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SOVIET ATOMIC ENERGY

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TECHNOECONOMIC ASPECTS OF THE REALIZATION OF CENTRALIZED HEAT SUPPLY FROM ATOMIC BOILER UNITS

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It is now quite clear that it is desirable and necessary to widen the sphere of application of nuclear energy resources, with primary emphasis being given to the use of nuclear reactors for residential community and industrial heat supply. This is due to the fact that out of the total quantity of fuel used, 30-40% of fuel resources are devoted to the needs of heat supply, i.e., ~ 50% more than is used for producing electrical power. While electrical power is generated mainly by burning low-grade coal, the requirements of heat supply are met mainly by high-quality gas fuel.

Experience in operating atomic central heating-and-power plants (ACHPP) (Bilibinsk in the USSR and "Agesta" in Sweden) shows without a doubt that it is technically possible to construct radiation-safe and operationally reliable nuclear sources of heat supply. Studies have now been completed [1-3] which have determined the principles of organization and the conditions needed to ensure economic efficiency of centralized heat supply on the basis of ACHPP. These questions have not yet been nearly as thoroughly analyzed in their application to atomic boiler units (ABU).

Principles of the Organization and Application of Organic-Fuel Boiler Units. In spite of the well-known advantages to be gained by combining the production of heat and electrical energy with central heating-and-power plants (CHPP) (these are becoming more widely used with the improvement of the central-heating energy cycle), heat supply from boiler units using organic fuel has been widely developed because of several reasons. The main reason is that raising the coefficient of centralization of the heat supply is made possible by the economic efficiency gained when heat from a common source is supplied to relatively small users, for whom heat supply by CHPP is economically undesirable or impossible because of:

a low level of the total heat loads of individual cities, residential communities, and industrial enterprises, resulting in a reduction of the unit output of CHPP energy units and the deterioration of their technico-economic indicators;

a low territorial density of heat usage in a given region, resulting in highly capital-intensive heat systems with large amounts of waste when incorporation of users is pushed to the level of total heat load appropriate to CHPP operation;

the impossibility of locating CHPP's in a given region because water resources are lacking, or there are no areas suitable for construction (e.g., in the center of large cities).

Table 1 gives data which illustrate the importance of a centralized heat supply for relatively small heat users relative to the fuel and energy economy of our country. Table 1 shows the structure of the concentration of heat loads and the shares of various levels of concentration in the total rate of fuel consumption of the country [4].

The use of regional boiler units (RBU) as a means of supply is economically efficient for small heat users due to the specific characteristics of the former as sources of centralized heat supply:

the relative simplicity of their main equipment compared with CHPP and their resulting far lower specific cost for small attached heat loads (the capital component of the cost and of the specific reduced expenditures when heat is produced by boiler units is 5-15%);

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TABLE 1. Illustrative Characteristics of Heat Load Concentration in the USSR

Mean heat loads of cities, Gcal/h	1970			Growth since 1971			1980		
	No. of cities	total heat load		No. of cities	total heat load		No. of cities	total heat load	
		Tcal/h	%		Tcal/h	%		Tcal/h	%
> 10	2	40	14	1	30	10	3	70	12
5—10	3	20	7	14	95	32	17	115	20
3—5	17	65	23	12	45	15	29	110	19
1—3	66	105	37	57	105	35	123	210	35
0.5—1	81	55	19	31	25	8	112	80	14
Subtotal	169	285	100	115	300	100	284	585	100
< 0.5	5335	190	40 *	511	50	14 *	5846	140	29 *
Total	5504	475	100	626	350	100	6130	825	100

*Percent of the total.

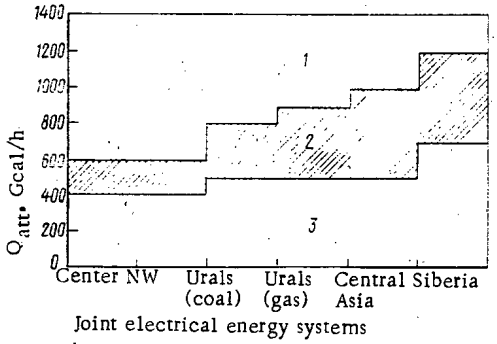

Fig. 1. Zones of economic efficiency of a CHPP (1); equal efficiency (2); the efficiency of the separate scheme (KES + RBU) (3).

according to a graph of heat loads, the possibility of operation with extreme discharge or total shutdown for prolonged nonheating period (up to 3000-4000 h/yr);

the lack of a need for circulating water and the possibility of placing them in the center of large population masses (when using petroleum residue fuel gas);

the simplicity of operation, the small number of servicing personnel, and the relatively low level of qualifications required of them.

Technoeconomic studies have shown that the decisive factor involved in choosing the type of source of centralized heat supply using organic fuel (CHPP or boiler units) is the level of concentration of the heat loads. Naturally, this level will be quantitatively different in the various regions of the country (Fig. 1) which have different conditions of heat supply [5]. In terms of the level of concentration of heat loads and the required (by the necessary conditions for reliable supply) number of units ($n_u = 4-6$ and more), the unit output of regional boiler units is from 30 to 150-180 Gcal/h.

The Economic Prerequisites for the Establishment of Atomic Boiler Units. At higher specific capital investments, atomic boiler units (ABU) can be competitive because of their lower fuel components of reduced expenditures on the production of heat compared with boiler units using organic fuel. The maximum permissible investment on ABU depends on the cost of organic fuel, the fuel component of the reduced expenditures, and the placement conditions. The data shown in Fig. 2 characterize the conditions for replacing ABU and CHPP using organic fuel. For typical price levels of organic fuel during the period prior to 1990 (30-35 rubles/ton ideal fuel) and realistic values of the fuel component of the reduced expenditures, unit investments on ABU can be 2.5-5.5 times as large as those for boiler units using organic fuel.

If ABU are oriented toward additional heat loads above 400-600 Gcal/h, the economic conditions for their construction must be determined by making a comparison with CHPP using organic fuel. Because of the higher efficiency of the latter, the maximum possible unit investments on ABU are lowered. If we refer these to the unit investments on boiler units using organic fuel, the maximum allowable excess of the unit investments on ABU is between 1.5 and 5 times larger.

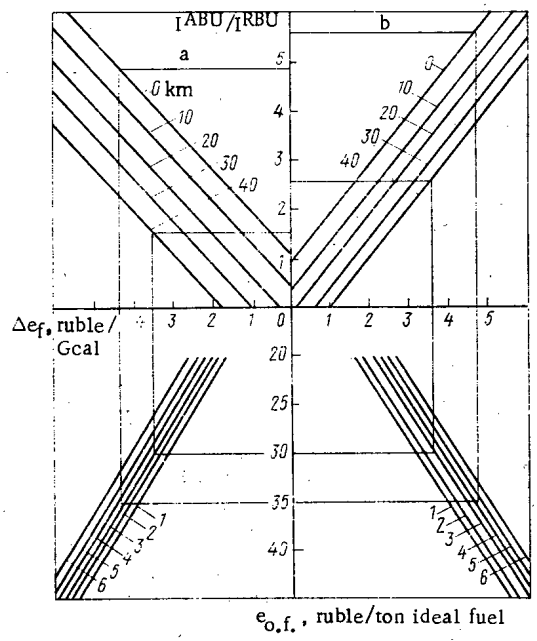

Fig. 2. Conditions for the economic competitiveness of ABU: a) ABU replace CHPP; b) ABU replace RBU; 1-6) the fuel component of the cited expenditures on ABU, equal to 2, 1.8, 1.6, 1.4, 1.2, and 1 ruble/Gcal, respectively; I^{ABU}, I^{RBU}) the specific investment in ABU and RBU; $e_{o.f}$) closing expenditures on organic fuel; Δe_f) the difference between ABU and RBU of the fuel component of the cited expenditures per unit of heat delivered to users. The distance the heat is transported is indicated on the curves.

Technical, Performance, and Economic Characteristics of ABU. Atomic boiler units can be established on the basis of the familiar types of nuclear reactors (water-cooled–water-moderated power reactors (VVÉR), RBMK, AMB, and VK), allowing for the possibility of reducing the coolant temperature at the reactor outlet and the necessity for reducing their unit output. Atomic boiler units can also be based on specialized nuclear reactors oriented toward ABU operating conditions. For example, they can be based on nuclear reactors cooled by high-boiling organic liquids which have good potential for effecting pressure reduction in the reactor loop. It should be noted that in every case the ABU will be quite different from organic-fuel boiler units on account of the specific properties of the nuclear reactor as an energy source.

The structure and nomenclature of the necessary services and equipment (control and safety rods, gas management, technical waste control and purification, supply and storage of fresh fuel, burial or transportation of radioactive waste, etc.) in the reactor section and fuel management of ABU is almost the same as in the reactor equipment of atomic electric power plants and atomic central heating-and-power plants (ACHPP). Unlike boiler units fired with petroleum residue fuel gas, ABU require much larger areas, to allow for the disposition of the services and equipment noted above and a zone for purposes of health protection. Also, access must be provided to these areas (most likely by railroad) in order to supply fresh fuel and remove the spent fuel. This complicates the problem of locating the ABU in or near densely populated areas. The maintenance of ABU is not altogether clear at this time. They require a large number of highly qualified personnel, and in order to be used only in their heat-supply capacity, must either be stopped or discharged down to 10-20% of their nominal power for a nonheating period (3000-4000 h/yr). Analyses of actual data on condensing atomic electric power plants and calculated figures for ABU (at the engineering proposal and design sketch stage) have shown that the specific cost of reactor installations largely depends on their unit output, while the constant component takes up 60-80% of the total expenditures on the production of heat by ABU.

Increasing the unit output of reactors and raising the utilization factor of their installed capacity thus enhances the economic competitiveness of ABU much more than of boiler units using organic fuel.

The unit heat output of ABU (Q_u^{ABU}) and their utilization factor are rigorously related to the attached heat load (Q_{att}), the required number of units (n_u), and the fractional participation of the reactors in the cover-

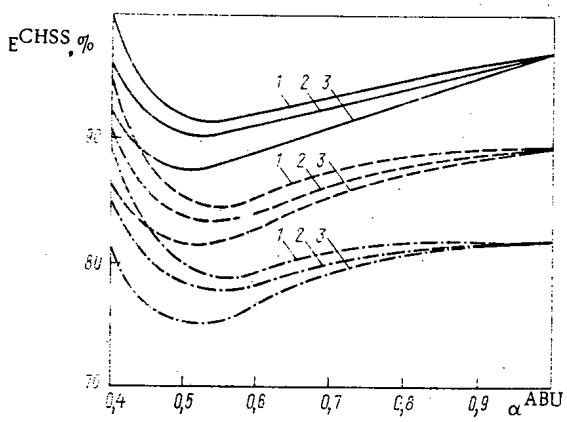

Fig. 3. Effect of α^{ABU} on the specific reduced expenditures on heat supplied to users in a CHSS for closing expenditures on organic fuel of 40 (1), 35 (2), 30 rubles/ton ideal fuel (3) and fuel components of reduced expenditures on ABU of 2 (——), 1.5 (– – –), and 1 ruble/Gcal (–·–·–·–).

age of the added heat load (α^{ABU}). Increasing the utilization factor of the installed capacity (which is possible because of a reduction in the fractional participation of the reactors in the coverage of the added heat load) necessarily reduces the unit output of ABU reactors and vice versa. A decrease in α^{ABU} from 1 to 0.5 increases the annual number of hours of usage of the installed capacity of ABU reactors by a factor of 1.5 (from 3000-3500 to 4500-5000), and the unit output of the reactors is reduced by a factor of two.

The choice of a rational value for α^{ABU} is made by a process of technoeconomic optimization. Figure 3 shows the results of economic estimates which were made for a centralized heat supply system (CHSS). The expenditures per unit of heat supplied to users are plotted vs α^{ABU} for a calculated regional heat load of 2000 Gcal/h and for various values of the cost of organic fuel and the fuel component of the cited expenditures on ABU.* The result is that for CHSS based on ABU it is appropriate to include the peak reserve organic-fueled boiler units, which considerably reduces the cited expenditures per unit of heat (≈8-10%). The optimal value of α^{ABU} depends on various factors, such as the cost of organic fuel, the unit capital investment in peak reserve boiler units, the length of the transit network from the ABU to the centers of heat usage, etc. Calculations show that depending on the combination of these factors, the optimal value of α^{ABU} lies within the range 0.5-0.8.

In order to make it possible to establish 500-1000 MW (the output of the power reactors of contemporary atomic electric power plants is 3-6 times as much) in the structure of heating ABU with 2-4 units for a reactor with unit heat output, attached heat loads of from 1500 to 6000 Gcal/h are required.

These technical, operational, and economic characteristics of ABU make it doubtful as to whether they can be considered as the direct analog of boiler units using organic fuel.

Economic Competitiveness and Use of ABU. The main factors which bear on the economic competitiveness of ABU are cost of organic fuel, attached heat load (which influences the unit output of ABU reactors and their technoeconomic characteristics), disposition relative to centers of heat usage, length of the heat conductors running to the points of heat usage, the fractional participation of the attached heat load, and the utilization factor of the installed capacity.

Studies were carried out to determine the effect of these factors on the economic competitiveness of ABU and their usage; some of the results are plotted in Fig. 4. The figure gives a plot of the ratio of the maximum allowable and anticipated actual specific investments (I) in ABU, as a function of the attached heat load (and the corresponding unit heat output of the ABU reactors), the cost of organic fuel, and the distance from centers of heat usage.

For organic fuel costs below 40 rubles/ton ideal fuel, ABU are thus unable to compete with boiler units which use organic fuel wherever the latter are employed (Q_{att} = 100-600 Gcal/h), because the anticipated spe-

*The range assumed for the variation of the relative investment in ABU (allowing for their dependence on the output of the unit) is indicated in Fig. 4.

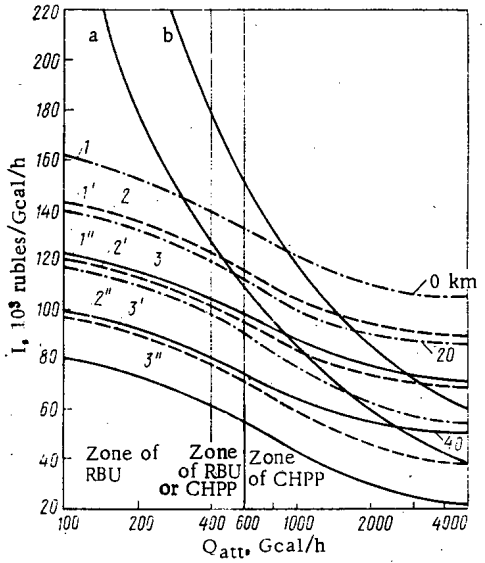

Fig. 4. Economic competitiveness of ABU, for completing expenditures on organic fuel of 40 (–.–.–), 35 (– – –), and 30 rubles/ton ideal fuel (——); a, b) actual specific investments in the ABU; 1, 1', 1", 2, 2', 2", 3, 3', 3") maximum allowable investments in ABU.

cific investment is more than the maximum allowable value. For the level of completing expenditures on petroleum residue fuel gas which is projected for the period up until 1990 (30-35 rubles/ton ideal fuel), ABU can compete only with basic CHPP using organic fuel; ABU become competitive for attached heat loads of 1500 Gcal/h and above. This value is higher than the minimum allowable figure for ACHPP (1000 Gcal/h and higher), since the unit heat output of the ABU reactors cannot be lower than 500 MW. Unlike boiler units using organic fuel, which in CHPP provide a way of extending the application of centralized heat-supply sources at the expense of providing heat to smaller users, ABU using low-temperature nuclear reactors can only supplement ACHPP in supplying heat to large users. Therefore, the factors which are crucial in deciding on the advisability of their construction may restrict the choice of areas to accommodate ACHPP in a given region (taking into account the fact that they cannot be farther from the heat users then the economically admissible distance). These factors may also hinder the economically appropriate introduction of electricity-generating power based on the fact that ACHPP will have a large associated condensed output.

Rational disposition of the ABU with respect to heat users is an important factor in making the ABU economically competitive. If ABU are located in the immediate vicinity of cities and areas of high population density, there will be higher specific expenditures on land confiscation, preparation of the surface, radiation safety measures (compared with the existing norms for planning "remote" nuclear energy installation). If ABU are located near populated areas, additional or reinforced protective shells may need to be constructed to prevent ejection of activity into the surroundings, and it may be necessary to locate the reactor section underground and construct additional facilities for treatment and transportation of radioactive wastes and spent fuel. Preliminary estimates and an analysis of the variation of the economic characteristics of nuclear energy installations with their distance from centers of population show that construction and operating expenditures incurred when ABU are located in the immediate vicinity of cities and populated areas may be 20-35% higher than the alternative expenditures incurred when they are located at a distance of 15-20 km or more. Calculations taking this into account show that in certain cases ABU with an overall attached heat load of 1500 Gcal/h and above can be economically feasible and efficient when they are located 20-25 km from cities and large population centers, in spite of the increased cost of the heat delivery networks. In this case the ABU continue to be competitive with CCHP (see Fig. 4).

Atomic–Chemical Boiler Units Based on High-Temperature Reactors (HTR). As was noted above, one of the most important ways to make nuclear sources of heat supply economically competitive (including ABU) is by increasing the unit output of the nuclear reactors which enter into their structure. When liquid coolants are used in heat transport systems, the possibilities for concentrating the heat outputs of ABU nuclear energy sources are limited. The economically feasible limits to the distances of heat transfer using hot water in a two-pipe system and assuming the possible elevation of temperature of the water in the supply mains of up to

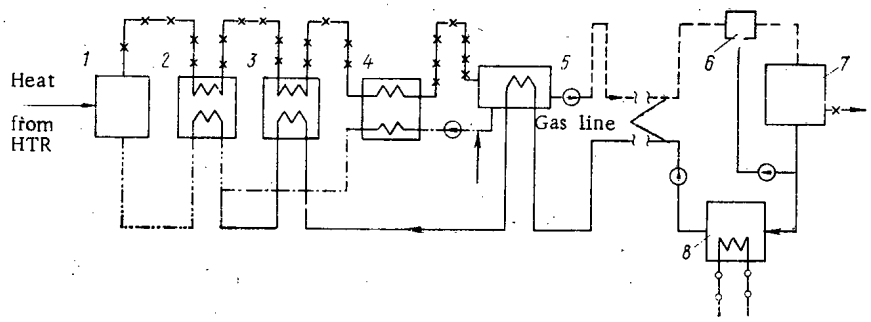

Fig. 5. Diagram of heat flow in an HTR atomic–chemical boiler unit: 1) device for steam–water conversion of methane; 2) heating of steam–gas mixture; 3) heating of methane; 4) steam generator; 5) condenser; 6) regenerator; 7) methanator; 8) boiler (steam generator); – – – –) H_2+CO mixture; –×–×–) H_2+CO+H_2O mixture; ———) CH_4; – ·· – ·· –) CH_4+H_2O; –○–○) circulating water and process steam; —— · —— ·) water and water vapor.

180-200°C are about 30-40 km; for a single-pipe system they are 50-70 km. Heat in the form of steam can be transported over distances of as much as 10 km, but this does not allow for centralization of the heat supply of relatively small heat consumers who are separated from each other by distances of more than 100 km. In order to fit nuclear heat supply sources (primarily ABU), it therefore would be useful to explore qualitatively new heat-transfer systems which would be designed for economically efficient heat transfer over distances of 100 km or more. This problem may possibly be solved by using systems in which the transfer is accomplished with "cold" heat-transfer agents, which transport the heat in a chemically bound state. As an example, such a heat conductor could be composed of a gaseous mixture of H_2, CO_2, and CO, which can be obtained by the endothermic vaporous conversion of methane [6]. A gaseous mixture (H_2+CO_2+CO) is transported in the cold form along a pipe to the heat-usage centers, where methane is exothermally synthesized from H_2, CO_2, and CO in special devices (methanators). The heat liberated in the reaction at temperatures as high as 400-650°C can be used to supply heat to residential and industrial users. The methane cooled in the process of heating the circulating heat conductor is sent to a conversion center. Since steam–water conversion of methane is accompanied by the absorption of large quantities of heat at temperatures of 850-900°C, the source of energy for the conversion center must be the HTR (Fig. 5).

Preliminary estimates show that heat in a chemically bound state can be transported with economic efficiency from nuclear conversion centers to heat-usage centers located over 100 km away. This creates new possibilities for concentrating heat outputs and enlarging the unit outputs of the nuclear energy sources of an atomic–chemical boiler unit by providing a large number of relatively small users with a centralized heat supply (Q_{att} =100-600 Gcal/h).

ABU for Heat Supply to Underdeveloped and Remote Regions. The experience of successfully operating the Bilibinsk ACHPP (which has three 36-MW central-heating power units) with a total design heat output of 75 Gcal/h as well as experience with the ARBUS and VK-50 installations indicates that it is possible to construct reliable and economically efficient low-power (5-100 MW) atomic heat-supply systems (based on channel and box-type nuclear reactors cooled by boiling water and organic liquids. Along with combined heat-supply installations (ACHPP), in some cases it may be useful to construct special-purpose atomic heat-supply systems (ABU) for underdeveloped areas and regions with difficult access. Studies have shown that under the conditions of fuel supply which prevail in these regions, ABU can be competitive if their specific investments are not more than 22-25 times those of boilers using organic fuel. As shown by preliminary estimates of the specific costs of ABU, such conditions can be ensured for unit nuclear reactor outputs in the several megawatt range. At the present time a design has been worked out for an organic–organic reactor with 15 MW of heat output (the ATU-15 system) for ABU.

CONCLUSIONS

Together with ACHPP, atomic boiler units can in this way be effectively applied to centralized heat-supply systems. In the European part of the country, a centralized heat supply to relatively small consumers of heat cannot be achieved with ABU based on the types of reactors which have been perfected. This is not the case with boilers using organic fuel. Atomic boiler units can supply heat mainly to large users (more than

1500 Gcal/h) and particularly to regions where it is impossible or unfeasible to construct ACHPP. In underdeveloped and remote areas, atomic boiler units can be economically efficient with nuclear reactors operating at heat outputs levels of several megawatts. With the successful development and perfection of high-temperature nuclear reactors and systems for transporting heat in a chemically bound state, the outlook is that atomic–chemical boiler units will be created which will permit nuclear energy resources to be much more widely applied as efficient centralized heat supply sources for smaller heat users, whose share of total heat consumption is as high as 40%.

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MULTICRITERIAL OPTIMIZATION OF THE DEVELOPMENT OF NUCLEAR POWER IN THE FRAMEWORK OF THE PERSPECTIVES OF COMECON

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UDC 621.039.003

The nuclear power of the COMECON member-countries composes an intricate system. On the one hand, national programs for nuclear power development form a basis for predicting the nuclear power development of the COMECON countries as a regional system; on the other hand, these programs are drawn up by taking into consideration the demands placed on the COMECON nuclear power system. Such a system approach to national economic tasks makes it possible to bring out the effect of socialist collaboration in the area of nuclear power – the effect of integration.

In view of the large number of factors affecting the development of the COMECON nuclear power system, its future is hard to predict because of difficulties related to its dynamics and the multistage character of its processes, its multilevel and hierarchic structure, the uncertainty of its conditions and parameters and the choice of optimization criteria.

The choice of such criteria (goal functions) for the solution of optimization problems involved in studying the structure of nuclear power in perspective leads to the so-called multicriterial optimization. The transition to the latter also comes from the attempt to estimate a property of solutions assumed from different standpoints. The structure of developing nuclear power can be simultaneously optimized according to several criteria by applying the method used to solve multicriterial problems.

Multicriterial problems arise in the following situations [1] (see Fig. 1):

a) an object is regarded as a single element with several criteria;

b) an object consists of several elements which are optimal according to one criterion;

c) an object can contain many elements, each being in a multicriterial situation.

The differences in these situations (one multicriterial element, many single-criterial elements, and many multicriterial elements) is basic, because the methods used in the solution of one case are not applicable to another.

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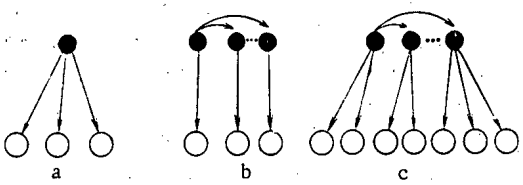

Fig. 1. Situations which lead to multicriterial problems.
●) elements; ○) criteria.

Until now, the optimization of the development of nuclear power engineering within the scope of COMECON has been carried out according to only one minimizing criterion – the integrated requirements for natural uranium. This was due to: 1) the importance of such a criterion for this new and important branch of energy engineering, just beginning to come into use; 2) the fact that there were no theoretical principles in the economics of nuclear power engineering (the first article in this area [2] was published in 1969); 3) the fact that the COMECON countries did not have a common currency, which is now the trade ruble.

In order to make a more complete study of the effect of the various factors on our concepts of the possible outlook for the development of nuclear power engineering with mathematical modelling, economic criteria such as reduced expenditures can also be used.

Application of the methods of solution for multicriterial problems to nuclear power engineering makes it possible both to combine these most important optimization criteria (integrated requirements for natural uranium and reduced expenditures), and to take into account other criteria (labor costs, costs of metal, etc.) as well.

In solving multicriterial problems, one looks for such a solution which would be optimal for the set of goal functions $f=\{f_i\}$ $(i=1, 2, \ldots, M)$. In the majority of cases, however, no such solution exists simultaneously for all f_i. Therefore, a solution of a multicriterial problem is understood to be such a solution which may not be optimal for any of the goal functions, but which is the most acceptable for the whole set of goal functions. Such a solution of the multicriterial problem is called a compromise.

We now consider those problems of the structure of nuclear power engineering in which the methods of multicriterial optimization find important application.

Problem 1. To optimize the nuclear power engineering structure of the COMECON member-nations as a unified system, according to two criteria: that of minimum expenditure on natural uranium, and that of minimum reduced expenditures in conversion rubles. This corresponds to the "one multicriterial element" situation. Constraints are imposed on imports of generating machinery, on electrical power out, and on the distribution of plutonium.

Problem 2. In optimizing the future structure of nuclear power engineering, it must be kept in mind that by 1990, after which time the large-scale introduction of breeder reactors is expected, large quantities of plutonium will be required.

The problem is therefore solved with the following goal functions: minimum expenditure on uranium, minimum reduced expenditures, and maximum plutonium operating time. Constraints are imposed on generating machinery imports and electrical power output. This situation is that of "one multicriterial element."

The solution of such a problem for a 40-50-yr forecast will make it possible to bring out the role of breeder reactors more fully and to show the necessity of their development within the structure of nuclear power engineering in the framework of COMECON.

Problem 3. On account of such factors as the large scale of nuclear power engineering, the construction of a large number of industrial nuclear power stations and the opening up of new types of nuclear power stations, and the necessity of cooperation between COMECON member-countries in the construction of individual nuclear power plants, a number of problems are encountered in studying the future of nuclear power engineering. These relate to the mechanical-engineering base, expenditures on metals, and labor costs. It is therefore appropriate to optimize the structure of nuclear power engineering according to the following criteria: minimum reduced expenditures, minimum expenditures on metals, and minimum labor expenditures. In this case we meet the most complicated situation, that of "many multicriterial elements."

Problem 4. In solving the first two problems, one optimizes the structure of the nuclear power engineering of the COMECON member-countries as a whole. In studying the structure and perspectives for the develop-

ment of nuclear power engineering separately for each country, it is useful to optimize the structure of nuclear power engineering (regarding the nuclear power engineering of the COMECON countries as a unified system) at first according to minimum expenditure in each country, and at the next stage according to minimum total expenditure in the system. This corresponds to the situation of "many single-criterial elements." Constraints on imports of industrial machinery to each country are imposed on the elements. The plutonium balance equations provide a relation between the elements. The solution of such a problem is of great interest in relation to the study of strategies for developing nuclear power in COMECON countries.

The methods that are used effectively in solving multicriterial problems include the method of the leading target function, the generalized goal function, and the method of successive relaxations. Other methods exist which are also used to solve multicriterial problems. These, however, do not correspond as well to the problems of optimizing nuclear power development.

Method of the Leading Goal Function [3]. This specifies, as its name implies, the optimization of the problem according to one goal function. The other goal functions are considered as additional constraints during the separate steps of the calculations, which are done by the simplex method. The multicriterial problem thus reduces to the traditional single-criterial one. We will apply this method to multicriterial problems with linear goal functions and linear constraints. It can be used in solving Problem 3.

A variant of the leading goal function method is the Radzikovskii method [3], in which the optimization according to each criterion is stipulated at the first step. As an example, we consider a three-criterial problem. Let the optimal values of the criteria be f_1^0, f_2^0, f_3^0 (for the sake of definiteness, we will assume them to be maximal in the optimization with respect to each of them separately). We select the most important criterion, e.g., f_3. The optimal values obtained for the remaining criteria are compared with given quantities D_1 and D_2. At the next step, we carry out an optimization with respect to the most important criterion f_3, with the additional constraints

$$D_1 \leqslant f \leqslant f_1^0; \quad D_2 \leqslant f_2 \leqslant f_2^0.$$

This method suffers from the defect that the present problem cannot have allowable solutions, due to the introduction of the additional constraints.

Method of the Generalized Goal Function. Such a function can be formed in various ways. By choosing weighting coefficients (α_i) for M inequivalent goal functions, a generalized goal function can be introduced:

$$F(x) = \sum_{i=1}^{M} \alpha_i f_i(x).$$

If the criteria are equivalent, the integrated relative loss [4] can play the role of generalized goal function. The former is equal to the sum of the relative deviations from their optimal values of the set of optimality criteria:

$$F(x) = \sum_{i=1}^{m} \frac{f_i^0 - f_i(x)}{f_i^0} + \sum_{i=m+1}^{M} \frac{f_i(x) - f_i^0}{f_i^0},$$

where f_i^0 is the optimal value of the i-th criterion. The first sum is written for the maximized criteria; the second, for the minimized criteria. Such a generalized criterion can be used for criteria with different dimensionality, e.g., in the case of Problems 1 and 3.

In case the criteria are not equivalent, the generalized goal function is modified. In such a case, let there be given a preferred arrangement of the goal functions, in order of importance: $f_1 \succ f_2 \succ \ldots \succ f_M$ ($\succ$ denotes the preference). In so doing, this preference is given quantitatively in comparison with the least important criterion by the quantities $\rho_1, \rho_2, \ldots, \rho_{M-1}$. The quantities ρ_i indicate how many times the optimal solution of the problem according to the i-th criterion is more preferable than the solution according to the M-th criterion. The generalized criterion then takes the form

$$F(x) = \sum_{i=1}^{m} \frac{\rho_i}{f_i^0} [f_i^0 - f_i(x)] + \sum_{i=m+1}^{M-1} \frac{\rho_i}{f_i^0} [f_i(x) - f_i^0] + \frac{f_M(x) - f_M^0}{f_M^0}.$$

Method of Successive Relaxations [5]. We will also use this method in the case of inequivalent criteria. Let all of the criteria be already distributed in order of their decreasing importance: $f_1 \succ f_2 \succ \ldots \succ f_M$. For the sake of simplicity, we will assume that all of them must have a maximum value. The following is an algorithm for finding the compromise solution. An optimal solution according to one criterion f_1 is found first.

A certain "relaxation" Δf_1 is assigned next. This determines the loss which it is possible to yield according to this criterion in order to obtain a better result according to the next most important criterion f_2. We now find the optimal solution according to the criterion f_2 with the additional constraint

$$f_1(x) \geqslant f_1^0 - \Delta f_1,$$

where f_1^0 is the optimal value of f_1.

Next, we again designate a relaxation Δf_2 which determines the loss from the optimal value of the second criterion obtained in solving this problem, in order to then obtain the optimal solution of f_3, etc.

The method of solution of multicriterial problems can be used for arbitrary goal functions with arbitrary constraints. But we cannot apply this method directly in a number of cases because of difficulties which arise which are related to the solution of optimization problems in which additional constraints are taken into account. It should be noted that in using the methods indicated, it is necessary to assign a "weight" to the goal functions or to make preferences among them according to their importance. For this it is useful to apply the methods of expert estimates [6], this being particularly important in such a new field as nuclear power engineering because they make it possible to include the practical experience, knowledge, and intuition of specialists in formulating correct models of the behavior of a system.

The most widely used methods for making expert estimates are the method of preference and the method of rank.

In the method of preference, the experts distribute the given criteria according to their decreasing importance, determining the position of each criterion.

Let R_{ij} be the position of the i-th criterion as determined by the j-th expert. Then the weighting coefficients of the criteria are determined by the expression

$$\alpha_i = \sum_{j=1}^{n} R_{ij} \Big/ \sum_{i=1}^{m} \sum_{j=1}^{n} R_{ij},$$

where m is the number of criteria and n is the number of experts.

In the method of ranks, the expert estimates the importance of the criterion on a quantitative scale running from 0 to 10. The weighting coefficients of the criteria are in this case determined by the expression

$$\alpha_i = \sum_{j=1}^{n} W_{ij} \Big/ \sum_{i=1}^{m} \sum_{j=1}^{n} W_{ij},$$

where $W_{ij} = \gamma_{ij} \sum_{i=1}^{m} \gamma_{ij}$ (γ_{ij} is the rank of the i-th criterion given by the j-th expert).

Let us consider a variant of the nuclear power engineering system of the COMECON countries: Bulgaria, Hungary, the German Democratic Republic, Poland, Rumania, the USSR, and Czechoslovakia. We will solve optimization problem 4 with a common store of the plutonium produced and with economic criteria. We will thoroughly discuss the solution of this problem by the method of total relative losses. The choice of the method is determined by the linearity of the goal functions and the constraints, the comparative simplicity of the calculations, and the possibility of using goal functions of various dimensionality. The generalized goal function and the goal functions of each of the COMECON member-countries can be represented either in the national currency or in trade rubles.

The following constraint is introduced for this problem [7]: The balance equation of the input power for each country

$$\sum_{f=1}^{n} x_{f,\tau,i} = N_{\tau,i},$$

where i = 1, 2, ..., M (M is the number of COMECON member-countries); $x_{f,\tau,i}$, quantity introduced to the i-th country during time τ of the output of a nuclear power station of type f; $N_{\tau,i}$, necessary growth during time τ of the output of all nuclear power stations in the i-th country.

In this variant, the structural connection between the nuclear power station outputs introduced into the different countries is expressed by the equation for the balance between the secondary nuclear fuel – plutonium (the uranium–plutonium cycle is considered in the present problem) and the requirement of its breeder reactors and the operating time accrued by all of the nuclear power stations. This condition can be formulated

in the following way: At each instant of time, the plutonium store of the system which is ready for loading into the breeder reactors must be nonnegative.

We introduce the following notation: F_a, set of nuclear power station types using uranium as fuel, F_b, set of nuclear power station types using plutonium, $g^0_{f_b,\tau}$, specific yield of plutonium on the initial loading of the nuclear power stations (per unit installed capacity) with f_b reactors of the type ($f_b \in F_b$) introduced in the time interval τ; $g_{f,\tau}$, specific plutonium yield per unit of electrical energy produced in time τ; $g_{f_b,\tau}$, specific expenditure on plutonium per reloading per unit of electrical energy produced in time τ; r_b, inertia of the exterior fuel cycle, i.e., the time between unloading the fuel elements to their loading after processing in the reactor; α_τ, store of plutonium ready for loading which is available in the system up to the end of the interval τ; $h_{f,j,\tau}$, annual number of hours of usage of the output of nuclear power stations of type f in the j-th regime during the interval τ; and $G_{\tau - r_b}$, plutonium produced in the interval $\tau - r_b$ by existing nuclear power stations at the beginning of the reference period.

We consider two operating regimes of nuclear power stations: with the least possible number of hours of usage in the first year of the introduction of output (j = 1) and with the least possible number of hours of usage (j = 2) in the following years of operation of the nuclear power stations. In the first year of operation of nuclear power stations with breeder reactors, the fuel expenditure per reloading is equal to zero. The quantity of plutonium produced by all types of nuclear power stations in the interval r is determined by the equation

$$P_r = G_{r-r_b} + \sum_{f=1}^{n} \sum_{j=1}^{\alpha} \sum_{\tau=1}^{r-r_b} \sum_{i=1}^{M} g_{f,\tau} h_{f,j,\tau} x_{f,\tau,i}.$$

The quantity of plutonium loaded in nuclear power stations of the f-th type introduced in the interval r is equal to

$$P^0_r = \sum_{f_b \in F_b} \sum_{i=1}^{M} g^0_{f_b,\tau} x_{f_b,\tau,i}.$$

The expenditure on plutonium per reloading in the interval r is determined by the equation

$$\bar{P}_r = \sum_{f_b \in F_b} \sum_{j=1}^{2} \sum_{\tau=1}^{r-1} \sum_{i=1}^{M} g_{f_b,\tau} h_{f_b,j,\tau} x_{f_b,\tau,i}.$$

The condition for the nonnegativity of the store of commercial plutonium in the system is obtained in the form

$$P^0_r + \bar{P}_r - P_r - \alpha_{r-1} + \alpha_r = 0,$$
$$\alpha_r \geqslant 0, \quad r = r_0, \ \ldots, \ R'.$$

Here we use the economic criterion expressed in its monetary form, it being the one which most fully answers to the problem posed in this study:

$$E_i = \sum_f \sum_t (I_{f,t,i} + C_{f,t,i})(1 + \sigma_{red})^{1-t},$$

where σ_{red} is a norm for reducing the nonsimultaneous expenditures; $I_{f,t,i}$, investment in nuclear power stations of the f-th type in the year t in the i-th country; and $C_{f,t,i}$, cost of operating nuclear power stations of the f-th type in the year t in the i-th country.

The first stage of the solution consists in solving the problem M times for each country without going beyond the national scope; i.e., using constraints on the introduction of power to the given country and with their own reserves of plutonium produced. This corresponds to the actual resources of each country. From the minimization we obtain M optimal solutions which determine the optimal structure of nuclear power engineering in each country.

At the second stage of the solution, we minimize the goal function, which is written for the entire system as a whole as

$$E = \sum_{i=1}^{M} \frac{E_i - E^0_i}{E^0_i}.$$

Such a form for the goal function ensures minimum deviations from the optimal national plans for developing nuclear power engineering in the COMECON countries, while making the goal function dimensionless and enabling the E_i to be expressed in conversion rubles as well as in the national currency.

In case it is necessary to give preference to a particular country because it is desirable to make it the location of a large number of a certain type of nuclear power station, preference coefficients ρ_i are inserted into the goal function. These can be determined by the "expert" methods.

In this case, the goal function is

$$E = \sum_{i=1}^{M-1} \rho_i \frac{E_i - E_i^0}{E_i^0} + \frac{E_M - E_M^0}{E_i^0}.$$

In conclusion, it should be noted that multicriterial optimization problems, like all optimization problems, should be solved not by just one but by several methods owing to the lack of an adequate mathematical model for the problem of an actual national economy. Therefore, the solutions obtained must be multivariant in assessing their features by means of the assumed solution.

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AN ESTIMATE OF THE UNCERTAINTY FACTOR FOR PREDICTING THE DEVELOPMENT OF NUCLEAR POWER ENGINEERING

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Experience has shown that the use of mathematical modeling is the most effective way to forecast the development of nuclear power engineering. A model which takes into account the economic criteria of a system has been developed [1, 2] for making long-range forecasts. The model introduces: a) a set of linear constraints of the form

$$Ax = b, \quad x \geqslant 0, \tag{1}$$

(where **x** is the desired strategy for development of the system; A, a matrix; and **b**, constraint vector) and b) the linear functional being minimized in region (1), with

$$cx \to \min, \tag{2}$$

where **c** is the specific expenditures vector.

The desired strategy **x** is found for given A, **b**, and **c** by one of the methods of linear programming.

Known models [3-8] with specific characteristics have the same mathematical statement of the problem (equivalent to the problem of linear programming) as in expressions (1) and (2). A characteristic of any linear programming model is its determinacy, i.e., the single-valuedness of all exogenetic data of the model which enter into A, **b**, and **c**.

However, the situation encountered in forecasting nuclear power engineering is such that many important characteristics are not known with precision. Since they cannot be precisely known, in modeling they are repre-

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sented in an inexact form, usually in the form of intervals of possible values. Several factors determine the uncertainty of the characteristics; e.g., the instability of the object studied (this appears, in particular, in the ambiguity in the directions of technical progress), the limited nature of the knowledge process, the psychological nature of the persons who are the sources of the forecasts, etc. In the present problem the uncertain factors are the investment in fast-reactor nuclear power plants, cost of mining and reserves of natural uranium, the rates of long-term growth of nuclear power, the dates for bringing fast reactors into operation, etc. Under conditions of uncertainty, it is therefore not possible to apply the optimization criteria in the form of expression (2). The attempt to overcome this defect of the model of linear programming is multivariant, but using determinate data in calculations is of little help [9]. In most cases this uncertainty results in ambiguous forecasts for the structure of nuclear power engineering and makes it necessary to develop a different methodological basis and a different way of optimization which takes the uncertainty into account.

It should be noted that the values of the majority of the factors which are represented in an indeterminate form at the time of prediction are not formed by a random mechanism. This pertains, e.g., to such indicators as natural uranium reserves, the dates for putting commercial fast reactors into operation, the investment in them, etc. Since the uncertainty does not reduce to randomness in the problem of forecasting the structure of nuclear power engineering, it is not appropriate to use stochastic programming methods [10].

It has also been pointed out [8] that the factors which most strongly affect the structure of nuclear power engineering are concentrated in the vector **c**, i.e., in the coefficients of the functional of the determinate optimization problem. These factors are the investment in nuclear power stations with fast and thermal reactors. We will consider this case of optimization under conditions of uncertainty, with the vector **b** and the matrix A given in determinate form, and the vector **c** given in an indeterminate form. We shall refer to the problem of finding the optimal strategy in an indeterminate situation specified in this way as a problem of the first kind [11].

Solution of the Problem of the First Kind. Let the set of allowable strategies for development of a system of nuclear power engineering be given by Eqs. (1). Let there be also given the set D of possible values of the vector **c**. It is assumed that it is impossible to control the choice of any vector **c**, i.e., the existence of the set D reflects the incompleteness of the knowledge of the character of the future development of the system, on the one hand, and the objective indeterminacy of the development of the system at the time of prognosis, on the other hand. In such a situation it is impossible to use directly criterion (2) of minimum reduced expenditures, since it has no meaning for set D. It is further assumed that D is a convex set. The method described in [11] does not operate with a continuous set, but with a discrete subset of vectors $\mathbf{c}_1, \mathbf{c}_2, \ldots, \mathbf{c}_S$ of this set. Henceforth, the set of limiting points will be used as the discrete set. It is proven that by discretizing the convex set D by its limiting points, and applying a method proposed in [11] to this discrete set, one can determine the strategies that are optimal on the whole set D. In light of the above, a method for taking the indeterminacy into account is characterized in the following way.

1. The method of optimization: For the optimization criteria, we use the set of criteria of the theory of statistical solutions, viz., the criteria of Wald, Savage, Laplace, and Hurwitz. According to the Wald criterion, one calculates a strategy which in comparison with every other has the least expenditure under the worst conditions defined on the set D which are possible for the given strategy. According to the Savage criterion, a strategy is found with the least maximal risk. According to the Laplace and Hurwitz criteria, strategies are found with the least average and least weighted expenditures, respectively. In the last case, the minimal and maximal expenditures are "weighted" by a coefficient which is specified a priori and which is a constant for each country.

2. The optimal strategy is determined on the whole set of allowable strategies (see Eq. (1)) and on the whole set of possible values of the vector **c** – the set D.

3. No arbitrary rejection of part of the data occurs when this method is used, nor is nonessential data included unjustifiably (e.g., on reducing the indeterminacy to randomness).

4. The method is oriented toward the linear programming approach.

The following order of steps in the calculations is recommended.

First Stage. Using a specially developed method, the work is organized to formulate the exogenetic data of the model.

Second Stage. A separation is made between the essential and nonessential data, consistent with the goals of the calculation. The basis of the separation can be past experience, expert estimates, or special cal-

TABLE 1. Parameters of Special Linear Problems in Allowing for the Factor of Indeterminacy

Criterion	Formula of the criterion	Linear functionals	Constraints	No. of linear problems / No. of constraints
Mean expenditures (Laplace)	$(1/S)\sum_{i=1}^{S}(\mathbf{c}_i, \mathbf{x})$	$(1/S)\min\left(\sum_{i=1}^{S}\mathbf{c}_i, \mathbf{x}\right)$	$A\mathbf{x}=\mathbf{b};\ \mathbf{x}\geqslant 0$	$1/m$
Max. expenditures (Wald)	$\max_{i=1,\ldots,S}(\mathbf{c}_i, \mathbf{x})$	$\min y$	$A\mathbf{x}=\mathbf{b};\ y\geqslant(\mathbf{c}_i, \mathbf{x});\ i=1, 2, \ldots, S;\ \mathbf{x}\geqslant 0;\ y\geqslant 0$	$1, m+S$
Max. risk (Savage)	$\max[(\mathbf{c}_i, \mathbf{x})-z_i];\ z_i=\min_{\mathbf{x}:A\mathbf{x}=\mathbf{b}}(\mathbf{c}_i\mathbf{x})$	$\min y$	$A\mathbf{x}=\mathbf{b};\ y\geqslant(\mathbf{c}_i, \mathbf{x})-z_i;\ i=1, 2, \ldots, S;\ \mathbf{x}\geqslant 0;\ y\geqslant 0$	$S+1/m+S$
Weighted expenditures for a given a of the expenditure (Hurwitz)	$a\min_i(\mathbf{c}_i, \mathbf{x})+(1-a)\times\max_i(\mathbf{c}_i, \mathbf{x})$	$\min_i \widetilde{K};\ \widetilde{K}=\min[(a\mathbf{c}_i, \mathbf{x})+(1-a)y];\ i=1, 2, \ldots, S$	$A\mathbf{x}=\mathbf{b};\ (\mathbf{c}_i, \mathbf{x})\leqslant(\mathbf{c}_j, \mathbf{x});\ j=1, 2, \ldots, i-1, i+1, \ldots, S;\ y\geqslant(\mathbf{c}_k, \mathbf{x});\ k=1, 2, \ldots, S;\ \mathbf{x}\geqslant 0;\ y\geqslant 0$	$S, m+2S-1$

culations based on a linear programming model. Nonessential parameters in the subsequent calculations are represented by mean values. The essential parameters, which are given by a range of values, form a convex set (in the process, the interdependence of the essential parameters should be taken into account, consistent with the specification of the problem). Instead of the continuous set, the set of peaks of the convex set of the essential parameters is selected for subsequent calculations.

Third Stage. For each peak $\mathbf{c}_i$, the following problem is solved on the linear programming model with constraints (1): the linear functional $\mathbf{c}_i\mathbf{x}$ is minimized. The purpose of this stage is to find the numbers $z_i = \min_{\mathbf{x}} \mathbf{cx}$.

Fourth Stage. The special problems of linear programming are formed and solved in accordance with Table 1 [11]. The purpose of this stage is to find the set of the optimal strategy of development under the conditions of indeterminacy.

Fifth Stage. The optimal strategies found are analyzed, taking into account the specific content of the problem being solved. The purpose of this stage is to narrow down, as much as possible, the recommended strategy.

Sample Calculation. The structure of nuclear power engineering was calculated for 50 years. The effects of the indeterminacy of the specific investments in nuclear power stations with fast and thermal reactors and the cost of mining natural uranium were studied. Each of them was defined initially as a range of possible values and the set together constituted a convex polyhedron. In accordance with the method used, the polyhedron was replaced by the set of limiting points. Each limiting point, conventionally designated by a state of the medium, is characterized by three numbers: I_f and I_t are the investments in nuclear power stations with fast and thermal reactors, and C_{nat} is the cost of mining natural uranium. The characteristics of the peaks of the set D, which formed the calculated set of states of the medium, is shown in Table 2 (maximum and minimum values are denoted by bars above and below respectively).

We consider fast breeder reactors, fast-converter reactors, thermal reactors using enriched uranium, and thermal reactors which will be converted to plutonium fuel in the middle of the 21st century. Converter reactors are reloaded with uranium only during the first 3 years, and then are converted to a breeder regime. The hypothesis that the regime of reloading such reactors is efficient has been confirmed [9]. It has been proposed that fast reactors will be put into service by the beginning of the 1990s and at that time will be widely introduced in power engineering (the model distinguishes between the concepts of possibility and advisability: The optimal time for introducing breeder reactors on a wide scale is determined on the model, i.e., is an endogenic variable). Possible improvements in technology were taken into account in assigning the reactor characteristics. The thermal reactors considered were of the light-water type.

Each peak of the set was assigned a number for which is used the following notation: 1.1, 1.2, 1.3, . . . , 1.8. We also refer to the peaks of the set D as limiting states.

Along with the exact method for optimizing under conditions of indeterminacy, which is considered here, other, less accurate methods for determining the optimal strategy are also possible. According to the simplest and least accurate method (but which has become widely used; see, e.g., [12]), the optimal strategies are determined from the number of strategies found on a linear programming model; i.e., from the number of strategies

TABLE 2. Variants of the States of the Medium – the Peaks of the Set D

State	Indicator	Date of putting nuclear power stations into service				Indicator	Rel. cost of mining natural uranium, by categories		
		1986–1990	1991–1995	1996–2005	after 2005		1	2	3
1.1	$\overline{I_f}$	—	440	330	265	$\overline{C}_{nat}$	1	1,76	2,4
	$\underline{I_t}$	325	325	270	245				
1.2	$\underline{I_f}$	—	440	330	265	$\underline{C}_{nat}$	0,75	1,32	2,4
	$\underline{I_t}$	325	325	270	245				
1.3	I_f	—	440	330	265	$\overline{C}_{nat}$	1	1,76	2,4
	$\underline{I_t}$	265	265	220	190				
1.4	$\overline{I_f}$	—	440	330	265	$\underline{C}_{nat}$	0,75	1,32	2,4
	$\underline{I_t}$	265	265	220	190				
1.5	$\underline{I_f}$	—	355	315	235	$\overline{C}_{nat}$	1	1,76	2,4
	$\overline{I_t}$	325	325	270	220				
1.6	$\underline{I_f}$	—	355	315	235	$\underline{C}_{nat}$	0,75	1,32	2,4
	$\overline{I_t}$	325	325	270	220				
1.7	$\underline{I_f}$	—	355	265	215	$\overline{C}_{nat}$	1	1,76	2,4
	$\underline{I_t}$	265	265	220	190				
1.8	$\underline{I_f}$	—	355	265	215	$\underline{C}_{nat}$	0,75	1,32	2,4
	$\underline{I_t}$	265	265	220	190				

in the third stage. In our work, this approximate method was used to determine to what degree application of the more precise method would be justified. The approximate method can be termed a first-approximation method, and the strategy obtained with it can be said to be optimal in the first approximation.

In order to distinguish the first-approximate strategies, we used the designations of those corresponding limiting stages with which the present strategies were determined, enclosing the designation of the strategy in double brackets. For example, $\langle\langle 1.1\rangle\rangle$ denotes a strategy found by optimizing the linear programming model under the assumption that the limiting state 1.1 is realized. Similarly, a strategy which is optimal on the linear programming model with the limiting state 1.2 is denoted by $\langle\langle 1.2\rangle\rangle$. In conformity with this, eight such strategies are defined: $\langle\langle 1.1\rangle\rangle$, $\langle\langle 1.2\rangle\rangle$, ..., $\langle\langle 1.8\rangle\rangle$. Four characteristic values were calculated for each of them: the maximal expenditures and risks, and the average and weighted-average expenditures. The development strategies which were optimal in the first approximation were then chosen in accordance with the four criteria used under the conditions of indeterminacy. A subsequent analysis and calculations showed that all of the strategies $\langle\langle 1.1\rangle\rangle$, $\langle\langle 1.2\rangle\rangle$, ..., $\langle\langle 1.8\rangle\rangle$ have the following basic properties:

fast reactors are present in all optimal structures of nuclear power engineering for the calculated period;

from a certain stage which is characteristic for each strategy, fast reactors begin to displace thermal reactors;

from the beginning of the 21st century, thermal reactors are gradually converted to plutonium fuel;

the increase in the cost of mining natural uranium, which was used in the calculations, is a more important factor than the uncertainty of the mining cost within the categories; the latter uncertainty did not appear to have a long-term effect on the structure of nuclear power stations;

the ratio α of the investment in nuclear power stations using fast and thermal reactors has a significant effect on the solution obtained in optimizing the date for which widespread construction of fast reactors is initiated.

With minimal E (the states 1.5 and 1.6), i.e., under the most economically favorable conditions for the development of breeder reactors, the optimal date for introducing fast reactors would already have to be at the beginning of the 1990s, and their share by the end of the century would come to about 30%. These data approximate the results which would have been obtained by optimizing the structure of nuclear power engineering according to the criterion of minimum requirement for natural uranium. The share of thermal reactors using enriched uranium would decrease by 1995 to 35% (by the end of the century to 7%), but the share of thermal reactors using plutonium fuel would increase. Their share at the beginning of the 1990s would be 30%, and by the end of the century would amount to 50%. By the beginning of the 2020s their share decreases to 15%. The difference in the cost of natural uranium with respect to the first two categories of reserves in these states does not have any effect on the structure of nuclear power engineering.

TABLE 3. Matrix of Expenditures and Criterial Values

Strategy	Expenditures								Numerical value of the criterion			
	Medium state											
	1.2	1.2	1.3	1.4	1.5	1.6	1.7	1.8	Wald	Laplace	Hurwitz	Savage
Optimal accord. to the first-approximation method:												
«1.1»	85,8	85,0	79,7	78,8	82,3	81,3	72,0	72,5	85,8	79,5	73.9	2,6
«1.2»	86,1	84,8	79,5	78,5	82,3	81,3	72,2	71,0	86,1	79.5	74,1	2,4
«1.3»	88,4	86,9	77,6	76,1	85,7	84,1	75,1	73,5	88,4	81,0	76,4	4,4
«1.4»	88,4	86,9	77,6	76,1	85,7	84,1	75,1	73,5	88,4	81.0	76,4	4,4
«1.5»	86,9	86,0	81.9	81,0	81,7	80,9	72,6	71,7	86,9	80.4	74,8	4,9
«1.6»	87,0	86,0	81,9	81,0	81,7	80,9	72.6	71,7	87,0	80.4	74.8	4.9
«1.7»	85,8	85,0	79,8	78,8	82,3	81,3	72.0	71,0	85,8	79.5	73.9	2.6
«1.8»	86,0	85,0	79,5	78.5	82,3	81,3	72,0	71,0	86,0	79.5	73,9	2,4
Optical accord to exact method of taking uncertainty into account:												
≪Wald≫	85,8	85,0	79,7	78,8	82,3	81,3	72.0	71.0	85.8	79.5	73.9	2,6
≪Laplace≫	86,0	85,0	79,2	78,2	82,6	81,6	72.0	71,0	86,0	79.4	73.9	2.1
≪Hurwitz≫	86,0	85,0	79,5	78,5	82,5	81,4	72,0	71.0	86.0	79,5	73,9	2,4
≪Savage,≫	86,1	85,0	78,6	77,3	83,2	81,9	72.6	71.4	86.1	79,5	74,4	1.5

Under the least favorable conditions (states 1.3 and 1.4), the optimal date for introducing breeder reactors should not be until the end of the first decade of the 21st century, when $\alpha = 1.4$. It is clear that such a late introduction of fast reactors means that they cannot have much effect on the structure of nuclear power engineering until the end of the first decade of the 21st century. Intermediate structures will evolve with the limiting states 1.1, 1.2, 1.7, and 1.8. For all of these variants, the optimal time for beginning widespread construction is the middle of the 1990s. Their share increases rapidly by the end of the century. The results cited here show, e.g., that a question of such paramount importance as when to begin the widespread construction of fast reactors cannot be solved by a simple multivariant calculation without using the criteria assumed in the solutions which take uncertainty into account.

The results of calculations made with the two methods (exact and first-approximation) are shown in Table 3. It is in the form of a matrix divided into four quadrants. The elements of the table which are located in the two left-hand quadrants correspond to the reduced total expenditures (in monetary units) on the development of a nuclear power station system within the period being forecast, using a strategy whose designation is given to the left of the row with a medium constant as indicated at the top of the column. For example, there is the number 85.8 (underlined) in the upper left-hand corner of the table. It shows that the first-approximate strategy ⟪1.1⟫ requires an expenditure of 85.8 monetary units, if I_f, I_t, and C_{nat} are found to be appropriate to the state of the medium 1.1 (see Table 2). Similarly, to the right of the previous value, the number 85.0 indicates the expenditures of the strategy ⟪1.1⟫ for the state of the medium 1.2, etc. The top half of the table thus contains results obtained by the first-approximation method, with the lower half containing results using the exact method developed above. Strategies designated as ⟨⟨Laplace⟩⟩ are thus obtained by the latter method, with the system being optimized according to the Laplace criterion. The ⟪Savage⟫, ⟪Wald⟫, and ⟪Hurwitz⟫ strategies are similarly defined. The lower left quadrant contains results which enable a comparison to be made between the expenditures of each strategy for different limiting states.

On the right side of the table (the columns correspond to the Wald, Laplace, Hurwitz, and Savage criteria) are listed the expenditures and risk consistent with the meaning of the criterion indicated by the column designation. For example, the number at the intersection of row ⟪1.2⟫ and the Laplace column shows that the expenditure for strategy ⟪1.2⟫, averaged over all possible states of the medium, equals 79.5 monetary units; the number at the intersection of the ⟪Hurwitz⟫ row and Savage column shows that the maximal risk of the ⟪Hurwitz⟫ strategy amounts to 2.4 monetary units. The remaining data of the table are similarly interpreted.

Let us compare the first-approximation strategies. The best of these, according to the Wald criterion, are the ⟪1.1⟫ and ⟪1.7⟫ strategies, with maximal expenditures 85.8. The equality of these indicators means that strategies ⟪1.1⟫ and ⟪1.7⟫ are economically equivalent according to the Wald criterion. Continuing this analysis for the remaining criteria, we discover that the ⟪1.1⟫, ⟪1.2⟫, ⟪1.7⟫, and ⟪1.8⟫ strategies are economically equivalent according to the Laplace criterion; the ⟪1.1⟫, ⟪1.7⟫, and ⟪1.8⟫ strategies are equivalent according to the Hurwitz criterion; and the ⟪1.2⟫ and ⟪1.8⟫ strategies are economically equivalent according to the Savage criterion. From this we conclude that in the first approximation there is no strategy which is

TABLE 4. Indicators of the Structure of Nuclear Power Engineering

Strategy	Optimal year for introducing breeder reactors	Share of breeder reactors	
		end of 1990s	beginning of 21st century
≪Savage≫	2001	—	10
≪Laplace≫	1996	20	40

absolutely best when compared with the others according to all of the optimality criteria under the conditions of uncertainty. The ⟪1.2⟫ strategy is optimal according to two criteria, the ⟪1.1⟫, ⟪1.7⟫, and ⟪1.8⟫ strategies are optimal according to three, while ⟪1.2⟫ is worse than ⟪1.8⟫ with respect to the whole group of criteria. There remain three strategies which are optimal under conditions of uncertainty in the first approximation: ⟪1.1⟫, ⟪1.7⟫, and ⟪1.8⟫. A comparison of the corresponding structures according to the results of the calculations shows that there is little difference between them. Their characteristic common indicators are the following: The optimal time for introduction of breeder reactors is in the mid-1990s, their share in the structure of nuclear power engineering at the end of the 1990s is about 25%, amounting to about 40% around the middle of the first decade of the next century.

Using the results contained in the lower right-hand corner of Table 3, let us now compare the ⟪Wald⟫, ⟪Laplace⟫, ⟪Hurwitz⟫, and ⟪Savage⟫ strategies. We can exclude from further consideration the ⟪Hurwitz⟫ strategy, since according to all criteria it has worse indicators than the ⟪Laplace⟫ strategy. If we proceed in a strictly formal manner, we cannot set up any order of preference among the remaining three strategies (⟪Wald⟫, ⟪Laplace⟫, and ⟪Savage⟫). However, such a formal approach seems excessively dogmatic, and the analysis of the strategies can be taken somewhat further. Let us really compare ⟪Wald⟫ and ⟪Laplace⟫. According to the Hurwitz criterion, they are the same. The second strategy is somewhat better than the first (about 0.13%) according to the Laplace criterion and is inferior to the first (by about 0.23%) according to the Wald criterion. However, ⟪Laplace⟫ is considerably better than ⟪Wald⟫ (by 19%) according to the Savage criterion. Because of this, we will henceforth consider only two strategies; ⟪Laplace⟫ and ⟪Savage⟫. They are roughly the same according to the Wald and Laplace criteria, although according to the Hurwitz and Savage criteria the first strategy is slightly better (about 0.1%) and they are different. According to the Hurwitz criterion, the difference is to the advantage of the ⟪Laplace⟫ strategy (amounting to about 0.7%), but the maximal value of the risk is 60% greater than in the ⟪Savage⟫ strategy. This difference apparently gives the preference to the ⟪Savage⟫ strategy (Table 4). To bring out additional considerations in support of the ⟪Savage⟫ strategy, we can compare the indicators for different limiting states (the lower left-hand quadrant of Table 3). Analysis shows that the ⟪Laplace⟫ strategy is more economical than the ⟪Savage⟫ strategy for the 1.5, 1.6, 1.7, and 1.8 states, which are favorable for the development of fast reactors. On the other hand, it is inferior to the latter, with the unfavorable conditions found in the 1.3 and 1.4 states, which appear to be more likely at present. The ⟪Savage⟫ strategy must therefore be regarded as optimal within the scope of the present model and method used above. Comparison with the first-approximation strategies shows that the solutions and recommendations adopted in accordance with their fundamental structural characteristics will be intrinsically different. This completely justifies the comparatively small additional costs involved in applying the more complicated, yet more precise method for optimizing under conditions of uncertainty.

Solution of Problems of the Second Type. In case there is considerable indeterminacy not only in the vector **c**, but in the remaining exogenetic parameters of the model as well, the present method is not applicable. One must approach the formulation of the problem with a somewhat different attitude. We note that although the model of nuclear power engineering described by Eqs. (1) was dynamic, time was not properly taken into account. In reality, time is nonuniform with respect to the problem of forecasting. In fact, although the forecast encompasses 50 years, from a practical point of view the solutions of principal interest are those which must be adopted in the time interval closest to the moment at which the forecast is made. But in the course of time, new solutions may occur, and at the moment of making the forecast one should not adopt solutions for the whole subsequent period of time. In a method for taking the indeterminacy factor into account, it is therefore useful to consider the possibility of adopting solutions in the future. In addition, it is necessary to take into account the fact that any solutions are adopted in a hierarchically connected system and must therefore be adopted at their own hierarchical level. A method allowing for these requirements and characteristics of dynamical development has been devised and discussed in the literature [11, 13]. The next task is to apply it in predicting the development of nuclear power engineering.

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METHOD OF COMPUTING LARGE PERTURBATIONS OF REACTIVITY BY DIFFERENCE ITERATIONS IN THE MONTE CARLO METHOD

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In calculating reactivity perturbations by the Monte Carlo method it is desirable to use information obtained earlier regarding the neutron flux and quality distribution in the unperturbed reactor. However, known methods of calculating large reactivity perturbations [1, 2] reduce to one method: Even when one has complete information on the unperturbed reactor, the methods require the asymptotic distribution of neutron flux or quality to be established in the perturbed reactor, i.e., they require the homogeneous problem to be solved. Only after this is done is the desired reactivity perturbation calculated using the correlation sample [3], correlation weights [4, 5], or even, the so-called "perturbation source" [2, 6, 7], i.e., the nonuniform population equation is solved with a source obtained by solving the homogeneous equation. In the final analysis the error in the calculated reactivity perturbation is determined by the error in solving both the inhomogeneous and the homogeneous problems. Therefore, in particular, the efficiency of calculating large reactivity perturbations drops sharply with increase in the degree of localization of the perturbation in the parameters, even though local estimates are used in solving the inhomogeneous problem.

The present paper describes a method for calculating large reactivity perturbations due to variation in the macroscopic constants; the method does not require asymptotic distributions in the neutral flux or quality in the perturbed reactor to be modeled by the Monte Carlo method. In addition, when the three-dimensional distribution of sources or of the quality of the fission neutral spectrum (FNS) in the unperturbed reactor and the unperturbed value of the effective neutron multiplication coefficient K_{ef} are given, the difference in the asymptotic distributions of sources, or correspondingly, in the quality of the FNS in the perturbed and the unperturbed reactors is modeled directly. The desired reactivity perturbation is calculated during modeling of this difference distribution.

For the unperturbed reactor let K_{ef} and the three-dimensional distribution of FNS sources $Q(\mathbf{r})$ be given, satisfying the critical equation [8] conventionally:

$$K_{ef} Q(\mathbf{r}) = \int Q(\mathbf{r}') P(\mathbf{r}' \to \mathbf{r})\, d\mathbf{r}', \quad (1)$$

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where $P(\mathbf{r}' \to \mathbf{r})$ is the FNS number, appearing at the point $\mathbf{r}$ for fissions induced by neutrons, which originated from a single FNS, launched at the point $\mathbf{r}'$ (the FNS Green function).

The asymptotic distribution $Q'(\mathbf{r})$ of FNS sources in the perturbed reactor satisfies the analogous equation

$$K'_{ef} Q'(\mathbf{r}) = \int Q'(\mathbf{r}') P'(\mathbf{r}' \to \mathbf{r})\, d\mathbf{r}', \tag{2}$$

where $P'(\mathbf{r}' \to \mathbf{r})$ is the FNS Green function of the perturbed reactor and K'_{ef} is the perturbed effective neutron multiplication coefficient. The integration is carried out over the reactor volume.

Multiplication of Eqs. (1) and (2) by a certain weight function $G(\mathbf{r})$, belonging to a class of functions in which the integral operators of Eqs. (1) and (2) are defined, integration over the reactor volume, and subtraction of one equation from the other leads to the relation:

$$\Delta K_{ef} \equiv K'_{ef} - K_{ef} = \frac{\langle\langle Q' P' G\rangle\rangle - \langle\langle QPG\rangle\rangle - K_{ef} \langle \Delta QG\rangle}{\langle Q'G\rangle}, \tag{3}$$

where the angle brackets denote integrals of the type

$$\langle QG\rangle = \int Q(\mathbf{r})\, G(\mathbf{r})\, d\mathbf{r};$$

$$\langle\langle QPG\rangle\rangle = \int\int Q(\mathbf{r}')\, P(\mathbf{r}' \to \mathbf{r})\, G(\mathbf{r})\, d\mathbf{r}'\, d\mathbf{r},$$

and the function $\Delta Q(\mathbf{r}) = Q'(\mathbf{r}) - Q(\mathbf{r})$.

The concept of the difference iteration method is based on representing the successive neutron generations [8] and is as follows:

at some instant of time in the unperturbed and perturbed reactors the zero generation FNS sources are switched on, with the same source density $Q_0(\mathbf{r}) = Q'_0(\mathbf{r}) = Q(\mathbf{r})$;

the difference distribution $\Delta Q_1(\mathbf{r})$ is modeled, which describes the difference between the first generation FNS sources $Q'_1(\mathbf{r})$ and $Q_1(\mathbf{r})$, due to the action of the perturbation in the zero generation:

$$\Delta Q_1(\mathbf{r}) = \int Q(\mathbf{r}')\, \Delta P(\mathbf{r}' \to \mathbf{r})\, d\mathbf{r}',$$

where $\Delta P(\mathbf{r}' \to \mathbf{r}) = P'(\mathbf{r}' \to \mathbf{r}) - P(\mathbf{r}' \to \mathbf{r})$;

the difference distribution ΔQ_1 obtained is taken as the initial distribution in an ordinary iteration process in the perturbed reactor:

$$\Delta Q_{N+1}(\mathbf{r}) = \int \Delta Q_N(\mathbf{r}')\, P'(\mathbf{r}' \to \mathbf{r})\, d\mathbf{r}',$$
$$N = 1, 2, 3, \ldots \tag{4}$$

In Eq. (4) it is enough to perform $N = N_{as}$ iterations, where N_{as} is the number of neutron generations after which the asymptotic distribution $Q'_N(\mathbf{r})$ (but not $\Delta Q_N(\mathbf{r})$!) of FNS sources would be established with the required accuracy in the iteration process in the perturbed reactor:

$$Q'_{N+1}(\mathbf{r}) = \int Q'_N(\mathbf{r}')\, P'(\mathbf{r}' \to \mathbf{r})\, d\mathbf{r}', \quad Q'_0(\mathbf{r}) = Q(\mathbf{r}),$$
$$N = 0, 1, 2, \ldots \tag{5}$$

The value of N_{as} is determined by the given accuracy in calculating ΔK_{ef}. All the functionals of the asymptotic distributions $Q(\mathbf{r})$ and $Q'(\mathbf{r})$, appearing in Eq. (3) for ΔK_{ef}, are expressed in terms of the distribution $\Delta Q_N(\mathbf{r})$ in Eq. (4).

It can be shown that for any N

$$Q'_N(\mathbf{r}) - Q_N(\mathbf{r}) = \sum_{n=0}^{N} K_{ef}^{N-n} \Delta Q_n(\mathbf{r}),$$
$$\Delta Q_0(\mathbf{r}) = 0, \quad N = 0, 1, 2, \ldots;$$
$$Q_N(\mathbf{r}) = K_{ef}^N Q(\mathbf{r}).$$

Substitution in Eq. (3) of the asymptotic distribution $Q'_N(\mathbf{r})$ for $N \geq N_{as}$, following some elementary transformations, gives

$$\Delta K_{ef} = \frac{K_{ef}^{-N} I_{N+1}}{I + \sum_{n=0}^{N} K_{ef}^{-n} I_n}, \tag{6}$$

where $I = \langle QG \rangle$ and $I_n = \langle \Delta Q_n G \rangle$. Because of the asymptotic nature of the distribution Q'_N for $N \geq N_{as}$, the value of ΔK_{ef}, determined by Eq. (6), is independent of $N \geq N_{as}$, and therefore, for an increase in the neutron population number, only the normalized distribution $Q'_N(\mathbf{r})$ varies, determined with an accuracy up to a multiplying factor.

For $N < N_{as}$ Eq. (6) gives an approximation of varying level. For example, for $N = 0$ we obtain the so-called small perturbation formula, corresponding to the assumption $Q'(\mathbf{r}) = Q(\mathbf{r})$:

$$\delta K_{ef} = \langle\langle Q \Delta P G \rangle\rangle / \langle QG \rangle = I_1 / I, \tag{7}$$

i.e., the first-order perturbation theory formula. For $N = 1, 2, 3, \ldots$ we obtain perturbation theory formulas of order 2, 3, 4, etc.

It should be noted that the distribution of FNS sources $Q(\mathbf{r})$, generally speaking, is a weaker function of the perturbation than is the differential neutron flux. Therefore, Eq. (6) of perturbation theory of order $(N+1)$ usually corresponds to classical perturbation theory formulas of higher order. For example, perturbations that are small in the sense of it being possible to neglect the difference between the asymptotic distributions of $Q(\mathbf{r})$ and $Q'(\mathbf{r})$ of the FNS sources may not be negligible in classical perturbation theory, where the smallness of a perturbation is reckoned from the possibility of neglecting the difference in the asymptotic distributions of differential flux or differential neutron quality.

As a weight function $G(\mathbf{r})$ it is desirable to choose a function which is close to the distribution $Q^{+'}(\mathbf{r})$ of the FNS in the perturbed reactor. A choice of $G(\mathbf{r}) = Q^{+'}(\mathbf{r})$, according to [8], would lead to convergence in Eq. (6) even in the first iteration, i.e., Eq. (7) would be exact. Since knowledge of $Q^{+'}(\mathbf{r})$ is equivalent to knowledge of K_{ef}, in practice one must make do with some a priori information on this distribution. Frequently a good approximation is the distribution $Q^{+}(\mathbf{r})$ of FNS quality in the unperturbed reactor, which satisfies the equation in [8] . In this special case of the choice of weight function, the original Eq. (3) takes the simple form

$$\Delta K_{ef} = \langle\langle Q' \Delta P Q^{+} \rangle\rangle / \langle Q' Q^{+} \rangle. \tag{8}$$

Substitution into Eq. (8) of the asymptotic distribution $Q'_N(\mathbf{r})$ with $N \geq N_{as}$ leads to

$$\Delta K_{ef} = \frac{I_1 + \sum_{n=0}^{N} K_{ef}^{-n} \Delta I_n}{I + \sum_{n=0}^{N} K_{ef}^{-n} I_n}, \tag{9}$$

where $\Delta I_n = \langle\langle \Delta Q_n \Delta P Q^{+} \rangle\rangle$.

For $N < N_{as}$ Eq. (9), as was true for Eq. (6), gives an approximation of different level, and for $N = 0$ it coincides with Eq. (7).

The distribution $Q^{+}(\mathbf{r})$ can be calculated as either a direct sample [7] or a conjugate sample [9] (if it is not known beforehand). Additional time spent in calculating $Q^{+}(\mathbf{r})$ is devoted to calculating a series of perturbations due to the fast convergence of the difference iterations.

Equation (9) is, in fact, another form of Eq. (6) with $G(\mathbf{r}) = Q^{+}(\mathbf{r})$. However, when the Monte Carlo method is used, the form in which the functional is written frequently dictates the necessary or most desirable algorithm for calculating it. For example, for calculating the functional I_{N+1} in the numerator of Eq. (6), one must consider the trajectories of all neutrons of the N-th generation ($N \geq 1$) generated by the source $\Delta Q_N(\mathbf{r})$, whether they passed through the perturbed region in the given generation or not. But the numerator of Eq. (9) should be calculated by separating the main part of I_1 and using correlation technology, the correlation sample, correlation weights, or the perturbation source, to estimate the corrections ΔI_n of higher order of smallness, in tracking neutron trajectories in the perturbed reactor from the source $\Delta Q_1(\mathbf{r})$ and the daughter sources $\Delta Q_n(\mathbf{r})$. Here the contributions to the corrections only come from neutrons which have passed through the perturbed region in the n-th generation.

As a result of dispersion of the estimate ΔK_{ef} from Eq. (9), the dispersion will be less than that in the estimate using Eq. (6), for $G(\mathbf{r}) = Q^{+}(\mathbf{r})$ and $N \geq 1$.

In fact, the mathematical expectation of the sum of contributions from neutrons which have not passed through the perturbation region, in the estimate I_{N+1} is approximately I_1, while the dispersion of this sum is

larger than that in the estimate of the functional I_1, since it includes terms from the dispersion of the random sample and the source dispersion $\Delta Q_N(\mathbf{r})$. The dispersion in the sum of the contributions of neutrons which have passed through the perturbation region is usually larger than that in the estimate of the sum obtained with the correlation methodology.

If one does not use correlation methods, the dispersions in the estimates of ΔK_{ef} from Eqs. (6) and (9) with $G(\mathbf{r}) = Q^+(\mathbf{r})$ coincide.

It is desirable to use Eq. (9) for $G(\mathbf{r}) \approx Q^+(\mathbf{r})$, e.g., when $Q^+(\mathbf{r})$ is given as a histogram. In this class of case Eq. (9) gives an estimate of ΔK_{ef} with a bias equal to the difference between Eqs. (6) and (9). This bias is usually less than the error in estimating ΔK_{ef}, even for the choice $G(\mathbf{r}) = Q(\mathbf{r})$, but it is difficult to obtain an estimate by the Monte Carlo method, since it is the difference $(\Delta K_{ef} - \Delta \tilde{K}_{ef})$ between two close and weakly correlated quantities.

The difference iteration method makes it possible to efficiently calculate large reactivity effects from local perturbations. In fact, by special combinations of direct and conjugate samples of neutrons one can successfully pass all the zero-generation trajectories through the perturbation region. Here all the trajectories contribute to the model distribution $\Delta Q_1(\mathbf{r})$ and the functional I_1, and therefore, also to the production of all the successive N_{as} neutron generations. Even if the conjugate sample is not used in the zero generation, in the first and subsequent generations one tracks only the history of neutrons which contributed to $\Delta Q_1(\mathbf{r})$ and I_1 in the zero generation, i.e., which carry information on the perturbations.

The name of the difference iteration method stems both from the fact that it is based on the idea of iterating the difference distribution $\Delta Q_1(\mathbf{r})$ formed in the zero generation, and also from the fact that the distributions $\Delta Q_N(\mathbf{r})$ are differences in the distributions in two successive generations of the nonlinear process

$$Q^*_{N+1}(\mathbf{r}) = \int Q^*_N(\mathbf{r})\,[P'(\mathbf{r}' \to \mathbf{r}) + (1 - K_{ef})\,K_{ef}^{-N}\delta(\mathbf{r}' - \mathbf{r})]\,d\mathbf{r}',$$

$$Q^*_0(\mathbf{r}) = Q(\mathbf{r}), \quad N = 0, 1, 2, \ldots,$$

which for $K_{ef} = 1$ is simply Eq. (5) for establishing the asymptote in the perturbed reactor, i.e.,

$$\Delta Q_N(\mathbf{r}) = Q^*_{N+1}(\mathbf{r}) - Q^*_N(\mathbf{r}).$$

Quite analogously, we can construct difference iterations based on the perturbed and unperturbed conjugate equations for the quality $Q^{+'}(\mathbf{r})$ and $Q^+(\mathbf{r})$, using a conjugate sample of neutrons as basic, and applying a direct sample in the zero generation with local perturbations. The resulting formulas for ΔK_{ef} and δK_{ef} will have the same form as in Eqs. (6), (7), and (9), if we understand the following integrals for those functionals appearing in them: $I = \langle Q^+ G \rangle$, $I_n = \langle G \Delta Q_n^+ \rangle$, $\Delta I_n = \langle\langle G \Delta P \Delta Q_n^+ \rangle\rangle$, and in Eq. (9) $G(\mathbf{r}) = Q(\mathbf{r})$. Generally speaking, convergence of the difference iterations in neutron quality will be different.

The method of difference iterations for FNS sources was developed in 1975 in a group of PIR programs, written in translator input language with ALGOL TA-1M for the M-222 BÉSM-4 computers. Within the framework of application of the correlation weight method, the program calculates arbitrary variations in reactivity in reactors of R − Z geometry in the multigroup transport approximation with isotropic transitions. Local reactivity effects are calculated using a special technique for linking the direct and conjugate neutron samples. The PIR complex can also calculate the distribution of quality (using the technique of [7]) and NFS sources in histogram representation. The complex is attached to the ARAMAKO catalog [10] through the ARMONT complex [11], which has 26-group macroscopic zone constants. The time for calculating a single neutron trajectory depends on the average number of collisions, i.e., on the specific model. The average speed of calculation is about 0.03 sec per collision.

The PIR complex was checked on different models by comparing with data obtained by the S_N method and the method of integration with respect to a parameter [12] (two-dimensional one-group tests), the diffusion program method (six-group R−Z model [13], separate calculations and by the small perturbation theory), and by the ARMONT and the MMK-2 complexes [14, 15] (26-group R−Z models).

The results of verification of the complex and the experience accumulated of using it to solve practical problems indicate that: 1) to reach an accuracy of 5-10% in the reactivity effect (within the n-th approximation adopted) one requires to track $\sim N \cdot 10^3$ neutron histories; 2) one is practically never required to examine approximations on the order $N > 3$; 3) the statistical approach for calculating perturbations is concurrent in the sense of expenditure of machine time and the use of computer resources, even in two-dimensional problems in the case where the diffusion approximation is applicable; 4) it is most efficient to use difference iterations to estimate reactivity effects from local perturbations, allowing for distortion in the source distribution.

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CORROSION PRODUCTS IN MAIN TECHNOLOGICAL SYSTEMS OF ATOMIC POWER PLANTS WITH AN RBMK-1000 REACTOR DURING OPERATION

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The chemical conditions of the system of an atomic power plant are determined by the set of specific properties of the aqueous working medium and of the structural materials, the operating conditions, and their reciprocal influence. Analysis of the chemical conditions in the technological system consists of an evaluation of the corrosion state of the materials of the system and observations of the parameters of the aqueous medium.

The present paper is devoted to a study of the corrosion state of the surface of structural materials of the systems and equipment of an atomic power plant with an RBMK-1000 reactor. In the study samples were taken directly from the system (during the operation of the atomic power plant) and from the outer surface of the equipment (during shutdown after 21,000 h of operating at power). The samples of deposits were taken in five segments (Table 1) of the main circuit of the atomic power plant (see Fig. 1): in the drum-separator, in the multiple forced-circulation circuit (MFCC), in the steam circuit (steam pipes, high- and low-pressure cylinders (HPC and LPC)), and in the condensate-decontamination circuit (deaerator, mechanical filter).

The chemical isotopic and phase composition were determined in order to identify the compounds in the samples (Tables 2 and 3). Techniques for determining the chemical composition were described earlier [1, 2]. The phase composition of low-activity deposits was determined on a URS-50 NM x-ray apparatus with an SSD stand and the phase analysis of the radioactive deposits was made by the γ-ray resonance method on a YaGRS-4 spectrometer [3, 4].* The spectra were taken in a transmission geometry in a constant rate mode. The en-

*For more details of the technique and some results of this analysis, see the paper by A. I. Moskvin et al., (this issue, p. 27).

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TABLE 1. Characteristics of Sampling Segments

Site of deposit sampling	Material of equipment	Medium and its parameters
Drum-separator*	Steel 22K, ÉI-898 (plating)	Desalinated water, steam–water mixture (moisture content 14-17%), steam (moisture content 0.1%), $p = \delta^2$ 7 MPa, t =284.5°C
"Shadr" flowmeter	Kh18N10T	Desalinated water, p = 7 MPa, $t = \delta^2$ 280°C
EFPT†	Kh18N10T	Desalinated water, t = 50-70°C, contact with nitrogen p = 0.15 MPa
Steam pipes emerging from drum-separator	Grade 20 steel	Steam (moisture content 0.1%), t = 280°C, p = 5.5 MPa
HPC, third stage	Rotor 43KhM1A steel, vane 1Kh13 steel, diaphragms 0Kh13 steel	Steam, p = 0.53 MPa, t = 160°C
HPC, fifth stage	Ditto	Steam, p = 0.35 MPa, t = 138°C, moisture content 15.9%,
Deaerator‡	Walls grade 3 steel, end plates grade 20 steel	p = 0.6 MPa, t = 164°C
Mechanical feedwater filter	Body grade 20 steel, filter cartridge steel Kh17N13M3T	t = 165°C

*Surface area 230 m^2.
†280 m^2.
‡165 m^2.

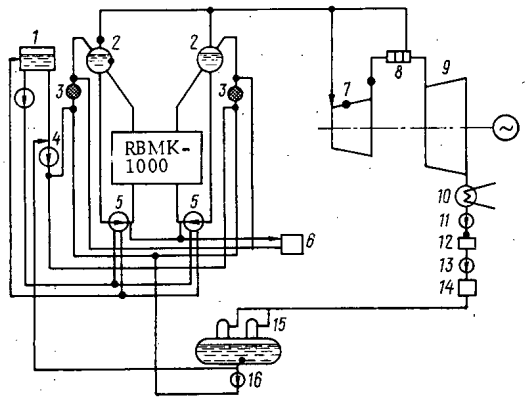

Fig. 1. Diagram for sampling deposits from equipment of atomic power plant with RBMK-1000 reactor: 1) EFPT; 2) separators; 3) mechanical filters; 4) emergency feed pump; 5) TsÉN-7 pumps; 6) bypass decontamination; 7) HPC; 8) separators–steam preheaters (SSP); 9) LPC; 10) condensers; 11, 13) condensate pumps CPI and CPII; 12) condensate decontamination; 13, 14) low-pressure preheaters; 15) deaerator; 16) electric feed pumps; ●) sampling site.

TABLE 2. Chemical and Phase Composition of Loose Corrosion Products from Surfaces of Equipment of RBMK-1000 Reactor

Sampling site	Phase composition, wt. %				Chemical composition, wt. % in sample							Insoluble residue
	Fe_3O_4	γ-Fe_2O_3	α-Fe_2O_3	γ-FeOOH	Fe	Cr	Ni	Mn	Cu	Zn	Si	
Drum-separator (from control samples)	54	46	—	—	49.7	1.85	1.8	1.23	0.07	—	—	—
Drum-separator (from bottom of apparatus)	—	—	—	—	44.6	1.64	0.52	0.43	0.43	0.13	—	18.9
EFPT	—	—	—	—	46.2	1.65	0.6	0.6	0.3	0.1	—	4.0
Suspensions in coolant (from drum-separator)	30	—	70	—	—	—	—	—	—	—	—	—
Steam pipes of drum-separator	30	—	70	—	65.0	0.09	0.014	0.56	0.14	0.078	0.003	—
Steam pipe before check-regulating valve	—	—	100	—	60.3	0.32	0.12	0.49	0.1	0.1	—	—
Diaphragm of third stage of HPC	—	—	100	—	67.8	0.13	0.006	0.011	0.07	0.091	0.7	—
Steam pipe of HPC – SSP (at steam outlet from fifth stage of HPC)	15	15	70	—	67.3	0.088	0.059	0.014	0.085	0.20	1.6	—
Steam pipe of HPC – SPP: upper layer of deposits (black)	70	—	30	—	73.6	0.03	—	0.004	—	—	—	—
lower layer of deposits (cherry-colored)	—	—	100	—	68.1	0.1	—	0.003	—	—	—	—
LPC, at steam outlet from fifth stage (black)	70	20	—	10	60.5	0.1	—	0.17	—	—	—	—
LPC, at steam outlet from fifth stage (reddish-brown deposits)	—	—	—	100	49.2	0.05	—	0.64	—	—	—	—
Suspension from condensate decontamina.	39	15	17	24	60	0.3	0.05	0.3	3.4	—	—	—
Deaerator tank	28	30	42	—	66.4	0.09	0.05	0.013	0.016	0.002	—	—
Mechanical feedwater filter	—	—	100	—	65.9	0.27	0.1	0.13	0.01	0.1	—	—

TABLE 3. Radiochemical Composition of Corrosion Products, Ci/g

Radionuclide	Sampling site						
	drum-separator (from bottom of apparatus)	steam pipe ahead of HPC (ahead of check-regulating valve)	from diaphragms of HPC	steam pipe of HPC – SSP at steam outlet from fifth stage of HPC	steam pipe after LPC	deaerator	Mechan. filter for feedwater decontamin.
^{51}Cr	$(9,4\pm1,3)\cdot10^{-4}$	—	$(8,0\pm2,8)\cdot10^{-8}$	—	—	$(1,9\pm0,3)\cdot10^{-8}$	—
^{54}Mn	$(2,2\pm0,7)\cdot10^{-5}$	—	—	$(4,7\pm2,8)\cdot10^{-9}$	—	$(9,7\pm1,1)\cdot10^{-9}$	$(1,5\pm0,7)\cdot10^{-7}$
^{59}Fe	—	—	$(9,2\pm3,8)\cdot10^{-9}$	—	—	$(2,2\pm0,9)\cdot10^{-9}$	—
^{58}Co	$(4,4\pm0,9)\cdot10^{-5}$	—	$(5,0\pm2,8)\cdot10^{-9}$	—	—	$(1,9\pm0,5)\cdot10^{-8}$	$(1,9\pm0,9)\cdot10^{-7}$
^{60}Co	$(7,2\pm0,9)\cdot10^{-5}$	$(5,1\pm3,3)\cdot10^{-10}$	$(1,4\pm0,3)\cdot10^{-8}$	$(2,4\pm0,4)\cdot10^{-8}$	—	$(1,7\pm0,2)\cdot10^{-8}$	$(3,5\pm0,7)\cdot10^{-7}$
^{65}Zn	$(9,4\pm1,7)\cdot10^{-5}$	—	$(2,3\pm0,6)\cdot10^{-8}$	$(3,7\pm0,8)\cdot10^{-8}$	—	$(6,1\pm1,2)\cdot10^{-9}$	$(4,5\pm2,0)\cdot10^{-7}$
^{95}Zn	$(4,7\pm0,5)\cdot10^{-4}$	—	$(2,4\pm0,7)\cdot10^{-8}$	$(2,8\pm0,6)\cdot10^{-8}$	$(2,8\pm1,8)\cdot10^{-9}$	$(2,7\pm0,9)\cdot10^{-9}$	$(1,1\pm0,2)\cdot10^{-6}$
^{95}Nb	$(9,1\pm0,9)\cdot10^{-4}$	$(3,0\pm0,9)\cdot10^{-9}$	$(2,9\pm0,7)\cdot10^{-8}$	$(5,6\pm0,8)\cdot10^{-8}$	$(3,0\pm2,0)\cdot10^{-9}$	$(6,2\pm1,1)\cdot10^{-9}$	$(2,0\pm0,3)\cdot10^{-6}$
^{103}Ru	$(4,1\pm1,2)\cdot10^{-5}$	—	—	—	—	—	$(1,8\pm1,1)\cdot10^{-7}$
^{106}Ru	$(4,7\pm0,9)\cdot10^{-4}$	—	—	—	—	$(7,4\pm3,7)\cdot10^{-9}$	$(2,0\pm0,7)\cdot10^{-6}$
^{136}Cs	—	—	—	—	—	$(1,3\pm0,8)\cdot10^{-9}$	—
^{137}Cs	—	—	$(2,7\pm1,6)\cdot10^{-9}$	—	$(1,2\pm0,7)\cdot10^{-9}$	—	—
^{140}Ba	—	—	—	—	$(1,2\pm0,4)\cdot10^{-8}$	$(7,5\pm1,4)\cdot10^{-9}$	—
^{141}Ce	$(2,1\pm0,9)\cdot10^{-5}$	—	$(6,3\pm0,8)\cdot10^{-8}$	$(1,7\pm0,3)\cdot10^{-8}$	—	$(8,5\pm1,0)\cdot10^{-9}$	$(2,5\pm1,0)\cdot10^{-7}$
^{144}Ce	$(9,5\pm4,8)\cdot10^{-4}$	—	—	$(1,2\pm0,2)\cdot10^{-7}$	$(7,2\pm0,3)\cdot10^{-9}$	—	$(3,2\pm0,5)\cdot10^{-6}$
Total	$(4,1\pm0,2)\cdot10^{-3}$	$(2,6\pm0,1)\cdot10^{-8}$	$(4,9\pm0,3)\cdot10^{-7}$	$(3,6\pm0,3)\cdot10^{-7}$	$(5,9\pm0,6)\cdot10^{-8}$	$(1,7\pm0,1)\cdot10^{-7}$	$(1,2\pm0,1)\cdot10^{-5}$

ergy spectrum of the deposits was recorded with a DGDK-40 semiconductor detector with an intrinsic resolution of 2.7 keV for the 1332-keV γ line of ^{60}Co with an AI-4096-ZMV-100 analyzer. The spectra were analyzed on an M-6000 ASVT by procedures described earlier [5].

The data from visual examination of equipment surfaces are given in Table 4. The GTsN delivery pipes displayed corrosion damage in the form of pits and pockholes up to 2 mm deep, situated along the periphery in the region of weld seams. The pockholes are coated with a dense oxide film ranging from light brown to black in color; black oxides also appear at places where the surface of the pipes has been treated mechanically

TABLE 4. Corrosion Condition of Equipment Surfaces

Equipment	Characteristics of surface condition
Drum-separator	Surface of manhole cover has no damage after removal of loose deposits. The bulk of the corrosion products on the bottom of the apparatus.
"Shadr" flowmeter	Covered with dense, thin, gray and black film tightly adhering to metal.
EFPT	Considerable quantity of dark brown deposits on bottom of tank. Tank walls are 70% covered with loose corrosion products, located mostly in the region of weld seams.
Steam pipes of drum-separator	Surface completely covered with fine pits up to 7 mm in diam. and up to 0.5 mm in depth. Waterlines appear in horizontal segments of pipes. Surface of lower part damaged more intensively than surface above waterline. Most large pits on surfaces of pipes are in lower part.
Steam pipe before check-regulating valve	Superficial brown deposits adhering tightly to surface of metal. No visible flaws can be seen on surface after removal of deposits.
Steam pipe (receiver) of HPC–SSP (4 m from fifth stage of HPC)	Dense two-layer superficial deposits on walls of steam pipe. Upper layer consists of black corrosion products which flake off in scales of up to 0.2-0.3 mm thick. The cherry-colored bottom layer adjoining the metal reaches a thickness of 0.5 mm. The surface under the deposits has no visible corrosion damage.
HPC	When the loose brown deposits are removed the steam pipe leading to the HPC shows no visible damage. Surfaces of steam pipes and diaphragms of HPC are smooth, without traces of corrosion damage. The packing segments and their mounting springs are covered with dense black oxide films coated with a loose brick-covered red deposit. The vanes of the first and second stages are covered with pockholes and pits as a result of metal erosion. The vanes of the fourth and fifth stages have a dense oxide film on the side opposite to the steam flow. At the steam outlet from the HPC the surface of the steam pipe and guide vanes are covered with pits of different shapes with even vertical edges, diam. of 2 to 20 mm, and depth of up to 1 mm. At the bend in the steam pipe (with a change in direction of flow) the surface is shiny, as if glazed.
LPC	The surface of the turbine housing is covered with loose deposits ranging from reddish brown to black in color. The surface of the trailing edges of the rotor vanes is honeycombed with numerous pits and spots.
Deaerator	In the deaerator tank a waterline dividing the steam and water phases appears on the walls at a distance of 0.2 diam. from the bottom. Below the waterline the surface is reddish brown, monochromatic, and completely covered with pits and spots up to 1 mm in diam. and 0.5 mm in depth. Above the waterline there are loose brick-red deposits under which fine spots and pits with a diam. of 2-3 mm are observed. The condition of the weld seams does not differ from that of the surface of the tank walls. The surfaces of the deaerator column, delivery steam pipes, and grids are coated with dark (greenish-violet iridescent) dense oxide film.
Mechanical feedwater filters	The filter covers, housing walls, and guide vanes of filter cartridges are covered with brick-colored corrosion products. Pits with a diam. of up to 1-2 mm appear on the surface of the cover when the loose deposits are removed. There are very many corrosion spots, as a result of which the surface is rough. The stainless steel parts bear no traces of corrosion damage.

A considerable quantity of loose corrosion products accumulated in the drum-separator of the MFCC, concentrated primarily on the bottom of the separator; the walls were coated with a thin layer of deposits. The surface of the manhole cover had no visible damage (after removal of the loose deposits). The emergency-feed pump tank (EFPT) had up to 70% of its surface covered with loose corrosion products. When the steam parameters in the steam circuit are reduced the composition of the loose corrosion products changes. Thus, two-layer deposits are observed even in the HPC; the blackish upper layer is a mixture of magnetite and hematite whereas the lower layer consists only of hematite (see Table 2).

According to the data from electrochemical polarization measurements [6], the continuous film on the housing of the "Shadr" flowmeter (MFC circuit) has a spinel composition: $(Fe, Ni)O-(Fe, Cr)_2O_3$. The loose corrosion products from the bottom of the drum-separator contain (in percent of weighed sample) 45-50 Fe, up to 2 Cr, up to 0.6 Ni, and up to 0.5 Mn.

In systems of pearlitic steels (grade 20, grade 3, and grade 22K steels) the deposits are due entirely to iron oxides (65-68%), those of other elements being < 1%. Phase analysis (see Table 2) shows that iron is present in the systems in the form of hematite α-Fe_2O_3 (0-100%), magnetite Fe_3O_4 (0-30%), maghemite γ-Fe_2O_3 (0-40%), and lepidocrocite γ-FeOOH (up to 24%). At places where there is increased buildup of loose corrosion products their phase composition is multicomponent (e.g., the condensate-decontamination filters contain Fe_3O_4, α-Fe_2O_3, γ-Fe_2O_3, γ-FeOOH). This indicates that oxides with different structures are formed under various conditions and are moved through the technological system in segments with reduced hydrodynamic characteristics. The conditions affecting the formation of different oxides can be changed both during a change in the operating conditions of the entire atomic power plant (transient conditions) and inside the system because of local differentials in the temperature, pressure, and concentration of the various substances. The largest quantity of loose corrosion products is observed in the deaerator and the drum-separator (see Tables 2 and 4).

Apparently, for the most part it is oxides, formed in the process of corrosion of the deaerator tank walls (grade 22K steel), which are accumulated in the deaerator. Moreover, the fine fraction of the suspended corrosion products as well as the colloidal particles pass through the condensate-decontamination apparatus and, partially growing larger during the motion, settle on the segment of the condensate-decontamination circuit from the deaerator to the drum-separator.

The buildup of enlarged suspensions occurs in the mechanical filters. Since the radioactivity of the suspensions trapped in the filter is high (10^{-5} Ci/g), then probably they are mainly corrosion products of the steam circuit which grew larger during the motion after shutdown of the condensate decontamination. The fine suspended particles which are not trapped by mechanical filters are borne along with the feedwater into the drum-separator. This is why there is a large buildup of corrosion products in it, although no traces of significant corrosion damage are observed on the examined surfaces of either the MFCC or the drum-separator.

The proportion of Ni, Mn, and Cr content in the sediments (see Table 2) is the same. The suspensions in the drum-separator of the RBMK are probably mainly corrosion products of the steam and condensate-decontamination circuits. The radioactivity of the corrosion products from the drum-separator and the mechanical filter is due to the same radionuclides.

The radiochemical composition of the loose deposits is given in Table 3. The radioactivity is due primarily to ^{51}Cr, ^{54}Mn, ^{60}Co, $^{95}Zr-^{95}Nb$, ^{144}Ce, and ^{106}Ru. Thus, the content of radionuclides in the corrosion products of the MFCC, steam, and condensate-decontamination circuits is, respectively: 20, 30, and 30% ^{124}Ce; 10, 7, and 10% ^{95}Zr; 10, 15, and 17% ^{106}Ru. The maximum specific radioactivity of $(4.1 \pm 0.2) \cdot 10^{-3}$ Ci/g is displayed by deposits in the drum-separator. As the distance of the sampling site from the reactor in the flowsheet increases, the specific radioactivity of the deposits diminishes to $(5.9 \pm 0.6) \cdot 10^{-8}$ Ci/g (steam pipe after LPC).

Analysis of the data obtained shows that there is an insignificant amount of deposits of only corrosion origin in the principal systems of an atomic power plant with an RBMK-1000 reactor. The phase composition of the corrosion products depends on the variations in the operating conditions of the system. The working surfaces of the system and the equipment are in a satisfactory state.

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MÖSSBAUER SPECTROSCOPIC DETERMINATION OF PHASE COMPOSITION OF CORROSION PRODUCTS OF STRUCTURAL MATERIALS OF PRIMARY CIRCUIT OF RBMK-1000 REACTOR WITH NEUTRAL WATER CONDITIONS

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When problems pertaining to the organization of water conditions of an atomic power plant, deactivation of equipment, and prediction of the radiation conditions are to be solved, there arise problems of ascertaining the contamination of the coolant and the surfaces of the structural material with their corrosion products. Basically, the corrosion deposits and suspension consist of a mixture of radionuclide-contaminated oxide compounds of iron whose form depends on the temperature, the pH, and the velocity of the coolant, the oxygen concentration in the coolant, the thermal flux, and the intensity of the radiation [1]. Analysis of the corrosion products is facilitated considerably by the application of nondestructive methods of analysis. The Mössbauer or nuclear γ-ray resonance method has a number of advantages. The Mössbauer spectra parameters have individual values (for a number of iron compounds) and it is thus possible, when analyzing such spectra, to determine the qualitative composition and initial structure of the corrosion products. The linear (within certain limits) dependence of the area of Mössbauer spectra on the iron content in one form or another makes it possible to directly determine the quantity of various structural components of corrosion deposits. In ordinary thermal power engineering, along with electron diffraction and x-ray structural analysis Mössbauer spectroscopy has recently attracted increasing attention [2-4]; the main advantage of this technique in comparison with x-ray structural analysis, e.g., lies in the great amount of information contained in the Mössbauer spectra and the possibility of automating their processing by computer. The application of the Mössbauer method in nuclear power engineering is complicated by the presence in the corrosion products of the reactor materials of a considerable quantity of long-lived radionuclides whose radiation results in a lower signal-to-noise ratio and causes difficulties in processing Mössbauer spectra. This is evidently explained by the fact that no published data are available on the application of Mössbauer spectroscopy to the study of corrosion products of reactor materials.

The corrosion products were sampled in various segments of the primary circuit of the V. I. Lenin Leningrad Atomic Power Plant (LAÉS): from the drum-separator, the condensate-decontamination apparatuses, the deaerator, and the system for drawing off live steam.* As the samples of corrosion products formed under the conditions of the drum-separator we took corrosion deposits on steel control specimens held in the coolant and the steam–water phase for 13,500 h. The control specimens in the coolant of the drum-separator constituted an assembly of plates of grade-20 steel and 08Kh18N10T steel, measuring 20 × 20 × 3 mm; in the steam–water phase the assembly consisted only of plates of 08Kh18N10T steel of the same size. In the course of reactor start-up samples were taken of suspended corrosion-product particles after a sample of coolant from the drum-separator had been allowed to settle for 24 h in a glass vessel. In the case of the deaerator and system for drawing off live steam the samples of corrosion deposits were scraped right off the surface with a stainless steel scraper. Samples of corrosion products from the condensate-decontamination apparatus were obtained by loosening the cation exchange resin with desalinated water. Under normal reactor operating conditions the temperature in the drum-separator is maintained at 285°C whereas under the conditions of condensate decontamination it varies from 30 to 40°C.

*Paper by V. M. Sedov et al. (this issue, p. 22).

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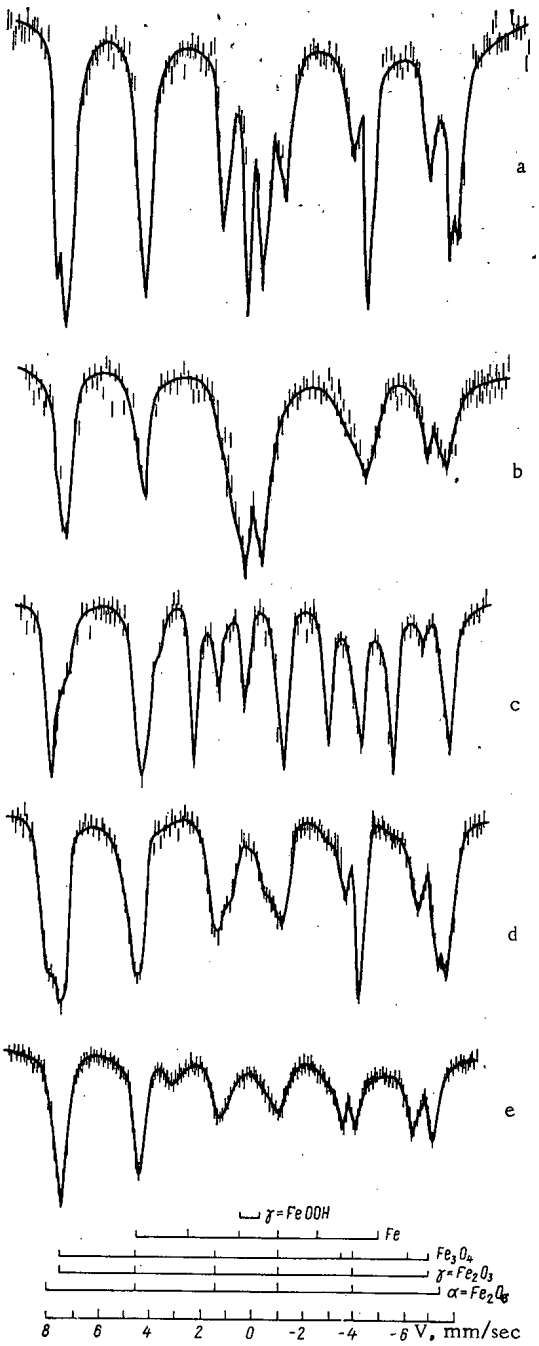

Fig. 1. Mössbauer spectra of some samples of corrosion products of structural materials of primary circuit of RBMK-1000 reactor, measured at 295°K: a, b) loose deposits from condensate decontamination (apparatuses 3 and 4, respectively); c) tightly adhering deposits on grade-20 steel in coolant in drum-separator; d, e) loose deposits from deaerator and system for drawing off live steam.

The samples of corrosion deposits were arbitrarily subdivided into loose and tightly adhering. The category of loose deposits comprised those deposits which flaked off the surface as the result of a sharp blow to the assembly of control specimens. The tightly adhering deposits were taken from the surface of the grade-20 steel with a scalpel after the loose deposits had been removed from it. All the operations conveying and grinding the corrosion-product samples and preparing absorbers from them for measuring the Mössbauer spectra were carried out with due observance of radiation safety measures. The spectra (see Fig. 1) were taken at 295°K on a YaGRS-4 spectrometer operating in constant-acceleration mode. We used ^{57}Co in chromium as a source of resonance γ-ray quanta and a gas-filled proportional counter as a radiation detector.

TABLE 1. Parameters of Mössbauer Spectra of Iron and Hydroxides Synthesized by Well-Known Methods

Compound	Temp., °K	IS_{Fe},* mm/sec	QS,* mm/sec	H_{eff},* kOe
α-FeOOH	80	0.37±0.04	0.21±0.02	493±5
	295	0.49±0.01	0.36±0.06	355±7
α-Fe_2O_3	295	0.44±0.01	0.21±0.05	518±10
β-FeOOH	80	0.37±0.04	0.17±0.03	460±10
	295	0.39±0.01	0.75±0.01	0
γ-FeOOH	80	0.33±0.01	0.78±0.01	0
	295	0.33±0.01	0.78±0.09	0
γ-Fe_2O_3	295	0.39±0.05	0.1±0.1	505±10
δ-FeOOH	80	0.36±0.05	0.00±0.08	483±10
	295	0.36±0.05	0.54±0.05	0
Fe_3O_4	295			
	Fe(III)	0.39±0.05	0.1±0.1	505±10
	Fe(II)+ + Fe(III)	0.65±0.05	0.08±0.05	475±10

*IS_{Fe} is the isomeric shift relative to metallic iron; QS is the quadrupole splitting, and H_{eff} is the effective magnetic field in the nucleus.

TABLE 2. Phase Composition of Iron Compounds in Corrosion Products of Structural Materials of RBMK-1000 Primary Circuit Found by Mössbauer Spectroscopy, wt. %

Corrosion-Product Sampling Site	Corrosion-Product Sample	Fe_3O_4	α-Fe_2O_3	γ-Fe_2O_3	γ-FeOOH	Fe_{met}
Drum-separator	Loose deposits on assembly of 08Kh18N10T steel in steam–water phase	54	–	46	–	–
	Loose deposits on grade-20 and 08Kh18N10T steel in steam – water phase.	–	–	100	–	–
	Tightly-adhering deposits on grade-20 steel in coolant	17	51	22	–	10
	Suspension from coolant	21	79	–	–	–
Condensate decontamination	Suspension from apparatus 3	39	17	15	24	–
	Suspension from apparatus 4	34	22	34	9	–
Extraction of live steam	Loose deposits	50	–	50	–	–
Deaerator	Loose deposits	28	42	30	–	–

To reduce the load on it from high-energy pulses (from activation products) the pulse-height analyzer was operated in the coincidence mode in the energy range close to 14 keV. The shape of the Mössbauer spectra of some corrosion-product samples is shown in Fig. 1. The spectra were processed on an M-220 computer by the least squares method. For qualitative identification of iron-oxide compounds, we compared the parameters of the Mössbauer spectra of corrosion products and iron-oxide compounds synthesized by the standard technique (Table 1). Quantitative calculations of the individual oxide phases were performed by the procedure of Meisel [5]. The results of the processing of the Mössbauer spectra of the corrosion products from the primary circuit of the RBMK-1000 are given in Table 2.

It is seen from Fig. 1 and Table 2 that the phase composition of the iron oxides is complex and depends on the conditions under which the corrosion products of the reactor materials were formed and deposited. For example, in the high-temperature zone (drum-separator) the corrosion-product suspensions consist of particles of magnetite (Fe_3O_4) and hematite (α-Fe_2O_3) whereas at reduced temperatures (condensate decontamination) the composition of the suspensions is more complex: In addition to the oxides mentioned above maghemite (γ-Fe_2O_3) and lepidocrocite (γ-FeOOH) are observed. The loose deposits in the steam phase consist of Fe_3O_4 and γ-Fe_2O_3 (drum-separator, extraction of live steam) whereas the composition of loose deposits on the surface of the steels in the coolant depends on the site of the sampling: Only γ-Fe_2O_3 is observed in the drum-separator whereas

Fe_3O_4, α-Fe_2O_3, and γ-Fe_2O_3 are observed in the deaerator. The same oxides also form the tightly adhering deposits on the surface grade-20 steel in the coolant of the drum-separator.

At the given stage of accumulation of experimental data one can attempt to give only a qualitative interpretation of the results obtained. The quite high oxygen content in the coolant ($\geq$10 μg/liter) increases the probability of the occurrence in the coolant of ionic forms of iron (III), the primary products of the dissolution of the metallic surface. It is well known that at a low temperature in a neutral medium hydrolysis of the ionic forms of iron ends in the formation of insoluble particles of lepidocrocite (α-FeOOH) [6]. A rise in temperature leads to the dehydration of the α-FeOOH to γ-Fe_2O_3, which at a temperature > 300°C goes over entirely into α-Fe_2O_3. Hematite can also go over into the coolant from corrosion deposits on overheated surfaces [1]. In addition to the above-mentioned oxide compounds of iron, magnetite can also appear in the aqueous coolant, e.g., by the Schikorr reaction [7]:

$$3Fe + 4H_2O \rightarrow Fe_3O_4 + 8H^+. \quad (1)$$

Thus, the coolant of an operating reactor may be expected to contain Fe_3O_4 and α-Fe_2O_3 at a high temperature and lepidocrocite and an intermediate form of its dehydration, maghemite, at a low temperature. The phase composition of the corrosion products in the coolant of the drum-separator at 285°C and the condensate-decontamination apparatuses at 30-40°C (see Table 2), found by the Mössbauer-effect method, confirms the validity of our assumptions.

As for the high-temperature corrosion deposits, with our sampling technique the composition of the corrosion-product samples called loose deposits may prove to contain corrosion products formed on steel of the external epitaxial layer of the protective corrosion film [8]. Moreover, it may also contain particles of corrosion deposits carried by coolant or steam from other parts of the circuit as well as hydrolysis products of ionic forms of iron, formed in the coolant itself. The tightly adhering deposits on grade-20 carbon steel may presumably consist of oxide compounds formed on the surface of the protective corrosion film consisting of an inner (topotaxial) and an outer (epitaxial) layer [8]. This may in part receive magnetite particles which are tightly bound up with the metallic surface; this magnetite is formed when the metal reduces the deposited hematite [1]:

$$Fe + 4\alpha = Fe_2O_3 \rightarrow Fe_3O_4. \quad (2)$$

In the case of stainless steels a protective film is built of ferritic or chromium–nickel spinels with the general formula AB_2O_4, where A are ions of divalent metals (Fe^{2+}, Mn^{2+}, Zn^{2+}, Ni^{2+}, ...) and B are ions of trivalent metals (Fe^{3+}, Cr^{3+}, Al^{3+}, ...) [8]. The protective film on carbon steel (depending on the oxygen content in the coolant) consists of either magnetite (or maghemite) in the pure form or successive layers Fe_3O_4–γ-Fe_2O_3–Fe_3O_4 [1, 9]. The superficial deposits, as a rule, consist of a mixture of magnetite and hematite [1]. Under our conditions one would expect magnetite, maghemite, and hematite in the loose deposits and magnetite and maghemite in tightly adhering deposits on grade-20 steel. It is seen from Table 2 that in actual fact this assumption holds only for loose deposits in the deaerator. The presence of only γ-Fe_2O_3 in loose deposits on the assembly in the coolant of the drum-separator is evidently due to the fact that only the dehydration products or lepidocrocite from the coolant enter these deposits. Bearing in mind that the transport of particles by the steam is much smaller than that by the coolant [10], the similarity of the composition of the loose deposits on stainless steels in the steam–water phase of the coolant and in the system for extraction of live steam can be attributed to the local origin of these deposits. Unfortunately, with the sampling technique employed, bits of the metal surface enter the tightly adhering deposits, in spite of the protective film formed by the oxides (Fe_3O_4 and γ-Fe_2O_3); this explains why the Mössbauer spectra of the tightly adhering deposits contain the lines of metallic iron (see Fig. 1). The presence of hematite in the tightly adhering deposits may be due to the incomplete oxidation of Fe_3O_4 to α-Fe_2O_3 [1].

It has thus been shown that Mössbauer spectroscopy can be used to determine the phase composition of iron compounds in the activated corrosion products of the structural materials of the RBMK-1000 primary circuit. The phase composition of the oxide forms of iron depends on the temperature of the circuit segment from which the samples of corrosion products were taken.

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ACTIVITY OF RADIONUCLIDES IN THE COOLANT OF THE SECONDARY LOOP OF A NUCLEAR POWER PLANT WITH VVÉR-440 REACTORS

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Radionuclides in the water and steam supplied to boilers and turbines by a nuclear power reactor present a certain radiation hazard. As a result of small defects in the piping, cooling water in the primary loop can leak into the water and steam of the secondary loop.

We have used data from the Kola nuclear power plant. The first unit of this plant operated more than 3 years before January, 1977 and the second unit more than 2 years. The first unit generated $5.2 \cdot 10^9$ kWh of electrical energy (520 effective days) during two runs, and the second unit generated $3.0 \cdot 10^9$ kWh (298 effective days) during one run.

Each VVÉR-440 commercial type reactor in the Kola nuclear power plant operates with six steam generators (SG) which supply steam to two turbogenerators (TG). The pressure of live saturated steam in the secondary loop reaches $4.7 \cdot 10^6$ N/m^2 at 260°C, and the circulation of water is 2700 tons/h. The pressure of the coolant in the steam generators in the primary loop is $12.5 \cdot 10^6$ N/m^2 at an average water temperature of 282°C.

After passing through the high-pressure cylinder (HPC) and two low-pressure cylinders (LPC) of a turbogenerator the spent steam enters the condensers. After passing through two heating stages and a deaerator the condensate returns to the steam generators (Fig. 1).

To remove contaminants, part of the water at a flow rate of 15-16 tons/h is drawn from the steam generators and passed through a special purification unit (SVO-5).

The total volume of water in the secondary loop is ~1250 m^3, including 106 m^3 in 10 low-pressure condensate heaters (LPH), 240 m^3 in two deaerators, 600 m^3 in two condensers, and 256 m^3 in six steam generators.

As a result of bleeding off steam for in-plant needs, the loss of water through leaks in the secondary loop and the water sampler, the discharge of steam through expansion tank vents, the circulation system ejectors and various excess pressure relief valves, the secondary loop has to be supplied chemically desalted water at the rate of ~20 tons/h through the condenser. Half of this loss is from steam consumed for technological needs.

Thus, the replacement fraction of the volume of water in the secondary loop is $20/1250 = 1.6 \cdot 10^{-2}$/h. As a result of the irrevocable loss of water and the use of fresh water the coefficient n_1 characterizing the removal of nongaseous radionuclides from the water is half this value, i.e.,

$$n_1 \approx 0.8 \cdot 10^{-2}\ \text{h}^{-1}.$$

The water in the secondary loop is further freed of radionuclides by the ion-exchange filters of the water purification unit (SVO-5). If the radionuclides were completely trapped by these filters the decontamination factor n_2 would be $15/1250 = 1.2 \cdot 10^{-2}$/h.

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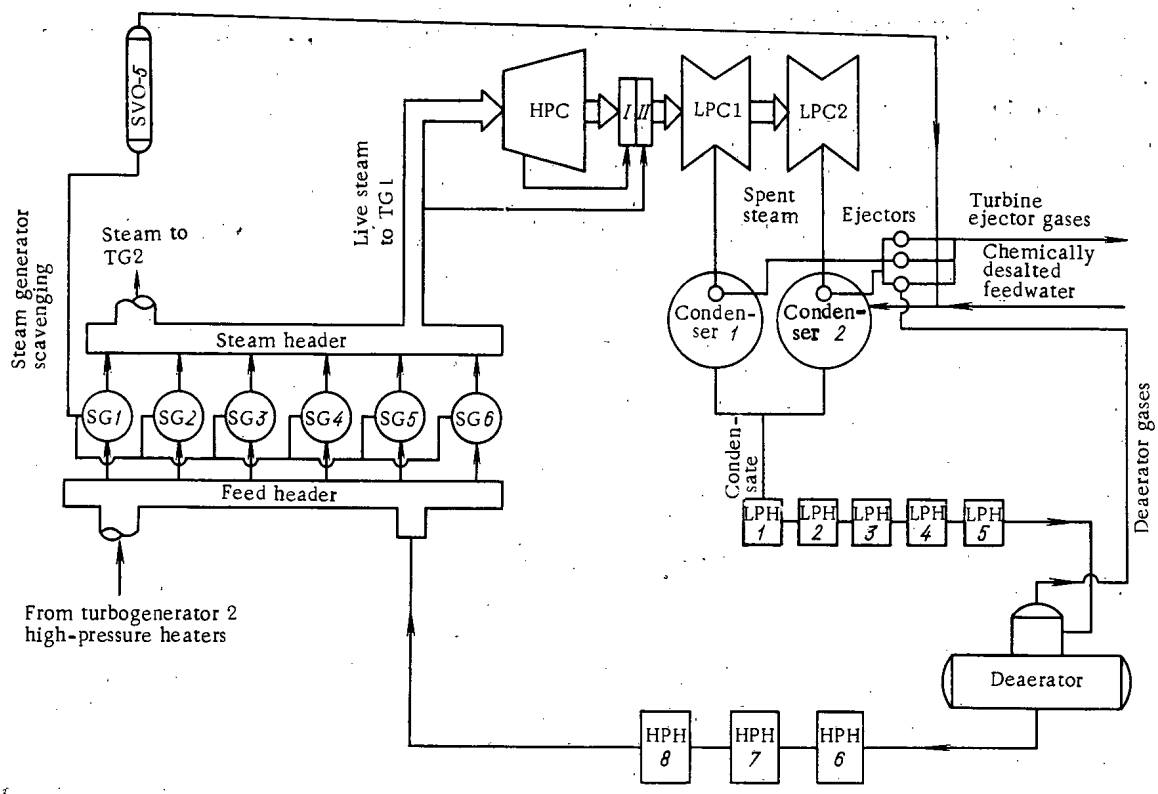

Fig. 1. Activity of radionuclides in coolant of secondary loop of a nuclear power plant.

TABLE 1. Average Values of Measured Concentrations of Certain Radionuclides in the Water of the First and Second Units of the Kola Nuclear Power Plant, Ci/liter

Radionuclide	First unit		Second unit	
	primary loop	boiler water	primary loop	boiler water
^{24}Na	$5{,}9\cdot10^{-5}$	$< 1\cdot10^{-12}$	$1{,}6\cdot10^{-5}$	$< 1\cdot10^{-12}$
^{42}K	$7{,}6\cdot10^{-6}$	» »	$1{,}2\cdot10^{-6}$	» »
^{18}F	$5{,}6\cdot10^{-6}$	» »	$1{,}0\cdot10^{-6}$	» »
^{131}I	$6{,}1\cdot10^{-6}$	$5{,}2\cdot10^{-11}$	$5{,}0\cdot10^{-5}$	$8{,}9\cdot10^{-11}$
^{133}I	$3{,}3\cdot10^{-5}$	$4{,}4\cdot10^{-11}$	$1{,}5\cdot10^{-4}$	$8{,}1\cdot10^{-11}$
^{135}I	$2{,}9\cdot10^{-5}$	—	$1{,}1\cdot10^{-4}$	—
^{85m}Kr	$2{,}0\cdot10^{-5}$	Not observed	$9{,}0\cdot10^{-5}$	Not observed
^{88}Kr	$3{,}0\cdot10^{-5}$	The same	$1{,}1\cdot10^{-4}$	The same
^{133}Xe	$2{,}8\cdot10^{-4}$	» »	$2{,}1\cdot10^{-3}$	» »
^{135}Xe	$1{,}8\cdot10^{-4}$	» »	$6{,}4\cdot10^{-4}$	» »
Total activity of gases	$5{,}1\cdot10^{-4}$	» »	$2{,}8\cdot10^{-3}$	» »

Notes. 1) From dosimetry monitoring data the total activity of gases in the exhaust of the turbine ejectors is less than $5\cdot10^{-10}$ Ci/liter on the average, but sometimes increases to $2\cdot10^{-9}$ Ci/liter. 2) Radiometry data show that the average total activity of the dry residue of boiler water on the steam generators is about 4.1 times smaller than the ^{131}I or ^{133}I activities.

If we introduce a correction for the measured efficiency of trapping ^{131}I and ^{133}I entering the water of the secondary loop together with radioactive gases, n_2 is decreased to $0.7\cdot10^{-2}$/h. Gaseous radionuclides are not retained by the water purification filters.

The accumulation and equilibrium content of a radionuclide in the water of the secondary loop is described by the equation

$$dQ/dt = gA - (\lambda + \lambda_r + n_1 + n_2)\,Q, \qquad (1)$$

where Q is the activity in the water of the secondary loop, Ci; A, the concentration in the water of the primary loop, Ci/liter; g, the rate at which water leaks from the primary into the secondary loop, liters/h; λ, the decay

TABLE 2. Calculated Leakages of the First (g_1) and Second (g_2) Units, g/h

Radionuclide	λ	n_1	n_2	g_1	g_2
^{131}I	$3.6 \cdot 10^{-3}$	$0.8 \cdot 10^{-2}$	$0.7 \cdot 10^{-2}$	41	12
^{133}I	$3.3 \cdot 10^{-2}$	$0.8 \cdot 10^{-2}$	$0.7 \cdot 10^{-2}$	17	7
Av. value				29	9.5

constant and λ_r is a constant describing the removal of activity with the steam. The deposition of ^{131}I and ^{133}I on the surfaces and equipment of the secondary loop can be neglected in view of the low value of the sorption constant. The solution of this equation is

$$Q = (gA/\lambda + \lambda_r + n_1 + n_2)\,[1 - \exp - (\lambda + \lambda_r + n_1 + n_2)\,t]. \quad (2)$$

Under steady-state conditions for a constant leakage of the primary loop and a constant supply to the secondary loop, Eq. (2) simplifies to

$$Q = gA/\lambda + \lambda_r + n_1 + n_2. \quad (3)$$

The unknown in Eq. (3) is g – the leakage of water from the primary into the secondary loop. The remaining parameters are monitored during the operation of the power plant and are known to a certain accuracy.

Table 1 lists the average concentrations of certain radionuclides in the water of the primary loop of both units of the Kola nuclear power plant. The averaging was performed over a three-month interval preceding shutdown for refueling and routine maintenance, i.e., toward the end of the run of each unit. The radionuclide composition of the steam generator boiler water is presented for the secondary loop. The specific activity of the dry residue of a sample of this water is generally $< 4 \cdot 10^{-10}$ Ci/liter, which is the limit of sensitivity of the method of measurement used. The incomplete data of the secondary loop result from the low level of activity and the difficulty of spectrometric analysis at this level.

It is clear from the data of Table 1 that the activity of the water in the primary loop is determined mainly by the gases ^{24}Na, ^{42}K, ^{18}F, and $^{131-135}I$, and that of the secondary loop by ^{131}I and ^{133}I. These data show that the number of fuel elements with leaking cladding does not exceed 5% of the admissible value [1].

In the steam generation process gases are completely removed from the water, but about 99% of the iodine remains in it. The iodine is carried along with the steam and returns to the loop as the steam condenses, except for the part which escapes with the irrevocable steam.

In the Kola nuclear power plant the irrevocable loss of water as steam is ~10.0 tons/h, mainly as a result of boiler losses (6.0 tons/h), safety valves (1.6 tons/h), turbine ejectors (2 tons/h), and the circulation system (0.4 tons/h). Then the constant for the removal of ^{131}I and ^{133}I from water, λ_r, will be $10.0 \cdot 10^{-5}$/h, i.e., $\lambda_r \ll \lambda$, n_1, n_2, and in subsequent calculations λ_r for iodine isotopes can be neglected.

The leakages $g_{1,2}$ of the first and second units (g/h), calculated by using the above assumptions, are shown in Table 2. It is clear from the table that the leakages estimated from the ^{131}I and ^{133}I activities in the water of the primary and secondary loops lie in the range (7-41) ± 12 g/h and are 19 ± 63% g/h on the average.

To estimate the leakage from the values in Table 1 for the sum of the activities of the gases in the water of the primary loop and in the gas–air mixture of the turbine ejectors we assume $n_2 = 0$ for gases. The removal constant for gases must be appreciably larger than n_1, since gases escape completely from water into steam, but in contrast with iodine are not returned to the water when the steam condenses. Each generator produces ~450 tons of steam per hour, and the whole unit produces ~2700 tons/h. Hence the constant for the removal of gases from water $\lambda_r = 2700/1200 \approx 2.2$/h.

Thus, the activity of the gas in the water of the secondary loop will be extremely low, and since λ is much smaller than λ_r, its value from Eq. (3) will be $Q = gA/\lambda_r$. Then the rate at which gases enter the steam is the product $Q\lambda_r = gA$; i.e., it is completely determined by the leakage rate g.

The initial concentration of gases in the steam volume of the steam generators and live steam supplied to the turbogenerators (pressure $4.5 \cdot 10^6$ N/m^2, specific volume $v = 0.045$ m^3/kg) can be obtained from the relation

$$Q/M = gA/M\lambda_r, \quad (4)$$

where $M = V/v$ is the mass of live steam, $V = 190\ m^3$ is the total steam volume of all the steam generators (SG) and six steam pipes (SP) feeding the turbogenerators ($V_{SG} = 6 \times 25\ m^3 = 150\ m^3$, $V_{SP} = 6 \times 6.8\ m^3 = 40\ m^3$, $V/v = 190/0.045 = 4 \cdot 10^3$ kg).

After passing through the turbogenerators the steam from the turbine enters the condensers (740 tons/h), where the gases contained in it are separated and removed by the turbine ejectors. These ejectors also draw off the gases from the deaerator which receives about 1000 tons of condensate water per hour. The increase from 740 to 1000 tons/h is the result of the aerator receiving condensed steam from the low-pressure heaters (LPH) and from other sources. The rest of the steam (350 tons/h) goes to the high-pressure heaters (HPH) and for other needs.

The steam generators supply 1350 tons of live steam per hour to each turbine. All the escaping gases are drawn off from at least 1000 tons of this steam per hour and sent to the exhaust, where their concentration is measured by a radiation meter. Thus, the intake of radioactive gases into the volume of the ejector gases monitored by the radiation meter is at least $(1000/1350)ga \approx 0.75\,gA$. The rate of discharge of the gas–air mixture under normal pressure by a turbine ejector is 30–40 kg/h, i.e., 35 m^3/h on the average.

If we assume that the volume V of the ejected gases monitored in the Du-200 tube by the radiation meter is $2\pi(0.1)^2 \approx 6.3 \cdot 10^{-2}\ m^3$, the gas exchange constant n_3 of this volume will be $35/(6.3 \cdot 10^{-2}) \approx 5.5 \cdot 10^2$/h. Then from Eq. (3) the equilibrium concentration K in the volume being measured is $K = 0.75\,gA/n_3V$.

Hence we calculate the leakage of the steam generators of each unit from the relation

$$g = Kn_3V/0.75A. \tag{5}$$

Using the data of Table 1 we obtain for the first unit

$$g_1 = 5 \cdot 10^{-10} \cdot 5.5 \cdot 10^2 \cdot 63/0.75 \cdot 5,$$
$$1 \cdot 10^{-4} \approx 5.10^{-2} \text{ liters/h}$$

and for the second unit

$$g_2 = 5 \cdot 10^{-10} \cdot 5.5 \cdot 10^2 \cdot 63/0.75 \cdot 2.8 \cdot 10^{-3} \approx 1 \cdot 10^{-2} \text{ liters/h}.$$

These calculations show that the leakages of water through defects of the steam generators are between 10 and 50 g/h, i.e., on the average close to the estimate made from the ^{131}I and ^{133}I activities. The error of the calculations using gases is clearly larger than that from the activities of the radionuclides mentioned because gases may be lost through leaks in the secondary loop. The dilution of radioactive gases by nonradioactive gases contained in the feedwater and drawn off by the turbine ejectors from the steam and the deaerator together with the radioactive gases is no less important, but is difficult to take into account.

From the radiation monitoring data taken in the second run of the first unit the activity of the water in the secondary loop as determined from the dry residue increased to $2 \cdot 10^{-9}$ Ci/liter (i.e., by a factor of 100) in 26 cases of 5400 measurements. In the first run of the second unit there were about 40 such cases.

The increase in activity of the ejector gases to $2 \cdot 10^{-9}$ Ci/liter, which was observed repeatedly, is equivalent to an increase in leakage, as monitored by the gases, to four times the standard values, i.e., $4 \times 5 \times 10^{-2} = 0.2$ liter/h. and when air losses and other errors are taken into account – to 1 liter/h.

For a concentration of ejector gases of $5 \cdot 10^{-8}$ Ci/liter, i.e., for a leakage 100 times larger than usual, the value of $g = 19 \times 100 \approx 1.9$ liters/h is comparable with the disorganized leakage of the primary loop.

The probability of the appearance of such a leak during the two runs of 1975-76 was $1.2 \cdot 10^{-1}$ (reactor eff. days)$^{-1}$ or $1.2 \cdot 10^{-8}$ (kWh)$^{-1}$ of generated electrical energy. From the data of the last two runs of 1976-1977 the probability of such a leak does not exceed $1.6 \cdot 10^{-2}$ (reactor eff. days)$^{-1}$ or $1.6 \cdot 10^{-9}$ (kWh)$^{-1}$ of generated electrical energy.

CONCLUSIONS

Analysis of the values presented for the concentrations of certain radionuclides measured in the water of the primary and secondary loops shows the presence of a constant leakage of water from the primary into the secondary loop equal to 19 ± 12 g/h on the average for a pressure drop of $\sim 7 \cdot 10^6$ N/m^2 between the loops.

The probability of this leakage increasing temporarily by a factor of 100 to ~1 liter/h is $1.2 \cdot 10^{-1}$–$1.6 \cdot 10^{-2}$ (reactor eff. days)$^{-1}$ or $1.2 \cdot 10^{-8}$–$1.6 \cdot 10^{-9}$ (kWh)$^{-1}$ of generated electrical energy.

The data obtained can be used to analyze the radiological consequences of such leaks for various heat-transfer schemes in nuclear power plants for supplying heat.

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MEASUREMENT OF THE $^{240}Pu/^{235}U$ AND $^{242}Pu/^{235}U$ FISSION CROSS-SECTION RATIOS FOR 0.127-7.4-MeV NEUTRONS

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The present article continues the series of papers on the study of the fast-neutron fission cross sections of a number of uranium and plutonium isotopes. The method of investigation and the main details of the experimental procedure are described in detail in [1-3]. Preliminary information was presented in [4].

The measurements were performed at electrostatic accelerators with neutrons from the $^{7}Li(p, n)^{7}Be$ ($E_n = 0.127$–0.342 MeV), $T(p, n)^{3}He$ (0.313-3.4 MeV), and the $D(d, n)^{3}He$ (3.6-7.4 MeV) reactions. By using two back-to-back ionization chambers [1] the $^{240}Pu/^{235}U$ and $^{242}Pu/^{235}U$ fission cross-section ratios were measured simultaneously.

The work was performed in two stages. In the first stage the energy dependence of the fission cross-section ratios was studied and the components of the neutron background were measured. The targets were layers of uranium and plutonium oxides deposited on thin (0.07-0.1 mm) platinum or aluminum backings. The isotopic purity of the ^{240}Pu was 99.49%, ^{242}Pu 99.92%, and ^{235}U 99.9955%.

Great care was taken to account correctly for the spontaneous fissions of ^{240}Pu and ^{242}Pu since these account for the sharp decrease in induced fission at neutron energies below threshold. Spontaneous fissions of ^{240}Pu and ^{242}Pu reached 28 and 58% respectively of the total number of fissions at $E_n = 0.127$ MeV, and did not exceed 0.6-1% for $E_n > 1.2$ MeV except between 3.6 and 4.2 MeV where they were 2.5-6% of the total. The components of the neutron background were measured as described in [1].

The correction for the background in the laboratory was 0.6-0.8% near threshold, varied slowly with energy, and did not exceed 0.4% for $E_n \geq 1$ MeV. The correction for neutrons entering the detector after scattering from the target structure reached 9-12% for $E_n \leq 0.342$ MeV, was 3-5% in the 1-2 MeV range, and did not exceed 0.5% for $E_n > 4$ MeV. The correction for the background of accompanying reactions took account of the fission of nuclei by neutrons from parasitic (p, n) and (d, n) reactions on molybdenum and titanium which enter into the composition of the targets. The maximum number of ^{235}U fissions by neutrons from accompanying (p, n) reactions did not exceed 4.4-5% ($E_n = 3.4$ MeV), but reached 30% for (d, n) reactions. The corrections of the $^{240}Pu/^{235}U$ and $^{242}Pu/^{235}U$ fission cross-section ratios did not exceed 1.8-2.1% for (p, n) reactions ($E_n \leq 3.4$ MeV) and were 1.5-4.5% for (d, n) reactions ($E_n = 5.5$–7.4 MeV).

In the second stage of the work the absolute values of the $^{240}Pu/^{239}Pu$ and $^{242}Pu/^{239}Pu$ fission cross-section ratios were measured for $E_n = 0.975$, 1.5, 2.0, 2.5, and 3 MeV. The use of ^{239}Pu instead of ^{235}U in this stage enabled us to apply the "method of isotopic admixtures" to determine the ratios of the numbers of fissionable nuclei in the layers being investigated and in the reference layers. This method can be used when the cross section under study has a threshold character. For example, highly accurate measurements of the absolute values of the $^{238}U/^{235}U$ fission cross-section ratio were made in this way [1]. The main advantage of the method is that while the work is proceeding at the accelerator both the ratios of the numbers of fissionable nuclei in the layers, and the absolute values of the fission cross section ratios can be determined directly

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with an ionization detector. This eliminates the necessity of determining the absolute efficiency of the fission chambers, and requires only a small correction for the change in the ratio of the chamber efficiencies as a function of neutron energy.

In this part of the work layers of ^{240}Pu and ^{242}Pu containing 6.7 and 5.8% admixtures of ^{239}PU respectively were specially prepared and subjected to careful mass spectrometric analysis. Layers of ^{239}Pu with an isotopic purity of 99.9% were used as standards. Under irradiation by slow neutrons with energies below the ^{240}Pu and ^{242}Pu fission thresholds, fission of ^{239}Pu nuclei was observed in the layers under investigation and in the reference layers. The ratios of the fission rates by slow neutrons, in combination with the data on the relative ^{239}Pu content in the ^{240}Pu and ^{242}Pu layers measured in the mass spectrometric analysis, permitted the determination of the ratios of the numbers of fissionable nuclei in the ^{240}Pu/^{239}Pu and ^{242}Pu/^{239}Pu layers. Since the slow neutrons induced ^{239}Pu fissions in both the layers under investigation and in the reference layers, the neutron spectrum can be arbitrary. Measurements were made in a flux of neutrons from the accelerator target slowed down by a 10-cm-thick polyethylene block. The number of fissions produced when the detector was covered with a 0.5-mm-thick layer of cadmium was subtracted from the number of fissions when the detector was bare. The cadmium ratio was 18-20, and the corrections of the ratio of fission rates did not exceed 0.4-0.5%.

Measurements with slow neutrons were performed repeatedly with neutrons from the Li(p, n) and T(p, n) reactions having maximum energies of 150 and 500 keV respectively. The agreement of the results within 1-1.5% showed that they are independent of the energy of the neutrons before moderation. To eliminate the depression of the slow-neutron flux in the backing, the targets were irradiated alternately from both sides to average the neutron flux through the layers.

After the measurements with slow neutrons a series of measurements was performed with fast neutrons using the same layers and recording procedure. The combination of experiments with slow and fast neutrons permitted the determination of the absolute values of the ^{240}Pu/^{239}Pu and ^{242}Pu/^{239}Pu fission cross-section ratios for $E_n = 0.975$, 1.5, 2.0, 2.5, and 3 MeV. Since the ^{235}U fission cross section was used as a standard in the measurements of the energy dependence of the fission cross section ratios over the whole energy range (0.127-7.4 MeV), the absolute values were also reduced to this standard by multiplying the ^{240}Pu/^{239}Pu and ^{242}Pu/^{239}Pu fission cross-section ratios by the value of $\sigma_f^{239}\text{Pu}/\sigma_f^{235}\text{U}$ obtained by the authors [2].

The values of the energy dependence of the ^{240}Pu/^{235}U and ^{242}Pu/^{235}U fission cross-section ratios measured in the first stage of the work were normalized to the absolute values at the neutron energies indicated above. In spite of the fact that the error of the absolute values in this method of calibration includes the uncertainty in the ^{239}Pu/^{235}U fission cross-section ratio (1.4-1.5%), the "method of isotopic admixtures" is not inferior in accuracy and is more reliable for ^{240}Pu and ^{242}Pu nuclei than other methods, such as the measurement of the ratio of the α activities of layers.

Thus, the error in determining the ratio of the number of fissionable nuclei was determined largely by the uncertainty of the ^{239}Pu content in the ^{240}Pu and ^{242}Pu layers. Therefore the mass spectrometric analysis of the specially prepared mixtures received particular attention. Isotopic analyses of the mixtures were performed on samples of $\leq 10^{-8}$ g by counting ions and using the program of displacement of mass peaks. In processing the results corrections were made for the background of scattered ions and isotopic discriminations (by means of the accelerating potential and in the ion detector). The validity of the displacement of mass peaks and the stability of isotopic discriminations were monitored by the rhenium ratio. Check experiments showed no observable peaks of foreign materials with mass numbers from 239 to 242 and no effects of fractionation in the evaporation process. The measurement of the ^{238}Pu concentration was hampered by a small admixture of ^{238}U present in the mixtures, and therefore an upper limit is indicated for the ^{238}Pu content. Table 1 presents the results of the analysis, averaged over many mass spectra for various evaporation conditions. The error indicated is the standard deviation. To increase the reliability of the data and to check the accuracy of the corrections introduced, mixture 2 in Table 1 was analyzed independently on two mass spectrometers. The two analyses agreed within the limits of error.

In the analysis of the mixtures an estimate was made of the content of ^{238}U and nuclides with mass numbers 243 and 244. Their content was estimated as $< 10^{-1}$ and $< 10^{-3}$ at. % respectively. Immediately before the mass spectrometric analysis the samples were purified of americium in order to avoid the superposition of peaks corresponding to mass number 241. A similar method of mass spectrometric measurements was employed in [1] to analyze mixtures of ^{238}U and ^{235}U.

A number of corrections were calculated for the measured values of the ^{240}Pu/^{235}U and ^{242}Pu/^{235}U fission cross-section ratios. As a result of the high isotopic purity of the samples the correction to take account of

TABLE 1. Isotopic Composition of Mixtures Used to Measure Absolute Values of Fission Cross-Section Ratios, at. %

Mixture	^{238}Pu	^{239}Pu	^{240}Pu	^{241}Pu	^{242}Pu
1	$\leqslant$0,0705	6,7435±0,0145	92,7611	0,2657±0,0012	0,1592±0,0010
2	$\leqslant$0,0047	5,7975±0,0269	0,0596±0,0005	0,0592±0,0005	94,0790

TABLE 2. Absolute Values of Fission Cross-Section Ratios

E, MeV	E, keV	$\sigma_f{}^{240}Pu/\sigma_f{}^{239}Pu$	$\sigma_f{}^{242}Pu/\sigma_f{}^{239}Pu$	$\sigma_f{}^{239}Pu/\sigma_f{}^{235}U$ *	$\sigma_f{}^{240}Pu/\sigma_f{}^{235}U$	$\sigma_f{}^{242}Pu/\sigma_f{}^{235}U$
0,975	31	0,827±0,009	0,763±0,011	1,443±0,021	1,194±0,022	1,101±0,022
1,50	49	0,809±0,009	0,707±0,010	1,572±0,022	1,271±9,023	1,112±0,022
2,00	60	0,841±0,010	0,730±0,010	1,154±0,022	1,307±0,024	1,134±0,020
2,50	75	0,854±0,010	0,713±0,010	1,547±0.022	1,321±0.024	1,103±0,022
3,00	84	0,884±0,011	0,734±0,010	1,556±0,022	1,375±0,025	1,142±0,022

*Results from [2].

TABLE 3. Corrections and Experimental Errors in the $^{240}Pu/^{235}U$ and $^{242}Pu/^{235}U$ Fission Cross-Section Ratios, %

No.	Sources of corrections and errors	$^{240}Pu/^{235}U$		$^{242}Pu/^{235}U$	
		Correction	Error	Correction	Error
1	Ratio of numbers of fissionable nuclei	–	0.6	–	1.0
2	Difference of neutron fluxes through layers	2.1	0.2	1.4	0.2
3	Error of ratio of fission cross section [2]	–	1.41	–	1.41
4	Statistical error of measurements with fast neutrons	–	0.5	–	0.4
5	Fission of minority isotopes	8.0	0.2	8.0	0.2
6	Energy dependence of ratio of fission chamber efficiencies	1.6	0.4	1.6	0.4
7	Neutron background in laboratory	0.3	0.2	0.3	0.2
8	Background of neutrons scattered by target	0.3	0.3	0.2	0.3
9	Background of neutrons from accompanying (p, n) reactions	0.8	0.4	0.6	0.4
10	Scattering of neutrons by backings	–	0.5	–	0.5
11	Total error	–	1.84	–	2.0

the fission of minority isotopes by fast neutrons did not exceed 0.3% for the energy dependence data even near the threshold. The correction was 8-8.5% for the absolute values. Since the main part of the correction was related to the admixture of ^{239}Pu in ^{240}Pu and ^{242}Pu, and since we measured the ratio of the fission cross-sections of these nuclei, the correction was calculated sufficiently accurately by performing two or three iterations. The correction for the dependence of the ratio of the fission chamber efficiencies on neutron energy, taking account of the motion of the center of mass and the angular anisotropy of fission, did not exceed 2-2.5% over the whole range studied.

The difference in fast-neutron fluxes through layers separated by backing was 2.1 and 1.4% for the absolute values of the $^{240}Pu/^{235}U$ and $^{242}Pu/^{235}U$ fission cross-section ratios respectively, and for the results of the energy dependence this difference was taken into account by normalizing to the absolute values. No corrections were introduced for elastic and inelastic scattering of neutrons from backings, but upper estimates of these effects (0.3-0.5%) were included in the total error of the results.

Table 2 lists the absolute values of the $^{242}Pu/^{235}U$ and $^{240}Pu/^{235}U$ fission cross-section ratios. The total error listed is the root-mean-square sum of all the uncertainties. The error in the determination of the ratios

TABLE 4. $^{240}Pu/^{235}U$ and $^{242}Pu/^{235}U$ Fission Cross-Section Ratios

E_n, MeV	ΔE_n, keV	$\frac{\sigma_f^{240}Pu}{\sigma_f^{235}U}$	$\Delta\frac{\sigma_f^{240}Pu}{\sigma_f^{235}U}$, %	$\frac{\sigma_f^{242}Pu}{\sigma_f^{235}U}$	$\Delta\frac{\sigma_f^{242}Pu}{\sigma_f^{235}U}$, %	E_n, MeV	$E\Delta_n$, keV	$\frac{\sigma_f^{240}Pu}{\sigma_f^{235}U}$	$\Delta\frac{\sigma_f^{240}Pu}{\sigma_f^{235}U}$, %	$\frac{\sigma_f^{242}Pu}{\sigma_f^{235}U}$	$\Delta\frac{\sigma_f^{242}Pu}{\sigma_f^{235}U}$, %
0.127	19	0,0534	4.8	0.0100	5,7	1,900	63	1,296	2,1	1,121	2,2
0,150	19	0,0544	4,3	0.0136	4,6	2,00	60	1,314	2,0	1,135	2,2
0.180	19	0,0574	3.6	0,0159	5,3	2,10	68	1,309	2,0	1,125	2,3
0,210	18	0,0639	3.8	0.0193	5.3	2.20	70	1,316	2,0	1.107	2,2
0.240	17	0,0696	3.9	0,0293	5.4	2.30	71	1.318	2,0	1,106	2.6
0.270	18	0.0817	3.6	0,0310	4.9	2.40	72	1.330	2.0	1,100	2.2
0.300	18	0.0898	3,6	0,0401	3.7	2.50	73	1.332	2.0	1,107	2,2
0,313	40	0,107	3,3	0,0533	4,0	2,60	77	1,337	2,0	1,119	2,3
0,342	38	0,119	2.6	0.0623	4.2	2.70	78	1,336	2,0	1,127	2,5
0.365	37	0.140	2.9	0,0781	3.5	2.80	79	1.357	2,0	1,144	2.2
0.404	36	0,174	2.3	0.0958	3.0	2.90	82	1.363	2.0	1.144	2.2
0,444	35	0.224	2.2	0,130	2.6	3.00	84	1.369	2,1	1,148	2,2
0,483	34	0.306	2.2	0.173	2.3	3,10	86	1,377	2,0	1,159	2,2
0,523	33	0.393	2.2	0,209	2.4	3,20	88	1,375	2,1	1,153	2,2
0,562	33	0.498	2.2	0.284	2.5	3,30	91	1.384	2,0	1,153	2,2
0.601	32	0,598	2.2	0.359	2,3	3,40	93	1,380	2,0	1,147	2.2
0,641	32	0,695	2,2	0,428	2,3	3,60	192	1,383	2,0	1,152	2,2
0,680	32	0,754	2,2	0,491	2,5	3,80	182	1,382	2,0	1,150	2.2
0,720	32	0.818	2.1	0.557	2.2	4,00	146	1,397	2,1	1,152	2,3
0,759	31	0.905	2.2	0.670	2.3	4,20	141	1,404	2.2	1,145	2.2
0.798	31	0,976	2.1	0.781	2,3	4.40	132	1.402	2.2	1,147	2,2
0,836	31	1,060	2,2	0.881	2,3	4,60	131	1,394	2,2	1,146	2,3
0,877	31	1,130	2.1	0,977	2,3	4,80	125	1,403	2,2	1,142	2,3
0.926	31	1,144	2.1	1.006	2,3	5,00	126	1,413	2,2	1,147	2,2
0.975	31	1.189	2.1	1.104	2.2	5,20	129	1,432	2.2	1,156	2.6
1.025	34	1,243	2,1	1.154	2.3	5,40	131	1,450	2,2	1,182	2,3
1.074	37	1.261	2.1	1.197	2,2	5,60	135	1,468	2.3	1,209	2,3
1.123	40	1.259	2.1	1.199	2,2	5.80	138	1,487	2,3	1,239	2,2
1,172	42	1.274	2.1	1.187	2,2	6,00	142	1,488	2,3	1,279	2,3
1.221	44	1,257	2,1	1.161	2,5	6,20	147	1,453	2,3	1,274	2,3
1,270	45	1.270	2.1	1.143	2,4	6,40	152	1,406	2,4	1,273	2,4
1,320	46	1.270	2,2	1,138	2,2	6,60	160	1,380	2,6	1,263	2,6
1.400	47	1.268	2,1	1.141	2,3	6,80	167	1,344	2,5	1,226	2,8
1.500	48	1.275	2.0	1.100	2,2	7,00	173	1,330	2,5	1,190	2,8
1.600	49	1.294	2,1	1.111	2,2	7,20	178	1,305	2,6	1,178	2,9
1,700	60	1.300	2,2	1,113	2,2	7,40	183	1,283	2,8	1,167	3,0
1,800	61	1,304	2,1	1.123	2,2						

of the numbers of fissionable nuclei for $^{240}Pu/^{235}U$ and $^{242}Pu/^{235}U$ were 0.6 and 1% respectively. It includes the uncertainties of the admixtures of ^{239}Pu in the ^{240}Pu and ^{242}Pu layers (0.22 and 0.46%), the statistical error of the measurements with slow neutrons (0.47 and 0.86%) determined from the spread of the results, and small errors (0.3 and 0.1%) related to taking account of the slow-neutron fission of ^{241}Pu in the mixtures (Table 1). Table 3 shows the structure of the characteristic corrections and errors for measurements performed at E_n = 3 MeV. Table 4 lists the measured results for the whole range of energies studied. The total error listed is the root-mean-square sum of the average error of the absolute values of the $^{240}Pu/^{235}U$ and $^{242}Pu/^{235}U$ fission cross-section ratios (1.8 and 2.0% respectively), the uncertainty of the results of the energy dependence (from 0.7 to 4.5-5.3%), and the error in normalizing the energy dependence data to the absolute values (0.25%).

The main contribution to the uncertainty of the results of the energy dependence for $E_n < 1.2$ MeV comes from the statistical error in taking account of the spontaneous fission of ^{240}Pu and ^{242}Pu (from 0.5 to 4.5-5.2%) and the error in the corrections for the background of neutrons scattered by the target (0.4-1.2%). For $E_n > 5$ MeV the uncertainty in taking account of the background of accompanying (d, n) reactions (1-2%) was considerable. For other values of the energy the errors in the measurements of the energy dependence were close to those listed in lines 4-10 of Table 3.

The total error of the measured values of the $^{240}Pu/^{235}U$ and $^{242}Pu/^{235}U$ fission cross-section ratios was 2-2.5 and 2.2-2.8% respectively over most of the energy range studied, and increased to 4.8-5.7% for $E_n = 0.127$ MeV. The systematic part of the total error, correlated over the whole energy range, was 1.8 and 2% respectively for the $^{240}Pu/^{235}U$ and $^{242}Pu^{235}U$ fission cross-section ratios.

$\sigma_f^{240}Pu/\sigma_f^{235}U$. There is good agreement with the data in [5] near threshold. The results in [6] are in good agreement on the shape of the energy dependence curve but are displaced in energy by 20-30 keV. The data in [7, 8] tend to be lower, and the results in [9] in the 100-180 keV region are 20% higher than our values. Above 1 MeV the best agreement is with the results in [6]. The data from other experiments [7-11], including the results in [5], where we have noted good agreement for $E_n < 1$ MeV, are systematically lower by 5-10%.

$\sigma_f^{242}Pu/\sigma_f^{235}U$. Near threshold our results are in good agreement with the data in [6], but displaced in energy by 20-30 keV, as is the case for the $^{240}Pu/^{235}U$ fission cross-section ratios. For $E_n > 1$ MeV our results agree with those in [6] both in absolute magnitude and in the shape of the energy dependence. The data in [12] are systematically lower over the whole energy range studied, with the difference reaching 20% in the plateau region.

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ESTIMATES OF GLOBAL ^{85}Kr RADIATION SAFETY

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Sources releasing radioactive inert gases (RIG) into the atmosphere include nuclear explosions, atomic-power-station nuclear reactors, and nuclear-fuel reprocessing plants. Most of the RIG activity is composed of short-lived radionuclides.

Of the long-lived RIG radionuclides, the one which arouses the most concern is ^{85}Kr ($T_{1/2} = 10.76$ yr, $E_\beta = 0.672$ MeV), which is characterized by a high mobility in the atmosphere and, being an inert gas, is involved in metabolic processes to an extremely small extent. Because of its low solubility in water (0.034 mg/g at 37°C [1]) and competition with stable (natural) krypton, ^{85}Kr is not washed out by atmospheric precipitation to any significant extent; the corresponding coefficient is $4 \cdot 10^{-11}$ sec^{-1} [2]. The oceans contain only ~3% of the total ^{85}Kr content in the atmosphere [3]. Likewise, the absorption of ^{85}Kr on aerosol particles, with subsequent precipitation, is not of any great importance. It is thought that the interaction of krypton with the soil is irreversible. The transfer of krypton into plants is also extremely insignificant, amounting to 10^{-10}-10^{-9} cm^3/sec [4].* Thus, the ^{85}Kr concentration in the air decreases almost entirely as the result of dispersal and radioactive decay.

In natural conditions, ^{85}Kr is formed in insignificant amounts as a result of the spontaneous fission of uranium and the interaction of cosmic-radiation neutrons with atmospheric krypton. The equilibrium of the radiokrypton cycle is disrupted as a result of nuclear explosions in the atmosphere (in a 1-kton explosion, 25.6 Ci of ^{85}Kr is formed). Underground nuclear explosions are also accompanied by the release of RIG to the atmosphere: This occurs through cracks which form in the rock as a result of the underground explosion [5, 6]. According to the generalized data of the UN Scientific Committee on the Effect of Atomic Radiation (UN SCEAR), ~3 MCi of ^{85}Kr have been injected into the atmosphere as a result of all nuclear explosions [7]. Information on the power of explosions in nuclear-weapons tests and the ^{85}Kr concentration in the atmosphere

*P. Wolleque and J. Fix, Transfer of Airborne Krypton-85 to Vegetation (cited in [4]).

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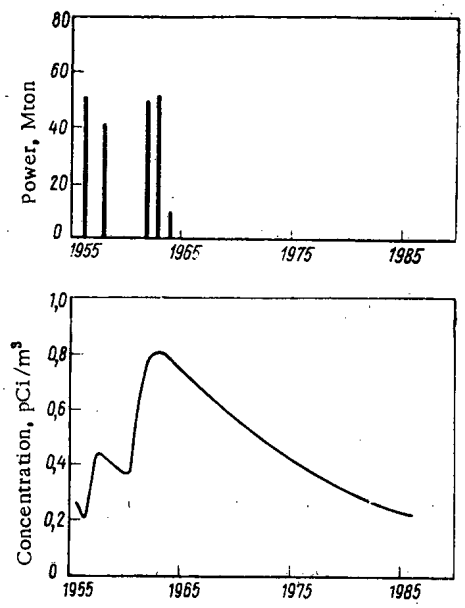

Fig. 1. Nuclear-explosion power and ^{85}Kr concentration in air (the total fission power is 203 Mton).

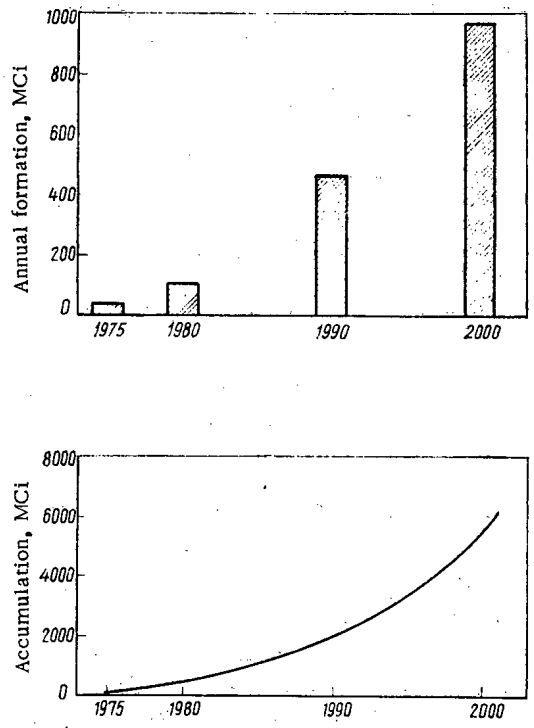

Fig. 2. Expected annual formation and total accumulation of ^{85}Kr.

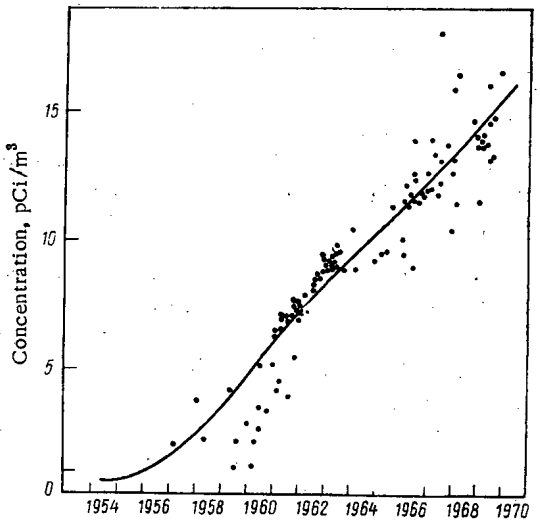

Fig. 3. ^{85}Kr concentration in the atmosphere of the Northern Hemisphere [16].

are shown in Fig. 1 [8]. The production of plutonium has made a more important contribution to the global accumulation of ^{85}Kr. For example, in 1966 alone, the contribution of US commercial reactors was 15 MCi.* According to the generalized UNSCEAR data [9], RIG release from PWR and BWR reactors was 10-20 and 1300 Ci/MW(elec.)·yr, respectively (mainly short-lived RIG isotopes). In the conditions of normal atomic-power-station use, the radiation dose to the population is insignificant [10, 11]. For example, in the territories adjacent to the Novovoronezh atomic power station, the radiation does did not exceed 2 mrem/yr [12], i.e., significantly below the permissible level of irradiation of the population, 500 mrem/yr [13].

The main sources of atmospheric ^{85}Kr contamination are nuclear-fuel reprocessing plants. In the existing technology, short-lived radionuclides decay in the course of storage of the treated fuel, and practically

*D. Bernhardt, A. Moghissi, and J. Cochran, Atmospheric Concentrations of Fission Product Noble Gases (cited in [4]).

TABLE 1. Human Radiation Dose Due to ^{85}Kr Present in Atmospheric Air, mrad/yr

Organ or tissue	1977	2000	2050
Skin	$42 \cdot 10^{-3}$	8,4	630
Lungs	$6 \cdot 10^{-4}$	$13 \cdot 10^{-2}$	9,6
Tissue epithelium	$1 \cdot 10^{-3}$	$2 \cdot 10^{-1}$	15
Internal organs	$3 \cdot 10^{-4}$	$7 \cdot 10^{-2}$	5
Gonads	$3 \cdot 10^{-4}$	$7 \cdot 10^{-2}$	5

TABLE 2. Possible Somatic and Genetic Consequences of Irradiation Due to ^{85}Kr (No. of cases per year per million people)

Type of disorder	1977	2000	2050
All malignant neoplasm (except the skin)	$4,5 \cdot 10^{-5}$	$9 \cdot 10^{-3}$	0,7
Lung cancer	$1 \cdot 10^{-5}$	$2 \cdot 10^{-3}$	0,1
Skin cancer	$6 \cdot 10^{-5}$	$1,2 \cdot 10^{-2}$	1
Leucosis	$1 \cdot 10^{-5}$	$2 \cdot 10^{-3}$	0,1
Congenital defects*	$4 \cdot 10^{-5}$	$8 \cdot 10^{-3}$	0,6

*Per million births.

the only RIG remaining in the fuel is 375 Ci/MW(elec.)·yr of ^{85}Kr [9]. As a result of economic and technological difficulties, radiokrypton traps are not yet in use, and the radiokrypton is released to the atmosphere through the gas-escape tube. In 1975, the ^{85}Kr release from all the world's atomic-industry installations was ~30 MCi. For the predicted development of nuclear power, the formation of ^{85}Kr in 1980, 1990, and 2000 may reach 103, 461, and 988 KCi/yr, with a total accumulation of 339, 2070, and 6280 MCi, respectively [14] (Fig. 2). According to the data of [15], the release of ^{85}Kr in 1985, 1995, 2010, and 2020 may reach 191, 639, 2076, and 3360 MCi, respectively.

The ^{85}Kr concentration in air is rising continuously (Fig. 3) [16]. Above the territories of the USSR in 1975 it was 0.016 pCi/liter [16]. By the year 2000, the radiokrypton concentration may increase by a factor of 100-200, to reach 2-4 pCi/liter [4, 17]. It is thought that after 80-100 yr the concentration may reach 300 pCi/liter [18].

The global ^{85}Kr radionuclide is a source of constant external and internal irradiation of the human population. The biological safety of the existing and predicted ^{85}Kr concentrations will now be estimated.

The main contribution to the radiation dose is that of the radionuclide's β radiation. Since the maximum path of ^{85}Kr β particles in soft tissues is 2.52 mm, the whole structural formation of the skin is exposed to intense irradiation (the thickness of the human skin is 2-2.2 mm). The maximum absorbed β-radiation dose is formed in the basal layer, which is the critical layer of the skin (in terms of radiosensitivity, the skin structure may be ranked in the following order: the basal layer of the epidermis and the endothelium of the vessels, hair follicles, the sebaceous glands, muscle, erectile hairs, the sweat glands). The radiation dose to the dermal layers due to γ radiation from ^{85}Kr ($E = 0.517$ MeV) is approximately two orders of magnitude lower. For γ radiation, the gonads are the critical organs.

Radiokrypton inhaled with air is also a source of internal irradiation (the intake of ^{85}Kr through the skin is no more than 0.4% of the amount inhaled [19]). In conditions of steady equilibrium, the distribution coefficient for radiokrypton in fatty tissue, blood, muscle, and other tissues of the human organism is 0.46, 0.046, and 0.047 ml/g, respectively. The half-life of ^{85}Kr in the lungs and blood is 30 sec, in muscle and other tissue it is 8 min, and in fatty tissues 2.7 h [19].

Estimates of the possible annual bodily radiation doses are given in Table 1. The calculations use the dose coefficients in [20]. The maximum exposure is that of the skin and, of the internal organs, the epithelium of the respiratory organs. Similar values for the dose were given in [4]. The ^{85}Kr radiation doses are extremely small at present, and form an insignificant fraction of the dose corresponding to the natural background radiation (100-150 mrad/yr). For comparison, the radiation doses from other sources are as follows: global radionuclide precipitation, 5 mrem/yr; flights by reactive airplanes, 3 mrem per flight; radioisotope investigations using RIG, tens to hundreds of mrad [21].

In the years to come (especially after the year 2000), the radiation dose due to global ^{85}Kr will rise considerably, and the irradiation of exposed portions of the body will exceed the background by a factor of 5-6, but these doses are very small, and can cause neither acute nor chronic radiation damage. However, the possible carcinogenic and genetic consequences of such irradiation on the population as a whole deserves special consideration (Table 2). In view of the modern radiobiological ideas of UN SCEAR and the International Commission on Radiological Protection (ICRP), the estimate of the radiation risk is based on the concept of thresholdless radiation effects and a linear dose–effect relation. The integral radiation dose is found as the product of the radiation dose and the size of the population. This approach is conservative in the familiar sense. But, at present, it is the best guarantee of reliable radiation protection of the human population.

The contribution of ^{85}Kr to the incidence of additional cancers and genetic disorders is extremely slight in comparison with their natural frequency. Fatalities from spontaneous tumors in the U.S.A. amount to ~1600 cases per year, and in the USSR to 1500 cases per year per million people [22]. The natural frequency of genetic disorders has been estimated at 60,000 cases per million births [23].

At present UN SCEAR and ICRP do not give estimates of the skin-cancer risk due to radiation, since the epidemiological data are extremely contradictory. The lack of an unusually high incidence of skin cancer among the survivors of atomic bombardment and also those irradiated in other circumstances suggests that the skin is less sensitive to the carcinogenic action of radiation than the bone marrow, the bronchial system, the mammary gland, the thyroid gland, and bone tissue. This is also indicated by the low natural incidence of skin cancer in comparison with other organs. The yield of skin tumors is approximately an order of magnitude lower than for more radiosensitive tissues.

Experimental data allow a rough estimate to be made of the yield of skin tumors. Using ^{90}Sr and ^{90}Y, at β-radiation doses of 100-1600 rad, the tumor yield in rats was 120 cases per 10^6 rad [24, 25]. Taking into account that in humans the exposed areas of skin which are exposed to ^{85}Kr β radiation constitute ~10% of the total body surface, and that exposure is chronic, it may be expected that the tumor yield in humans will not exceed 1-2 cases per 10^6 man-rem. If this value of the risk of skin cancer development in humans is used for rough calculations, the tumor yield due to global ^{85}Kr in 1977, 2000, and 2050 may be $6 \cdot 10^{-5}$, $1.2 \cdot 10^{-2}$, and 1 case over the lifetime of the population per million people, respectively. These figures represent only a tiny fraction of the deaths from skin cancer over the lifetime of the population per million people (1050 cases). It is also necessary to take into account that skin cancer is readily diagnosed and relatively simple to treat. This concludes the evaluation of the radiation safety of global ^{85}Kr.

It should be emphasized that the evaluation of the local radiation dose to the population in the vicinity of reprocessing plants requires special consideration. This will soon be dealt with in a report by the US National Commission on Radiological Protection [26].

The biological effect of ^{85}Kr on the skin is increased as a result of UV irradiation. It is thought that UV radiation is a carcinogenic and mutagenic agent; this could be the explanation, e.g., for the high incidence of skin cancer among pale-skinned peoples. There are certain differences in the mechanisms by which penetrating and UV radiation act. For example, UV radiation leads to the formation of thymine dimers in DNA, while the action of penetrating radiation is associated with bond rupture. It is not yet clear whether the combined effect is simply additive or involves synergy. Special investigations are necessary.

Thus, analysis of the available material indicates that the development of nuclear power will not lead to any significant increase in the number of malignant neoplasms and genetic disorders due to global ^{85}Kr in comparison with their natural frequency.

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CONTACT CONDUCTIVITY BETWEEN A UO_2 CORE AND CLADDING

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As nuclear engineering develops, more and more detailed in-pile processes must be studied. However, setting up experiments in research reactors is very complicated and requires an appreciable expenditure of time and resources. A large number of factors affecting the process must be taken into account. Under such conditions the intuition of the experimenter plays a smaller role.

Recently methods of empirical search have been extensively developed, and therefore the problem of extracting the maximum information for limited expenditure of time and materials is very urgent. One of the most effective methods of solving this problem is the use of mathematical modeling of an experiment. The success of such an approach depends on how accurately one succeeds in formulating the criterion for optimizing the model for solving a definite class of problems. A number of criteria have been developed for optimum modeling which enable the investigator to choose the locations of the experimental points and process the results of observations.

We present the results of the statistical processing of experimental data reported by various authors [1, 2] on contact conductivity between a UO_2 fuel element core and cladding of the container type. In processing the passive experiment we succeeded in choosing a design corresponding to a complete factorial experiment [3].

Choice of Controlling Factors. An analysis of various processes which occur in a fuel element did not enable us to determine the quantitative effect of individual factors on the contact conductivity α. The number of these factors is very large and their interactions complex. Such multifactor systems are called diffuse.

The choice of the optimization criterion and the complexity of the model depend on the choice of the main factors in the development of the mathematical model describing the physical process. A mathematical model is an equation relating an output parameter to the controlling factors:

$$y = f(\tilde{x}_1, \tilde{x}_2, \ldots, \tilde{x}_k) = f(\tilde{x}). \quad (1)$$

In our case the output parameter is the value of α. We list below the factors which initially were assumed to have a significant effect on the output parameter, together with the ranges of their variation [1, 2]:

1.	Linear power, W/cm	100—500
2.	Initial gap, mm	0.045—0.18
3.	Pressure of filler gas, N/m^2	$(1-10)\cdot10^5$
4.	Thermal conductivity of gas, W/cm · deg C	0.000165—0.0015
5.	Porosity of UO_2, %	2.85—6.0
6.	Microhardness of cladding, kg/mm^2	10—40
7.	Diameter of cladding, mm	9—17
8.	Thickness of cladding, mm	0.35—1
9.	Pressure of coolant, N/m^2	$(10-50)\cdot10^5$

In processing the results of a passive experiment it is impossible to choose an optimum design from such a large number of factors, but if it is assumed that some of them are not significant, such a design may emerge. The statistical relation of each factor to the output parameter can be estimated from the values of the coefficients of pair correlations. Thus, factors 5-9 in the above list were discarded because of a lack of correlation with the output parameter. It was confirmed later in processing the experimental results that these factors are not significant. Factors 1-4 were chosen as controlling.

It is obvious that the dependence of the contact conductivity on the main factors chosen cannot be determined without taking account of their interaction. For example, the thermal conductivity of the gas depends linearly on the temperature, which in turn is determined by the power developed in the fuel element. The pressure of the gas in the fuel element must be related to the power. It was shown in [1] that the effect of accommoda-

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TABLE 1. Design Matrix for a 2^4 Complete Factorial Experiment

Expt. No.	x_0	x_1	x_2	x_3	x_4	x_1x_2	x_1x_3	x_1x_4	x_2x_3	x_2x_4	x_3x_4	$x_1x_2x_3$	$x_1x_3x_4$	$x_2x_3x_4$	y_e	y_c	$(y_e-y_c)^2$
1	+	−	−	−	−	+	+	+	+	+	+	−	−	−	0.13	0.082	0.0023
2	+	+	−	−	−	−	−	−	+	+	+	+	+	−	0.33	0.376	0.0022
3	+	−	+	−	−	−	+	+	−	−	+	+	−	+	0.045	0.093	0.0023
4	+	+	+	−	−	+	−	−	−	−	+	−	+	+	0.13	0.074	0.003
5	+	−	−	+	−	+	−	+	−	+	−	+	−	+	0.18	0.154	0.0007
6	+	+	−	+	−	−	+	−	−	+	−	−	+	−	0.36	0.384	0.0006
7	+	−	+	+	−	−	−	+	+	−	−	−	−	+	0.045	0.069	0.0005
8	+	+	+	+	−	+	+	−	+	−	−	+	+	−	0.13	0.103	0.0007
9	+	−	−	−	+	+	+	−	+	−	−	+	+	−	0.13	0.179	0.0014
10	+	+	−	−	+	−	−	+	+	−	−	−	−	+	0.73	0.682	0.0023
11	+	−	+	−	+	−	+	−	−	+	−	−	+	−	0.14	0.092	0.0023
12	+	+	+	−	+	+	−	+	−	+	−	+	−	−	0.24	0.286	0.0021
13	+	−	−	+	+	+	−	−	−	−	+	+	−	−	0.93	0.957	0.0007
14	+	+	−	+	+	−	+	−	−	−	+	−	+	−	1.5	1.474	0.0007
15	+	−	+	+	+	−	−	+	−	−	+	−	−	+	0.23	0.205	0.0006
16	+	+	+	+	+	+	+	+	+	+	+	+	+	+	0.5	0.525	0.0006

Notes. y_e is the experimental value of α; y_c is the value of α calculated from Eq. (5); + is the upper level of variation of the factor (corresponds to $x=+1$); − is the lower level ($x=-1$).

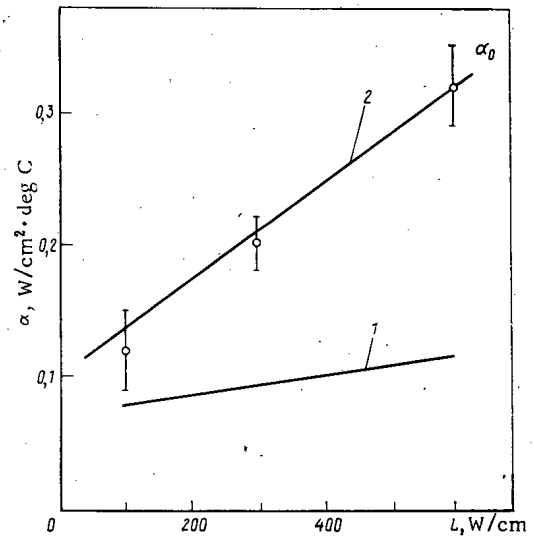

Fig. 1

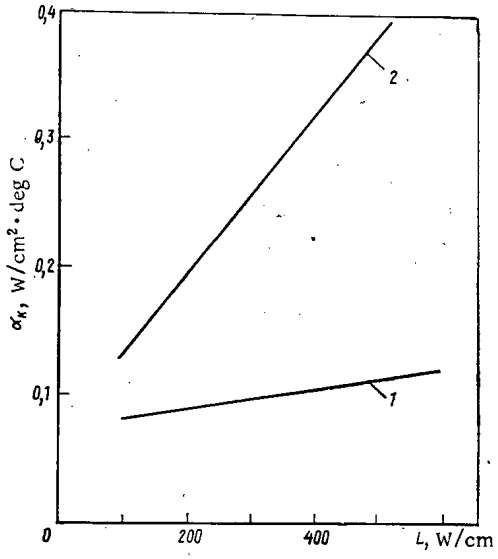

Fig. 2

Fig. 1. Contact conductivity α between UO_2 core and cladding as a function of linear power of fuel element: 1) α_k calculated by Eq. (5) for $\sigma=168\ \mu$, $\lambda=0$, $P=0$; 2) for $\sigma=168\ \mu$, $\lambda=0.0015$ W/cm$^2\cdot$deg C, $P=10^5$ N/m^2; ○, experiment [2].

Fig. 2. "Solid" component as a function of linear power of fuel element: 1) calculated for $\sigma=168\ \mu$, $P=0$, $\lambda=0$; 2) calculated for $\sigma=10\ \mu$, $P=0$, $\lambda=0$.

tion depends on the pressure. It is very complicated to take account of these interactions and to calculate them from physical principles. Therefore it is necessary to choose a design of processing which permits an independent determination of all the interactions among the main factors. The coefficients in such a model will have a physical meaning which permits an estimate of the contribution of the various factors and their interactions to α.

Choice of Design. It is known from the theory of mathematical design of experiments that in a passive experiment the influence of one or another effect cannot be estimated independently since the "design" matrix in this case is not orthogonal [3], i.e.

$$\sum_{u=1}^{N} x_{iu}\, x_{ju} \neq 0,\ i \neq j;\ i,\ j=0,\ 1,\ \ldots,\ K. \qquad (2)$$

In order to be able to make an independent estimate of all the linear effects and the effects of interactions it is necessary to have a design of a complete factorial experiment. We succeeded in choosing such a design

from existing experimental data [1]. Table 1 lists the design of a 2^4 complete factorial experiment. The factors were varied from +1 to −1. The coding was performed by the expression

$$\tilde{x}_i = \frac{2(x_i - \bar{x})}{x_{max} - x_{min}}, \quad (3)$$

where $\tilde{x}_i$ is the coded value of the i-th factor (i = 1, 2, 3, 4); x_i, the running value of the factor; $\bar{x}_i$, the average value of the range of definition of the i-th factor; and x_{max} and x_{min} are the maximum and minimum values in the range of definition. The design includes all double and triple interactions of the factors.

The coefficients were calculated by the expression

$$B = (x^T x)^{-1} x^T y, \quad (4)$$

where B is an estimate of the coefficients in the regression equation; x, the 2^4 complete factorial problem design matrix; x^T, the transposed matrix; and y, a column vector of the experimental values of the contact conductivity α, W/cm^2 · deg C.

The following regression equation was obtained:

$$y = 0.3594 + 1.136x_1 - 0.1769x_2 + 0.125x_3 + 0.1906x_4 - 0.0631x_1x_2 + 0.0075x_1x_3 + 0.0619x_1x_4 - \\ - 0.0813x_2x_3 - 0.0956x_2x_4 + 0.115x_3x_4 + 0.0137x_1x_2x_3 + 0.01x_1x_3x_4 - 0.0713x_2x_3x_4. \quad (5)$$

A test of the significance of the coefficients in Eq. (5) by Student's criterion showed that all the interactions are significant. The model turned out to be adequate for the experimental results. Figure 1 compares the values calculated by Eq. (5) with experimental results reported in [2]. There is good agreement even for low values of α. The unbiased estimate of the mean square deviation from the model is $\sigma = 0.1$ W/cm^2 · deg C.

Test of Adequacy of Model Chosen. The adequacy of the computational model (5) for the range of variation of factors listed above was tested by the Fisher criterion [3]. However, the model turned out to be adequate over an even wider range of variation of the main factors. Thus, values calculated by Eq. (5) agree well with experiment [2] up to a linear power of ~800 W/cm. The experiments did not differ from the regression equation by more than 3%. Such agreement presupposes the possibility of extrapolating with respect to other factors also, e.g., x_3 and x_4. Similar calculations were performed for $x_3 = -1.22$ and $x_4 = -1.22$, which correspond to zero thermal conductivity and zero gas pressure (Fig. 2). If the relations shown in Fig. 2 are considered as the contribution of the "solid" component α_k to the total conductivity of the gap α, certain conclusions can be drawn about the effect of linear power and the initial gap between fuel and cladding on the value of the solid component α_k. Figure 2 shows that the initial ("cold") gap has a significant effect on the dependence of α_k on the linear power of the fuel element. For an initial gap of ~160 μ the contribution of the solid component varied from 0.07 to 0.12 W/cm^2 · deg C depending on the linear power. However, for an initial gap of ~10 μ the contribution of α_k increased to 0.4 W/cm^2 · deg C.

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SPATIAL DISTRIBUTION OF SLOW HYDROGEN AND HELIUM ATOMS INTRODUCED IN A SOLID

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The investigation of the spatial distribution of hydrogen and helium ions introduced in a solid is an important problem of radiational materials science. However, the great mobility of light ions in metals and their interactions with crystal defects, including those of radiational origin, complicate the interpretation of the experimental results. Many data recently accumulated indicate that there is a correlation between the distribution of the introduced ions and the distribution of radiation defects formed by these or other ions [1-3]. If this relation is determining, it may be possible to obtain the true distribution of introduced ions by calculating the steady distribution of atoms and defects, and hence to calculate the diffusion of introduced ions and the dispersion of the surface. Such distributions have been calculated by the "enlarged"-collision version of the Monte Carlo method [4]. In this case, the calculation involves the Lindhardt elastic-scattering cross section and the mean specific energy losses in inelastic processes, which predominate in the total losses of light ions, obtained experimentally. By this means, the maximum agreement between calculation and experiment is obtained, although neglecting the crystal structure may lead to discrepancies. The distribution calculated for 6.7-keV deuterium ions introduced in carbon (Fig. 1) provides a sufficiently good description of the experimental data [5]. The discrepancy at the "tail" of the distribution may be the result of diffusional blurring, and of the transport of deuterium directly by the incident beam. A characteristic feature of the calculated distribution (the distribution of projective paths) is its asymmetry with respect to the maximum R_p. With adequate accuracy, the calculated distribution n(x) may be approximated by a Gaussian curve

$$n(x) = n_0 \exp[(x - R_p)^2/2\sigma^2]. \quad (1)$$

The asymmetry of the distribution is taken into account by introducing the dispersions σ_1 and σ_2: The first describes the region $x \leq R_p$ and the second the region $x > R_p$. It is necessary to note that the part of the profile for $x > R_p$ is cut off at $x = R_p^{max}$. The parameters n_0, R_p, R_p^{max}, σ_1^2, and σ_2^2 may be calculated for different incident-particle energies from the empirical relation

$$R_p(R_p^{max}, \sigma_1^2, \sigma_2^2, n_0) = AE^{\alpha}. \quad (2)$$

Values of the constants A and α for some elements are shown in Table 1. In determining the maximum concentration (for $x = R_p$), an error of ± 10% is possible, as a result of normalization of the curves to take account of the reflection coefficient. The degree of asymmetry of the distribution decreases with increase in mass of the incident particles and the atomic number of the target atoms. A quantitative estimate of the role of radiation defects in the stabilization of introduced atoms may be made by calculating the point-defect concentration $C_d(E)$ as a function of the incident-particle energy E

$$C_d(E) = N\Phi \int_{\bar{T}_d}^{T_m(E)} \nu(T) \frac{d\sigma}{dT} dT, \quad (3)$$

where $d\sigma/dT$ is the cross section for energy transfer T by the recoil atom (an analytic expression for $d\sigma/dt$ corresponding to the power-law potential $V(r) = A_2/r_2$ is used); $\bar{T}_d$, the mean threshold energy of displacement; $\nu(T)$, a cascade function; N, the number of target atoms per cm^3; and Φ, the incident-particle flux. To obtain the distribution of the defect concentration over the depth of the irradiated target, it is necessary to average $C_d(E)$ over the particle spectrum f(E, x)dE at the given depth, calculated by the Monte Carlo method. With an error of 15-20%, the calculations of $C_d(x)$ for niobium ($\bar{T}_d = 36$ eV [6]) may be represented in the form ($\Phi = 1$ cm^{-2})

$$C_d(x) = B \exp[-b(x/R_p^{max})^{\beta}]. \quad (4)$$

The constants B, b, and β have the following values: for H^+ when E = 5-15 keV, $B = 7.2 \cdot 10^4 E^{0.4}$ cm^{-3}, $b = 4.5 - 0.11(E-5)$, $\beta = 1.8 + 0.04(E-5)$; for D^+ when E = 2-12 keV, $B = 2.4 \cdot 10^5 E^{0.23} cm^{-3}$, b = 4.1, $\beta = 2.1$; for He^+

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TABLE 1. Values of A and α for the Calculation of the Distribution in Eq. (1)

Particle	n_0		R_p		R_p^{max}		σ_1^2		σ_2^2	
	A, cm^{-3}	α	A, Å	α	A, Å	α	A, Å^2	α	A, Å^2	α
					Silicon					
H^+	$2{,}1\cdot10^5$	$-0{,}6$	320	0,74	550	0,7	$3{,}8\cdot10^4$	1,3	$2{,}6\cdot10^4$	1,3
					Iron					
H^+	$6{,}9\cdot10^5$	$-0{,}8$	120	0,8	220	0,7	$5{,}5\cdot10^3$	1,3	$1{,}4\cdot10^3$	1,7
He^+	$8{,}0\cdot10^5$	$-0{,}7$	67	0,85	145	0,74	$1{,}8\cdot10^3$	1,5	$2\cdot10^3$	1,3
					Niobium					
H^+	$5{,}2\cdot10^5$	$-0{,}6$	120	0,8	240	0,7	$6{,}5\cdot10^3$	1,3	$3{,}5\cdot10^3$	1,3
D^+	$4{,}3\cdot10^5$	$-0{,}6$	115	0,9	280	0,7	$3{,}8\cdot10^3$	1,7	$5{,}0\cdot10^3$	1,3
He^+	$5{,}7\cdot10^5$	$-0{,}7$	75	0,9	175	0,8	$4{,}5\cdot10^3$	1,3	$4{,}0\cdot10^3$	1,3
					Gold					
H^+	$5{,}7\cdot10^5$	$-0{,}7$	95	0,86	210	0,7	$1{,}7\cdot10^3$	1,9	$1{,}5\cdot10^3$	1,75

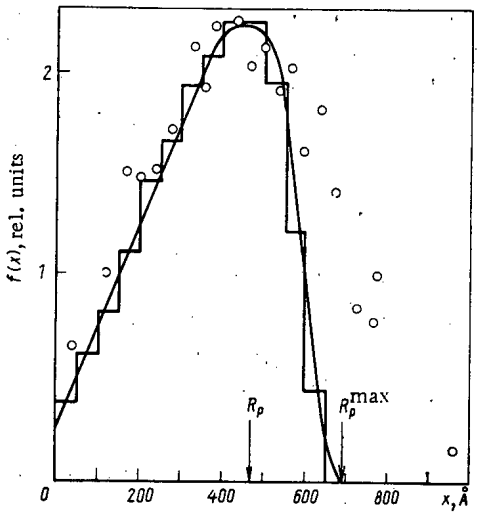

Fig. 1. Distribution of 6.7-keV deuterium ions introduced in carbon: ○) experiment; ——) approximation by Eq. (1); the histogram shows the results of Monte Carlo calculation.

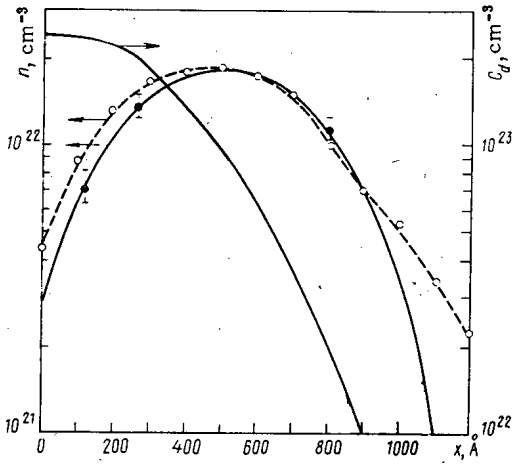

Fig. 2. Distribution over the sample depth of concentration of 9-keV ^{3}He ions introduced in niobium with $\Phi = 1.5 \cdot 10^{17}$ cm^{-2} (left-hand scale) and of defect concentration (right-hand scale): ——) calculation; ●) statistical error of the calculation; ○) experiment [7].

when E = 2–12 keV, $B = 1.2 \cdot 10^6 E^{0.15}$ cm^{-3}, b = 4.1, $\beta = 2.1$. The value of R_p^{max} is calculated on the basis of the data in Table 1.

Calculations by Eq. (3) show that when $E \leq 2$ keV, helium ions produce a high concentration of defects in the region $x < R_p^{max}$, sufficient for effective interaction with the atoms introduced in the material. Two limiting cases may be considered. In the first, the defect concentration $C_d(x)$ markedly exceeds the concentration of introduced ions n(x) over the whole region $x \leq R_p^{max}$ and stabilization of the introduced atoms would be expected, i.e., the calculated profile should be sufficiently close to the experimental result. This is in fact the case when ^{3}He of energy 9 keV is introduced into niobium [7], as shown in Fig. 2 ($\Phi = 1.5 \cdot 10^{17}$ cm^{-2}). The small discrepancy in the tail of the distribution is probably associated with channeling of the introduced ions (the samples are polycrystalline) and diffusion of helium. In the second case, $C_d(x) < n(x)$ for all $x < R_p^{max}$, with the possible exception of the surface region of the target. This case corresponds to protons of energy < 15 keV and deuterons of energy < 5 keV introduced in niobium and tungsten, when there is a small probability of interaction with defects and hydrogen diffuses rapidly, resulting in a characteristic distribution with a concentration maximum at the surface. Such a distribution has been observed experimentally in tungsten irradiated by 7.5-keV protons [8]. However, for tritium, as shown by the calculations of [9], at least partial stabilization would be expected. At the same time, it is necessary to take into account that, when a target is irradiated by a mixed flux of hydrogen and helium ions, it is possible for the stabilization conditions to change as a result of defects produced by the helium ions. In this case, the above distributions of the atomic and defect concentrations may be used as initial data for calculations of the diffusion and stabilization of the introduced atoms.

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ALGORITHM TO ESTIMATE THE LOCAL PERTURBATIONS OF LINEAR RADIATION-FLUX FUNCTIONALS USING THE MONTE CARLO METHOD

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The calculation of small perturbations of linear flux functionals by means of a direct random walk, using correlational weights, was considered in [1, 2]. An analogous approach is also possible for a conjugate random walk: Pseudoparticles are emitted from the detector region and perform a random walk in the unperturbed region up to the source, and a single set of trajectories is used to calculate the unperturbed functional in the normal way (see [3-7], for example) and its perturbation, using conjugate correlations weights as the mathematical expectation of the random quantity

$$\xi_{\Delta}^{+}=\sum_{i}^{l} S(x_i) W_i^{+}\left[\frac{S'(x_i)}{S(x_i)}\prod_{j=0}^{i}\frac{K'(x_j\to x_{j+1})}{K(x_j\to x_{j+1})}\cdot\frac{\varphi'(x_0)}{\varphi(x_0)}-1\right], \tag{1}$$

where l is the number of collisions on the pseudoparticle trajectory; W_i^+, the current weight of an ordinary conjugate walk; $S(x)$, the distribution of particle sources; $\varphi(x)$, the detector sensitivity function; and $K(x_j\to x_{j+1})$, the probability density of pseudoparticle transition from point x_j to x_{j+1} of the phase space $X=\{\mathbf{r}, E, \Omega\}$. (Here and below, a prime denotes a parameter of the unperturbed medium.)

The estimate in Eq. (1) is unbiased if $K(x'\to x)\neq 0$ when $K'(x'\to x)\neq 0$ and $\varphi(x)\neq 0$ when $\varphi'(x)\neq 0$, i.e., it is required that all the pseudoparticle trajectories which are possible in the perturbed medium must also be possible in an unperturbed medium. If these conditions are not satisfied, it is always possible to select an intermediate medium in which all the trajectories of the unperturbed and perturbed problems are possible and to perform the random walk in that area. The corresponding averaged random quantity is

$$\eta_{\Delta}^{+}=\sum_{i=0}^{l} W_i^{*}\left[S'(x_i)\prod_{j=0}^{i}\frac{K'(x_j\to x_{j+1})}{K^{*}(x_j\to x_{j+1})}\frac{\varphi'(x_0)}{\varphi^{*}(x_0)}-S(x_i)\prod_{j=0}^{i}\frac{K(x_j\to x_{j+1})}{K^{*}(x_j\to x_{j+1})}\frac{\varphi(x_0)}{\varphi^{*}(x_0)}\right]. \tag{2}$$

(An asterisk denotes a parameter of the intermediate medium.)

The estimates in Eqs. (1) and (2) are effective when the perturbation region covers the detector or constitutes a large part of the volume of the given system in the case of an unlocalized source. The analogous direct-walk estimates of the perturbation are effective when the perturbation region and the detector are unlocalized.

The present work considers an effective means of estimating the local perturbations in the case of an unlocalized source and detector, on the basis of the combined use of conjugate and direct random walks. The direct and conjugate walks are connected at the surface of the perturbed volume.

It may be shown that, in the system of sources $S(x)$, the flux $F(x)$ is equal to the sum

a) of the flux $F_0(x)$ generated in a system with a blackbody V_0 of arbitrary form;

b) of the flux $F_\Gamma(x)$ generated in the system of sources $S_\Gamma(x)$ at the surface Γ of the volume V_0:

$$S_\Gamma(x)=\begin{cases}-(\mathbf{n}\Omega)F_0(x) & \text{when } \mathbf{r}\in\Gamma \ (\mathbf{n} \text{ is the external normal in x}),\\ 0 & \text{otherwise;}\end{cases}$$

c) and of the flux $F_V(x)$ generated in the system of sources

$$S_V(x)=\begin{cases}S(x) & \text{when } \mathbf{r}\in V_0,\\ 0 & \text{otherwise,}\end{cases}$$

i.e., $F(x)=F_0(x)+F_\Gamma(x)+F_V(x)$.

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To estimate the perturbation of the linear functional $L = \int F(x)\,\varphi(x)\,dx$ it is evidently sufficient to model only the components $F_\Gamma(x)$ and $F_V(x)$ of the flux $F(x)$ by the Monte Carlo method, taking as V_0 the perturbed volume of the system. This may be achieved by means of algorithm 1, as follows.

1. The perturbed volume is replaced by a blackbody.
2. Uniformly over the surface Γ of the body, a random choice is made of the coordinates of the pseudoparticle-generation point $\mathbf{r}_0$.
3. The pseudoparticle energy E_0 is chosen randomly from the a priori spectrum $\tilde{F}(E)$ on the surface Γ.
4. A random choice of the direction of flight Ω_0 is made isotropically in the hemisphere $(\mathbf{n}\Omega) > 0$.

   The coordinates $x_0^+ = (\mathbf{r}_0, E_0, \Omega_0)$ thus obtained are selected values from the distribution $S_\Gamma(x) = \tilde{F}(E)/2\pi\Pi$, where Π is the area of the surface Γ.
5. By conjugate random walk, the trajectories of N^+ pseudoparticles with initial weight $W_0^+ = 1$ are traced from the point x_0^+ and an estimate is obtained for the flux

$$F_0(x) = M\hat{F}_0(x_0); \quad \hat{F}_0(x_0) = \sum_{i=0}^{l} W_i^+ S(x_i); \quad x_0 = (\mathbf{r}_0, E_0, -\Omega_0),$$

   where l is the number of collisions on the pseudoparticle trajectory; M denotes the mathematical expectation.
6. The medium in the perturbed volume is restored.
7. By direct random walk, the trajectory of a particle with initial current weight

$$W_0 = (\mathbf{n}\Omega_0)\,\frac{2\pi\Pi}{\tilde{F}(E_0)}\,\frac{1}{N^+}\sum_{j=1}^{N^+}\hat{F}_{0j}(x_0)$$

   and initial correlation weight $W_B = 1$ is traced from the point $\mathbf{x}_0$.
8. Analogously, the initial coordinates and weight are chosen for particles which model the flux component $F_\Gamma(x)$, and the N_Γ particle trajectories are traced.
9. In the usual way, a random choice from the volume source $S_V(x)$ is made of the particles which model the flux component $F_{V_0}(x)$ and, by direct random walk, the N_V particle trajectories are traced. The perturbation of the functional L may be estimated on the $N_\Gamma + N_V$ trajectories by the method of correlated choice, etc. An analogous algorithm was used in [8] to calculate small local perturbations of nuclear-reactor reactivity.

Using algorithm 1 it is possible to estimate the effects of relatively homogeneous local perturbations. Very often, a weak perturbation in a relatively large volume V_1 is found in combination with a strong perturbation in a relatively small volume V_2. In this case, the effects of each perturbation may be very close and of opposite sign. The simplest examples of this type of perturbation are the compression of an irradiated sample and the compensation of the Doppler effect in the active zone of a reactor by adjustment of the control rods. In such situations, algorithm 1 is able to model only the flux components $F_V(x)$ and $F_\Gamma(x)$, and is relatively ineffective in estimating the effect of a strongly localized perturbation. In this case, it is expedient to express the flux $F(x)$ as the sum

$$F(x) = F_{\Gamma_2}(x) + F_{V_2}(x) - F_{\Gamma_1}(x)\,|_{V_2} + F_{V_1}(x)\,|_{V_2} + F_0(x),$$

where $F_{\Gamma_1}|V_2$ and $F_{V_1}|V_2$ are the fluxes generated by the corresponding sources in a medium with a blackbody in volume V_2. Each component of the flux may be effectively modeled using algorithm 2, as follows.

1-4. These operations are as in algorithm 1, except that the perturbed volume is understood to be V_2.

5. By conjugate random walk, the trajectories of N_2^+ pseudoparticles are traced from the point x_0^+, and estimates are obtained for the fluxes

$$F_2(x_0) = M\hat{F}_2(x_0); \quad \hat{F}_2(x_0) = \sum_{i=0}^{l} W_i^+ S(x_i);$$

$$F_2'(x_0) = M\hat{F}_2'(x_0);$$

$$\hat{F}_2'(x_0) = \sum_{i=0}^{l} W_i^+ S'(x_i) \prod_{j=0}^{i} \frac{K'(x_j \to x_{j+1})}{K(x_j \to x_{j+1})}.$$

The difference between F_2' and F is due to the perturbation in V_1.

6. The medium in V_2 is restored.

7. By direct random walk, the trajectory of the particle with initial current weight

$$W_0 = (n\Omega_0)\frac{2\pi\Pi_2}{\tilde{F}(E_0)}\frac{1}{N_2^+}\sum_{j=1}^{N_2^+}\hat{F}_{2j}(x_0)$$

and initial correlation weight

$$W_B = \sum_{j=1}^{N_2^+}\hat{F}'_{2j}(x_0)\Big/\sum_{j=1}^{N_2^+}\hat{F}_{2j}(x_0) \approx M\frac{F_2'(x_0)}{F_2(x_0)}$$

is traced from the point x_0.

8. Analogously, the initial coordinates and weight of the particles which model the flux component $F_{\Gamma_2}(x)$ are chosen and the N_{Γ_2} particle trajectories are traced.

9. In the usual way, a random choice from the volume source $S_{V_2}(x)$ is made of the particles which model the flux component $F_{V_2}(x)$ and, by direct random walk, the N_{V_2} particle trajectories are traced.

10. Algorithm 1 is used to deal with the perturbed volume V_1; volume V_2 is replaced by a blackbody. As a result, the flux components $F_{\Gamma_1}(x)|V_2$ and $F_{V_2}(x)$ are modeled on $N_{\Gamma_1}+N_{V_1}$ trajectories.

The perturbation of the functional L is then estimated on $N_{\Gamma_2}+N_{V_2}+N_{\Gamma_1}+N_{V_1}$ particle trajectories by the method of correlational weights.

Depending on the features of the specific perturbation and perturbed functional, it is simple to construct an adequate calculational scheme on the basis of the given algorithms.

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POROSITY DISTRIBUTION IN NICKEL FOLLOWING ARGON BOMBARDMENT

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We have investigated the depth distribution of porosity in annealed samples of nickel (99.9 at. %) following exposure to 1-MeV Ar^+ ions at 625°C. The dose used was $7 \cdot 10^{16}$ ions/cm^2 at a current density of 4-6 μA/cm^2. After exposure, surface samples were removed by vibropolishing, their thickness determined, and electron micrographs of the reverse side were obtained.

Blisters were not observed during these investigations as the critical dose for their formation is $(1.5\text{-}2) \cdot 10^{17}$ ions/cm^2 under our experimental conditions [1]. The electron micrographs show that pores are formed at depths of 0.2-0.65 μm and, at depths below 0.45 μm, dislocation loops are observed along with pores. The fractional pore volume as a function of depth is shown in Fig. 1. In accord with well-known calculations [2], the range of 1-MeV Ar ions is ~ 0.6 μm. However, the greatest density of radiation damage is at a depth of ~ 0.5 μm and, for the dose given, peaks at a value of ~ 40 displacements/atom. All the argon atoms are therefore localized in the rather narrow layer (~ 0.2 μm) at the end of their range, and the concentration of gas in that layer is 3-4 at. %. If it is assumed that all the argon is found in the pores, then three argon atoms are contained in each pore; that is, the observed pores appear to be empty, with only a negligible quantity of gas.

In the microphotographs shown in Fig. 2, the concentration of pores at a depth of 0.55 μm is $\approx$13 times greater than that at 0.35 μm, but their mean diameter is only half as large. In Fig. 3 we show the experimentally determined pore-size distribution at various depths and in Table 1 we give the derived parameters. Evidently, swelling at 0.55-μm depth is due to the large pore concentration while at 0.35-μm depth it is due to the large pore dimension.

The difference between the dimensions and concentrations of pores at 0.35 and 0.55 μm is explained primarily by the large concentration (3-4 at. %) of argon atoms at 0.55 -μm depth. It is known that gas introduced into the sample before exposure influences the concentration and dimensions of the pores, so that the process

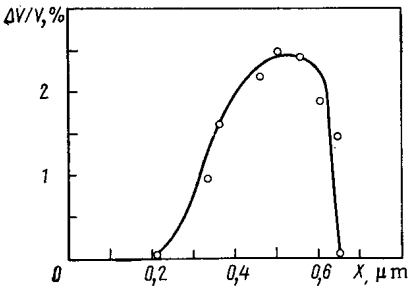

Fig. 1. Fractional pore volume at various depths in the irradiated sample.

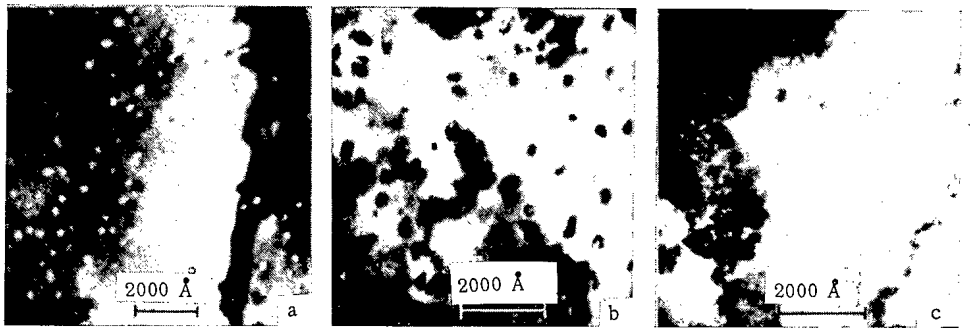

Fig. 2. Microstructure of irradiated nickel at a depth of: a) 0.25 μm; b, c) 0.55 μm.

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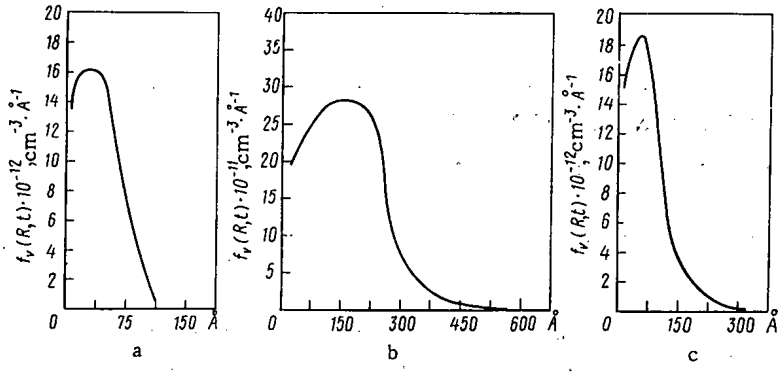

Fig. 3. Pore-size distribution at various depths in the sample: a) 0.2 μm, b) 0.35 μm, c) 0.55 μm.

TABLE 1. Results of Sample Analysis

Sample depth (μm)	Dose (displacements/atom)	N_V, cm^{-3}	$\langle d_V \rangle$, Å	$\frac{\Delta V}{V}$, %	N_l, cm^{-3}	$\langle d_l \rangle$, Å
0,2	5	$1{,}1 \cdot 10^{15}$	50	0,02	—	—
0,35	20	$1{,}5 \cdot 10^{15}$	210	1,7	—	—
0,55	40	$2 \cdot 10^{16}$	110	2,5	$6 \cdot 10^{14}$	300

Notation. N_V, N_l are the pore and loop concentrations; $\langle d_V \rangle$, $\langle d_l \rangle$ are the mean pore and loop diameters; $\Delta V/V$ is the fractional pore volume.

of pore formation accelerates and they are stabilized at an early stage [3]. This change of chemical composition may also lead to changes in such important parameters as defect energy, diffusion coefficient, surface energy, etc. Alternatively, the differences between the pore-size distributions and concentrations may be explained by the varying rate of displacement creation and by the variation of overall exposure with sample depth: at a depth of 0.35 μm these are half as large as at a depth of 0.55 μm.

In summary, we have irradiated nickel with 1-MeV argon ions. At a depth of 0.55 ± 0.1 μm (which corresponds to the region of greatest radiation damage) a great number of argon atoms accumulate along with the radiation damage. They are not mobile at the temperature of exposure, 625°C, and are probably trapped by pores. This is associated with reduced interstitial vacancy recombination and with increased pore generation. The resultant enhancement of the interaction of interstitial atoms with each other explains the formation of comparatively large (300 Å) dislocation loops (see Fig. 2b).

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^{135m}Ba YIELDS IN ^{133}Cs(α, pn)^{135m}Ba AND ^{139}La(p, αn)^{135m}Ba NUCLEAR REACTIONS

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The nuclide ^{135m}Ba ($T_{1/2}$ = 28.7 h) is an extremely efficient medical radionuclide. Through the isomeric γ transition (E_γ = 268.24 keV) it decays into the ground state ^{135}Ba (see Table 1). The data given in Table 1 were taken from [1].

In the present study we measured the dependence of the ^{135m}Ba yield on the energy of the bombarding particles during irradiation of cesium with α particles and lanthanum with protons. We irradiated the compound CsCl; the factor for the conversion of the ^{135m}Ba yield to pure cesium is 1.4; lanthanum was irradiated in the form of metal specimens. The energy of the bombarding particles was varied with aluminum retarding foils. The procedure for irradiating the specimens and measuring the ^{135m}Ba activity and the integrated charge of the beam with a copper monitoring foil was similar to that described in [2]. The ^{135m}Ba activity was measured by the photopeak of the 268.24-keV γ-ray lines. The results of the measurements of the ^{135m}Ba yields are given in Fig. 1. The error of the experimental values of the yields is ±10-12% and is due in the main to the systematic errors in the measurement of the ^{135m}Ba activity and the integrated charge of the beam.

TABLE 1. Characteristics of ^{135m}Ba Quantum Radiation

Radiation energy, keV	Transition probability, %	Quantum yield, %
γ 268.24	100	15.6
K_α 32.06	49.9	44.4
K_β 36.56	11.2	9.9
L_X 4.47	74.1	11.4

Note. For γ radiation α_K = 3.91, α_L = 1.23, and α_N = 5.42.

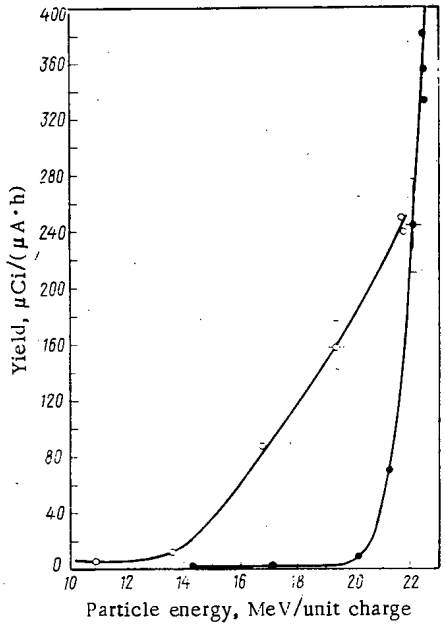

Fig. 1. Plot of ^{135m}Ba yield vs energy of particles during irradiation of thick targets of cesium with α particles (○) (2×) and of lanthanum with protons (●).

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The measured yield curves graphically demonstrate the role of the Coulomb barrier for protons and α particles emitted in (α, pn) and (p, αn) reactions. The effective threshold of the A(a, bxn)B reaction can be calculated from the formula $E_p^{eff} = (-Q + U_B)(A + a/A)$, where Q is the thermal effect of the reaction, $U_B = 0.96 Z_B Z_e/(B^{/3} + e^{1/3})$ is the Coulomb barrier of the emerging particle. In the given case $E_p^{eff} = 22.6$ MeV for the (α, pn) reaction and $E_p^{eff} = 18.8$ MeV for the (p, αn) reaction. It is precisely at these values of the energy of α particles and protons that the yield begins to grow rapidly (see Fig. 1), although the energy threshold of these reactions are 15.5 and 2.66 MeV, respectively.

It is of particular interest to obtain ^{135m}Ba by irradiating lanthanum with protons. This method has been proposed for the first time in the present paper. In the cyclotron of the Physics and Power Engineering Institute at Obninsk (FÉI) at maximum particle energy the ^{135m}Ba yield for the La +p method is four times that for the CsCl +α method and, moreover, under the conditions of commercial production of ^{135m}Ba a target of lanthanum metal can withstand a higher (4-5 times higher) beam power than a cesium chloride target can. This means that the output of the La +p method is roughly 15-20 times that of the CsCl +α method. It is seen from Fig. 1 that when the proton energy is increased by a mere 1-2 MeV the ^{135m}Ba yield grows severalfold.

Let us note that radioisotopically pure ^{135m}Ba is formed in both methods. This conveniently distinguishes ^{135m}Ba from ^{133m}Ba used in medicine ($T_{1/2} = 38.9$ h) which always contains a ^{133}Ba impurity ($T_{1/2} = 10.35$ yr).

Published results of measurements of the functions of excitation of (p, αn) reactions on nuclei of ^{26}Mg [3], ^{69}Ga [4], ^{38}Sr [5], and ^{56}Fe [6] to a proton energy of 40 MeV show that the cross section for this reaction is quite large and in some cases reaches a maximum of 100 mb or more. In the presence of a proton beam of appropriate energy and intensity, therefore, the (p, αn) reaction can be used for commercial production of radionuclides. For example, from the ^{52}Mn, ^{65}Zn, and ^{84}Rb yield curves obtained by integrating the excitation functions of the (p, αn) reaction, as measured in ^{56}Fe [6], ^{49}Ga [4], and ^{88}Sr [5], respectively, it follows that as the proton energy increases to 36 MeV [the effective threshold of the ^{139}La(p, α 3n)^{113}Ba reaction] one might expect the ^{135m}Ba yield to increase roughly 50 times in comparison with the yield when $E_p = 22.5$ MeV.

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PARAMETERS OF ^{60}Co NEUTRON LEVELS

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The ^{59,60}Co cross sections must be known in order to ascertain the optimal conditions for the accumulation of ^{60}Co during irradiation of metallic cobalt in a nuclear reaction. At the present time there is only one paper on the measurement of $I_{n\gamma}$ and σ_{th} in which use is made of activation analysis [1] and no data at all are available on the total cross sections.

In the present paper the total ^{60}Co cross section was measured by the time-of-flight method on the neutron spectrometer of the SM-2 reactor [2]. A battery of helium counters served as the neutron detector. The measurements were performed on a 4096-channel time analyzer. The best resolution of the analyzer with a 91.7-m flight base was 20 nsec/m.

Radioactive ^{60}Co ($T_{1/2}$ = 5.27 yr) was obtained by irradiating metallic cobalt (99.98% ^{59}Co) in the SM-2 reactor. To reduce the effects of self-screening of the material under irradiation, the cobalt specimen was assembled from 15 metal plates (each measuring 1 × 1.28 × 8 mm), separated by 1-mm aluminum spacers. The irradiation was carried out in the vertical channel of the SM-2 reactor, the density of the neutron flux being $\varphi_{th} = 2.14 \cdot 10^{14}$ neutrons/cm^2 · sec. The ^{60}Co content was determined with irradiated indicator monitors placed over the entire surface of the specimen. The number of nuclei in the irradiated specimen was $(1.25 \pm 0.03) \cdot 10^{23}$ nuclei/cm^2 in the case of ^{59}Co and $(7.9 \pm 1.2) \cdot 10^{21}$ nuclei/cm^2 in the case of ^{60}Co. The principal error in the determination of the amount of ^{60}Co was introduced by the self-screening correction.

The transmission of the specimen was measured in the range from 0.01 to 3000 eV on the neutron spectrometer with an arrangement for performing experiments with highly active specimens. The statistical error in transmission on the limbs of resonances was 1-2% and the neutron background did not exceed 2%. Four levels pertaining to ^{60}Co were detected in the range of neutron energies studied. The resonance parameters Γ_n^0 calculated by the "method of areas" are 55 ± 8, 80 ± 15, 490 ± 70, and 97 ± 15 MeV for neutron levels at energies of 820, 1740, 2020, and 2505 eV, respectively (in this case, $\bar{\Gamma}_\gamma = 500$ MeV). Measurement of transmission in the thermal range made it possible to estimate the cross section for $E_0 = 0.0253$ eV for ^{60}Co within limits of 6 b. The resonance capture integral for ^{60}Co, calculated from the parameters found, is equal to 4.0 ± 0.7 b.

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TESTS ON ZIRCONIUM SUPERHEATING CHANNELS IN THE FIRST UNIT AT THE KURCHATOV BELOYARSK NUCLEAR POWER STATION

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One form of improvement to the RBMK reactors is nuclear steam superheating, which is envisaged, e.g., in the RBMKP-2400 design, whose output is to be 2400 kW [1]. Tests have been made on zirconium superheating channels denoted by PPK-Ts in the first unit at the Beloyarsk nuclear power station in order to check the design decisions and to complete the development of individual items in the channel and TVS fuel-pin assemblies.

The experimental PPK-Ts is of welded construction, viz., a body containing two TVS, one above the other, together with six rod-type fuel elements (Fig. 1). The body consists of outer and inner tubes. The upper part (outside the core) of the outer tube is made of stainless steel, while the lower part (in the core) is made of zirconium alloy. The steel tube and the tail are joined to the zirconium tube by means of inserted joints formed by a special welding technique.

The inner tube also consists of two parts: The upper part is made of stainless steel, while the lower (in the core) is made of zirconium alloy. The parts are joined in a similar fashion. The temperature of the zirconium alloy is reduced by a thin-walled heat screen made of stainless steel inserted on the inside of the tube. Graphite sleeves are fitted on the body.

The coolant enters the PPK-Ts via the inlet tube and passes through the annular gap between the outer and inner tubes to enter the cavity in the tail, from which it rises along the space between the fuel pins and then passes out through the outlet tube. The annular gap between the thermal screen and the internal tube receives only a small amount of steam, and this should reduce the heat leak from the TVS to the zirconium tubes substantially.

The PPK-Ts units were set up in working sections of an AMB-100 reactor in the first unit at the station and were connected to the circulation loop in the same way as standard PPK units. Figure 1 [2] shows the essence of the way the PPK-Ts is connected to the loop.

Neutron-Physics Tests. These were performed in accordance with a special program, and the following conclusions can be drawn.

1. Measurements on the neutron flux near the PPK-Ts for various types of channel have shown that it has virtually no effect on the neutron flux in adjacent cells. Therefore, the change in neutron flux on installing the PPK-Ts can be neglected.

2. The axial gap at the junction between the upper and lower TVS units shows a stepout in the neutron flux; this stepout at the center of the gap (maximum value) is about 35% if there is no water in the channel, while the stepout at the boundary of the fuel pins is about 13%. Water in the channel increases the stepout at the boundary of the fuel pins to about 20%.

3. The neutron distribution over the perimeter of the fuel pins has its maximum at the point most remote from the center of the channel. The nonuniformity factor for a channel containing no water is 1.18, while the maximum differences between the neutron fluxes for the different pins are represented by $K_{max} = 1.35$. Water in the channel increases the first quantity by a factor 1.1, whereas K_{max} is increased by a factor 1.2.

Flushing the PPK-Ts. The PPK-Ts was run up to power in accordance with the rules for starting up the first unit [3]. An important operation here is to flush the PPK, namely to transfer from water cooling to steam cooling and run the reactor with steam superheating. The reactor parameters for the PPK flushing were as

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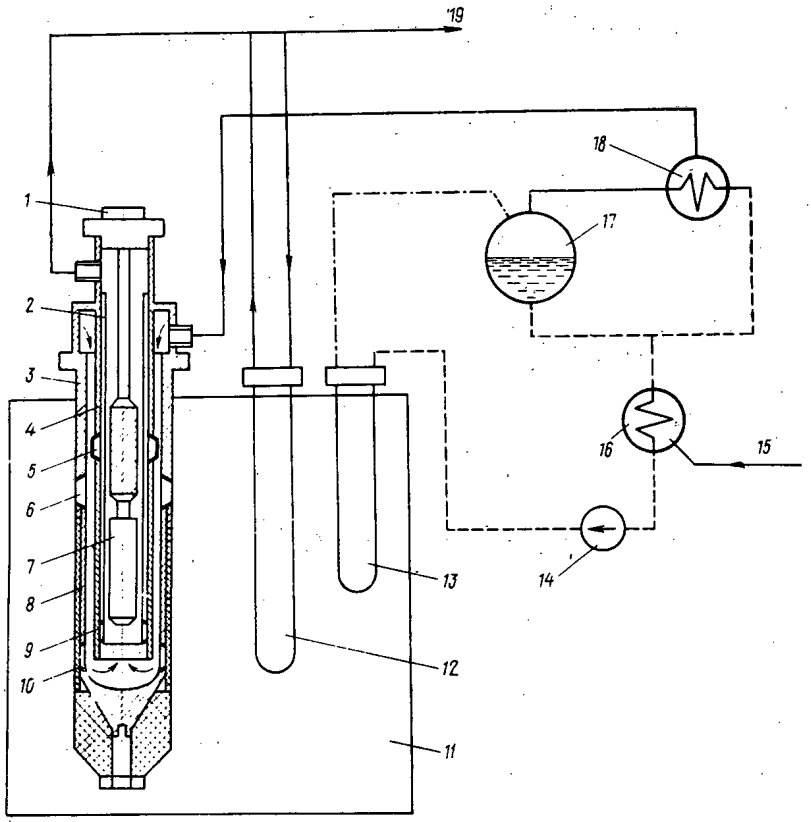

Fig. 1. Major aspects of the channel design and connection to the circulation loop: 1) PPK-Ts; 2) thermal screen; 3, 4) outer and inner tubes; 5) joint in inner tube; 6) upper joint on outer tube; 7) TVS; 8) graphite sleeves; 9) seal between thermal screen and inner tube; 10) lower joint of outer tube; 11) reactor; 12) standard PPK; 13) evaporator channel; 14) principal circulation pump; 15) feedwater; 16) preheater; 17) separator; 18) evaporator; 19) steam to turbine; ——) steam; – – –) water; —· ——) steam–water mixture.

follows: power level 10% of nominal, water flow rate in each PPK 1-1.5 tons/h, feedwater temperature 190°C, pressure in the generator evaporators 25 kgf/cm^2 above the saturation temperature, and water temperature after the PPK 260-265°C.

During the PPK flushing, the reactor power is reduced to 2.5%, while the feedwater input to the loop is stopped, and the flow rates along the channels are increased on account of the accumulated heat in the loop, which displaces the water from the principal steam pipes into a special vessel. After 20-25 min, water is admitted to the evaporators and the temperature of the superheated steam leaving each PPK rises, which indicates that the flushing is complete. The power level is then gradually raised.

Raising Power Level. When the PPK has been flushed, any rise in power level is accompanied by a sharp increase in the steam temperature at the exit from the PPK-Ts. The maximum rate of temperature rise immediately after flushing occurred in the inner tube of the PPK-Ts (20°C/min) and the rate of temperature rise in the steam at the outlet from the PPK-Ts was 3.3°C/min, while the standard PPK gave a rate of change of temperature in the steam at the outlet of not more than 1.3°C/min. Temperature measurements on the tail of the TVS showed that the substantial heat transfers between the rising and descending flows may be due to unsatisfactory operation of the seal around the heat screen in the lower part of the channel. As a consequence, the power loss in the cooler part of the channel between the top of the core and the outlet steadily increases as the PPK-Ts heats up (to 25 kW). Considerable power losses occur also in the region between the outlet and the superheated-steam collector (11 kW). The reactor power level at that time was 11.5-12.2% of nominal, while the steam temperature in the inner tube of the PPK-Ts had attained its maximum value. This time interval (that required to raise the power level after flushing from 2 to 12%) is one in which there is a rapid rate of temperature rise in the PPK-Ts consequent on the rise in power level due to the low steam flow rate in the channel. A similar effect, but rather smaller, occurs for the rate of temperature rise for the standard PPK.

Any further rise in reactor power did not cause any further temperature increase in the PPK-Ts, and in fact there was a fall. Operation at 80% power and transient run-up in the range 80-100% power gave rise to a steam temperature at the outlet of the PPK-Ts of 520°C, which was virtually the same as the mean temperature of the superheated steam from the standard channel.

CONCLUSIONS

The experimental PPK-Ts units built to test the design principles for the RBMKP-2400 draft are viable; the PPK-Ts channels provide nuclear superheating of the steam to 520°C at a pressure of 90 kgf/cm^2. The channels are transferred from water cooling to steam cooling in accordance with the standard rules for starting up the first unit. Experience with starting up the first unit containing the PPK-Ts has shown high rates of rise in the steam temperature at the exit from the channel after flushing the PPK-Ts on account of large heat leaks within the channels. This feature must be eliminated in finalizing the design of the PPK-Ts and defining the working conditions more closely.

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IN MEMORY OF YURII ARONOVICH ZYSIN

On Oct. 25, 1978, Professor Yurii Aronovich Zysin died. He was a prominent scientist, a Lenin and State Prize Laureate, and a doctor of physicomathematical sciences.

Zysin was born on Nov. 5, 1917, in Leningrad. In 1933 he entered the physics department of Leningrad State University (LGU). He did his undergraduate work in the newly completed cyclotron laboratory of the Radiev Institute. After graduating from LGU in 1939 he did his graduate work at the Institute of Chemistry and Physics of the Academy of Sciences of the USSR, where he began his multifaceted career. In this period, Zysin published several computational and experimental papers. In June 1941 Zysin volunteered for the front. In 1943 he was recalled to Moscow to participate in work on our country's first radiolocation stations. In this connection, in 1943-44 he developed the first gas-filled trigatron in the USSR, defended his candidate's dissertation, and published several articles.

A significant stage in Zysin's career is connected with nuclear physics. He carried out a great cycle of measurement of elementary and group constants which are necessary for the development of nuclear engineering. His experiments were marked by originality and a particular thoroughness. He proposed and worked out a study of fission products. In 1955 he defended a doctoral dissertation on this theme.

Yu. A. Zysin (along with A. A. Lbov and L. I. Sel'chenkov) correlated the results of his measurements and the data available in the literature in the reference work Fission Product Yield and Its Mass Distribution (1963), which was published in other countries.

Zysin's scientific interests were very broad. Along with research into nuclear physics, he worked much and with great success in the field of creating physical devices, new research methods, and measuring instruments. He did much for the development of a new area – the physics of high energy densities.

Zysin actively propagandized scientific achievements, often came forth with scientific-technical and sociopolitical questions, and was one of the first to popularly describe, in "Nature" (1945), the possibilities of using atomic energy. In science and in life he was modest and a convinced, principled communist. He was characterized by his broad viewpoint, his goodwill, and his sensitivity in his relations with people. He seriously enjoyed

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art and literature, and did much drawing. Zysin was a marvelous scientific organizer, an educator of youth, and a brilliant and interesting lecturer. His students, among which are more than 50 candidates and doctors, successfully work in many scientific and educational institutions in our country.

The government had highly valued the services of Yu. A. Zysin, having awarded him one Order of Lenin, two Orders of the Red Banner of Labor, the order "Badge of Honor," and many medals.

Yu. A. Zysin lived a brilliant life. Pleasant memories of this extraordinary scientist and human being will always be preserved in the hearts of all who had the opportunity of meeting him.

COLLABORATION DIARY

The Sixteenth Meeting of the Scientific–Technical Coordinating Council (STCC) on Radiation Technology of the COMECON Permanent Commission on Atomic Energy was held in Moscow on Sept. 5-8, 1978. The solution of problems raised by the Commission's resolutions was examined. In connection with this in the future, basic construction of radiotechnological installations, as well as methods and apparatus for monitoring and control of technological processes, are proposed to be worked out on a mutually agreeable basis between interested organizations of the COMECON member-nations, which usefully manage the unification of groups of radiotechnological and radiotherapeutic installations for remote and intraenclosure irradiation, and to work out recommendations regarding their industrial manufacture and exploitation, etc. In particular, suggestions were examined, which will be submitted in the first half of 1979 for the planning of timetables of work in the development of:

installations with an electron accelerator for the radiation modification of rolled goods (fabrics, glass-plastics, leather goods);

gamma installations for the manufacture of concrete–polymer materials and products from radiation-modified timber;

gamma installations for the radiation sterilization of medical products;

installations with an electron accelerator for the radiation cleansing of communal and industrial sewage waters;

standard installations for the nondestructive monitoring of networks and parts of nuclear-energy systems by using neutron radiography.

The STCC on Radiation Technology submitted the proposals for the draft of the Commission's working plan in the field of radiation engineering for 1979-1980, among which is a proposal for standardization; it examined the results of comparisons of electron-irradiation dosimeters manufactured in the German Democratic Republic, the Polish People's Republic, and the USSR; it recommended a unified dosimetry system, developed in the USSR, comprising electron-irradiation dosimeters of type DRD-2M, DPP-2/10, DRPK-TTKh, TsDP-B4S-2, PDP-S480-2, and DPTs-2/25, as well as a color indicator of absorbed dose of type TsVND-3.

* * *

The Meeting of Specialists on the Radiation Sterilization of Materials and Products Intended for Medical Use was held in Moscow on Sept. 5-8, 1978. It supplied a summary of scientific–technical collaboration in this field in 1976-1978, and recommended use of the following standard procedural documents in the COMECON member-nations:

recommendations on the heat-radiation methods of sterilizing several types of medical products;

gaschromatic procedure for determining the residual composition of a monomer (methylmethacrylate, styrene) in polymer materials on the basis of acrylates, styrenes, and their copolymers;

procedure for monitoring pyrogenicity of radiation-sterilized single-use medical instruments having come into contact with blood;

spectrophotometric procedure for evaluating the quality of plastic products which have undergone radiation sterilization;

procedural recommendations on the prognostication of sanitary-chemical properties of irradiated products made of polymer materials intended for medical purposes;

recommendations on the choice of a sterilization radiation dose.

The specialists evaluated drafts of several other standard procedural documents and heard information regarding research conducted in this area in the COMECON member-nations.

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The Ninth Meeting of the STCC on Research Reactors was held in Warsaw on Sept. 12-15, 1978. Data of the COMECON Secretariat on the completion of the Commission's activities and its working organs were evaluated. A report was submitted on the results of plan fulfillment in the development and improvement of research reactors and of the work done on them, among which is the enumeration of completed work and proposals for the completion of ongoing work. Plans were discussed and submitted regarding safety regulations for research reactors and critical facilities and will be presented to the COMECON Permanent Commission on Atomic Energy for examination. A first draft of regulations for safety procedures at research reactors and critical facilities (general principles, departmental licenses, monitoring, and regulation) was examined; this draft had been prepared by Hungarian specialists.

Specialists of the COMECON member-nations exchanged information on work done in 1978 on several aspects of the problems "Development and improvement of research reactors and the conduct of work using them in the field of reactor physics and technology" and on proposals for introducing finished work into practice. Reports were presented by Czechoslovakian and Polish specialists on "Basic characteristics of the VVR-S Rzhezh reactor after its reconstruction" and "The 20-yr operation of the VVR-SM reactor in Poland."

Participants of the meeting approved a draft of a working plan for the COMECON Permanent Commission on Atomic Energy for 1979-1980 in the area of research reactors.

* * *

The Tenth Meeting of the Council of International Departmental System of Scientific and Technological Information on Electric Energy was held in Nesebur (Bulgaria) on Sept. 26-29, 1978. The condition of the International Departmental System of Scientific and Technological Information on Electric Energy (MOSNTIÉÉ) was discussed and the long-range development program for 1979-1985 was ratified. This plan anticipates the creation of an International Departmental Automated System of Scientific and Technological Information (MOASNTI) on Electric Energy and an International Departmental Data Retrieval System on Atomic Energy (MOAFIPS). The system will expedite the collection, processing, storage, retrieval, and dissemination of data on basic technological and technioeconomic parameters (constant and variable) for nuclear power plants with water-moderated—water-cooled (VVÉR) reactors, and "accident data." This will considerably lower the expenditures for processing and transmitting such information and will serve to increase the quality and operativeness of supplying information on scientific—technicoeconomic problems in the development of nuclear power engineering among the COMECON member-nations.

* * *

The Twenty-Seventh Meeting of the Working Group on Reactor Science and Technology and Nuclear Power Engineering was held in Hermsdorf (GDR) on Oct. 3-6, 1978. The procedure and time limits were discussed for the preparation of draft agreements and working plans regarding problems in nuclear power engineering. These were included in a long-range target program of collaboration to secure the economically grounded energy, fuel, and raw-materials requirements of the COMECON member-nations. Information was submitted on the results of cooperation, during the elapsed portion of 1978, in the area of reactor science and technology and nuclear power engineering, as well as on completed works and proposals regarding their introduction into practice. These materials have been presented for inclusion in the draft of the report to the COMECON Permanent Commission on Atomic Energy on the work done in 1978.

Proposals were submitted on collaboration in the creation (as a result of the joint efforts of the COMECON member-nations) of a standing base for the experimental testing and improvement of instruments and equipment for power-reactor installations. The progress in the preparation of drafts of agreements and working plans were discussed on selected questions that are included in the submitted plan of the many-sided, integrated measures of the COMECON member-nations for 1976-1980. The working plan of the COMECON Permanent Commission on Atomic Energy for 1979, in the area of reactor science and technology, was discussed and approved.

The Working Group considered the proposals of the Republic of Cuba related to the accelerated development of science and technology in that country.

* * *

The Fourteenth Meeting of the STCC on Fast Reactors was held in Moscow on Oct. 3-6, 1978. Materials were submitted on the progress in fulfilling the working plan of cooperation for the development of a high-

power, fast-neutron reactor installation with a sodium heat carrier for the period September 1977-August 1978, and results of the symposium on the heat physics and fluid mechanics of the reactor cores and steam generators for fast reactors were discussed. Proposals directed toward the solution of basic problems in this area were examined. The participants of the meeting heard with interest from Soviet specialists, who discussed the progress in the assembly of the BN-600. Bulgarian specialists reported on the progress in the preparation of an atlas of calculated nuclear data, and on the sequence and times of its publication.

The participants discussed drafts of an agreement on collaboration in the fulfillment of scientific-research and experimental-construction work being conducted within the framework of the COMECON Permanent Commission on Atomic Energy on the development of high-power, fast-neutron reactor installations, and on programs for the conduct of work along the lines of this agreement. The progress in preparing an agreement and a plan of cooperation for the development of a steam generator for a nuclear plant with a BN-1600 was discussed.

Plans for the work of the COMECON Permanent Commission on Atomic Energy in the area of fast-neutron reactors and of the STCC for 1979-1980 were submitted, as were plans for the pace of work on the theme "Measurement of neutron spectra in fast-neutron systems."

MEETING OF THE WORKING GROUP ON POWER ENGINEERING

M. B. Agranovich

The Fourteenth Meeting of the Working Group on the Organization of Cooperation in the Field of Power Engineering Management of the GDR–USSR Intergovernmental Commission on Economic and Scientific–Technical Cooperation was held in Moscow on Oct. 17-20, 1978. The Soviet delegation was led by the Minister of Power and Electrification, P. S. Neporozhnii; the German delegation, by Minister of Power and the Coal Industry, K. Zibol'd.

The state of power management in both countries was discussed, as were perspectives of scientific–technical cooperation in 1979-1980. The fruitful and purposeful cooperation between specialists of the Soviet Union and the German Democratic Republic in the equipping and operation of thermal and nuclear power plants in the GDR was noted. Information was heard from German specialists on the operational experiences of the first three units, on the completion of assembly and preparation for start-up-related adjustments of the fourth unit, and on the pace of construction of the fifth through eight units of the "Nord" atomic energy plant. Scientific–technical cooperation for 1977-1978 was approved in the following areas:

working out of technological and design decisions for large energy units with the aim of further raising the reliability of their operation and repair;

improvement of assembly technology for new, powerful thermal and atomic energy plants and adoption of new construction assemblies for atomic power plant structures and installations.

The working group supported the results of the cooperation between the NVAÉS Atomic Energy Plant and the "Nord" atomic plant, and confirmed its interest in its further development and broadening.

The working group delegated to experts from interested organizations of both sides the responsibility of preparing proposals on the theme and program of joint work, from which, at the request of German specialists, must originate the realization of scientific–technical cooperation in the following areas in 1979: improving nuclear safety in the operation of nuclear power plants; using noise emission for the monitoring of materials behavior during operation; development of automated monitoring systems for safety and protection; inclusion of systems, developed in the GDR, of noise diagnostics in the design of 440- and 1000-MW reactor installations; development of new procedures and equipment for deactivating the primary loop of an atomic reactor, as well as methods of inspecting equipment, tools, and devices for monitoring the equipment in the primary loop while the block is not operating. The German proposals are oriented toward cooperation in the construction of reactors of 1000-MW power, with minimum unit and construction expenditures, the development of construction–technological methods and equipment for the construction of reactor shells, the utilization of discarded heat from an atomic plant, and other problems of power engineering management.

During the meeting of the working group, a smaller meeting took place between representatives of the State Planning Commission of the Council of Ministers of the USSR (Gosplan) and the State Planning Commission of the GDR on long-range questions, in particular, on the erection of power engineering equipment in the GDR within the framework of the five-year plans for 1981-1985.

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CONFERENCES AND SEMINARS

CONFERENCE ON THE 30th ANNIVERSARY OF ISOTOPE PRODUCTION AND USE IN THE USSR

A. K. Kruglov and N. A. Matyushina

This conference was held in October 1978 in Obninsk and was concerned with the production and use of isotopes, with emphasis on the last 10 years. There were about 400 participants from various organizations under the State Commission on Atomic Energy, the Academy of Sciences of the USSR, the academies of the union republics, the All-Union Academy of Agricultural Sciences, the Academy of Medical Sciences of the USSR, various ministries and departments, and representatives of other socialist countries. There were 42 papers presented at the conference. On the first day, papers were read on isotope production and the preparation of radioisotope radiation sources, thermal isotope units, labeled compounds, and radiopharmaceuticals.

Over 160 radioisotopes covering most of the elements in the periodic system are not utilized, which range from tritium to californium inclusive, which are derived by extraction from fission products (^{90}Sr, ^{137}Cs, etc.), by irradiation of stable materials in nuclear reactors (^{14}C, ^{60}Co, etc.), or from cyclotrons, etc. The production of ^{238}Pu has also begun. Over 700 types of α, β, γ, and neutron sources of radioisotope type have been produced for radiation-treatment systems, flaw detection, x-radiography, activation analysis, and various engineering monitoring instruments, in addition to about 1000 standard or reference-series sources for special purposes.

The production of heat sources based on ^{90}Sr and ^{238}Pu has been organized in order to meet a rapidly developing demand; also, 76 radiopharmaceuticals have been produced, which provide means of performing about 130 diagnostic tests and 20 radiotherapeutic treatments. The extension of the range of these in 1968-1978 was accompanied by a considerable increase in the volume of production, which doubled in money terms in that period or by more than a factor 2.5 by number of items, which has attained 80,000 per year.

The general use of radioisotope methods in molecular biology, biochemistry, immunochemistry, and other areas would have been impossible without the organization of production of biologically significant compounds labeled with T, ^{14}C, ^{32}P, and other isotopes. Some 150 new compounds labeled with these radioisotopes have been produced in the past ten years, in addition to about 100 deuterated compounds; further, the volume of production has almost doubled. Major advances have also been made in the accumulation, isolation, and use of substantial amounts of the transplutonium elements, including ^{252}Cf, which are derived from a high-flux reactor. Miniature californium sources giving neutron yields of 10^6-10^{10} neutrons/sec have been utilized prospecting for and surveying deposits of oil, gas, and other mineral resources, as well as in neutron radiography and neutron-activation analysis. Also, ^{252}Cf sources have been used in medicine in improving the treatment of malignant tumors. Participants from abroad also gave reports on advances made in the production and use of isotopes derived from reactors and cyclotrons.

The second day was concerned with papers dealing with radiation engineering and the use of isotopes and nuclear radiations in various areas in engineering, agriculture, and biology.

These papers indicated that radioisotope methods and equipments have become a powerful means of advancing science and engineering; the following major areas have developed in radiation engineering in the last ten years: equipment design, instrumentation, and energy production. Prototype or working industrial systems have been designed or commissioned involving about 50 processes in radiation engineering. The following equipments have been designed or commissioned: The rV-1200 for the production of heat-resistant electrical insulating materials, the GUD-300 for the production of components made of modified wood, the Sterilizatsiya II and III units for industrial radiation sterilization of plastic medical items, equipments for producing wires and cables with irradiated polyethylene insulation, a series of systems for hardening coatings, the Kolos units for irradiating seeds before sowing, the Pant system for the synthesis of dibromylthiotoluene, the Poplin system for radiation-chemical treatment of cloth, and the Plasma system for removing hydrogen from chlorine. Design work is in hand for new generations of radiation-engineering systems de-

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signed particularly for chemical and biological processes, environmental protection, medical-instrument sterilization, modification of various materials, and processing agricultural and food products.

Radioisotope instruments have become a common means of monitoring and automation in industry; the demand for these increases every year. In 1968, ferrous-metallurgy plants operated about 1200 different radiation instruments, but the number almost doubled in the next five years. Some 80,000 radioisotope instruments were supplied to industry over the complete decade.

Many studies have also been performed to improve methods and means of flaw detection with ionizing radiation, particularly in monitoring materials, components, assemblies, and welded joints. These developments have met the needs of major areas in industry for methods and means of nondestructive testing. A new trend in flaw detection is the use of spatially encoded sources of ionizing radiation, which can provide three-dimensional information.

Methods derived from nuclear physics have become widely adopted in laboratories, industrial organizations, and geological bodies for the analysis of materials by means of radioisotopes. The past decade is essentially the period when these methods attained wide industrial use. One of the most promising trends in the development of analysis methods is flow monitoring, which provides a basis for comprehensive automation of many industrial processes. Detailed studies have made it possible to produce economic thermal and electrical power sources on the basis of ^{90}Sr. Over 300 of these devices have now been manufactured and put into use, and the total working time of these is about 9 million h. Sources under the designation Beta-M are in routine production, whose electrical output is 10 W and which are used in the main to supply automatic meteorological units in areas of very harsh climate. These units have proved to be the only reliable ones for such conditions. They have also been used to advantage in navigational equipment in major areas of the Arctic Ocean, the Baltic, the Barents Sea, and the Sea of Okhotsk (radio buoys, lighted buoys, and magnetic-variation stations).

Radioisotopes and ionizing radiations have become essential research tools in various branches of agricultural science (research on the metabolism of biologically important elements in plants and animals, and research on the production of meat and milk by livestock). Radiation-induced mutations have also provided a basis for regionally adapted agricultural plants.

Considerable interest was aroused by papers on the use of stable isotopes in agriculture and on the use of radioisotopes in molecular biology. The first of them dealt with the major role of ^{15}N in research on nitrogen fertilizers. These results have produced a radical alteration in the views of agrochemists on the practical significance of individual processes in the conversion of nitrogen fertilizers in the soil–plant system. Researches are in hand for ways of improving the return from nitrogen fertilizers. The second paper stated that some major discoveries in molecular biology had been obtained in the last ten years by means of isotope methods that could not have been made by any other method.

The state of the art and the development prospects for isotopes in medicine were dealt with extensively in papers presented by the leading institutes of the Ministry of Health, the Ministry of the Medical Industry, and the Academy of Medical Sciences of the USSR on the third day of the conference. These indicated that radiotherapy is one of the main means of treating malignant tumors. Today, radiotherapy is applied to over 70% of cancer patients as an independent treatment for the tumors. Design work has been completed on the AGAT-V1, AGAT-V2, and AGAT-V3 intracavity irradiators, which have been put into routine production, since these represent a new and progressive approach in radiation treatment. Clinical radioisotope diagnostic techniques have become essential in the proper evaluation of many diseases or functional states. In 1977, over half a million patients were examined by radioisotope means, including diagnosis of myocardial infarcts, ischemic heart disease, hypertension, malignancies, and the like. Also, radioisotope methods have begun to play a substantial part in space medicine (blood redistribution under conditions of weightlessness, shifts in calcium metabolism, and the like).

Considerable interest was aroused by a paper on the state of the art and development prospects in the use of implanted devices having radioisotope power supplies employing ^{238}Pu. These biomedical ^{238}Pu sources are of high specific electrical output and very high stability, and they are often the only fully acceptable power source for cardiac pacemakers. Several basic types of Soviet cardiac pacemakers now employ ^{238}Pu sources, and dozens of them are now in use. Researches are also in hand on the use of radioisotope power supplies for other implanted systems such as artificial hearts, while there are many other applications of radioisotopes in research in chemistry, physical chemistry, metallography, nuclear physics, and techniques based on isotope analysis.

The president of the Academy of Sciences of the USSR, A. P. Aleksandrov, addressed the conference and stressed the progress made by the USSR and other socialist countries in this major area. In particular, he pointed out that the production of hundreds of radioactive and stable isotopes began during the past decade, as well as of thousands of sources and compounds employing radioisotopes. Highly valuable processes in radiation engineering have been designed and implemented, as well as radioisotope methods of engineering monitoring and high-sensitivity analysis methods based on activation and x-ray fluorescence. On the other hand, there are some organizational deficiencies that have held back the exploitation of the advantages of radioisotope methods in industry, science, and medicine. Major advances have been made in the synthesis of labeled compounds, but here Aleksandrov stated that the range still does not meet the requirements of molecular biology and medicine. The most important way of overcoming this deficiency is to extend cooperation and specialization in the production of labeled compounds and preparations within the COMECON framework.

Another major shortcoming is the low rate of progress in the design, engineering, and production of nuclear-physics equipment in the USSR, particularly as required for operations with isotopes. There has been very poor coordination of efforts in the production and use of isotopes and isotope technology, as the work has been widely spread between various ministries and departments, and prompt and effective organizational measures are required to overcome this.

In conclusion, Aleksandrov wished the participants fresh successes in the production and use of isotopes.

SEMINAR ON REACTOR ENGINEERING

R. R. Ionaitis

This session of the regular seminars on the engineering and economic aspects of nuclear power was held May 31, 1978 in Moscow to celebrate the 25th anniversary of the Power Engineering Research and Design Institute. The meeting dealt with aspects of advances in nuclear power engineering and research reactor construction, as well as with major research and design aspects of this area in science and technology generally. The seminar was under the direction of Academician N. A. Dollezhal and was attended by 150 leading workers from 47 organizations; 9 papers were read and discussed.

Dollezhal opened the seminar with a paper entitled "Scientific and engineering problems in nuclear power development," in which he sketched the history of this branch of engineering, the current state of the art, and the existing major problems, which are as follows: 1) cost reduction and accelerated construction for nuclear power stations by increasing the unit power of reactors and other major installations, along with improvement in the efficiency, standardization of equipment and components, the use of cheaper materials (without adverse effects on the reliability), and particularly, reduction in the amount of metal required;

2) Provision of high reliability and economy in nuclear power stations by the use of higher-grade grade equipment, reduction in downtime due to reloading and repair operations, closer observance of planned operating schedules, automation of reactor management, and equipment of cores with new measuring instruments;

3) Exploitation of unused reactor power consequent on fresh research in hydrodynamics, heat physics, reactor physics, and materials science, particularly with a view to operation of components under highly stressed conditions;

4) Design of nuclear power stations with features such as operation with highly variable loading, in particular, by the design of fuel pins capable of withstanding long-term cyclic loading, which involves research on the reliability of many components under variable-load conditions; and

5) Design of reactors for district-heating and high-temperature techniques.

I. Ya. Emel'yanov presented a paper on the development prospects for water–graphite channel reactors (in particular, reactor dynamics and control). Researches on the latter aspect were begun under Kurchatov's direction with the first comparatively small reactors. Key aspects governing progress in modern power re-

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actors, particularly the RBMK type, are as follows: optimal core structure, dynamics of the energy production over the core volume, optimization of core dynamic characteristics, synthesis of automatic systems for stabilizing and regulating the power production in the core, utilization of control systems already designed, comprehensive automation of reactor and power-station management, and extension of researches on existing reactors. Emel'yanov pointed out the effects of fuel-cycle characteristics on the energy distribution, particularly the complexity of the spatial and time trends in the latter, which means that three-dimensional models are essential in calculations. There are also problems in determining the nonstationary distortion of the power production and in defining the discrepancies between theory and experiment. It was pointed out that instability in the energy distribution is inherent in high-power reactors, and the processes are characterized by periods of seconds as well as of hours. As yet, there is no rigorous theory or synthesis method for the control systems needed for such reactors, and considerable reliance must be placed on the development of control systems by means of models, with subsequent checkout on actual reactors.

Therefore, the design of a new generation of reactors for high-power nuclear systems requires the solution of some essentially novel but very complicated problems in design and control. Studies in this area constitute a new stage in reactor design, and the scientific and engineering aspects of reactor design are basic to the construction of improved nuclear power stations of higher output. Channel reactors of output 1500 and 2400 MW have already been developed on this basis. The development prospects in these areas must involve improved theories and better computation methods, as well as extended and improved experimental techniques for researching the processes in existing reactors.

F. M. Mitenkov dealt with past results and development prospects for fast sodium-cooled reactors; these reactors represent a radical means of reducing the demand for uranium. The most important problems in this area are concerned with the design, manufacture, and operation of components of the sodium loops, as well as safe and reliable operation of the steam generators, radiation shielding, fuel-rod viability, and safety in nuclear power stations based on such reactors. The BN-350 was commissioned in the USSR in June 1973, and the equipment on the whole has operated reliably and in a stable fashion, although there have been difficulties on account of poor manufacture of certain components. The construction of the BN-600 has been completed, and design work is in progress for the main unit in a nuclear power station of output 1600 MW. The basic design philosophy of a fast sodium-cooled reactor is now reasonably well-defined, and appropriate experimental researches have been performed, while experience has been accumulated in design and manufacture of components, so optimization of general schemes and design details is now in hand.

Experience with the design, manufacture, and improvement of reactors of the VVÉR series was discussed by V. V. Stekol'nikov, These reactors routinely produce power outputs of 440 MW each. The economic parameters (low cost per kWh, high installed-power utilization factor) have been improved by increasing the unit power of the equipment, raising the working parameters of the coolants, use of chemical (boron) reactivity control, improved reliability in the power supply to the main circulation pumps from the use of special generators, and reduction in the number of buildings involved by placing most of the units in a single building and installing two reactors in a common building. New and advanced design features have been introduced into the VVÉR systems, which are particularly intended to improve the reliability and the working conditions, e.g., high-tensile steel components, integral reactor bodies without longitudinal welds, and so on. The utilization factor for the VVÉR is now commonly 0.7-0.8. Stekol'nikov also emphasized the disadvantages of nuclear power stations containing VVÉR reactors: The running-in period is 2-3 years, there are frequent failures in the circulation loops, and design errors have been discovered. Design and engineering measures are presently in hand to eliminate the shortcomings. The main VVÉR-1000 reactor will very shortly be completed, in which these improvements are expected to result in a very low cost per installed kWh. Particular attention is to be given in future in the design of power reactors of this type to further reduction in the construction costs, the use of multilayer containment vessels, reduction in the downtime for fuel changing, standardization and unification, reactor use in semipeak mode, increased specific energy output, reduction in the nonuniformity in the power production, and general optimization of the power station as a whole.

V. V. Orlov discussed the state of the art and development prospects in controlled thermonuclear synthesis; thermonuclear reactors could use inexhaustible fuel resources if the appropriate high temperature can be produced. It also seems that they may be cleaner than ordinary nuclear reactors, while the combination with ordinary reactors might be used to advantage in the exploitation of the high yield of high-energy neutrons. Designs and problems were discussed, in particular, toroidal reactors of tokamak type.

V. A. Sidorenko discussed mainly safety and environmental protection; the amounts of radioactive substances moving about the country increase every year on account of fuel shipping, reprocessing, and waste

storage. In spite of this, the statistics show that nuclear power is actually safer than thermal power. The discussion also touched on emergency states in nuclear power stations, reactors, and fuel rods. There was a particularly detailed discussion of the effects of the distance from the reactor to inhabited points under various circumstances (accidental breakdown, fire, aircraft crashes, and terrorist attacks).

B. B. Baturov discussed the operation of nuclear power stations under variable-load conditions; this has to be related to the demand in electricity systems, the increasing outputs from nuclear power stations, and the design features of Soviet reactors and power stations, particularly the static and dynamic features. It was demonstrated that nuclear power stations must participate in meeting the load under normal and emergency conditions, but the specifications for flexibility vary from one type of nuclear power station to another and cannot be the same as specifications for stations employing fossil fuel. The main factors restricting the flexibility of nuclear power stations containing standard high-power reactors are the inadequate fuel-rod viability on repeated cycling and the deformation of the power-distribution pattern arising when the power level is raised rapidly. Participation of nuclear power stations in meeting load curves and emergencies will require the design of specialized units. The papers dealt with the design principles for a semipeak reactor, particularly on the basis of experience in the design and operation of channel-type reactors and experiments on direct-flow fuel channels. It is anticipated that the economic parameters should be reasonably satisfactory.

Yu. M. Cherkashov stressed the importance of nuclear power sources for centralized heat supply, and conditions were laid down under which these could be economically competitive; the discussion turned to designs for nuclear combined district-heating and electrical power stations and boilers, and also to various reactor designs for high and medium power levels. Nuclear fuel applied to the supply of industrial and domestic heat can make more rational use of nuclear resources. It is possible to design nuclear power sources that supply heat and that are safe from the radiation viewpoints and highly reliable, as is clear from the experience with the Beloyarsk, Novy Voronezh, and Ul'yanovsk nuclear power stations and the Bilibinsk combined district-heating and electrical power station. It is also possible that nuclear stations will be built to supply heat alone in the range from one to several hundred MW near inhabited points. Nuclear heat sources become reasonably economic in remote and inaccessible areas at heat loads of 25-50 Gcal/h or outputs of 10-25 MW.

The development stages in MHD systems were considered by É. É. Shpil'rain. Trends in the development of open-cycle MHD systems were outlined. The Soviet Union has made major advances with prototype MHD power stations, where the U-25 system has been operated for several years with an output of 20 MW. The design of this station involved solving some major problems concerning the principal items of equipment in an MHD power station. The prospects for using MHD techniques in nuclear engineering were also discussed, along with the scope for using MHD in conjunction with thermonuclear systems.

All of these papers aroused considerable interest in the participants.

IAEA CONFERENCE ON LEAK DETECTION IN FAST-REACTOR STEAM GENERATORS

A. S. Mavrin

This conference was held in Dimitrovgrad in June 1978; the participants and observers were drawn from Britain, the Netherlands, Italy, France, the Federal German Republic, the USSR, the U.S.A., and Japan.

The purpose of the conference was to consider and discuss methods of detecting and locating leaks. Particular attention was given to accumulated experience, quantitative definition of criteria, and the actual sensitivities of the available methods.

The following topics were discussed in detail:

1. Methods and equipment for identifying and locating leaks, including concentration, helium, and electromagnetic ones;

2. Acoustic methods and equipment for identifying and locating leaks; and

3. Methods of checking and ensuring complete sealing in steam generators during manufacture, installation, and repair, in addition to methods of inspecting the states of tubes in such generators during periodic checks and in damage evaluation, particularly the evaluation of damage arising from leaks, and how these methods affect the generator design.

In all there were 14 papers. Most of them were devoted to the detection and examination of leaks during steam-generator operation.

Concentration Methods of Leak Detection. At present, the main method of detecting leaks involves measuring the hydrogen levels in the sodium and in the protective gas. The hydrogen concentration after diffusion through a nickel membrane may be determined in various ways: by mass spectrometry (France, with katharometers (Britain), or with ion pumps. The state of development of these instruments is now such that they have been put into routine production and are being delivered.

A hydrogen detector developed at Berkeley uses an electrochemical cell; this was described by Hayes (Britain). The solid electrolyte is a mixture of $CaCl_2$ and CaH_2, while the standard electrode is a saturated solution of LiH in Li. A background hydrogen concentration in the sodium of $0.1 \cdot 10^{-6}$ results in the output voltage altering by 1 mV when the hydrogen concentration alters by $3 \cdot 10^{-6}$. The response time is 5 sec. The instrument has been developed for research purposes, but it has been proposed also for use in steam generators.

An oxygen transducer with a solid electrolyte composed of ThO_2 and 7.5% Y_2O_3 has been developed in Britain, the U.S.A., and the Federal Republic of Germany; air is the standard electrode. The limit of detection is $40 \cdot 10^{-6}$ for the concentration of O_2 in sodium (a change in output voltage of 2 mV against a background concentration of 10^{-6}).

R. Magee and others (U.S.A.) described a measurement system that includes a hydrogen detector with an ion pump and two oxygen electrochemical transducers, which reduces the probability of a false-positive leak signal.

Many studies on the development of small leaks have been performed in certain countries; it has been found that there is a tendency for a leak giving a flow rate less than 5 mg/sec in a pearlitic steel to become blocked in the early stages, and therefore it may not be sensed for a long period (up to 30 days). After this the leak increases very rapidly (within 1-2 min), and this in turn may lead to rapid failure in adjacent tubes.

A report from France stated that there is a second incubation period, in which the leak rate increases to 0.1 g/sec over 10 min. After this there is a rapid increase in the leak. The probability of failure in a blocked leak increases if there is any thermal shock. J. Smedley (Britain) reported a calculation that demonstrated that a defect of this order must be observed within 5-7 sec in generators of PFR type if failure in adjacent tubes is to be avoided, on account of the lag arising from steam drainage. Concentration-dependent methods cannot provide such fast response, and therefore considerable attention is being given to acoustic systems.

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Acoustic Methods of Leak Detection. These methods appear very promising for detecting leaks of more than 1 g/sec. If they prove sufficiently reliable, they may be used in combination with automatic protection systems.

A French study concerned the background noise level in a model for a steam generator of output 50 MW for the Superphoenix reactor, which covered the frequency range from 0.2 to 16 kHz. The noise level increases exponentially with the generator load. The effect exists but is less pronounced in the frequency range 0.3-1.5 kHz, and this was the range used in the study. The model has been used to simulate leak rates of between 0.5 and 6 g/sec. Results have been obtained in which the signal-to-noise ratio fell from 8 to 1 as the load increased from 20 to 80% for a water flow rate of 0.5 g/sec. The results indicate that such a leak could be detected in this frequency range. A study in the Federal Republic of Germany was concerned with the background noise level given by an isothermal model; the best frequency band for detecting leaks was found to be that above 40 kHz. The 200-300 kHz band has been used in Japan. Leaks have been detected in the U.S.A. by cross correlation of signals from accelerometers. The computations were performed by a digital computer. The method allows one to detect a leak with a signal-to-noise ratio of 1/100, and the position can also be identified. A model for the steam generator in the nuclear power station at Clinch River has been fitted with 200 accelerometers operating in the 5-20 kHz band. It is also planned to extend studies to frequencies above 1 MHz. However, British workers stated that a leak of 1 g/sec was not observed even at a distance of 5 cm at a frequency of 1 MHz.

Therefore, existing acoustic measurements have been made under a variety of conditions with a variety of transducers, frequency ranges, processing methods, generator models, and test loops. The conclusion was that it is too early to decide which of the acoustic systems is the best.

Tube Inspection, Sealing, and Repair. Eddy-current methods are used for internal inspection of tubes and to identify leakage points, and ultrasonic methods are also employed, as well as traveling magnetic fields, helium and nitrogen tests, and methods involving the detection of alkalis on the water side of the tube.

Sanito and Kosugi (Japan) reported that inspection transducers have been developed that employ eddy-current ultrasonic techniques. Other measurements have been made by passing bursts of helium under pressure through the tubes.

An instrument has been developed in France for testing for thin spots in the walls of tubes of incalloy-800 of length up to 80 m, which also uses eddy-current methods.

The paper by van Westenbrugge and others (Netherlands) dealt with the detection of defects of size 0.1 mm by ultrasonic methods. Structural changes in the material were detected by traveling-field methods.

In spite of careful checks on tubes and welded joints at all stages of manufacture, it is impossible to rule out the occurrence of leaks. A leak in the body of a tube can arise from friction at the spacers or from corrosion on the water side. It is therefore essential to have access to the tubes or at least to the ends of the tubes. Replaceable tube bundles are planned for use in Britain and the Netherlands. The main method of repairing a steam generator when a small leak is detected is to shut off the relevant tube by means of a plug, which is later followed by welding by explosive techniques or ordinary methods. One of the means of repairing a tube assembly containing cracks is local milling followed by rewelding.

On the last day of the conference, the participants formulated some recommendations that will be published along with the presented papers by the International Atomic Energy Agency.

SIXTH INTERNATIONAL CONFERENCE ON HEAT EXCHANGE

V. S. Osmachkin

The conference was held in August 1978 in Toronto, Canada. Ten discussions were held and 35 reviews and 395 thematic reports were presented. The reviews are of interest to reactor specialists. We will examine some of them.

The report by A. Bergles (U.S.A) analyzed 13 methods of intensifying heat exchange in convection, boiling, and condensation, which are used in various branches of technology. It is noted that these methods, which lead to a marked economy in power and material expenditures, are an important area in theoretical and applied research.

Known methods of heat-exchange intensification can be divided into active and passive methods. The former, which do not require expenditure of external energy, assume different ways of treating the heat-exchange surfaces, the use of different devices for agitation or mixing the flow of the heat carrier, and the use of special additives in the liquids. Active methods include the use of forced vibration of the heating surface, pulsed flow, application of electric fields, and injection of liquids into the coolant flux (or suction from the flux).

The report presents examples of effective use of this or that method in different types of power equipment.

A review of the results of theoretical and experimental research in the area of heat-exchange crisis was presented in the report by G. Hewett (Great Britain). It is noted that, in connection with the actuality of critical problems in heat exchange in different branches of technology and particularly in reactor construction, the study of the mechanism of the phenomenon, the improvement of calculation techniques, and the study of critical phenomena applicable to actual assemblies are being intensely developed. In addition, the region of parameters that is studied is normally limited by a rather narrow range of value achievable in practice.

In spite of the large number of completed investigations, a full understanding has not, at present, been achieved of processes leading to heat-exchange crisis. The author of the report notes several characteristic reasons for the appearance of a critical situation. In the region of underheated water and low steam content, such reasons may lead to an accumulation of steam bubbles in the near-wall region which screen the heat-transfer wall; to a local preheating of the wall near a single steam-generation center; to the occurrence of local overheating of the heat-transfer wall upon appearance of steam blockages. In the region of the dispersive-annular flow regime, crisis in heat-exchange occurs when the dissipation in the liquid layer at the wall becomes zero as a result of excess dissipation of fluid from the layer due to outflow and evaporation on the influx of liquid as a result of turbulent diffusion. The author offered a schematic showing splitting of layers of steam-content dissipation into regions where, in his opinion, the mechanism of crisis occurrence corresponds to the noted reasons.

Heat-exchange research has become topical in connection with calculations of emergency processes in reactors with ruptured piping (report of F. Manger, Federal Republic of Germany). The report examines heat-exchange mechanisms and analyzes known methods of calculating the heat-transfer coefficient in the dispersive flow regime, and planar and transient boiling regimes. Comparison of the results of the calculations with the experimental data demonstrates, as a rule, their significant dissipation.

The report by L. S. Tong (U.S.A.) was devoted to heat-exchange problems arising from an examination of the safety of nuclear reactors. In particular, the report notes complex scientific problems applicable to reactor construction, which require careful study. These problems are related to such phenomena as heat-exchange crisis in nonstationary processes; repeated humidification of hot surfaces; heat exchange with rotation in the direction of motion of the heat carrier; instability of dissipation for discharge; condensation waves; heat transfer in fuel elements heat the humidification front; steam explosions, as well as heat exchange with ruptured fuel elements as a result of high surface temperatures.

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In recent years, the behavior of fuel elements under emergency conditions has been carefully studied: The change in effective admittance of the gas gap due to discharge of gaseous fission products, the cracking and compression of pellets, as well as deformation of coatings are being studied. Much attention is being devoted to working out methods of predicting conditions of fuel-element rupture. Research is being conducted on mounting racks on reactor loops.

The author notes phenomena related to heat-exchange problems in reactor fuel assemblies that must be studied in the future. These are related to mixing of one- and two-phase coolants in fuel assembly cells; the influence of a spacing grid on the thermohydraulic characteristics of the assembly; the boiling of underheated water in rod banks; heat-exchange crisis in nonstationary processes, among which is a sharp drop in pressure. A significant part of the report is reserved for the discussion of heat transfer for film boiling and the mechanism of repeated humidification of hot rods. This is important for the calculations of the degree of flooding of the reactor core after an accident stemming from loss of coolant. The differences in fuel-element-humidification mechanisms are examined for the flooding of the reactor core from below or from above, problems related to the determination of the humidification-front velocity, as well as the interaction of a fused fuel element with the coolant.

Nonstationary flow in two-phase mixtures was discussed in the report by S. Bernezh (Canada). The author examines two-phase models which are used in the solution of problems related to malfunction processes in reactors with ruptured piping. The homogeneous model of mixtures in thermomechanical phase equilibrium yields calculated results which do not agree sufficiently with experimental data. The mixture model, which takes account of thermal phase instability (given equal phase velocities), furnishes better agreement between calculated and experimental results. Examining models with thermomechanical phase instability brings forth purely mathematical obstacles, since the system of equations is insufficiently dependent and the roots of the characteristic polynomial are complex. The author derives differential equations of conservation of mass, motion, and energy for a one-dimensional flow of a two-phase mixture. A significant portion of the report is devoted to numerical methods of solving thermohydraulic equations for two-phase mixtures.

Thematic reports were presented at the conference in the form of placards and diagrams at exhibition "poster sessions," where authors answered questions and commented on their reports. The themes of the sessions touched on heat transfer: volume boiling and mixed convection, boiling two-phase flows, natural convection, condensation, etc.

One session was devoted to heat exchange in reactors. Thirty reports (of which six were from the USSR) were presented. The foreign reports touched on the study of coolant mixing between cells of the fuel-element bundles (J. Kelly and N. Todreas, U.S.A.); the experimental and theoretical investigation of water discharge when reactor piping has ruptured (S. Nesmi and V. Hennocks, Canada); programs for calculating nonstationary processes in channels (R. Lishkovsky et al., U.S.A.); investigation of heat emission in film boiling in repeated-humidification processes (D. Grounweld and G. Yan, Canada); heat exchange processes in pressure suppression systems in nuclear stations (M. Cumo et al., Italy); heat-exchange processes for UO_2 fragmentation under emergency conditions, and others.

English researchers (P. Valei et al.) presented a report on a method of predicting origination conditions of heat-exchange crisis, based on dispersive-annular flow of mixtures. The session of two-phase flows examined a Japanese report "Instability of two-phase flow in parallel channels" (K. Fukuda and T. Kobori), in which experiments on the stability of circulation in the channels of a boiling reactor are discussed; the experiments were carried out on a powerful heat-physics stand at the Oarai research center.

As was mentioned in the program, the organization of the exhibit sessions was adopted as an experiment. The form of presentation expedited the useful exchange of opinions between authors and interested participants of the conference and must be considered a success.

The conference was substantial in its scope. Its work shows it to have been a valuable source of information on the latest achievements of science in the field of heat transfer.

SEVENTH INTERNATIONAL IAEA CONFERENCE ON PLASMA PHYSICS AND CONTROLLED THERMONUCLEAR SYNTHESIS

V. S. Mukhovatov

The conference, held in Innsbruck, Austria in August 1978, was attended by 520 delegates representing 34 countries and two international organizations. The presented reports were divided into the following categories: magnetic confinement of plasma (110 reports), inertial confinement systems (20 reports), reactor design (11 reports).

Magnetic Confinement of Plasma. Among magnetic confinement systems, first place is occupied, as before, by tokamaks. About half of the reports presented at the conference dealt with their experimental and theoretical study.

Tokamak research is developing very successfully. In the two years since the previous conference, the plasma parameters at active installations (see Table 1) have improved significantly and three new and powerful installations have begun work: the FT in Italy, the TFR-60 in France, and the DOUBLET in the U.S.A. The announcement by Princeton University scientists regarding experiments using beams of fast neutrals to heat plasma in the PLT tokamak elicited great interest. In these experiments (injection power of 2.1 MW and density $\sim 2 \cdot 10^{13}$ cm^{-3}) the ion temperature was increased more than five times and reached a value of ~ 5.5 keV, which exceeds the "ignition" temperature of deuterium–tritium mixtures (4 keV) for an unlimitedly long plasma confinement time. In this experiment the energy lifetime of the plasma was small ($\tau_E \sim 25$ msec) and the Lawson parameter $n\tau_E \simeq 10^{12}$ cm^{-3}·sec was two orders of magnitude smaller than is required for the reactor. It is important, however, that according to collision rates, particles of plasma prove to be in the so-called "collisionless" region, characteristic for a reactor, where, in agreement with theory, there may occur new types of instabilities (the most dangerous of which is blocked-ion instability). In most of the experiments plasma destabilization was not observed, and the thermal conductivity of the ions did not differ greatly from neoclassical theoretical predictions. In an analogous experiment conducted on a smaller scale on the T-11 device, the collisionless regime among ions and electrons was also achieved and no deterioration of plasma confinement was observed. The required value for two-component reactors $n\tau_E \simeq 3 \cdot 10^{13}$ cm^{-3}·sec, was achieved on the ALCATOR device (U.S.A.), but at a plasma temperature lower (~ 0.9 keV) than was required. The next step – the achievement of $n\tau_E \gtrsim 3 \cdot 10^{13}$ cm^{-3}·sec and of a plasma temperature of 7-10 keV in one experiment – should be accomplished on the TFTR, JET, and JT-60 devices, which are presently under construction and are expected to be finished in 1981-1983. It should be noted that these parameter values can be also achieved on the recently completed DOUBLET III installation.

The possibility of obtaining hydrogen plasma with a low level of impurities was demonstrated earlier on the ALCATOR tokamak. Today, thanks to the use of several simple measures (cleaning the walls with a hydrogen discharge, spray-coating of titanium, utilizing diaphragms made of light materials, programming gas feed, etc.) in the majority of experiments the impurity content in a discharge can be significantly lowered and plasma with $Z_{ef} \simeq 1$ can be obtained.

Increasing plasma purity has allowed significant broadening of the range of parameters in which a discharge in a tokamak remains stable. Today, the value of the plasma density is 3-4 times that which 2 years ago was considered critical (ISX-A, T-11). It has become possible to reduce the stability margin $q = aH_0/RH_J$ to 1.7-2.5 without disrupting the thermal insulation of the plasma (DIVA, ALCATOR, T-11). Reducing the stability margin, i.e., reducing the longitudinal magnetic field H_0 with no change in the plasma flux, allows an increase in the parameter $\beta = 8\pi p/H_0^2$, upon which the economic indicators of the reactor depend significantly. On the T-11, in an experiment on plasma heating by a beam of fast neutral atoms, a value $\beta(0) \sim 4$-5% was achieved on the pinch axis; in the ohmic heating regime for the ISX-A, $\beta(0) \sim 2.2\%$. Research on the TOSCA device (Great Britain) showed that, in a tokamak with an elliptical or triangular magnetic surface, the plasma confinement time is larger than in a normal tokamak. In these experiments $\beta(0) \simeq 3\%$.

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TABLE 1. Plasma Parameters in Tokamaks at the Times of the Conferences at Berchtesgaden (1976) and Innsbruck (1978)

Indicator	1976	1978
Ion temperature $T_i(0)$, keV	1.8-1.9 (ORMAK' TFR)	5.5 (PLT)
Energy lifetime τ_E, msec	40-60 (T-10, PLT)	80 (PLT)
Lawson parameter $n(0)\tau_E$, $cm^{-3} \cdot sec$	$1.5 \cdot 10^{13}$ (ALCATOR)	$3 \cdot 10^{13}$ (ALCATOR)
Relation of the plasma pressure to magnetic field pressure on the pinch axis $\beta(0)$, %	1.2 (ORMAK); 1 (DOUBLET)	4-5 (T-11); 3 (TOSCA)
Effective ion charge Z_{ef}	3-5 (typical value); ~1 (ALCATOR, T-10)	1-2 (typical value)
Stability margin q_{min} (a_L)	3.5-5 (typical value); 2.2 (T-11)	2-2.5 (typical value); 1.9 (ALCATOR), 1.7 (DIVA)
Collision factor ν_i*	0.4 (TFR)	0.04 (PLT); 0.2 (T-11)

Thus, the parameter values obtained in present-day tokamaks are close to the values that are used in optimistic calculations for installations of the next generation.

Nearly 70% of the papers on the magnetic confinement of plasma were devoted to tokamak theories. Major attention was devoted to the magnetohydrodynamic theory of plasma stability with high β. Optimizing calculations have shown that MHD-stable states are possible in a tokamak with $\beta \simeq 10\%$. Refinement of the theory of microinstabilities has shown that plasma in a tokamak is, apparently, more stable than was considered previously. In particular, the collisionless drift instability, which seemed to be unavoidable and therefore was called "universal," is very effectively stabilized by broadening (in plane geometry) and many microinstabilities disclose a tendency toward stabilization under increased plasma pressure. A significant step has been taken in clarifying the nature of anomalous electron heat transfer in a tokamak.

Six experimental and 4 theoretical reports were devoted to research in stellarators. The plasma behavior mechanism in stellarators in the ohmic heating regime is basically the same as that for tokamaks. The ion thermal conductivity is close to neoclassical; the electron thermal conductivity, to anomalous. As in tokamaks, the energy lifetime increases upon increase in the plasma density. Scientists from Garkhing (FRG) and Kalem (Canada) maintain that the determining factor here is the relation between the directed (current) electron velocity and their thermal velocity. Such an interpretation does not contradict the results obtained on tokamaks; however, it can not be established unambiguously on tokamaks with ohmic heating because of the relation between the magnitude of the current and the plasma temperature. Neither is there certainty that, in stellarators with ohmic heating, such an interpretation is unique, but if it proves to be true, important consequences will follow. In the case of stellarators, the transition to currentless heating methods must lead to the improvement in plasma confinement. In large tokamaks presently under construction and intended for use in powerful supplemental heating and particularly in reactors, the relation of the drift velocity to the thermal velocity will be very small and therefore, it may seem that the anomalies presently being studied will play no role in future experiments.

After the resounding success of the 2XIIB announced at the previous conference, there appears to be a lull in the study of open traps. The search for ways of improving such traps and the progress in the technology of obtaining powerful electron and ion beams have led, in the United States, to the renewal of interest in the old idea of Kristofelos on the creation of traps with a rotating field on the basis of electron and ion rings. At Cornell University, the rotation of a field around the axis of an electron ring in ~1 msec was observed. Attempts to obtain rotation of a field in experiments on the 2XIIB at the Lawrence Livermore Laboratory have not yet met with success. Experiments are being conducted at Cornell University and at the Naval Research Laboratory on the rotation of a field with the aid of ion rings.

The construction of ambipolar traps, proposed 2 years ago by scientists from Novosibirsk and Livermore, has not yet been completed (with the exception of the Gamma-6 installation in Japan), yet the Novosibirsk researchers have found theoretically and have confirmed in a model experiment that the asymmetry of the magnetic field in such traps leads to an increase in transverse plasma losses as a result of neoclassical-type effects.

Inertial Confinement. Almost all major scientific centers where work is being done in laser and beam fusion presented reviews at the conference. These reports described the present state of research and programs for future investigation. In the United States in the past year, two large laser devices were completed, having a light-beam energy of 10 kJ: the 20-beam "Shiva" at Livermore with a neodymium–glass laser, and

the eight-beam "Helios" at Los Alamos with a CO_2 laser. In the first experiments on the "Shiva," which were on the compression of deuterium–tritium targets, an output of $\sim 7 \cdot 10^9$ neutrons/impulse was attained, which is a record for laser plasma. The research programs at these laboratories include the conduct of a thermonuclear demonstration experiment in the mid-1980s.

Reactors. Various aspects of tokamak reactors were examined in six reports, among which was a report on the design of a hybrid reactor being developed in the United States by Westinghouse in cooperation with several scientific centers. The design of a hybrid reactor on the basis of an ambipolar magnetic trap was examined in a report from the Livermore Laboratory. Part of the reports were devoted to alternate conceptions of thermonuclear reactors: reactors with a toroidal pinch and a reverse field, linear theta-pinch, linear compression, as well as toroidal systems with a sharp field geometry and with a corrugated magnetic field.

The conference showed that fusion research is developing with increasing speed. Devices are now being built which will, in the early 1980s, demonstrate the physical realization of fusion power. The results announced at the conference instill a feeling of certainty that this goal will be achieved.

SYMPOSIUM ON HIGH-CURRENT PULSE ELECTRONICS

G. A. Mesyats and E. B. Yankelevich

The third All-Union symposium organized by the Council on Plasma Physics of the Academy of Sciences of the USSR, the Council on Physical Electronics of the Academy of Sciences of the USSR, and the Institute of High-Current Electronics of the Siberian Branch, Academy of Sciences of the USSR, was held in Tomsk in June, 1978. Unlike two previous symposia which were devoted to emission electronics, this symposium dealt with the physics of the emission of charged particles from steady-state and transient plasma formations, the generation and transport of high-current electron beams, and the physics of the interaction of high-current electron beams with condensed media and gases. These subjects are related to the solution of such technical problems as the construction of generators of powerful electron beams for various purposes, generation of microwave oscillations and x rays, construction of powerful gas lasers, and the treatment of materials with high-current electron beams.

The papers presented at the symposium demonstrated the extension of the understanding of the physical processes occurring during field and explosive emission of electrons as well as in steady-state plasma sources of electrons. There has been a considerable expansion of the application of explosive-emission devices in areas in which thermoemission has been traditionally used (to obtain electron beams with a large cross section, to generate a tubular electron beam), work has begun on problems of plasma electron optics, development of inductive energy-storage devices is continuing and their use as accelerating-voltage sources for generators of powerful electron beams is being extended, etc.

Let us note the most important results.

Relations have been obtained between the limiting field-emission current density which does not destroy the cathode and the physical and geometrical characteristics of the cathode. The study of field emission at the limiting current density showed that when a point cathode is cooled to the temperature of liquid helium the limiting current density can be increased by a factor of two to three. In explosive emission in emission centers, whose lifetime does not exceed tens of nanoseconds, a transient Joule process takes place and the current density in the region of the metal-plasma interface is close to 10^9 A/cm^2. A study of the process by which the microrelief of the surface of the exploding cathode is formed showed that the morphology of that surface is changed through the extrusion of liquid metal from the region surrounding an emission center and subsequent cooling of that metal. Extrusion of liquid metal, resulting in the formation of microcraters, micropoints, and microdrops, occurs under the effect of pressure which is built up in the emission zone and which can reach 10^2–10^3 MPa. Such pressure causes the liquid metal to move with a velocity $\geq 10^4$ cm/sec. In the process, part of the metal leaves the cathode in the form of microdrops which in the case of a copper cathode carry approximately 25% of the mass lost per pulse. The study of the erosion of exploding cathodes under widely varying experimental conditions made it possible to establish materials (Pb, C, and Cu) which are preferable for the fabrication of durable cathodes and to propose a method for choosing optimal emitter parameters.

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Research has been carried out on the behavior of emission centers at a high current ramp rate in the diode ($di/dt = 10^8$–10^9 A/sec) in crossed electrical and magnetic fields. As shown by investigations, they form at a rate of up to 10^6 cm/sec. The formation of new emission centers is affected by dielectric and semiconducting inclusions and the film on the surface of a cathode operating in an exploding mode. Studies on the time characteristics of explosive emission in liquid-metal electrodes showed that there is self-reproduction of emitting centers. Explosive emission was detected in semiconductors.

Much attention was paid to research on the plasma formed in a diode and the effect of its parameters on the regularity of current tapping. It was established that current spikes in the diode appear when a plasma bunch forms on the cathode flare front, this bunch being charged to a considerably higher positive potential than the main part of the flare was. Note was taken of the possibility of modulating the electron beam in explosive-emission diodes with a pulse duration $\sim 10^{-4}$ sec.

Many papers discussed enhancement of the operating stability and efficiency of plasma emitters based on different types of gas discharge. The plasma emitters of electron and ion guns have already been built and continuously operating welding guns with a power of up to 20 kW are in service. Interesting work is being done on beam formation in a vacuum and in the presence of plasma in the accelerating gap. Electron beams with a cross section of up to 0.5 m^2, pulse duration 10^{-4} sec, and current up to 10-20 A have been formed in a vacuum.

Many papers were devoted to the formation of coaxial high-current beams in diodes with magnetic insulation and to the transport of such beams. The critical current in the transportation channel has been studied analytically with allowance for the accelerating region and it was shown that the beam current is always below the critical value. These calculations have been confirmed experimentally. Much attention was paid to investigations on the possibility of elongating the pulses and eliminating the reverse current in diodes with magnetic insulation. The investigation establishes that the pulse duration may be limited by longitudinal as well as transverse discharges. When the longitudinal discharge is taken into account, the pulse can be extended to 10 μsec.

Research is being continued on the transport of energy by high-current relativistic electron beams (in lines with magnetic insulation as well as in the absence of external focusing fields). The efficiency of transport of high-current electron beams through a neutral gas without an external magnetic field is low. It was found that electrons experience self-acceleration during gaseous focusing of a beam; this can be used to additionally increase the kinetic energy of part of the electrons in the beam during transport. The effect of dielectric walls in a vacuum channel on the efficiency of transport of high-current electron beams has been studied experimentally. The high efficiency of transport of beams in such channels is due to the formation of boundary-layer plasma from which the beam "extracts" ions necessary for power compensation. The composition of the plasma is determined not by the dielectric itself but by the layer of substances adsorbed on its surface. In this case ions move in the axial direction and their velocity determines the velocity with which the electron beam is propagated.

In systems with a floating anode ions in straight electron beams experience collective acceleration. Ions with an energy more than five times the potential difference applied to the diode were detected.

Advances have been made in improving the parameters of high-current electron accelerators. Accelerators for various purposes with a current exceeding 100 kA and an electron energy of up to 2 MeV are now under construction or being designed; these are the Voda-1-10, Akvagen, Tonus, Neva, the Angara-5 module, and the SÉP-1 and Sinus systems whose construction incorporates different circuit and design features. There has been a further development of accelerators with inductive energy storage; they are characterized by high specific indices and make it possible to generate electron beams with ultrahigh parameters with facilities of reasonable dimensions.

In recent years great attention has been paid to improvements in individual elements of generators of high-current electron beams and, in particular, commutators of various types (water, gas-filled, and flow commutators). Note should be taken of the achievements in this area. Thus, a firing stability of 10^{-8} sec relative to the triggering signal was obtained with multichannel spark gaps of the thyratron type.

Successes in the development of high-current accelerator technology predetermined the widespread use of high-current electron accelerators for constructing powerful gas lasers and devices for obtaining microwave radiation, for studying processes in solids, for hardening organic coatings, etc.

The study of the limiting regimes of field emission of electrons and investigations on explosive electron emission has made it possible to construct a new class of x-ray apparatuses, and x-ray sources which generate

intensive radiation, are reliable, have small dimensions, and serve diverse purposes. These apparatuses are being effectively introduced into the national economy.

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