta_0 = 30, 50, 70°$, ---) calculated.

Figure 3 shows the dependence of the angular distributions of reflected electrons on the angle of incidence for semiinfinite copper and lead targets st E = 25 MeV. The angular distributions of electrons reflected from copper and lead were measured at 15 and 20 MeV also, and for aluminum and cadmium at 15, 20, and 25 MeV. The curves obtained are similar to those shown in Figs. 2 and 3. As the angle of incidence is increased the angular distribution of reflected electrons becomes narrower for all elements over the whole energy range. Similar results were obtained for other thicknesses. The curves of Figs. 2 and 3 are plotted in relative units and are not normalized with respect to one another.

To investigate the possibility of a theoretical description of the results obtained the angular distributions of reflected electrons were calculated as in [3] by the equation

$$W(\theta_f, \theta_0, E_0, Z) = \int_0^{E_0} R_1^{\infty}(\bar{P}_f, E_f; \bar{P}_0, E_0) \frac{\cos\theta_f}{\cos\theta_0} dE_f, \tag{1}$$

where

$$R_1^{\infty}(\bar{P}_f, E_f, \bar{P}_0, E_0) = \frac{Z}{16\pi \ln\frac{2E_0}{I_z}} \frac{m}{E_0^2} \frac{\cos\theta_0 \cos\theta_f}{\cos\theta_0 + \cos\theta_f} \left[1 - \cos\oplus + \frac{Z \ln(183Z^{-1/3})}{4\pi \ln\frac{2E_0}{I_z}} \times \frac{m}{E_0} \ln\frac{E_0}{E_f}\right]^{-2} \frac{\left(\cos\theta_0 + \frac{E_0}{E_f}\cos\theta_f\right)^2}{\left[\cos\theta_0 + \cos\theta_f + (E_0 - E_f)\left(\frac{1}{E_f}\ln\frac{2E_f}{I_z} - \frac{1}{E_0}\ln\frac{2E_0}{I_z}\right)\frac{\cos\theta_0\cos\theta_f}{\cos\theta_0 \ln\frac{2E_f}{I_z} + \cos\theta_f \ln\frac{2E_0}{I_z}}\right]^2} \tag{2}$$

is the differential coefficient for the backscattering of relativistic electrons from a semiinfinite target, $E_0\bar{P}_0$ and $E_f\bar{P}_f$ are the energies and moments of the initial and final states respectively, θ_0 and θ_f are respectively the angles between the normal to the target and the directions of the incident and reflected electrons, Z is the atomic number of the target nucleus, I_Z is the ionization potential of the target atoms, m = 0.511 MeV is the electron mass, and $\cos\oplus = \bar{P}_0\bar{P}_f / P_0P_f$.

The expression for the differential backscattering coefficient was derived in [3] under the following restrictions:

on the initial energy of the incident electrons

$$E_0 < E_{cr} \approx \frac{1600m}{Z}; \quad (3)$$

on the final energy of the reflected electrons

$$P_f > 8mZ \frac{\ln (183Z^{-1/3})}{\ln \left(\frac{E_0^2}{mI_z^2}\right)}; \quad (4)$$

on the angle of incidence

$$\left(\frac{\pi}{2} - \theta_0\right)^2 \gg \frac{Zm}{E_0} \frac{\ln (183Z^{-1/3})}{\ln \frac{2E_0}{I_z}}. \quad (5)$$

An estimate of the limits of applicability shows that Eq. (2) can be compared with experimental data only for aluminum at small angles of incidence. However, restrictions 3, 4, and 5 are apparently not critical for the range of atomic numbers and energies considered, since the relative angular distributions of reflected electrons calculated by Eq. (1) agree satisfactorily with the experimental data for all elements in the energy range studied up to angles of incidence of 50°.

LITERATURE CITED

1. R. Dashen, Phys. Rev., 134, A 1025 (1964).
2. N. P. Kalashnikov and V. A. Mashinin, Zh. Éksperim. i Teor. Fiz., 59, 2025 (1970).
3. N. P. Kalashnikov and V. A. Mashinin, in: Second All-Union Symposium on the Interaction of Atomic Particles with a Solid [in Russian] (1972), p. 362.
4. V. I. Ermakov et al., At. Énerg., 29, 206 (1970).

TIME CHARACTERISTICS OF THE DISTRIBUTION OF SCATTERED GAMMA RAYS IN A UNIFORM AIR MEDIUM

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UDC 539.166.3

The nonstationary transport of gamma radiation in various media has recently become of interest. This is related both to the technical applications of gamma radiation [1] and to the study of phenomena arising from the action of short gamma pulses [2]. The published solutions of nonstationary transport problems mainly concern neutron radiation and frequently assume the interaction cross section is velocity-independent [3-5]. Information on nonstationary gamma ray distributions is very limited [6-8].

We study the propagation of gamma radiation in a uniform air medium from a pulsed source which can be described by a delta function; this is essentially an extension of [6]. The basic stationary characteristics of the distribution of scattered radiation were published earlier [9, 10].

Nonstationary distributions were obtained for source energies from 20 keV to 12 MeV and were normalized to a one-photon source. The time distributions of the intensity of the scattered radiation $I_s(E_0, \mu_0 R, \Delta t)$ and the energy absorbed per unit volume of standard density air $\varepsilon_s(E_0, \mu_0 R, \Delta t)$ were calculated for delay times $t = T - R/c$ of 2 nsec to 3 μsec. Henceforth, we discuss the distributions $I = 4\pi R^2 e^{\mu_0 R} I_s$ and $\varepsilon = 4\pi R^2 e^{\mu_0 R} \varepsilon_s$. Typical examples of the time dependence of the intensity are shown in Figs. 1 and 2. It is clear that the time behavior can be arbitrarily divided into two regions — a rapidly decreasing part for times less than ~0.1 μsec, and a more slowly decreasing part for $t > 0.1$-0.2 μsec. The rate of decrease of intensity in both regions, their duration, and the contributions to the time integral of the intensity depend on the distance $\mu_0 R$ and the source energy E_0. For all source energies a slowing down of the rate of decrease of intensity with increasing $\mu_0 R$ is observed, leading to an increase in the time the scattered

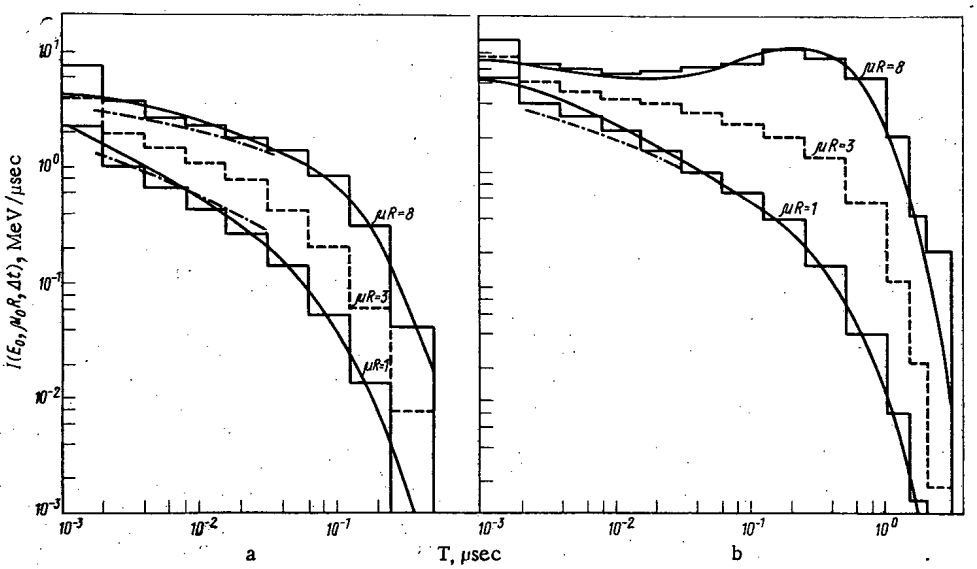

Fig. 1. Time dependence of intensity of scattered gamma radiation for source energies E_0 = a) 30 and b) 100 keV.

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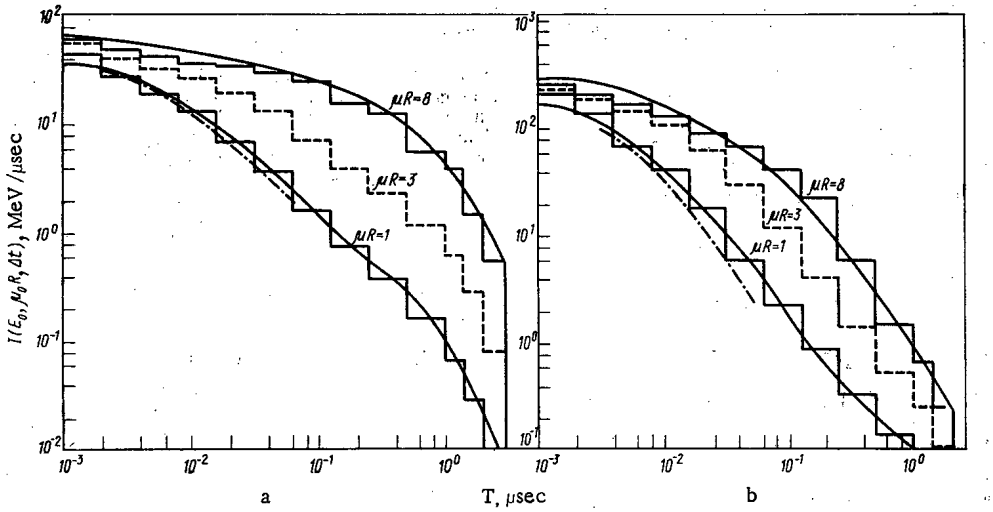

Fig. 2. Time dependence of intensity of scattered radiation from sources of hard gamma radiation with E_0 = a) 1 and b) 6 MeV.

radiation is effective. Moreover, for source energies from 50 to 250 keV, times ~0.1-0.25 μsec, and distances $\mu_0 R \approx 6$-8, an increase of intensity is observed (Fig. 1b).

The character of the change of intensity of the scattered radiation at short times can be found from an analytic treatment of the transport processes. To do this the time dependence of the intensity of singly scattered and doubly scattered radiation was computed by numerical integration. In the first scatter the angular and energy dependences of the scattering cross section were taken into account, and for $E_0 \leq 100$ keV Rayleigh scattering and the effect of screening in Compton scattering were also taken into account; in calculating the second scatter the scattering cross section was assumed isotropic. The total intensity of

TABLE 1. Coefficients in Eq. (1) for the Intensity of Scattered Radiation

$\mu_0 R$	A_1, MeV/μsec	α, μsec	A_2, MeV/μsec	τ_1, μsec	τ_2, μsec	τ_3, μsec	A_1, MeV/μsec	α, μsec	A_2, MeV/μsec	τ_1, μsec	τ_2, μsec	τ_3, μsec
	E_0 = 30 keV						E_0 = 40 keV					
1	1,05	0,0085	0,06	0,100	0,072	0,033	1,40	0,012	0,13	0,140	0,128	0,037
2	0,95	0,0230	0,18	0,120	0,075	0,068	1,20	0,034	0,47	0,160	0,128	0,075
4	0,88	0,0520	0,51	0,135	0,079	0,080	1,10	0,088	1,53	0,200	0,145	0,100
8	0,83	0,1150	1,86	0,200	0,082	0,100	1,00	0,160	5,42	0,280	0,150	0,152
	E_0 = 100 keV						E_0 = 500 keV					
1	2,70	0,022	0,46	0,200	0,285	0,077	11,00	0,014	0,71	0,170	0,458	0,060
2	2,25	0,066	2,07	0,250	0,295	0,137	8,10	0,051	2,46	0,220	0,510	0,111
4	1,80	0,170	11,53	0,350	0,305	0,270	6,70	0,130	9,53	0,320	0,555	0,190
8	1,60	0,220	70,49	0,480	0,330	0,560	5,40	0,180	42,80	0,420	0,570	0,250
	E_0 = 1 MeV						E_0 = 4 MeV					
1	21,5	0,009	0,75	0,140	0,550	0	140	0,0025	0,53	0,100	0,650	0
2	16,0	0,035	2,20	0,180	0,600	0	90	0,0110	1,37	0,120	0,670	0
4	13,0	0,090	6,30	0,255	0,650	0	65	0,0350	3,40	0,150	0,670	0
8	10,0	0,150	22,30	0,350	0,680	0	57	0,0670	8,00	0,180	0,670	0
	E_0 = 8 MeV						E_0 = 12 MeV					
1	460	0,00125	0,39	0,092	0,650	0	600	0,00095	0,33	0,090	0,650	0
2	260	0,0056	0,90	0,110	0,650	0	380	0,0048	0,72	0,110	0,650	0
4	190	0,0180	1,80	0,135	0,650	0	280	0,0140	1,53	0,140	0,650	0
8	150	0,0420	4,87	0,170	0,650	0	230	0,0370	4,10	0,170	0,650	0

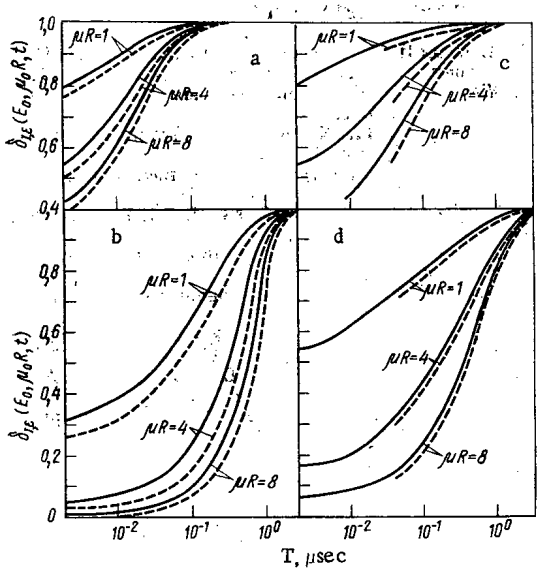

Fig. 3. Time buildup of the intensity and absorbed gamma energy for E_0 = a) 20 keV; b) 100 keV; c) 10 MeV; d) 1 MeV.

singly and doubly scattered radiation $I^{(1,2)}$ calculated in this way is shown in Figs. 1 and 2 by the dash — dot curves. Taking account of double scattering satisfactorily describes the change of intensity for times up to ~50 nsec for all initial energies when $\mu_0 R \leq 2$, and over the whole range of distances considered up to $\mu_0 R = 8$, for $E_0 \leq 40$ keV. For energies above 40 keV and distances $\mu_0 R > 2$ the values of $I^{(1,2)}$ are smaller than the values obtained in statistical calculations, which shows the necessity of taking account of higher orders of scattering, thus in general giving the problem a diffusion character. Since the contribution of singly scattered radiation to the total energy flux at short times, even for $\mu_0 R \approx 6$-8, amounts to several tens of percent, the behavior of I(t) is qualitatively similar to that of the energy flux of singly scattered radiation.

To make it more convenient to use the time distributions obtained, an analytic formula was chosen to approximate the time behavior of the intensity of scattered radiation from $t \approx 2$ nsec:

$$I(E_0, \mu_0 R, t) = A_1 e^{-t/\tau_1} \ln\left(1 + \frac{a}{t}\right) + A_2 e^{-t/\tau_2}(1 - e^{-t/\tau_3}). \quad (1)$$

The first term in Eq. (1) has a singularity at t = 0 corresponding to the singularity of singly scattered radiation. The second term represents the contribution of multiply scattered radiation. The values of the coefficients in Eq. (1) are listed in Table 1.

The time dependence of the radiant energy absorbed $\varepsilon(E_0, \mu_0 R, t)$ is practically the same as that of $I(E_0, \mu_0 R, t)$. The only difference is in the small increase in the energy release time, i.e., the time to decrease to a definite level, for example to 10^{-3} of the initial value, in comparison with the corresponding time for the intensity. However, in considering the time distribution of absorbed energy it must be kept in mind that the energy release time consists of the time of a random walk of a photon before the formation of a Compton or a photoelectron, and the time for the electron to transfer the energy to the air. The latter can amount to ~0.1 μsec for electron energies higher than 1 MeV, and can be comparable with or

TABLE 2. Coefficients in Eq. (2) for the Absorbed Energy

$\mu_0 R$	$A_1 \times 10^3$	$A_2 \times 10^3$	τ_1	τ_2	τ_3	$A_1 \times 10^3$	$A_2 \times 10^3$	τ_1	τ_2	τ_3	$A_1 \times 10^3$	$A_2 \times 10^3$	τ_1	τ_2	τ_3
	$E_0 = 30$ keV					$E_0 = 40$ keV					$E_0 = 50$ keV				
1	0,10	0,02	0,032	0,09	0,025	0,072	0,019	0,042	0,140	0,029	0,059	0,017	0,051	0,192	0,043
2	0,14	0,036	0,032	0,095	0,052	0,11	0,076	0,042	0,137	0,070	0,092	0,086	0,051	0,190	0,098
4	0,22	0,138	0,035	0,093	0,120	0,17	0,317	0,042	0,137	0,140	0,16	0,397	0,051	0,190	0,215
8	0,38	0,576	0,035	0,095	0,210	0,33	1,85	0,042	0,137	0,440	0,32	2,70	0,051	0,192	0,55
	$E_0 = 60$ keV					$E_0 = 80$ keV					$E_0 = 100$ keV				
1	0,053	0,016	0,054	0,24	0,054	0,054	0,017	0,054	0,30	0,086	0,062	0,018	0,054	0,335	0,090
2	0,084	0,084	0,054	0,24	0,130	0,086	0,088	0,054	0,31	0,189	0,097	0,089	0,054	0,347	0,182
4	0,15	0,48	0,054	0,24	0,290	0,15	0,638	0,054	0,31	0,485	0,17	0,705	0,054	0,355	0,535
8	0,26	3,42	0,054	0,24	0,700	0,27	5,22	0,054	0,31	1,270	0,33	5,96	0,054	0,380	1,410
	$E_0 = 150$ keV					$E_0 = 250$ keV					$E_0 = 500$ keV				
1	0,09	0,019	0,054	0,420	0,07	0,15	0,019	0,057	0,520	0,065	0,29	0,014	0,057	0,660	0,060
2	0,14	0,088	0,054	0,428	0,16	0,23	0,073	0,057	0,528	0,135	0,42	0,068	0,057	0,665	0,115
4	0,24	0,60	0,054	0,455	0,46	0,37	0,477	0,057	0,555	0,350	0,65	0,313	0,057	0,682	0,184
8	0,42	5,46	0,054	0,475	1,34	0,60	4,06	0,057	0,565	0,960	1,12	1,46	0,057	0,698	0,275

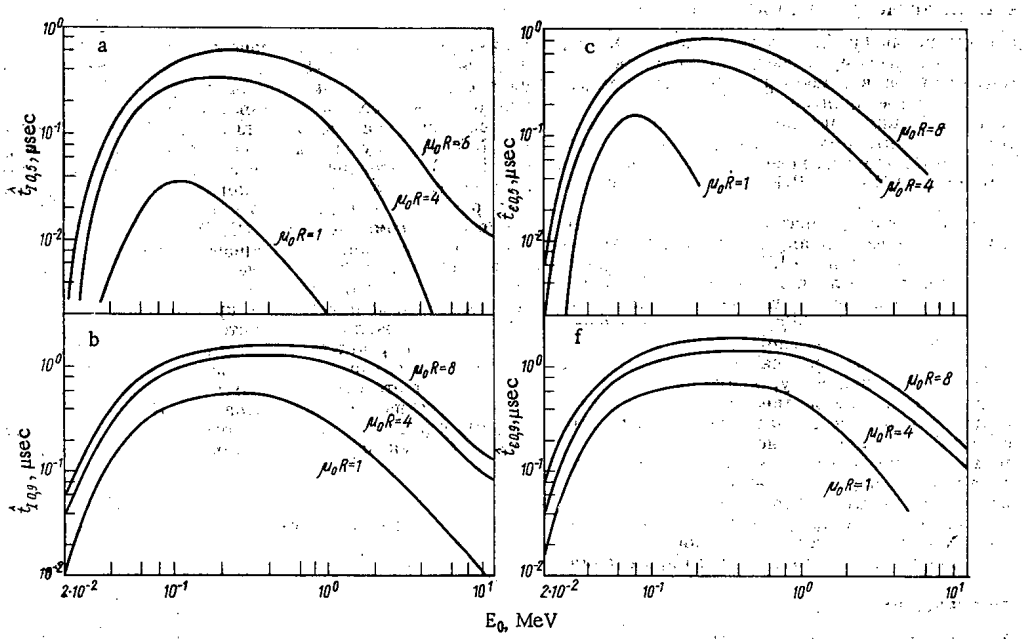

Fig. 4. Characteristic buildup times of the intensity and absorbed energy as functions of the source energy.

larger than the initial averaging intervals assumed in the calculation of the intensity. Therefore, the first minimum averaging interval for energy absorption, found under the assumption that the electrons are absorbed instantaneously, is 0-0.125 μsec.

The time dependence of the energy release is approximated by the expression

$$\varepsilon(E_0, \mu_0 R, t) = A_1 e^{-\frac{t-0.05}{\tau_1}} + A_2 e^{-\frac{t-0.05}{\tau_2}} \left(1 - e^{-\frac{t-0.05}{\tau_3}}\right) \text{ MeV/cm} \cdot \mu\text{sec}, \tag{2}$$

with the coefficients for $E_0 \leq 500$ keV given in Table 2. The coefficients are chosen in such a way that the average rate of energy release, including the energy release from unscattered radiation, agrees with that obtained from statistical calculations to within 10-15% in the first interval.

The average accuracy of approximate formulas (1) and (2) is 5-15%. At isolated points the differences may reach ~ 30% for extremely long and short delay times, which of course agrees with the statistical error in these intervals. Examples describing I (E_0, μ_0R, t) by Eq. (1) are shown in Figs. 1 and 2 by solid curves plotted on the histograms.

By using the differential distributions it is possible to trace the shaping of the radiation field in time and to calculate the time buildup factors B(E_0, μ_0R, t). Figure 3 shows the time dependence of δ_I defined by the relation

$$\delta_I(E_0, \mu_0 R, t) = \frac{B_E(E_0, \mu_0, R, t)}{B_E(E_0, \mu_0 R)} = \frac{E_0 + \int_0^t I(E_0, \mu_0 R, t)\, dt}{E_0 + \int_0^\infty I(E_0, \mu_0 R, t)\, dt}. \tag{3}$$

A comparison of the values of δ_I for various source energies shows that stationary values of the intensity are reached most rapidly for source energies of 20-30 keV and 6-12 MeV. The rate of formation of stationary radiation intensity levels can be characterized quantitatively, for example, by the buildup times to 50 or 90% of the total intensity: $\hat{t}_{I\,0.5}$ or $\hat{t}_{I\,0.9}$.

The change of characteristic times $\hat{t}_I$ for a change of the source energy E_0 is shown in Fig. 4a and b. The graphs have a maximum, as is shown particularly clearly for $\hat{t}_{I\,0.5}(E_0)$. We note that the largest value of $\hat{t}_I$ occurs at $E_0 \approx 100$ keV and above, i.e., for those energies for which the stationary buildup factors are maximum [9]. The values of $\hat{t}_I$ decrease with increasing source energy only for $E_0 = 1$-2 MeV, although for these energies the stationary buildup factors are decreased by more than an order of magnitude ($\mu_0 R = 8$) in comparison with the maximum values.

The rules for the formation of stationary distributions of the absorbed energy are very similar to those for the radiation intensity discussed above (cf. Fig. 3 where the values of $\delta_\varepsilon(E_0, \mu_0 R, t)$, defined similarly to δ_I, are plotted by open curves) although the characteristic times $\hat{t}_\varepsilon$ are 10-30% longer than the corresponding times $\hat{t}_I$, as can be seen from Fig. 4c and d.

In conclusion, we note that the characteristic buildup times $\hat{t}_{I,\varepsilon}$ can be used to determine the temporal properties of the source. Comparing $\hat{t}_{I,\varepsilon}$ with the characteristic time of development of the effect produced by the action of the source or with the time of its action it can be judged whether the source is instantaneous or steady with respect to the phenomena considered.

LITERATURE CITED

1. I. G. Dyad'kin, B. N. Krasil'nikov, and V. N. Starikov, At. Énerg., 35, 272 (1973).
2. A. V. Zhemerev et al., At. Énerg., 35, 438 (1973).
3. V. G. Morozov and S. A. Kholin, At. Énerg., 18, 62 (1965).
4. V. M. Kochetkov, Zh. Vychisl. Matem. i Matem. Fiz., 8, 1152 (1968).
5. W. Yingling and C. Bridgman, Nucl. Sci. and Engng., 53, 123 (1974).
6. O. I. Leipunskii et al., At. Énerg., 10, 493 (1961).
7. N. A. Seleznev, At. Energ., 31, 271 (1971).
8. A. I. Veretennikov, V. Ya. Averchenkov, and M. V. Savin, At. Énerg., 11, 177 (1961).
9. M. N. Vrubel', S. N. Sidneva, and A. S. Strelkov, At. Énerg., 34, 47 (1973).
10. S. N. Sidneva and A. S. Strelkov, At. Énerg., 36, 135 (1974).

REACTIONS OF OXYGEN AND OXIDES OF NITROGEN AND CARBON WITH O_2SbF_6, $XeFSbF_6$, AND $KrFSb_2F_{11}$

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UDC 546.29

A number of recently published studies have dealt with the chemical binding of noble gases released into the atmosphere during the operation of radiochemical plants and atomic power stations.

It was shown that the chemical trapping of radioactive radon and xenon is possible in theory. Thus, radon is bound in a solid, nonvolatile compound when it interacts with $ClF_2^+SbF_6^-$, $BrF_2^+SbF_6^-$, O_2SbF_6, and other compounds [1, 2].

Xenon is practically completely trapped when it is passed through a column filled with dioxygenyl hexafluoroantimonate [2]. Theoretical calculations show that such binding is possible by means of $N_2F^+SbF_6^-$; this assumption still requires experimental verification [3].

The reactant most widely used for binding radon and xenon seems to be dioxygenyl hexafluoroantimonate. The compound is thermally stable and relatively simple to obtain; the reaction with Xe and Rn yields oxygen.

In connection with the chemical binding of noble gases, it is of interest to study the reactions of dioxygenyl, xenon, and krypton complexes with gases produced during the operation of radiochemical plants (NO, NO_2, CO, CO_2, and O_2). Reactions with H_2 and H_2O were not studied, since the nature of this type of reaction is obvious.

Experimental Part

Initial Reactants. The dioxygenyl complex was obtained by photochemical means; the system $O_2 + F_2 + SbF_5$ was irradiated with light from a high-pressure mercury lamp. To speed up the process, the reaction mixture was heated to 70-80°C. The complex of xenon difluoride with antimony pentafluoride was obtained by a method described in [4]. The complex with composition $KrF_2 \cdot 2SbF_5$ was synthesized in the manner described in [5]. The synthesis of NO, NO_2, and CO was carried out by appropriate methods as described in [6]. In the present study we used industrial carbon dioxide gas and oxygen. The purity of the gaseous reactants was determined by means of infrared analysis.

The reactions of the complexes with NO, NO_2, CO, CO_2, and O_2 were carried out mainly in a quartz reactor. Some experiments were carried out in a reactor made of nickel. A prescribed quantity of dioxygenyl, xenon, or krypton complex (~1 g) was charged into the reactor in a dry argon box; then the reactor was connected to a vacuum stand and thoroughly vacuumed, and the appropriate reactant was introduced. An analysis of the reaction products was carried out by the method of infrared spectroscopy; the solid reaction products were subjected to hydrolysis. The results of the experiments are shown in Table 1.

Evaluation of the Results

Reactions with Nitrous Oxide. Each of the three complexes interacts actively with nitrous oxide. The most violent reaction is the one with the krypton complex. This is understandable, since $KrF_2 \cdot 2SbF_5$ is a more powerful fluorooxidizer than O_2SbF_6 and $XeF_2 \cdot SbF_5$.

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TABLE 1. Results of the Experiments

Reactants	Reaction products		Equation of the reaction	Remarks
	gaseous phase	condensed phase		
$O_2SbF_6+NO_2$	SiF_4 [a], O_2	$NO_2^+SbF_6^-$ [b]	$O_2SbF_6+NO_2 \rightarrow$ $\rightarrow NO_2SbF_6+O_2$	[a] SiF_4 is formed when intermediate reaction products act on the reactor wall [b] NO_2 is produced during hydrolysis
$XeFSbF_6+NO_2$	SiF_4, XeF_2, Xe	XeF_2 [c], NO_2SbF_6	$2NO_2+2XeFSbF_6 \rightarrow$ $\rightarrow 2NO_2SbF_6+Xe+XeF_2$	[c] In one of the experiments xenon difluoride was collected in substantial amounts and identified from its infrared spectrum and its reaction with water
$KrFSb_2F_{11}+NO_2$	SiF_4, Kr	NO_2SbF_6	$KrFSb_2F_{11}+2NO_2 \rightarrow$ $\rightarrow 2NO_2SbF_6+Kr$	—
O_2SbF_6+NO	SiF_4, NO_2	$NOSbF_6$ [d]	$O_2SbF_6+3NO \rightarrow$ $\rightarrow NOSbF_6+2NO_2$	[d] NO_2 is produced during hydrolysis
$XeFSbF_6+NO$	SiF_4, NO_2 [e]	$NOSbF_6$	$XeFSbF_6+2NO \rightarrow$ $\rightarrow NOSbF_6+Xe+NOF$ [f]	[e] NO is oxidized to NO_2 by the oxygen produced from the walls of the reactor [f] The infrared spectrum does not include the bands due to NOF absorption; this is probably due to the fact that NOF interacts actively with quartz
O_2SbF_6+CO	CO	O_2SbF_6	The reaction does not take place at room temperature	—
$XeFSbF_6+CO$	COF_2, Xe	SbF_5	$XeFSbF_6+CO \rightarrow$ $\rightarrow COF_2+SbF_5+Xe$	—
$O_2SbF_6+CO_2$	CO_2	O_2SbF_6	The reaction does not take place at room temperature	—
$XeFSbF_6+CO_2$	CO_2	$XeFSbF_6$	The same	—
$KrFSb_2F_{11}+CO_2$ [g]	CO_2	$KrFSb_2F_{11}$	The same	[g] p_{CO_2} = 300 torr
$KrFSb_2F_{11}^++O_2$	Kr, F_2	$O_2Sb_2F_{11}$	$2KrFSb_2F_{11}+2O_2 \rightarrow$ $\rightarrow 2O_2Sb_2F_{11}+Kr+F_2$	—

Infrared analysis of the gaseous products indicates in every case that the reaction products do not include any nitrogen oxyfluorides, but there is indirect evidence to show that oxyfluorides are formed as intermediate compounds in the reaction of NO_2 with the complexes of oxygen, xenon, and krypton. The appearance of silicon tetrafluoride in the reaction products is explained by the action of nitrogen oxyfluorides on the walls of the reactor. Another indication of the formation of NO_2F is the appearance of xenon difluoride in the products of the $NO_2+XeFSbF_6$ reaction. The reaction takes place in several stages:

$$2NO+XeFSbF_6 \rightarrow 2NO_2F+Xe+SbF_5; \quad (1)$$
$$NO_2F+XeFSbF_6 \rightarrow NO_2SbF_6+XeF_2;$$
$$NO_2F+SbF_5 \rightarrow NO_2SbF_6.$$

The summary equation of the reaction is

$$2NO_2+2XeFSbF_6 \rightarrow 2NO_2SbF_6+Xe+XeF_2. \quad (4)$$

Xenon difluoride is formed as a result of a reaction in which the cation XeF^+ is displaced by the cation NO_2^+ (2), since the latter has a lower ionization potential. Analogously, oxygen is produced in the $NO_2 + O_2SbF_6$ reaction. It is probable that the reaction of NO_2 with $KrF_2 \cdot 2SbF_5$ takes place in the same way, but krypton difluoride is an unstable and extremely reactive compound, so that it is natural that it is not found among the reaction products. The infrared spectrum of the solid powder formed as a result of the reaction of the complexes with NO_2 has a characteristic band in the 666 cm^{-1} region attributable to the oscillations of the Sb-F bond in the anion SbF_6^-. Hydrolysis of the powder takes place with the production of NO_2, which indicates that the cations O_2^+, XeF^+, and KrF^+ are displaced by the cation NO_2^+ in the corresponding complexes.

So far as trapping is concerned, this means that the presence of NO_2 is inadmissible at the stage of binding of radon, xenon, and (if this is possible) krypton; there must be a preliminary removal of NO_2 from the gases. The absence of chemically active substances from the gaseous products of the $NO_2 + O_2SbF_6$ reaction confirms the possibility of using the dioxygenyl complex for preliminary binding of the NO_2; this appears to be reasonable if the NO_2 content in the emitted gases is low.

Reaction with NO. The dioxygenyl and xenon complexes react with NO more intensively than with nitrous oxide, probably because of the higher chemical activity of nitric oxide; the mechanisms of interaction of nitrogen oxides with these complexes are apparently analogous. In the first stage there is fluoridation of the nitric oxide and breakdown of the complex. Then the nitrogen oxyfluoride which has been formed reacts with the free antimony pentafluoride and the remaining complex, forming with a cation containing nitrogen. In the reaction with O_2SbF_6 the oxygen formed as a result of the breakdown of the dioxygenyl complex oxidizes the NO to nitrous oxide.

Unlike the reaction with NO_2, when $XeFSbF_6$ reacts with NO, no formation of XeF_2 is observed, possibly because NO is more intensively fluoridated by xenon difluoride than NO_2 is. The infrared spectrum of the gaseous products of the reaction of O_2SbF_6 with NO is characterized by bands attributable to the absorption of NO_2. Nitrous oxide is formed in the reaction of $XeFSbF_6$ with NO as well, but in this case the oxidation of the nitrogen takes place as a result of the oxygen produced from the walls of the reactor. The hydrolysis of the solid reaction products takes place with NO_2 production.

Judging by the results of the experiments, if we are talking about the possibility of trapping noble metals, nitric oxide is just as inadmissible in the emitted gases as nitrous oxide. Moreover, reaction with the dioxygenyl complex produces nitrous oxide, which must be removed from the emitted gases.

Reactions with CO. The reaction of $XeFSbF_6$ with carbon dioxide is apparently a special case of fluoridation of CO to COF_2. The results of infrared analysis confirm this assumption. The equation for the reaction is

$$CO + XeFSbF_6 \rightarrow COF_2 + Xe + SbF_5. \quad (5)$$

At the same time, contact between CO and O_2SbF_6 does not lead to fluoridation of the carbon dioxide. The infrared spectrum of the gaseous phase contains only bands attributable to the absorption of CO. Carbon monoxide does not interact with O_2SbF_6. This indicates that CO, although it passes unimpeded through a layer of dioxygenyl complex, will break up the xenon complex, as a result of which radioactive xenon and a fluorine compound will be emitted into the atmosphere.

Reactions with CO_2 and O_2. Unlike carbon monoxide, carbon dioxide does not react either with O_2SbF_6 or with $XeFSbF_6$, nor even with such strong fluorooxidizers as $KrF \cdot 2SbF_5$. This fact can probably be utilized for removing carbon monoxide from emitted gases by oxidizing it to CO_2.

Oxygen is oxidized by the krypton complex to O_2^+ at room temperature. In the IR spectrum of a specimen of $KrF^+Sb_2F_{11}^-$ kept in contact with oxygen for 2-3 h we find an intense line in the 1,862 cm^{-1} region, which is characteristic of the oscillations of the O_2^+ cation.

Thus, most of the above-mentioned main components of emitted gases interact with the dioxygenyl, xenon, and krypton complexes. Consequently, for chemical binding of noble gases, in addition to searching for appropriate fluorooxidizers we must solve the problem of first removing from the emitted gases of radiochemical plants such impurities as water vapor, hydrogen, nitrous and nitric oxides, carbon monoxide, and oxygen.

LITERATURE CITED

1. L. Stein, Science, 175, 1463 (1972).
2. L. Stein, Nature, 243, 30 (1973).
3. J. Liebman, Nature, 243, 84 (1973).
4. V. A. Legasov et al., Preprint IAE-2163 (1971).
5. V. D. Klimov, V. N. Prusakov, and V. B. Sokolov, Dokl. AN SSSR, 217, 1077 (1974).
6. G. Brauer (editor), Handbook of Preparative Inorganic Chemistry, Academic Press (1965).

FROM THE COUNCIL FOR MUTUAL ECONOMIC AID (CMEA)

TWENTY-EIGHTH CONFERENCE OF THE PERMANENT COMMISSION FOR THE UTILIZATION OF ATOMIC ENERGY (PKIAÉ)

Yu. I. Chikul

The 28th Conference of the Permanent Commission for the Utilization of Atomic Energy, Council for Mutual Economic Aid (PKIAÉ SÉV) was held on June 17-20, 1975 in Rostock (DDR). Delegations from Bulgaria, Hungary, DDR, Cuba, Poland, Rumania, USSR, and Czechoslovakia participated in the work of the Commission. Representatives of "Interatominstrument," "Interatoménergo," the Joint Institute for Nuclear Research, and the Coordination Center for the Development of Medical Techniques were present by invitation of the Commission.

The Conference discussed measures taken by the Commission in the field of nuclear power generation, associated with carrying out the resolutions of the Executive Committee of the Council and also the committees of the Council for Mutual Economic Aid through collaboration in the field of systematic activity and scientific—technical collaboration. Proposals were considered for cooperation in the possible routes for future utilization of reactor technology, problems about the state of development and production of nuclear equipment in the period up to 1980, and the trends for its development in the member countries of CMEA up to 1990, cooperation in the field of radiation techniques and technology, including unified health regulations for the installation and operation of high-powered radioisotope γ facilities, and also problems concerning the standardization of instruments and products of nuclear technology.

The Commission considered a report on the work undertaken in 1974 and on future activities.

Appropriate recommendations and resolutions were accepted on all the problems discussed.

JOURNAL OF COLLABORATION

A meeting of representatives of PKIAÉ SÉV and the Permanent Commission for Electric Power was held on March 25-26, 1975 in Moscow. The meeting discussed the problem of the coordination of actions for ensuring radiation safety. This problem occurs in the "General comprehensive program of collaboration between member countries of CMEA and the Socialist Federated Republic of Yugoslavia (SFRYu) in the period up to 1980, in the field of protection and improvement of the environment and the rational utilization of natural resources allied to this." The meeting reviewed the complex of work carried out in the field of radiation safety within the framework of CMEA and approved the agenda periods of development and the discussion of the work plans on this theme.

A Symposium on the Scientific Aspects of Radiation Sterilization of Medicinal Products took place on April 7-9, 1975 in Brno (Czechoslovakia).

The Symposium created great interest and 75 specialists participated in its work. The participants in the Symposium heard and discussed 34 reports on the principal problems of radiation sterilization of medicinal products:

1. Technologies of processes in the sterilization of pharmaceutical preparations, antibiotics, vaccines, transplants, etc.

2. The choice of radiation dosage.

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3. Stability of preparations to the action of radiation.

4. The application of combined methods of development.

5. The choice of packaging materials, etc.

From the presentations of the papers at this symposium and the symposium of the IAEA on the utilization of ionizing radiation for the sterilization of medicinal products and biological tissues (Bombay, December 9-13, 1974) it follows that the problem of radiation sterilization of medicinal products and biological tissues occupies a notable place in the medicinal industry of many countries; the method of radiation sterilization is effective and sometimes also is the only method of guaranteeing the sterility of a number of medicinal preparations and transplants.

The Foundation Meeting of the Temporary International Collective Body on Radiation Sterilization of Materials and Manufactured Products for Medicinal Purposes (VMK-RS) took place on April 9-11, 1975 in Brno.

The participants at the meeting recorded that investigations into the radiation sterilization of materials and products for medicinal purposes are being conducted over a wide front and that the existing experience allows collaboration to be organized between the member countries of CMEA in this field: the development of unified methods for determining the sterilization dosages of radiation and monitoring of the irradiated output, carrying out its clinical tests, specialization and coordination of production, etc.

The radiation facility with a cobalt irradiator, operating in Czechoslovakia in the Faculty of the Scientific-Research Textile Institute (in Veverska Bitushka), which is equipped with the necessary apparatus, at present is a suitable base for carrying out the joint work.

The meeting considered and discussed the project of agreement on the setting-up of the VMK-RS, its program of work, problems concerning its financing and shared participation of its negotiating parties.

The Ninth Conference of the Commission for Scientific and Technical Collaboration — Radiation Techniques, was held on April 8-11, 1975 at Brno.

The Council considered and passed resolutions on a number of problems including:

Approval of the "Technical proposal concerning the organization for member countries of CMEA for the production of radiation-modified polymers and rubber — technical goods, and recommendation of its presentation for consideration at the next meeting of the PKIAÉ — SÉV.

Consideration and agreement of the final edition of the project Unified Health Regulations for the installation and operation of high-powered radiation γ facilities ("ESP-gamma"), and also preparation of the draft for submission of this problem to the PKIAÉ — SÉV.

Heard information from the Secretariat Division of CMEA concerning the course of exploitation for forecasting the development of radiation techniques and technology, considered it expedient, as a basis for exploiting the forecast, to set up an inventory of applied developments in radiation techniques and technology in the member countries of CMEA.

Endorsement of the information from the Czechoslovakia delegation concerning the results of technico-economic calculations in the field of radiation sterilization of materials and products for medicinal purposes.

Took note of the information from a number of delegations concerning the course of preparation of reference — procedural documents on the radiation sterilization of materials and products for medicinal purposes.

Refined the agenda and periods of presentations of data on a number of points, for the long-term complex plan for scientific — technical cooperation between member countries of CMEA in the field of radiation sterilization of materials and products for medicinal purposes.

Took note of the statement from the USSR delegation concerning the organization in October 1975 in Leningrad of a conference of specialists of member countries of CMEA, on the radiation sterilization of materials and products for medicinal purposes, and on working out unified reference documents in this field.

Discussed the results of the symposium of specialists from member countries of CMEA on the scientific problem of radiation sterilization of medicinal preparations which took place on April 7-9, 1975 in Brno, and noted its great importance.

Considered and approved the report of the Commission for Scientific and Technical Cooperation — Radiation Techniques (KNTS-RT) concerning the work in 1974 and future activities.

Approved the preliminary notification of the day of the Tenth Conference of the KNTS-RT, which is to take place from September 30 to October 3, 1975 in Minsk.

The Fifth Meeting of the KNTS on Research Nuclear Reactors took place on May 13-16, 1975 in Warsaw. Generalized proposals were considered for future collaboration in the field of research reactors in 1976-1980, prepared by the delegation of specialists from Rumania, based on information from the symposium on "Experience in the Operation and Utilization of Research Reactors" (October 1974, Predyal, Rumania). A review report from the delegation of specialists from Hungary was heard and discussed, concerned with the future collaboration of member countries of CMEA in the field of utilization of low-power research reactors, critical and subcritical assemblies for the training of personnel, and also reports from the country delegations on the course of the work in carrying out a program of cooperation on the theme: "Development and investigations in the field of intrareactor measurement techniques" and "Monitoring and control of research nuclear reactors." Specialists from the German Democratic Republic (DDR) informed the participants of the conference of the results of work on the noise diagnostics of reactors and intrareactor measurement techniques.

The preparation of work plans for scientific — technical cooperation between member countries of CEMA was also considered at the conference, on the theme of problems of the "Development and improvement of research nuclear reactors and the execution of work in them in the field of reactor physics and techniques." Agreed recommendations were accepted for all the problems discussed.

A Conference of specialists on Problems of the Safety of Sodium —Water Steam Generators was held on June 9-13, 1975 in Moscow. The following problems were discussed: the concept and prospects for the development of steam generators from the point of view of safety and reliable operation of nuclear power stations with fast reactors, schemes and designs of steam generators, numerical methods for estimating safety and optimization of structural layouts of steam generators, fundamental structural and technological solutions in respect of increased safety; the choice of structural materials for sodium — water steam generators; heat-exchange in steam generators, thermal fatigue and corrosion — thermal fatigue defects of units of the steam generator, irreversible change of form of materials under conditions of heat exchange and estimation of the resistance to irreversible change of form; the water cycle and corrosion; quality standards of water, organization and methods of flushing; technology and monitoring during manufacture and assembly of steam generators, assurance of efficient quality control; analysis of processes and situations leading to emergencies; development of hazards in steam generators, physicochemical processes in the case of hazards created by the interaction of sodium with water, the dynamics of development of leaks, stability of materials in the zone of interaction; leak-detection methods, existing and future systems for the detection of leaks, circuits and instrumentation for detection systems; experience in the operation of steam generators on test benches and facilities. It was noted that within the framework of cooperation on the problem being considered, considerable scientific-research and experimental-design work is being carried out. It was acknowledged to be essential to continue collaboration according to the agreed program.

The 10th Conference of "Interatominstrument" took place on May 13-17, 1975 in Warsaw. The Conference heard an account by the Director of the Society, Z. Tvardonya, about the work undertaken in 1974, the execution of the resolutions of the Council, passed at previous conferences, and about the progress in achieving the "Program of Actions of the Society in the Period 1974-1980." An account was heard also of the revised Commission and the results of financial — economical actions of the Society in 1974. The proposals of the director were considered and the finance plan for 1976 was approved.

Information presented by the director was discussed, concerning the specialization and coordination of production, information on the course of work for founding the faculties of the Institute of Nuclear Research in relation to the technical servicing of instruments and facilities for nuclear techniques, and the preparation of data which are essential for creating, within the framework of the Institute of Nuclear Research, a factory for the production of nuclear-physics equipment for the monitoring and control of nuclear power stations. The Council passed appropriate resolutions on these problems.

Preliminary notification was given for the data of the Eleventh Conference, which will take place on November 11-15, 1975 in Warsaw.

FOURTH ALL-UNION CONFERENCE ON DISSOCIATING GASES

V. B. Nesterenko

The fourth All-Union Conference on Dissociating Gases as Coolants and Working Substances in Nuclear Power Stations took place in Minsk on June 2-6, 1975. The Conference took place under the auspices of the Academy of Sciences of the USSR with the participation of the State Commission for the Use of Atomic Energy, the State Commission for Inorganic Fuel, the Department of Physicotechnical Power Problems (Academy of Sciences of the USSR), and the Scientific Council of the Academy of Sciences of the USSR on Heat Physics. Taking part in the work of the Conference were 350 representatives from 75 scientific-research institutes, undertakings, and organizations, as well as those of a number of Soviet Ministries and Departments.

Six review papers on the general tendencies of investigations into dissociating gases were read to the plenary sessions; the section meetings received 205 papers and communications.

Seven sections took part: These respectively considered atomic power stations, circuits, and cycles involving dissociating gases; the thermophysical properties of dissociating gases and liquids; heat transfer of dissociating coolants; gasdynamics of dissociating coolants; construction and fuel materials, working parts and technology of dissociating coolants; the physics of fast reactors incorporating dissociating coolants; the dynamics, control, and regulation of nuclear power installations using dissociating coolants.

Those present at the Conference discovered that, in the three years since the Third All-Union Conference, complex fundamental investigations had been carried out into the thermophysical, physicochemical, and technological properties of N_2O_4 in the temperature range 25-500°C at pressures of 1-170 atm and also into the properties of the construction materials used in conjunction with N_2O_4 up to 750°C. Considerable progress had been made in adopting N_2O_4 technology in more than 50 test systems of the Institute of Atomic Energy (Academy of Sciences of the Belorussian SSR), the State Institute of Applied Chemistry, and other institutes; two years experience had been gained in the use of a reactor loop system incorporating H_2O_4. In addition to fundamental scientific investigations relating to N_2O_4 and a large number of experiments, design work in connection with nuclear power stations based on dissociating gases had been carried out.

Cooperation between three Academies of Sciences (Ukrainian, Moldavian, and Belorussian) is actively developing in relation to the complex problem "Development of the scientific—technical foundations for the creation of fast-neutron reactors with dissociating coolants." The experimental base of the Institute of Atomic Energy (Academy of Sciences of the Belorussian SSR), the State Institute of Applied Chemistry, and other institutes have been supplied with new modern experimental equipment facilitating research and design on a high scientific level. A great achievement in the adoption of N_2O_4 technology is the creation and successful operation of an N_2O_4 reactor loop installation for more than 13,000 h.

The Conference noted the principal scientific achievements in the solution of the problems in hand. On the basis of complex research into the physicochemical and thermophysical properties of N_2O_4 up to 170 atm and 500°C, a reference book has been compiled for the thermophysical and physicochemical properties of N_2O_4. One great advance has been the scientific development of a new modified coolant "nitrine," based on N_2O_4 with certain technological additives, in the Institute of Atomic Energy (Academy of Sciences of the Belorussian SSR) and the State Institute of Applied Chemistry, and the execution of the first steps in studying the physicochemical and thermophysical properties of this coolant. Two- and three-dimensional mathematical models have been created for calculating the laminar and turbulent heat and mass transfer

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of N_2O_4 in a circular tube and an assembly of rods, allowing for the kinetic characteristics of the chemical reactions, and preliminary numerical results have been obtained. A number of experimental investigations have been carried out into single-tube and multiple-tube regenerators and condensers; the possibilities of intensifying heat transfer by a factor of two to five times in single-phase N_2O_4 flows, as well as during the condensation of N_2O_4 in a tube, have been established. Experimental work has been carried out on determining the flow coefficients and the velocity of sound in nozzles, and models of various stages in N_2O_4 gas turbines have been tested; new results have been obtained in tests carried out on bearings and sealing systems for N_2O_4 pumps and turbines. The IRT-R reactor has been reconstructed and its power increased to 4 MW; a gas loop installation has been constructed and successfully operated since 1973 — it has a thermal power of 100 kW and nuclear heating for the reactor; reserve (life) tests have been conducted in the latter on various construction materials, fuel composites, and fuel elements in an N_2O_4 atmosphere. Reserve tests have been carried out on construction materials subject to thermal cycling under stress and operating in the presence of phase transitions of the coolant; general laws have been established as to the interaction between these materials and N_2O_4, so providing a scientific basis for selecting and developing corrosion-resistant steels and alloys for N_2O_4 parts and components. Construction materials have been chosen, recommended, introduced, and successfully operated in active N_2O_4 test-beds.

Tests with experimental thermophysical test-beds in the reactor loop installation have revealed clear technological advantages of the new modified dissociating coolant (nitrine); considerable advances have been made in maintaining the required properties of the coolant by the periodic operation of a rectification column; chromatographic methods have been developed for operatively monitoring the composition of the dissociating coolant. Investigations have been conducted into the physical and thermophysical optimization of the parameters of gas-cooled fast reactors, together with comparative calculations for the physical characteristics of fast reactors respectively incorporating sodium, helium, and N_2O_4. The physical characteristics of the fast N_2O_4 reactor may well have a substantial effect in reducing the demand of developing nuclear power for natural uranium. As a result of the adoption of N_2O_4 at 150-170 atm and 450-500°C in the active test-beds, together with experimental investigations into the thermophysical characteristics of the coolant and tests on construction and fuel materials, it has been shown that it is technologically possible to obtain specific thermal intensities of 1000-1200 kW/liter in gas-cooled N_2O_4 fast reactors, together with high nuclear-fuel breeding characteristics (conversion ratio 1.5-1.6, T_2 = 4.5-5 years).

These investigations have facilitated the development of technical proposals for a single-circuit nuclear power station with an output of 1000-1200 MW (electrical), including a duly-justified choice of the parameters of the thermal system (150 atm 450°C) of the gas-cooled N_2O_4 reactor, the heat-transfer equipment, the gas turbine, pumps, and auxiliary systems; questions relating to the technology and safety of such power stations have also been elaborated.

The conference extensively discussed the projected characteristics of the BRIG-75 nuclear power station [maximum fast-neutron flux $7 \cdot 10^{15}$ neutrons/(cm^2 · sec), cassette dimensions, thermal power of the reactor 250 MW]. It was accepted that the parameters chosen were fairly representative for carrying out reserve tests on the fuel and fuel cassettes of a reactor of the BRGD-1000-1500 MW (electrical nuclear-power-station type, but it was suggested that the N_2O_4 gas turbine should be enlarged to 75-100 MW, while maintaining three or four reactor-cooling loop circuits (from considerations of safety in the single-circuit system of the nuclear power station).

It was noted in the plenary sessions, sections, and working groups of the Conference that the level of scientific-technical and design work, the degree of understanding in connection with the thermophysical and technological properties of N_2O_4, and the number of large thermal test-beds and reactor loop installations operating successfully over a number of years constituted a sound basis for the development of technical projects relating to the BRIG-75 nuclear power station. It was also noted that work on dissociating gases in the last few years had demonstrated a fruitful combination of fundamental investigations, scientific-technical development, and basically new planning decisions. The promising nature of continued work on nuclear power stations incorporating fast N_2O_4 reactors was acknowledged; so was the necessity of expanding scientific-research work in connection with a complex study of the thermophysical and technological properties of the modified dissociating coolant and the development of designs for a prospective nuclear power station with gas-cooled fast N_2O_4 reactors developing a power of 1200-1500 MW (electrical); recommendations were made that the creation of the experimental BRIG-75 nuclear power station should be expedited. The main lines of investigation for the tenth five-year plan were outlined and a date was set for the next Conference in 1978.

FIRST EUROPEAN NUCLEAR CONFERENCE

V. A. Sidorenko

The First European Nuclear Conference — the first conference of the European Nuclear Community — took place from April 21-25, 1975 in Paris. The motto of the conference was "The Maturity of Nuclear Power." About three thousand people from 47 countries participated in the conference. Papers were ordered as well as offered by the participants themselves. The ordered papers were heard at plenary and special sectional meetings, and the offered papers were heard at supplementary parallel sectional meetings.

The inaugural address at the opening of the conference was given by the French Prime Minister, Msr. J. Girac.

In the papers presented at the plenary sessions, the following problems were discussed: requirements on energy and resources; the role of nuclear power generation in the production of energy and achievements; disposition and the surrounding medium; safety and protection; provision of fuel (uranium, plutonium, thorium); irradiated fuel cycles; other applications (in addition to electric power generation); cost of construction of nuclear power stations; operating costs of nuclear power stations; heavy-water reactors, experience in their operation; light-water reactors, operating experience; gas-cooled reactors, operating experience; high-temperature reactors, operating experience; experience in the operation of the prototype fast reactors "Phoenix" and Dounraey; industrial and commercial application of fast reactors in various countries.

At the subsidiary sessions the following were considered: projects and designs; operating experience; construction of the active zone components; fuel handling; nuclear safety and protection; guarantee of quality and reliability; fuel manufacture; reprocessing, transportation and handling of nuclear wastes; low- and high-temperature technological heat, hydrodynamics and heat transfer. At the special sessions, with the invited speakers, the following were discussed; linearization, fusion; special applications (nuclear ships, explosions for peaceful purposes, etc); methods of enrichment; financing and insurance; the teaching and training of personnel.

In all, approximately 500 papers were presented to the conference, and the Organizing Committee selected about 350 papers. Before the start of the conference, a collection of comprehensive summaries of all papers was issued. The papers, or theses of the papers of the plenary sessions and complete texts of the papers at the subsidiary sessions, were not circulated, but in the autumn of 1975 it is proposed to issue the papers of the conference in two volumes, published by Pergamon Press.

The main interest of the assembly was concentrated on the plenary sessions. The subsidiary sessions simultaneously studied 11-13 different themes and it was possible for speakers to describe only briefly the data contained in the papers. The structure and content of the plenary sessions reveal the purposeful directivity of the conference and the nature of the principal problems of interest to specialists in nuclear generation in various and, first and foremost, in the European countries.

The greatest attention at the conference was on the successful scales of development of nuclear power generation and fuel resources, comparison of the different paths for the development of nuclear power generation and the role of breeder reactors, economics of nuclear power generation and its commercial supply, cooperation of different countries and standardization of technical resolutions and organization factors, safety of nuclear power generation, the effect on the surrounding medium, problems of fuel transportation, storage of waste, protection of stations from incidents and sabotage, problems of the fostering of favorable public opinion in relation to nuclear power generation.

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Mainly, practical workers and managers of nuclear power generation were assembled at the conference in order to demonstrate their achievements, to determine the most suitable forms of collaboration, and to conduct a broad propaganda in favor of nuclear power generation under conditions of extremely unfavorable public opinion and an increasing opposition in relation to it, in the majority of capitalist countries with the simultaneous objective necessity for increasing its role in industry, widening the sphere of application also by an actual increase of scales. The discussion of practical problems associated with the expansion of the scales of nuclear power generation were very specific and directed at a rational organization of production and operation (standardization, organization, and unification of plant maintenance and servicing, training of personnel, simplification of licensing procedures, etc.). In the report from the Director General of Euratom, Msr. F. Spaak, the tendencies are analyzed toward increasing the requirement for different energy resources in the world, increasing the energy requirements in different countries and regions of the world, stocks of the various energy sources, and a conclusion was drawn about the scales of development of nuclear power generation in the countries of the European Community by 1985. It had been assumed initially that the fraction of nuclear power generation amounted to 9%. At the present time, due to the changing conditions in the petroleum market, this figure has been reconsidered and increased to 13-16%. The Chairman of the French Commissariat a l'Energie Atomique (CEA), Msr. A. Giraud, based on the fact that under conditions in France, the derived cost of electric power from nuclear power stations will amount to 0.98 in 1977, and 0.51-0.55 in 1979-1980 of the cost of power obtained by the use of conventional fuel, considers various models of the development of nuclear power generation, the necessary resources of natural uranium, and the capacity of the concentration industry. Several variants are proposed in the report for the development of nuclear power generation, using only light-water reactors and different combinations of fast breeder reactors with light-water, high-temperature, and heavy-water reactors.

The report by A. Angelini (President of the Italian National Committee on Electric Power) was devoted to a consideration of all facets of the problem of nuclear fuel: resources, uranium enrichment, fuel element manufacture, the use of plutonium in thermal reactors, standardization of fuel, the role of fast breeder reactors in the change of requirement for uranium, in the uranium—thorium cycle. The total requirement on uranium by 1990 will increase to 1.5 to 2 million tons and by 2000 A.D. to 3.5 to 5 million tons. The estimated resources of uranium, with extraction costs of up to \$ 33/kg U_3O_8, are sufficient for confidently building up the capacity for nuclear power generation.

At the present time, the systematic searches for uranium are proceeding on a relatively small part of the earth's surface; further efforts, undoubtedly, will lead to the discovery of new deposits, although the cost of recovery of this uranium may be increased. In order that the existing resources should match the requirements, technical and financial efforts will be necessary. By 2000 A.D. several tens of millions of dollars will have to be injected into uranium extraction, although this amounts to a small percentage of the total expenditure on the development of nuclear power generation. The possible role of the uranium—thorium cycle in the next 15-20 years is estimated to be limited. With the assumed increase in cost of uranium, in the long-term plan an increased interest in thorium can be expected.

F. Mandel (Federal German Republic) analyzes the tendencies to change of the economic indexes of nuclear power stations, in the first place by the example of a nuclear power station in the Federal German Republic. By 1980 it is expected that the cost of electric power from the nuclear power station with a capacity of 1300 MW will be less by a factor of two than that from a solid-fueled thermal power station with a capacity of 730 MW. All the components responsible for the increased costs of installation of the nuclear power station are discussed. From 1968 to 1974 the cost of the various elements of the power station has increased by a factor of two to three. For example, from 1971 to 1974, the steam generator was dearer by 30%, the reactor hull by 60%, the containment shell by 30%, etc. An increase of the construction period for the nuclear power station by 15 months (from 63 to 78 months) increases the cost of electric power production in the nuclear power station by more than 10%, which corresponds to an increase of nearly 20% of the contract costs of the nuclear power station. A delay in the start of operation of the nuclear power station, with a capacity of 1300 MW, under conditions in the Federal German Republic, leads to an overexpenditure of 800,000 DM/day.

L. Minnick (USA) in his report on experience in operating light-water reactors in the USA draws attention to the fact that now, in the USA, permission has been granted for the operation of 51 power-generating water-reactors during which in the last eight years the average annual increase of nuclear power constituted 60%. This has created its difficulties. As a result, the average output per MW has been in operation (from the time of obtaining authorization) less than three years. This index reached the

maximum value (6 years) in 1967. Now in the USA it is commonly accepted that the maturity of nuclear power stations is achieved after the second fuel recharging. The average load factor of all stations from the start of their commercial operation amounts to 57% and for stations past the second recharging it is 74%. It is noted that during transition from the individual production of fuel elements and assemblies to mass production, their operating reliability has deteriorated; in particular, the contamination factor of the fuel has revealed foreign contaminants.

In other reports devoted to operating experience with light-water reactors, values are given for the load factors of nuclear power stations in countries (not in Europe) during their entire period of operation up to the end of 1974. Thus, for heavy-water reactors it is 69.7%; for gas-cooled reactors it is 65.1%; for pressurized light-water reactors it is 61% and for boiling light-water reactors it is 57.6%. Attention is drawn to the importance of solving the many operating problems: decontamination, reduction of the dosage received by personnel during maintenance, optimization of the principles of plant maintenance, provision of a maintenance basis, spare parts, instruction of operating personnel on training equipment. Various plant defects are discussed, including the various causes of defects in steam generators.

From the reports about operating experience with reactors of other types, it is observed that in the British heavy-water, channel, boiling reactor, six channels went out of action in the second year of operation. Start-up of the first commercial power station with this reactor, with a capacity of 660 MW, is expected by 1983.

In the channel reactor of the Pickering (Canada) nuclear power station, 17 tubes (channels) have been replaced. It is not expected in Canada that in the near future reactors will operate in a controlled load cycle. The Peach Bottom (USA) high-temperature reactor was shut down after 400 days of operation at a capacity of 30 MW. An increase of ^{137}Cs was observed. This reactor will not be operated in future because it is uneconomical. The Port St. Vrain high-temperature reactor, which for a long time after physical start-up did not come on power because of helium leakage through the circulator, has been running at 50% capacity. The feasibility of a further increase of capacity is being discussed.

In the report of the Director of the International Institute of Applied Systems Analysis, V. Heffel, the prospects of other applications of atomic energy are considered in addition to the production of electricity. The requirement for electric power as the ultimate form of energy amounted in 1970 to only 10% of the world's total energy requirement. It is expected that in 1985 this figure will amount to 14% and in 2000 A.D. to 21%.

At plenary sessions problems of the position of nuclear power stations, protection of the environment, and the safety and protection of nuclear power stations were discussed.

In addition to the conventional problems of safety (the reduction of thermal and radiation effects of nuclear pwer stations, choice of areas), great attention was paid to the protection of nuclear power stations from terrorists. The possibility of such a method of putting out of action a nuclear power station and the subsequent radiation effects on the surroundings and the population is being considered very seriously. It is probable that the acuteness of the posing of this problem is associated with the opposition to the development of nuclear power generation in many capitalist countries. As if to emphasize the seriousness of this problem, there was an incident (the explosion of two bombs) at one of the nuclear power stations in France several days after the conclusion of the conference.

The conference held undoubtedly was an important event in international cooperation in the field of nuclear power generation at the new stage of its development, which truly is characterized by the motto: "maturity of atomic energy."

INTERNATIONAL SYMPOSIUM ON THE RELIABILITY OF NUCLEAR POWER STATIONS

A. V. Karpov

An IAEA Symposium on the "Reliability of Nuclear Power Stations" was held from April 14-18, 1975 at Innsbruck (Austria). Two hundred and three specialists from 40 countries and eight international organizations participated in the work of the symposium. Forty-eight papers were read. The greater part of the papers (22 papers) was devoted to procedural problems of calculating the reliability and the application of reliability analysis during the planning of nuclear power station systems. The papers discussed methods of calculating the reliability characteristics in the absence of authentic statistical data, and the results of reliability analyses were presented, associated with ensuring the safety of nuclear power stations: emergency control systems, electric power supplies, emergency cooling, etc. In many countries, mathematical methods of reliability analysis of nuclear power stations are being developed and numerical programs for computers are being worked out. At present, there is still no unified and verified procedure for calculating the reliability of nuclear power stations. Because of the absence of authentic statistical data in relation to nuclear power station plant, when analyzing reliability it is recommended that statistical data be used for the corresponding plant in conventional power stations. In the planning of nuclear power stations not only qualitative but also quantitative analysis of all the systems and units of a nuclear power station is necessary, from the point of view of operating reliability. Such analysis permits the best alternatives for a scheme to be chosen and the critical components to be determined, whose replacement leads to a significant increase of the reliability index of the nuclear power station as a whole. Thus, the economic characteristics of a station can be improved considerably with the minimum of costs.

Statistical data were presented in 10 papers on the operation of plant and systems of nuclear power stations, the use of computers was discussed for the assembly, storage, and use of data concerning plant operation. It can be seen that in a number of countries, great attention is being paid to the collection of statistical data about the operation of nuclear power stations. In the USA, since the second quarter of 1974, reports are being issued quarterly with the data from 30 nuclear installations in the USA. Since 1970, in the IAEA, annual reports have been issued about the experience in operating nuclear power stations in the member countries of the IAEA.

In the IAEA report presented at the symposium, an analysis was given of the operation of 107 nuclear power stations in 15 countries. It showed that in 1973 the average load factor of a nuclear power station amounted to 62.1% and the operating factor at power was 72.9%. If the KKN* reactor is excluded, which was shut down in 1975 for economic reasons, the experimental power stations with a capacity of less than 100 MW(e) and also nuclear power stations which started up only in 1973, then the load factor of the remaining 74 reactors is 64.7% and the operating factor at power is 75.6%. It is noted that these figures are still far from the design load factors of 75-80%. Comparison of nuclear power stations with conventional power stations shows that the characteristics of the operating reliability of large-scale thermal power stations, with a capacity in excess of 600 MW(e), are one order or slightly better than the characteristics of nuclear power stations of the same capacity.

Problems of monitoring and servicing of systems and components of nuclear power stations during operation were discussed; first and foremost were methods of nondestructive monitoring of the metal of reactor hulls, operating under very high stress conditions. Work in this direction is starting to develop.

* Pressurized Boiling Channel Reactor.

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There have been some successes in France and the Federal German Republic in the construction of equipment for metal monitoring, based on the application of ultrasonics and acoustics. At the present time, defects with dimensions of one to two mm in steel sheets with a thickness of up to 350 mm can be determined. As regards the systematic monitoring of reactor hulls during operation, preparation for this work is now going ahead. Partial checking of the condition of a reactor hull has been carried out once in the Federal German Republic. An old defect of the hull was detected by means of ultrasonics which, as was shown, had not changed.

In the papers and discussions, the necessity was mentioned for increasing the sensitivity of monitoring equipment capable of operating at a high temperature and in intense radiation. Problems of optimization of technical servicing of systems and plants of nuclear power stations during operation were considered. Thus, it was noted that due to improvement of technical servicing of systems in the Hanford reactor, the readiness factor of the nuclear power station was increased significantly.

During discussions for ensuring the quality, reliability of operation and efficiency of nuclear power stations, the necessity for processing the complex system of requirements for nuclear power station projects was emphasized. It was found that the operating reliability of nuclear power stations depends on both the assurance of safety requirements and on a sufficiently high readiness factor of the nuclear power station. Economic optimization of nuclear power stations, from the point of view of safety and readiness factor, is essential. The readiness factor of nuclear power stations should be 75-80%. In 1974, in the USA, it amounted to 68-73% on nuclear power stations in operation.

Because of the initial mass construction of nuclear power stations and the bringing in to the work of a large number of organizations and people, the decisive factor is the operating reliability of the nuclear power station. A broad training for builders, designers, and operating personnel of the nuclear power station is necessary in methods of calculating the reliability of plant and the optimization of technical servicing.

EIGHTH INTERNATIONAL CONFERENCE ON LASER PLASMA FUSION

V. M. Korzhavin

The conference, held in Poland, ran from May 19 to May 23, 1975. It was organized by the Polish Academy of Sciences with the participation of some 100 scientists from eight nations. The largest delegations were from Poland, the USSR, the United States, France, and the Federal Republic of Germany. A total of 45 reports were read. Abstracts of the reports were printed in advance and distributed at registration. The conference proceedings will not be published. Arrangements were made for the participants to visit the Institute of Nuclear Research at Swierk and the Institute of Plasma Physics and Laser Microfusion at Warsaw and become acquainted with their work on quantum electronics, plasma physics, and controlled thermonuclear fusion.

A broad group of problems was considered: parameters and design principles of high-power laser systems for controlled thermonuclear fusion, the results of experimental and theoretical research on the interaction of powerful laser radiation with various targets, laser plasma diagnostics, and hybrid systems.

J. F. Holtzrichter (Livermore, USA) discussed the analysis of the properties of large neodymium glass laser systems. The main problem in raising the beam energy is the nonlinear interaction of laser light with the medium traversed by the beam, which results in wavefront distortion and impairment of focusing. In particular, when a certain threshold (~200 TW) is exceeded, the focal spot multiplies in diameter during a pulse (100 psec) or splits into several spots. The current Janus experiment (neodymium glass, 0.4 TW) at the Livermore Laboratory may serve as a standard component for the construction of laser systems of up to 20 TW. New materials based on BaF and phosphate glasses having a low nonlinear constant are being investigated at the same time. I. C. Guyot (Laboratories de Marcoussis, France) dealt with similar question; he investigated nonlinear properties of rod and disc specimens of neodymium glass. At energy flux densities above 0.5 J/cm^2 and 100 psec pulse length self-focusing of the radiation set in and the specimens broke down in both parallel and divergent beams.

The program of laser-system design for controlled fusion at Los Alamos (USA) was presented by R. I. Morse. Neodymium and gas lasers (CO_2, HF) are receiving equal attention. The Laboratory now has in operation a 500 J, 300 psec neodymium laser, a 1500 J, 1 nsec CO_2 laser, and a 2500 J, 35 nsec HF laser. It is expected that within the next two years 10^4 J, 1 nsec CO_2 and HF lasers will be built.

Problems in the development of high-power iodine lasers and approaches to their solution were discussed by K. Witte (Garching, Federal Republic of Germany). An iodine laser using a mixture of C_3F_7I and Ar has been built with flash-lamp pumping at energies of several hundred Joules and 1 msec pulse length.

Of special interest are experiments on the compression of spherical targets by high-power laser radiation, carried out at the Livermore and Los Alamos Laboratories and by KMS Fusion (USA). The targets were glass spheres 40-100 μ in diameter with 1 μ wall thickness, filled by a D — T mixture at a pressure of 10-100 atm. The energy and pulse duration were 20-100 J and 0.1-0.3. J. F. Holtzrichter (Livermore) and D. V. Giovannelli (Los Alamos) reported on the results of such experiments. Targets of the "sphere-on-disc" type were investigated with illumination from one side (neutron yield 10^4) and "sphere-in-disc" targets were studied under illumination from two sides (neutron yield 10^6). It was also mentioned that with special targets a neutron yield of 10^7 per shot had been recorded.

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The KMS Fusion company (P. Hammerling) is continuing the compression experiments first reported to the Fifth International Conference on Controlled Thermonuclear Fusion (Tokyo, 1974). Beam parameters and focusing conditions are being improved, and the dependence of neutron yield on target dimensions is being studied.

Yu. Merkul'ev (Lebedev Institute) reported on the technology of making, selecting, and monitoring spherical pellets of polystyrene and glass.

A large group of reports dealt with experiments on the interaction of high-power laser radiation with plasma from various targets. The results are often contradictory owing to difference in experimental design, laser-system characteristics, and the dynamics of the dispersing plasma. Particular attention was given to the way in which the coefficients of reflection and absorption depend on the intensity of the incident radiation. In their reports, R. Sigel (Garching, Federal Republic of Germany), E. B. Goldman (Rochester), and M. Dekruaze (Centre d'etudes de Limeil, France) concluded that the absorption of incident energy is classical in character. The reflection coefficient was 60-90% in these studies. Anomalous absorption was detected and investigated in experiments reported by D. V. Giovanelli (Los Alamos, USA), A. S. Shikanov (Lebedev Institute, USSR), V. V. Aleksandrov (Atomic Energy Institute, USSR), P. E. Dyer (University of Hull, UK), and E. Fabre (France). At fairly high incident-radiation power densities the reflection coefficient drops to 3-5%. These results are accounted for by the development of decay instabilities.

There were many detailed studies of x-ray and ion spectra, the intensity and polarization of harmonics of the fundamental, and other parameters. Numerical modeling of interaction processes for laser radiation and matter was considered from the viewpoint of the controlled-fusion problem. Hybrid systems using lasers were discussed. Thus, the Institute of Plasma Physics and Laser Microfusion (Poland) has carried out an experiment on heating of a dense plasma focus (capacitor-bank energy of 150 kJ) by radiation from a CO_2 laser (200 J, 30 nsec). Total neutron yield was found to increase by two or three times [from 10^{10} to $(2-3) \cdot 10^{10}$ neutrons per discharge]. Work is also being carried out on compression of specimens by an explosive detonation wave (50-60-fold compression) followed by heating with laser radiation (200 J). The possibility of generating ultrastrong magnetic fields (~1 MG) using explosive generators and confining a laser plasma by such fields is being studied. The Lebedev Institute is now preparing an experiment involving the use of a neodymium laser ($2 \cdot 10^3$ J) to heat a dense plasma focus.

The conference represents a step along the path toward realization of controlled thermonuclear fusion. The next conference will be held in Paris during November of 1976.

EIGHTH INTERNATIONAL CONFERENCE ON RADIOCHEMISTRY

V. V. Gromov

The Conference took place on April 28 to May 3, 1975 in Mariansk Lazne (Czechoslovakia). Its principal organizers were the Czechoslovakian Chemical Society, the Commission on Atomic Energy, and the Technical Prague University (Faculty of Nuclear Chemistry). One hundred and seventy four delegates from 16 countries attended. About 100 papers were presented, of which 21 papers were by Soviet scientists.

The subjects of the conference were extremely varied. This was reflected in the number and designations of the sections: ion exchange; radioactive indicators; stable isotopes and isotopic exchange; catalysis and diffusion; applications of radiochemistry and labelled compounds.

I. Zvara (Joint Institute of Nuclear Research, Dubna) explained the principles of application of thermochromatography for studying the super-heavy elements and cited the example of the estimation of the heat of sublimation of element 105. D. Moltsan reproted on the work being carried out in the Federal German Republic on the synthesis of the super-heavy elements in the region of predicted stability ($Z \geq 110$).

Certain technical problems were considered, which originate during the construction of a thermonuclear reactor and which can be resolved by radiochemical methods. S. Queim spoke about the investigations being undertaken in the Federal German Republic associated with the construction of a thermonuclear reactor. Various types of "neutronic" nuclear reactions are being studied and it was here that the reactions (n, H^3) and (n, He^3) were first investigated. A great deal of attention was paid to the formation of tritium by the irradiation of lithium by neutrons, which occurs in a thermonuclear breeder reactor.

The possibilities of using radiochemical methods for studying the kinetics of exchange reactions in complexes were demonstrated in the paper by H. Elias (Federal German Republic). The author has investigated the exchange of copper isotopes in chelates (about 100 compounds) and he determined the dependence of the rate of exchange on the structure of the ligands.

In the field of ion exchange, together with the usual work on the determination of the equilibrium coefficients of distribution, in a number of reports the results were given of investigations of the kinetic special features of sorption in the presence of complex-forming ligands (R. Dibozinskii, Poland; V. Koprda, Czechoslovakia) by the example of the separation of lanthanum and praesodymium.

The papers of the Czech scientists (S. Konechnyi and R. Tsaletka) merit attention on the synthesis by the method of the sol-gel process of a number of inorganic sorbents, possessing an increased mechanical stability and having the form of small spheres of finite size. For example, by treating a gel of titanium hydroxide with iron hexaferrocyanate, a cation-exchange granulated sorbent was formed with a capacity of ~2.5 mg · equ/g and with a ratio of titanium to $Fe(CN)_6$ from 0.15 to 0.71. This sorbent was used for the recovery of cesium from one-molar solutions of nitric and hydrochloric acid.

Several reports were devoted to the study of the state of various elements present in aqueous solutions in microconcentrations. The main attention here was paid to an investigation of the processes of equilibrium distribution of radioactive isotopes between different physicochemical forms of the corresponding stable elements (Ba, Fe, Se, Zn, Mn, Cr, Be, Ni, Co, etc.) in distilled, river, and also sea water (P. Benes, Czechoslovakia). For the rapid unification of states of a radioactive indicator and a defined element found in sea water, it is proposed to use the bioassimilation process of radionuclides of plankton organisms (multiple repetition of the adoption — development cycle). After 10-15 days this permitted the achievement of identicity of forms of radioactive and stable nuclides (V. V. Gromov).

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The formation of mononuclear and polynuclear forms (mainly products of hydrolysis) of such elements as Pu(IV), U(IV), U(VI), Fe(III), Cr(III), Y(III), and Eu(III) was discussed comprehensively, and also the conditions of formation of colloids of these elements in the range of microconcentrations. It was shown that at concentrations of less than 10^{-7}, true colloids of the elements studied may or may not be formed, because of adsorption of ion monomer and polymer forms on extraneous colloid contaminants, always present in the solutions (Yu. P. Davydov, USSR, F. Ishikawa, Japan).

The papers on the study of the forms of the volatile compounds of radioisotopes during chlorination or fluorination by different agents of uranium irradiated with neutrons attracted great attention (P. Hoffmann and I. Rudolph, Federal German Republic). The authors used high-temperature gas chromatography. The oxyhalides of Mo, Ru, Ta, I, Zr, and Nb were studied and the thermodynamic constants characterizing the compounds studied were calculated.

Some reports were devoted to the structure and physicochemical properties of deuterated compounds, the use of hydrogen and deuterium, and also certain aspects of the technique of investigating the isotopic exchange of hydrogen. For example, when investigating the isotopic exchange of hydrogen in the gas and liquid phases by the methods of kinetic infrared spectroscopy a time resolution of ~1 msec was achieved successfully, which exceeds the minimum recording times of isotopic exchange processes, using the conventional methods, by an order of two to three (S. F. Burenko, USSR). The use of ^{204}Tl was considered for investigating exchange reactions of Tl(I) and Tl(III) compounds. The mechanism of this exchange is suggested and theoretically substantiated through the formation of intermediate Tl(II) (R. Dodson, USA; I. Staryi, (Czechoslovakia).

In the field of extraction, special attention was paid to the determination of the composition of the extracted compounds and to the quantitative description of the extraction equilibria. The papers by I. Staryi et al. (Czechoslovakia) are interesting from this point of view; the formation is shown of complexes with mixed anions during the extraction of the halides of As(III) and Sb(III) in systems with diethyldithiocarbominates.

In the joint paper by P. Muller (DDR) and A. I. Khol'kin (USSR), systems are considered with monocarbonic acids, for which the formation of polynuclear complexes in the organic phase is characteristic. The effect of the nature of diluents is analyzed in detail by the example of the extraction of mercury from chlorine solutions, and a method of describing extraction systems with stepwise complex-formation in the aqueous phase is given (E. Hochfeld, Sweden).

Investigations of the extraction processes for the analytical determination of traces of metals are interesting. The feasibilities of using kinetic factors in the substoichiometric determination of traces of the rare-earth elements were discussed, including the different ion forms of these metals in aqueous solutions.

The main part of the papers on radiation chemistry was presented by the Soviet scientists. Thus, E. I. Saunin (USSR), in a paper on the study of the mechanism of radioluminescence of Mn^{+2} in radioactive $Ca^{*}SO_4$, showed experimentally that ion-radicals of SO_4^- and SO_3^-, formed by autoradiolysis of $Ca^{*}SO_4$, participate in luminescence processes. L. I. Barsova gave an account of the principles of the radiation method of analysis of oxygen-containing anion impurities in difficultly-soluble salts.

The extraprogram report of V. M. Byakov (USSR), "Investigation of the Mechanism of Radiation —Chemical Reactions Using the Positron Method," created great interest. Evidence was presented of the participation of a "dry" electron in chemical reactions in aqueous and alcohol media. In the reports of I. Kro (Poland), data were given on a study of the post-radiation destruction of electrons in polar systems, due to tunnel reactions of dry electrons with acceptors.

In a paper of practical importance related with diffusion (M. Neuman, DDR), the permeability of polymer materials (various foils, films, rubber, etc.) were studied in relation to radioactive isotopes (^{125}I, ^{35}S, ^{32}P, ^{137}Cs, ^{141}Ce, and ^{89}Sr). The diffusion coefficients of these isotopes were determined.

Problems of radiation heterogeneous catalysis also were represented. Thus, G. N. Pirogova (USSR) discussed the results of the use of technetium as a catalyst in dehydrogenation reactions of alcohols and hydrocarbons. The high activity and selectivity of technetium catalysts allow their use for the production of aldehydes and ketones.

The paper by A. F. Kuzina (USSR), on the electrochemical reduction of Tc(VII) and the production of different articles of metallic technetium and the properties of its compounds created great interest.

The paper by Ch. Neskovick (France), on the use of radioactive isotopes for studying the sorption capability of nickel ferrocyanide and the development of a procedure for separating certain pairs of elements on this sorbent, should be mentioned.

A number of reports referred to the development of procedures for the radiometric monitoring of the sorption of high-polar, low-volatile compounds in a chromatographic column during their separation (fatty acids, formic and acetic acids, etc.).

Some papers were devoted to the investigation of biological systems. Glycine, labelled with ^{14}C, and various oligopeptides were used for studying the mechanism of their interaction with nucleonic acids (K. Shtamyok, Federal German Republic).

The excellent organization, the creative environment of the conference, the supporting and useful discussions all contributed to the mutual enrichment of the participants with scientific ideas.

FIRST ALL-UNION CONFERENCE ON "PULSATION APPARATUS IN THE POPULAR ECONOMY OF THE USSR"

E. I. Zakharov

The Conference took place on May 26-30, 1975, in the "Atomic Energy" pavilion of the All-Union Center for the Popular Economy, having been organized on the initiative of the State Commission of Inorganic Fuel and the State Commission for the use of Atomic Energy of the USSR. Taking part in the work of the Conference were more than 260 specialists from the various Ministries as well as representatives of 150 scientific-research, engineering design, and industrial undertakings. Some 59 contributions relating to developments in pulsation apparatus and to the results of introducing the latter were received and discussed. The novelty and the promising nature of pulsation apparatus for a number of chemicotechnological processes were emphasized, as well as its considerable advantages over many types of existing apparatus, especially in processes associated with the use of toxic and corrosive media, and in setting up continuously-acting, high-efficiency installations. The leading role of the All-Union Scientific-Research Institute of Inorganic Materials in initiating the organization of research and creating the basic designs of pulsation systems was fully acknowledged.

The contributions may be divided into three groups: theoretical investigations into the influence of reciprocating vibrations (pulsations) in intensifying many technological processes (mass transfer, drying, evaporation, etc.), together with the calculation, simulation, and optimization of pulsation devices and systems (14 papers); the results of experimental and experimental-industrial tests on pulsation equipment in various chemicotechnological processes — extraction, sorption, lixiviation, synthesis, neutralization, gas purification, etc. (35 papers); experience in the industrial use of pulsation equipment in the undertakings of various Ministries (10 papers).

Among the papers of the first group, special attention should be paid to the work of the Institute of Inorganic Materials in studying the structure of flows in large-scale pulsation apparatus (diameter 1.5-3.4 m) of the column variety, and also to work carried out by the same Institute, together with the Bryansk Technological Institute, on the use of methods of classical hydrodynamics in studying the structure of flows and energy optimization, and also on the matching of pulsation systems. These investigations are of great practical value in laying the foundations for the calculation and simulation of pulsating equipment.

The large number of problems discussed in the second group of papers requires careful analysis and systematization; however, it may immediately be said that pulsations intensify many technological processes (they increase productivity, reduce the bulk of the equipment, and so on). Thus, according to the Institute of Inorganic Materials' data and the results of a number of other Institutes, pulsation equipment should greatly aid the development of continuous processes in the synthesis and polymerization of various materials (to replace existing periodic processes), and this should yield a significant economic effect.

Undertakings concerned with nonferrous metallurgy have found that pulsating sorption columns and apparatus used for washing, neutralization, and other operations have a two- or three-times greater efficiency than existing constructions, and greatly reduce both the load and the consumption of reagents. Some interesting communications were presented by representatives of the Chirchik Branch of the State Institute of the Nitrogen Industry, the Pyshminsk Copper-Electrolytic Combine, and others, on the use of pulsation equipment involving the participation of the gas phase in deposition and dissolution processes; this modification may reduce the volume of equipment required by a factor of five to ten times. Also

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deserving of attention are certain papers describing processes involved in the separation and washing of solid phases, the processing of metal powders, and the use of pulsations for intensifying heat transfer, drying, and crystallization.

Experience in the industrial use of extractors (mixer-settlers and columns), sorption columns, and large-scale reactors involved in mixing processes (maximum reactor volume 20 m^3) and lixiviation was described, as well as experience gained in the use of pulsating filter-coagulators and other equipment (All-Union Scientific-Research Institute of Inorganic Materials and others).

In practically all undertakings the possibility of reliably simulating industrial equipment in laboratory models has been fully established, together with the considerable economic advantage (10-50 thousand rubles for each item of equipment introduced) gained by the industrial use of such apparatus. According to the Institute of Inorganic Materials, the economic effect is at present 4.5 rubles for each ruble spent in research. In the tenth five-year plan the expected economic effect from the introduction of pulsation apparatus, by virtue of the present intentions of the Ministries of the Chemical Industry and Nonferrous Metallurgy alone, amounts to some 16 million rubles.

The Conference also made certain serious comments. Thus, the absence of specialized design organizations for the manufacturers of routine apparatus was noted; so was the necessity of standardizing existing industrial equipment, establishing a common coordinated working plan, and so forth.

Summaries of the papers have already been published [see Collection of Summarized Contributions to the First All-Union Conference on Pulsation Apparatus in the Popular Economy [in Russian], Moscow (1975)].

EXHIBITIONS

THE SPECIALIZED EXHIBITION "INTERATOMINSTRUMENT"

K. A. Nekrasov

For the first time in the USSR a specialized exhibition will be opened of the International Economic Group on Nuclear Instrument Design, "Interatominstrument," which will be taking place within the framework of the international exhibition "Physics-75" from November 25 to December 3, 1975 in Moscow.

The main importance of the exhibition is to demonstrate the increasing potential of the economic member organizations of Interatominstrument (IAI) in the development of nuclear instrument design. The exhibition should show the integration in the field of nuclear techniques within the framework of Interatominstrument, the assortment of manufactured articles provided by the members of the Group; the possibilities of the widespread use of efficient nuclear methods and instruments in science, industry, and medicine.

A seminar will take place during the exhibition, at which Soviet specialists will be able to familiarize themselves in more detail with the results of the application of instruments and equipment in industry, science, and medicine and also with the programmed activity of the member-organization of Interatominstrument which, as is well known, are a Society with limited responsibility "Elektroimpeks" (Sofia), the Amalgamated economic agency "Yadernaya Tekhnika" (Pleven), the combine "Gamma" (Budapest), EMG — electronic measuring instruments factory (Budapest), Hungarian external trade agency for instrument design products "Metrimpeks" (Budapest), Trade Agency for instruments and organization techniques "Migert" (Budapest), Finance and Economics Office RFT Mosselektronik "Otto Shen" (Dresden), Agency for external trade "Elektrotekhnik Eksport — Import" (German Democratic Republic), Agency for external and internal trade "Izokommerts" (German Democratic Republic), amalgamated factories for nuclear instrument design "Polon" (Warsaw), Agency for external trade "Metroneks" (Warsaw), All-Union group "Izotop" (Moscow), All-Union export — import group "Tekhsnabeksport" (Moscow), Agency for electronics and low-precision techniques "Tesla" (Prague) and External trade group "Kovo" (Prague).

The group "Interatominstrument" undertakes scientific research, experimental, planning-design, and production work, organizes scientific — technical production and commercial cooperation between economic organizations of the member countries of IAI, and also assists trade expansion with other countries. The activity of the group extends to manufactured products of nuclear instrument design and radiation-shielding techniques, equipment and apparatus with isotope radiation sources for irradiation and radioscopy, special devices and plant for nuclear technology and isotope sources.

It can be seen from the exhibition data that, as a result of the activities of IAI, production specialization of a considerable part of the products has been prepared, work has been carried out on the coordination of scientific — technical researches in the field of semiconductor detectors, certain types of nuclear — medical instruments, instruments for monitoring contamination of the surrounding medium, etc. Coordination in the field of external trade enabled the rate of increase of commodity turnaround between member organizations of IAI to be doubled, and in 1980 it should lead to an increase again by a factor of two, of the volumes of mutual deliveries. There exists technical servicing of nuclear equipment by the formation of servicing subsidiaries of IAI in Bulgaria, German Democratic Republic, and Poland. Work is currently under way on the formation of a servicing branch in the USSR. Information on the products of nuclear techniques is being extended by holding combined exhibitions and seminars, and by the publication of an information bulletin, "Novosti IAI."

The arrangement of the exhibits at the exhibition provides for seven sections: the first section is ionizing radiation detectors; dosimeters; radiometers, nuclear radiation spectrometers and nuclear-physical instruments for reactors; second section — nuclear-physical equipment; third section — radioisotope

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instruments; fourth section — nuclear instruments for medicine; fifth section — furniture for working with radioactive substances; sixth section — gamma-defectoscopy equipment; isotopic gamma facilities; a facility for the nuclear-physical analysis of a substance; radioisotope sources for electric power; seventh section — isotope nuclear radiation sources.

"Metroneks," "Kovo," "Mosselektronik," "Elektroimpeks," "Metrimpeks," "Izokommerts," "Gamma," and V/O Izotop participated in the provision of these manufactured products.

BOOK REVIEWS

V. V. Bugrovskii, V. P. Zhukov,
S. S. Preobrazhenskii, et al.

DYNAMICS AND CONTROL WITH A NUCLEAR ROCKET ENGINE*

Reviewed by V. M. Mikhailov

The book being reviewed is the first monograph on the dynamics and methods of control with nuclear rocket engines which for the last 20 years have attracted great attention everywhere from specialists in space technology. In the book by R. Bassard and R. Delaware (1967) "Nuclear Engines for Aircraft and Rockets," problems of dynamics and control were almost not considered.

In this monograph, the most promising types of nuclear rocket engines are considered. In the first chapter, a detailed classification is given of schemes for these engines and illustrations are provided. In chapters 2 and 4, problems of the mathematical description and development of control systems are studied in detail for an engine with a solid fissile substance. Problems of the mathematical description of non-steady-state processes are considered in quite some detail in the fifth chapter, for nuclear rocket engines with a gaseous fissile substance. The authors, from the basic mechanisms of neutron physics, heat transfer, gas- and magnetohydrodynamics, pass on to numerical relations which serve as the basis for studying the dynamics of nuclear rocket engines by means of computers, and for analytical investigations. In order to solve the problems arising from the mathematical description (basis of a simplified description, development of previously unknown mechanisms), unique methods of investigation have been developed.

A description of the dynamic properties of such important components of a nuclear rocket engine as the fuel element — coolant, fuel element — reactor moderator systems and other heat exchange equipment is given, and which takes account of the one-dimensional distributivity of the parameters. The expressions obtained by an analytical method for the transfer functions of these systems are illustrated by specific numerical calculations of the frequency characteristics. They are compared with the results of a zero-dimensional description (in lumped parameters) of these same systems. The comparison shows the admissibility in the majority of cases of substitution of the distributed description by the zero-dimensional description (at least, for sufficiently smooth effects).

Investigation of the dynamics of branched hydraulic schemes for rocket engines is fraught with great difficulties. Therefore, there is interest in the appearance of characteristic special features of these circuits and the development of methods based on this for their investigation, which are associated with lesser difficulties.

Two methods for investigating complex gas circuits are given in the book. The first method is extremely effective and reduces considerably the difficulties of the investigation and gives a total solution to the problem of describing quasi-steady-state processes in one class of complex gas circuits, with critical efflux at the outlet. The second method has a greater generality, and its application consists in that, in consequence of modification of the equations and the use of a contoured mathematical description of the circuits, the difficulties of its mathematical description and of the investigation are successfully simplified.

Useful results are given in the section on the mathematical description of neutron kinetics for the reactor and the circulating fuel. Based on the studies of a dynamic model of these processes and the effectiveness of various perturbations, a working dynamic model is obtained for this reactor.

* B. N. Petrov (editor), Atomizdat, Moscow (1974).

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Problems of the assignment of boundary conditions and the numerical calculation for certain magnetohydrodynamic flow schemes are interesting.

In order to justify the simplification of the mathematical description of gas flow dynamics associated with nonaccounting for sonic oscillations, a study of the asymptotic properties of gas flow at low Mach numbers is carried out in the appendices. The investigation shows that convergence of flow reactions by external effects (when the Mach number tends to zero) occurs only for quite smooth effects. Therefore, when estimating the errors introduced by the nonaccounting of sonic oscillations, the nature of the external effects on the system being studied must be taken into account.

Chapter 4 is devoted to problems of dynamics and control with nuclear rocket engines. Methods are described in it for analyzing the dynamic properties of nuclear rocket engines as a whole, under nominal and launch conditions, and also certain possible formulations and solutions of problems of control system synthesis under these conditions. These investigations utilize the present-day achievements of control theory and may be applied not only to nuclear rocket engines, but also to a wider class of vehicles.

In this same chapter, modifications are given for determining Lyapunov stability as applicable to systems with distributed parameters and a procedure for synthesizing a linear control system, possessing specified reactions to external effects. For a chosen set of control organs, measured coordinates of the vehicle and external effects, the necessary and sufficient condition is given for solvability of the synthesis, and a general form is derived for the control equations giving the solution of the problem of synthesis. The problem of providing the specified static relations for a closed system is considered separately, where the necessary quantity of integrated units in the controller is determined.

For control of nuclear rocket engines in the launch cycle, the formulation and solution of the problem of calculating the optimum control is derived, taking account of uncontrolled perturbations. This is new optimization formulation and it is particularly timely for reactor control, because small deviations of the law of control may cause a sharp breakdown of the constraints on the operating cycle of the reactor.

However, the application of the theoretical data of Chapter 4 to the problem of control of nuclear rocket engines has not been sufficiently completely presented in the book.

On the whole, the book reviewed is undoubtedly useful for quite a wide circle of specialists in dynamics and the control of nuclear power-generating plants, and also specialists in neighboring fields of technology. The book has been issued with a small circulation and has gone out of print rapidly. It will be advantageous, therefore, to republish it, taking account of the remarks mentioned above.

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