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CONTENTS

Declassified and Approved For Release 2013/02/15 : CIA-RDP10-02196R000800020004-1

Declassified and Approved For Release 2013/02/15 : CIA-RDP10-02196R000800020004-1

CONTENTS

(continued)

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Declassified and Approved For Release 2013/02/15 : CIA-RDP10-02196R000800020004-1

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CURRENT PROBLEMS OF RADIATION ECOLOGY AND HYGIENE IN NUCLEAR POWER

E. I. Vorob'ev, L. A. Il'in,
V. A. Knizhnikov, and R. M. Aleksakhin

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The growing worldwide attention to problems of ecology and hygiene has been due to the fact that at the present state of the art in the development of production forces man has become one of the most powerful factors with a global effect on the biosphere. According to some forecasts, maintenance of existing rates and tendencies in the development of economic activity can, by as early as the year 2000, create conditions unfavorable to human life on the entire planet [1, 2]. The principal source of this effect on the biosphere are the enterprises of the fuel energy cycle.

The imperceptibility of the effect of ionizing radiation clearly induces man to treat radiation contamination of the environment by nuclear power undertakings with much more caution than he does the usual ejections of combustion products from fossil fuels into the atmosphere. Meanwhile, the latter contain chemical substances, capable of causing cancer and genetic damage, as well as natural radionuclides [3]. There are objective grounds for concern in relation to possible harmful consequences from the development of nuclear power. The principal ones are due to the absence of information which would permit a quantitative comparison of the detriment to health and the environment from one form of power or another. Radiation ecology and hygiene should help obtain the appropriate information.

Papers published to date have assessed the risk from atomic power plants and from thermal power plants operating on organic fuel (coal, oil, gas) to human health [4-6]. In this case, however, the discharges from the atomic plants are taken to include all the main dose-producing components whereas the discharges from the thermal plants are taken to include only some macrocomponents and natural radionuclides [7, 8]. The radiation risk was calculated according to the maximum possible effects whereas the risk from many known dangerous agents, including carcinogens, contained in the discharges from thermal plants were not taken into account at all. At the same time, it is extremely important to assess alternative energy sources from the point of view of the carcinogenic hazard [9].

Assessment of the possible harm from various forms of power to the environment and human health should not be confined to comparison of power plants. In assessing the detriment to the health one must take account of the following stages in the complete cycle and the corresponding effect [8]:

prospecting for fuel – industrial accidents;

fuel extraction – industrial accidents and chronic ailments;

fuel treatment – industrial accidents and some risk to public health;

transportation of fuel – accidents with some risk to public health;

power generation – risk to public health and industrial accidents;

waste disposal – risk to public health and industrial accidents.

If all of the elements of this scheme are taken into account, then in accordance with the concept of "minimum-risk energy alternative" it is possible to compare the harmful consequences from the development of various forms of power and to determine those which are least hazardous to human life and health. It is a similar matter with assessment of the nuclear and alternative forms of power from the point of view of their impact on the environment.

An important area of research by radiation ecologists at the present time is unquestionably that of studying the migration of radionuclides under natural conditions from tailings and from fertilizers which can be obtained as a by-product during the production of nuclear fuel.

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This research is a joint task for radiation ecologists and hygienists since the final goal should be to develop standards designed to limit the entry of radionuclides into the human organism.

One of the main problems and fundamental questions of radioecology is that of scientifically substantiating allowable radiation burdens on nature. Research along these lines and elaboration of appropriate practical recommendations are hindered by the necessity of taking account of all aspects of the value of nature and the harm ensuing from damage to it. It is extremely important above all to ascertain the role and relation of the limitations imposed on discharges by hygiene standards and ecological requirements.

From the point of view of the use of the accumulated knowledge in the practice of designing and operating nuclear power enterprises, radiation ecology lags behind such a closely related discipline as radiation hygiene. As is known, radiation hygiene has developed a system of standards and regulations for the design, construction, and operation of nuclear power installations to prevent harmful effects on the health of the personnel and the public. All hygiene standards regulating the contamination of the environment in the final account have one goal, i.e., of protecting the health of man. The applied aspects of radiation ecology have different objectives, i.e., preventing damage to the environment.

It is logical to assume that whereas radiation-hygiene research has made it possible to develop legislative documents aimed at protecting human health, radioecological investigations should serve as a basis for appropriate legislative recommendations concerning environmental protection. As is known, hitherto there have not been any such recommendations of a legislative character. Hygiene standards do not take account of the "interests" of the environment as such. This situation should be attributed only to the "youthfulness" of radioecology. IAEA Publication No. 26 [10], which reflects present-day concepts of radiation protection, states that " ... radiation safety measures necessary for the protection of the human population will likely be sufficient to simultaneously protect all species of living organisms, although not necessarily all individuals of these species. The IAEA, therefore, assumes that if man is reliably protected against radiation, other species of living organisms will also be adequately protected." Thus, the cornerstone principle of setting standards for the radiation factor is asserted unambiguously, i.e., ensuring standards of radiation safety for man at the same time guarantees the radiation protection of living organisms in the environment and the biosphere as a whole. How valid and universal is this principle?

The rational basis for the radiation-hygiene criteria for standards for the effect of ionizing changes consists of two premises, i.e., biological (man is the most radiation-sensitive living organism on earth) and social (the protection of the health of man is a problem of paramount importance). Since man is classified among the most radiation-sensitive components of the biosphere (at least, his radio-sensitivity is no less than that of other mammals), the introduction of standards concerning human irradiation indeed guarantees radiation protection for all other living organisms if the effect of identical doses is considered. Thus the minimum absolute lethal dose (LD 100) is 450 rem. This indicator is also at the same level for other mammals which in respect of radiosensitivity stand above all other representatives of flora and fauna. The difference between the radiosensitivity of man and many other living organisms is not so very significant. For example, LD 100 for pines and a number of other conifers is 1000-2000 rd and even lower in the more sensitive phases of development. Thus, the radioresistivity of conifers which constitute the basis of forests on our planet is higher than that of man by a factor of only 2 to 4. Many other species of living organisms responsible for the normal activity of various forms of natural communities of flora and fauna and playing an important role in the existence of a number of the principal ecosystems of the biosphere (particularly, forests) have "reserves" of radioresistivity which exceed the LD 100 for man by 5-20 times.

As is known, under ordinary conditions when hygiene standards are established for the population account is taken of the genetic effects as well as late radiation effects, and not acute effects, i.e., values of roughly 1000-10,000 times smaller than LD 100. Observance of such standards automatically also guarantee protection of the ecosystem and the environment if radioactive discharges into the environment results in people and other living organisms being subjected to roughly equal irradiation. However, in actual fact in most cases the absorbed doses for objects in the environment (plants, wild and domesticated animals, etc.) are substantially higher than for man under the same ecological conditions. In this situation it is not clear whether in all cases ensuring the radiation safety of man guarantees the radiation safety of some groups of living organisms. For example, the absorbed dose in some critical organs of cereal plants when a fresh mixture of fission products falls upon them in the critical phase (critical in relation to irradiation) of the ontogeny of the plants can be 10-50 times higher (in extreme situations up to 100-250 times higher) than the dose of external γ rays at a height of 1 m (radiation-hygiene criteria of irradiation) [11, 12]. The difference between the absorbed doses in such objects of the natural environment as wild animals and man may be due to the different composition of their

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rations. In this respect the radioecological situation in the far north is characteristic. Here, under conditions when elk consume lichen, which is an efficient natural concentrator of radionuclides, and man eats the elk, the absorbed dose in bone tissue because of ^{90}Sr from all sources in the case of the elk is 100 times higher than in the case of man.

The difference is many times greater if one turns to those links in the ecosystem which are at low stages of evolutionary development (bacteria, fungi, yeasts, molds, algae, etc.). By virtue of the capability for concentrating radionuclides, these objects may be subjected to irradiation with a dose rate which is thousands, and not tens or hundreds, of times greater than in man in the same region. Such situations are most likely to arise when radionuclides fall into ponds, especially cooling ponds of atomic power plants, since the rise in the water temperature is conductive to more intensive absorption of radionuclides by practically all hydrobionts [13]. Thus, it can be concluded that, on the one hand, man is more sensitive to irradiation than are other living objects but, on the other hand, when radioactive substances enter the environment more living objects are subjected to irradiation with a much higher dose rate than is man.

With observance of hygiene standards living objects in nature will not be protected if the ratio of the radioresistivities of those objects and man (according to the indicators assumed in the hygiene standards) is lower than the ratio of the real dose burdens on these objects and man under the given contamination of the environment: $ED_{nat}/PD_{man} < RD_{nat}/RD_{man}$, where ED_{nat} is the endurable (harmless) dose of radiation of the natural object, PD_{man} is the permissible dose (radiation-hygiene standard) of irradiation of man, and RD_{nat} and RD_{man} are the real doses of radiation of the natural object and man, respectively, in the given situation.

Above we gave data on the comparative radioresistivity of man and natural objects according to LD 100, applicable for extreme situations. The hygiene standards limiting the radiation contamination of the environment proceed from small doses of chronic irradiation, comparable with the doses from natural background. These standards were designed to prevent (limit) such late stochastic effects as an increased incidence of cancer and genetic damage [10]. In accordance with present concepts, underlying these forms of pathology is damage at the molecular level (in the cell genome) which, if the repair and reproduction mechanisms are not actuated, results in the corresponding somatic (cancer) or genetic ailment. Thus, in order to approach the estimation of the comparative radiosensitivity of man and other living objects, not according to acute lethal effects but with account for parameters assumed in the hygiene standards, it is necessary to compare doses which cause in man effects taken into account in the elaboration of the hygiene standards and the level of doses causing undesirable effects in living nature and its ecosystems. Radiation doses capable of disrupting regeneration processes and thus facilitating the development of injury at various levels of biological integration, given in Table 1.

It must be emphasized that the hygiene standards, as follows from the concept of a no-threshold character and linearity, adopted by the IAEA [10] in essence regulate damage at the molecular level proceeding from the premise at permissible doses the protective mechanisms acting at the cellular and organisms levels are not suppressed and the frequency of damage to molecules is such that only in very rare cases does such damage escape elimination or repair and succeed in being realized through the formation of tumors or genetic damage. The objective of the hygiene standard is, to the extent possible, to protect each organism (individual); in protecting the environment, radioecology obviously can satisfy itself with protection at the population and biogeocenotic levels. The radical difference in the approach to protection of man and objects of nature, including farm and wild animals, is that hygiene is called upon, and strives, to protect each person from illness whereas ecology may not be interested in each individual but is concerned with protecting the population, the community. Thus, if for a group of reasons, including the limitedness of the food base, out of 100,000 fish eggs spawned in a body of water only 2000 individuals can develop and exist, then clearly there is no economic damage if, because of radiation, part of the fish eggs are destroyed or produce nonviable progeny since in this case 2000 individuals will still live in the body of water.

When the above is taken into account, it becomes understandable why under real conditions disruptions in the sizes of populations and biogeocenotic changes are recorded only at irradiation doses running to hundreds and thousands of rads. If these data and those in Table 1 are taken into consideration and compared with the hygiene standards regulating the irradiation of limited population groups (0.5 rem/yr according to radiation safety standard NRB-76) [14], then it can be concluded that living objects can suffer harm according to the indicators listed in Table 1 only at doses which are two to three orders of magnitude higher than the hygiene standard. It would be inadmissible to rule out the possibility of such situations with overirradiation of any objects of the environment. It may be that in some situations with observance of the technical standards for

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TABLE 1. Minimum Doses Acting on Mechanism of Repair of Radiation Damage at Various Levels of Biological Integration

Level	Regeneration mechanism and path	Eff. min. dose, rd
Molecular	Repair of biologically important molecules (e.g.,) repair of DNA by depletion-replacement; photoreactivation, postreplicative repair)	Units
Cellular	Amplification of processes of cell after the death of the radiosensitive part of the cell population	> 10
Organism	Repair of activity of individual systems of organism, leading to the complete regeneration of the organism, slowing down of some processes and amplification of others, ensuring complete regeneration	> 20
Population	Radioadaptation, radiation mutagenesis, elimination of relatively radiosensitive individuals from the population, selection by radiosensitivity	> 50
Biogeocenotic	Homeostatic regulation of biocenotic processes under radiation, regeneration of individual components of biogenosis (regeneration of woody stage vegetatively when reproductive organs are damaged, regeneration of herbaceous plants from dormant buds, etc.), the assembly of regenerative reactions in components of biogenoceses after uneven irradiation.	> 200

radioactive emission and discharges into the environment radiation protection of living objects cannot be guaranteed.

It must be deemed desirable to develop and introduce radioecological criteria for standardizing the radiation factor to supplement generally accepted radiation-hygiene standards. It may be that in certain situations the radioecological standards will be more stringent than are the radiation-hygiene standards, e.g., in situations when emissions and discharges of radioactive substances may reach the territories of national parks or game refuges. It is possible that nuclides of biogenic elements (e.g., ^3H, ^{14}C) will accumulate far from the point of discharge in quantities capable of leading in the final account to such changes in those systems that major economic or other consequences might ensue (e.g., a change in the food chains in the ocean, acceleration of mutation processes in pathogenic viruses and bacteria, etc.).

Unfortunately, present-day radioecology is still far removed from being able to present an orderly system of quantitative criteria of permissible dose burdens as has already been done in radiation hygeine for man. This is due primarily to the complexity and multiplicity of many years of ecological observations of the effect of small doses of radiation on natural ecosystems. As the first standardizing assumption for radioecological standardization of the radiation effect one could propose to use the concept of the "radioecological capacity of the environment." This term should be taken to mean the maximum permissible content of a radionuclide in a critical component of an ecosystem, a content such that the ecological harmony of the functioning of that ecosystem is not disrupted (for national parks and game refuges) or such that changes which are undesirable from the economic or other points of view do not occur in the ecosystem. In order for this assumption to be realized it is necessary to obtain concrete information providing a quantitative characterization of the radioecological capacity of the main types of natural terrains, bodies of water, etc., which is the only thing that can put the problem of ecological standardization on a scientific footing. An important aspect of the problem under consideration is that of clearly defining the objective and necessary degree of protection of some living objects or other in a given concrete situation since it is perfectly clear that the "injury—benefit" principle which is proposed for use in elaborating radiation safety standards for man can be used with all the more justification in elaborating radioecological regulations. The elaboration of such regulations and quantitative estimates of the damage to nature because of nuclear power can be of great assistance in mapping out the course of development of the power industry in general, and nuclear power in particular.

Some challenging problems confronting present-day hygiene have a direct bearing on the fate of energy programs. These problems are due to the fact that in such an important area as the effects of low doses, at

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Declassified and Approved For Release 2013/02/15 : CIA-RDP10-02196R000800020004-1

the level of the natural background, there is no unity of views as to problems of cardinal importance. Practically no information at all is available for making quantitative estimates of the danger caused by chemical carcinogenic components of emissions from thermal power plants [3]. Problems existing in this area must be presented clearly since one solution or other for these problems may have a significant influence on our assessment on nuclear and alternative forms of energy as well as on the entire system of standards regulating emissions.

Let us recall that present approaches to the regulation of dose burdens on man, as recommended by the IAEA and shared by other international organizations, proceed from the assumption that for stochastic effects (initiation of túmors and genetic damage) there is a linear thresholdless relation between the dose and the probability of the effect occurring [10, 15]. This approach makes it possible to assess the risk of stochastic effects under arbitrarily low doses and to establish standards for irradiation from the concept of acceptable risk. This approach served as the basis for nuclear powers to regulate irradiation of the population because of emissions from atomic power plants at an extremely low level, constituting only a small fraction of the fluctuations of the natural radiation background.

As is known, direct data about the capability of dose burdens at the level of the natural background to cause cancer and genetic damage are not available. The carcinogenic effects of irradiation were revealed in experiments or as the result of epidemiological surveys only at irradiation doses exceeding the annual doses from the natural background by a factor of hundreds and thousands. In numerous experiments and observations on animals and humans subjected to irradiation at a level of several tens of rem or less, no carcinogenic effect of radiation was detected [15]. Some specialists, basing themselves on the factual data from such investigations, assume that there exists a threshold of the carcinogenic effect of radiation. It cannot be ruled out, however, that a dose which proves to be ineffective when acting on a limited number of individuals, under the conditions of action on a large population it will display its carcinogenic effectiveness and the "threshold" will turn out to be imaginary [16]. Simple calculations show that usually applicable methods of experiments on animals do not permit the problem of the threshold and effectiveness of low doses to be solved. Thus, with an irradiation dose of 0.1 rd (the mean annual dose of the natural radiation background) it may be expected that if the concept of no-threshold character and linearity is correct, various forms of tumors will appear with an incidence on the order of 10^{-5}. Thus, in order to reveal the carcinogenic effectiveness of such a dose a group of animals numbering 100,000 is required. In actual fact, in view of the occurrence of spontaneous tumors in animals, for the detection of the effects of such low doses to be statistically reliable in comparison with a control there must be uniform test and control groups of animals with a much larger size; this makes it unfeasible to set up such studies and to solve the problem of whether or not a threshold exists on the basis of those studies.

However, there are also other ways of investigating the problem of the threshold and effectiveness of low doses. Thus, an important argument in favor of the existence of a "practical threshold" is the reliable establishment that the latent period of formation of some tumors increases as the irradiation dose decreases. This permitted the assumption to be made that at low doses the latent period may exceed the maximum lifespan [17]. This argument, however, can scarcely be taken into account when resolving the question of the threshold since the latent period of tumor formation is subject to considerable individual variations and with a reduction in the irradiation dose only the mean latent period increases. If in a small sample of animals at a given dose of irradiation tumors did not manage to develop, this does not mean that in a larger sample there will not be individuals which "manage" to give tumor growth. Thus, the given approach cannot serve as proof of the existence of a threshold.

The opinion concerning the absence of risk at irradiation doses within the limits of the fluctuations in the natural background is also argued with concepts based on consideration of processes of cancer initiation [18]. It turns out that at the level of the cell and in the whole organism there are mechanisms directed at repairing damage inflicted by radiation as well as at eliminating cells which have degenerated into cancerous cells. It is noted that as the result of this, only at doses of 100 rd or higher, which are capable of affecting the protective mechanisms, do the probability of multiplication of cancerous cells and their transformation into a tumor begin to exceed the incidence of spontaneous cancer [18]. Unfortunately, such an approach, from which it follows that doses of less than 100 rd do not constitute a carcinogenic hazard, cannot be accepted as satisfactory since it does not explain, and does not take suitable account of, the stochastic nature of tumor diseases and other (genetic) damage whose source lies in mutation processes. It is known that at a dose of 100 rd, affecting the protective properties of the organism, and at much higher doses not all irradiated individuals contract cancer. Conversely, it is perfectly clear that spontaneous cancer, which now causes the death of a large proportion of humans and animals, does not necessarily develop only with the protective mech-

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anisms suppressed by radiation. If the point is, however, that the protective mechanism are suppressed in all cancer patients, it is natural to assume that they are suppressed not by irradiation (only for an insignificant part of them did the irradiation dose exceed 100 rd) but by some other factors inherent in real living conditions. If this is the case, then the carcinogenic effect of low irradiation doses cannot be ruled out against such a real background since the capability of even the lowest doses to cause mutations does not arouse any doubts in anyone.

From the point of view of practical questions, numerous attempts to solve the problem of the threshold are stochastic in character and hardly have a direct bearing on the problem of setting standards for radiation effects. The point is that at the present time the mean individual radiation dose of the population of the USSR mainly from medical diagnostic procedures and the technogenic changed radiation background exceeds 250 mred/yr [19]. Inhabitants of large cities in industrially developed countries receive an annual whole-body dose of 0.3-0.5 rem and a dose of 1.0-1.2 rem to the lungs and 0.5-1 rem to the thyroid gland. It is not difficult to calculate that in 30-50 years of life these indicators may constitute doses which are very close to those which no one doubts are capable of increasing the incidence of cancer [15]. Since any effect subject to regulations is supplementary to that indicated, we assume that the IAEA position is substantiated; according to this position it is legitimate to calculate the risk from any arbitrarily small additional effect.

Thus, in summarizing the problem of estimating the effectiveness of low doses, it should be pointed out that adoption of the existing concept of the linearity and nonthreshold character of the stochastic effects and the possibility of calculating the risk as a result of low doses on their basis is quite justified. There is a need of further research and new methodological approaches to refine the dose-effect relation in the range of doses comparable with the dose rate of the natural radiation background.

Resolution of the question of the relative carcinogenic and genetic hazard of nuclear and alternative forms of energy depends to a certain degree on the carcinogenic effectiveness of the chemical agents present in emissions from thermal power plants operating on fossil fuel. At the present time, the mortality due to cancer in industrially developed countries is 1500 cases per year per million inhabitants. Using the concept of the no-threshold character and nonlinearity, we can estimate that because of the action of all sources of ionizing radiation, natural and artificial, out of one million p rsons 20-30 may die, i.e., no more than 2% of the total number of deaths due to cancer. In the opinion of many specialists, about 80% of the total number of deaths due to cancer are caused by chemical carcinogens, including those present in the emissions from power plants [3]. At the present time, it has been established experimentally (to some degree, this has also been observed for man) that some microcomponents of these emissions possess distinct carcinogenic properties for animals. These include 3,4-benzpyrene and other polycyclic aromatic hydrocarbons which are products of the incomplete combustion of fuel as well as a number of metals and their oxides, including nickel, chromium, nickel, iron, arsenic, and beryllium.

Moreover, it has been found that a carcinogenic and cocarcinogenic effect is exerted by sulfur dioxide which is emitted into the atmosphere in hugh quantities by coal-fired power plants as well as by the products of oil refining.

There is no direct proof that chemical carcinogens are responsible for a considerable part of the existing incidence of cancer, notwithstanding the importance of resolving this question. There are practically no regulations on the content of chemical carcinogens in the environment owing to the inadequate investigation of the hygienic aspects of carcinogenesis, especially the existence of a threshold of carcinogenic action by chemical substances. Since in the "nonradiation" areas of hygiene all standards have come to be established by proceeding from a threshold, the problem of setting standards for carcinogens in essence remains an open one [16, 17].

It is our contention that the discussion taking place on the existence or absence of a threshold in chemical carcinogens as in the case of ionizing radiation is more of academic rather than practical interest since in many places, if a threshold does exist for chemical carcinogens as it does for radiation, it has long been reached and surpassed. With the action of numerous chemical and radiation carcinogens characteristic of the present state of the environment, as a rule the action of the carcinogens combines and their effects add up. In other words, a pool of physical and chemical carcinogens which have reached threshold levels exists in the environment. As a result of this, funda ental importance is taken on by the problems of quantitative ascertainment of the dose-effect relations which, unfortunately, have hitherto remained practically uninvestigated, even for long-known highly active carcinogens. Clearly, it is an extremely urgent and promising task to develop a chemical equivalent of the rem, which would permit the degree of risk from the action of various carcinogens to be expressed in comparable units [3, 5, 16]. It is also important to study the effects of the linked and combined action of carcinogenic and cocarcinogenics factors. The solution of these problems will help find optimal ways

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of developing the power industry while preserving the natural environment and public health to the maximum extent.

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ACCELERATORS FOR INDUSTRY AND MEDICINE (CONTEMPORARY STATE AND PROSPECTS)*

V. A. Glukhikh

UDC 621.384.64 + 621.384.658

In recent years further significant expansion has occurred in the use of charged-particle accelerators in radiation technology, radiography, activation analysis, and medicine, along with improvement of engineering-economic and operational indicators of industrial accelerators and the assimilation of new directions of their

*This is the journal version of a lecture at the Third All-Union Conference on the Use of Charged-Particle Accelerators in the National Economy held in June of 1979 at Leningrad. More than 300 persons from more than 100 organizations of the Soviet Union and 40 foreign guests from nine countries participated in the conference. There were six working groups: Radiation engineering processes with the use of accelerators, accelerators for the national economy, the use of accelerators in medicine, radiography, activation analysis, and the formation and control of the parameters of the exit beam. About 200 lectures were delivered in three plenary and 15 section meetings. The conference showed that radiation technology utilizing accelerators has emerged onto a new qualitative level. If mainly accelerators designs were presented at the previous two conferences, lectures on testing the operation of radiation engineering accelerator facilities and their specific parameters and engineering-economic indicators predominated at this, the third, conference. This tendency was reflected in the plenary lecture of the president of the organizing committee of the conference, NIIÉFA Director V. A. Glukhikh, which is called to the attention of the readers.

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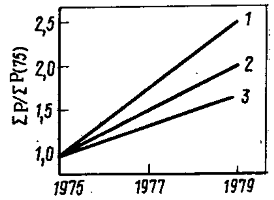

Fig. 1. Variation of the power of the electron accelerators installed in the industrial plants of (1) the Soviet Union, (2) the USA, and (3) Japan.

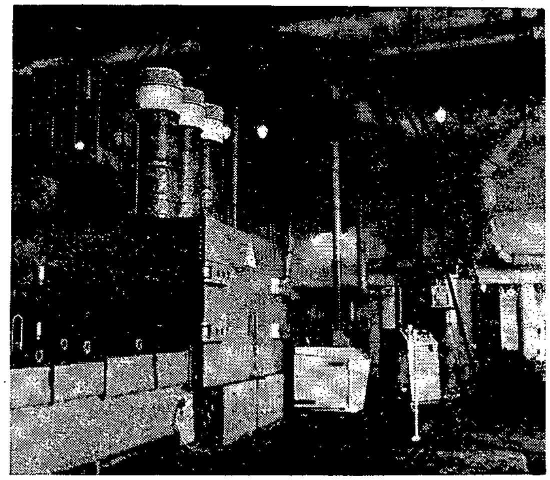

Fig. 2. Aurora II electron accelerator with individual protection.

application. At present the annual extent of the production of radiation facilities operated abroad exceeds 1 billion dollars [1]; more than 90% of it occurs for facilities in which accelerators are used. About 230 accelerators have been installed in industrial plants in the USA, 35 in Japan [2], and more than 40 in the Soviet Union. The growth of their use can be characterized by the change in the cumulative installed capacity [2, 3], which has doubled in the last 3-4 years (Fig. 1).

As previously, the most widespread industrial radiation processes are the treatment by accelerated electrons of various polymer materials, the insulation of wires and cables, and films, rubber vulcanization, hardening of lacquers and paints, seeding polymerization, and sterilization [4]. However, notwithstanding an annual increase by ~20%, the output of products and materials subjected to radiation processing still amounts to just a small fraction of the production volume of the corresponding branches of industry. This fact, together with the small compensation term of radiation engineering indicates the need for its broad introduction. The economy of radiation processing of materials in comparison with processing by traditional methods also acquires continously greater meaning in connection with the increase in the cost of energy in the last few years.

Significant successes have been achieved in the Soviet Union in the development and introduction of facilities with electron accelerators intended for the radiation modification of wires and cables with polyethylene insulation [5], the production of thermally seated products [6], the hardening of lacquer–paint coatings [7, 8], and radiation processing of fabrics and some other kinds of products and materials. New radiation processes have been developed, among which the use of electron accelerators for the protection of the environment is of special interest. One of the promising trends in this area is radiation purification of sewage, which permits giving up the traditional method of chlorination [9]. Preliminary economic evaluations of both methods [10] give comparable results; however, it is important that the undesirable action of a large amount of chlorine on natural conditions in the adjacent locality is excluded in the case of the radiation method. But accelerators whose total beam power amounts to several tens of megawatts are required for the purification of the sewage of a large city in this manner. The use of powerful electron accelerators is also possible for the purification of the gas exhausts of heavy industrial plants and thermal electric power plants of the oxides of sulfur and nitrogen contained in them. Results obtained with a test facility [11] permit concluding that in this case the necessary power of the electron beam for a plant with a capacity of 100 MW is about 4 MW, and the construction cost of such a facility is comparable to the cost of the usual type of purification equipment [10].

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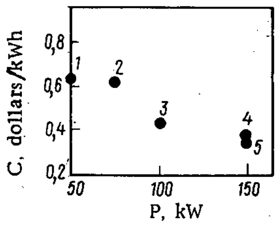

Fig. 3. Dependence of the cost of production processing for accelerators of the RDI firm on power for an energy in MeV of 1) 0.5, 2) 1, 3, 4) 3, and 5) 1.5.

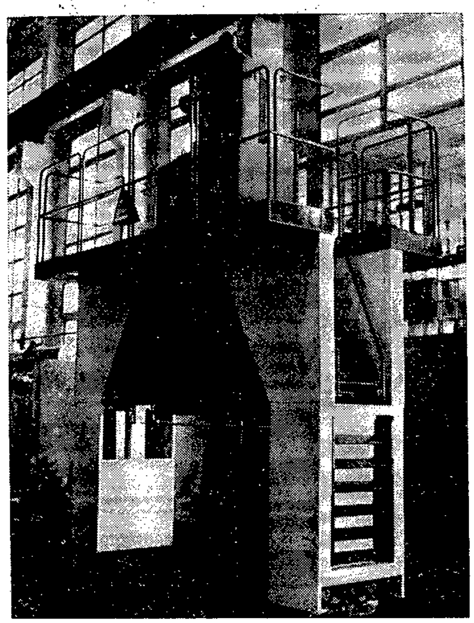

Fig. 4. Aurora III high-voltage electron accelerator.

In the majority of contemporary industrial radiation facilities high-voltage accelerators with an energy of 0.15-3 MeV are used, which permits processing materials and products up to 1.2 cm in thickness. For an energy up to 0.7 MeV accelerators with individual protection [7, 12] and with an extended cathode [13], which can be installed in ordinary production rooms along with the other engineering equipment of radiation facilities, have found widespread application (Fig. 2). The power of the largest of them, which can be discharged in series, has grown to 150 kW, and their reliability has simultaneously increased appreciably. As a result, the cost of processing materials with an electron beam (Fig. 3) has significantly decreased, and the prerequisites have been created for expansion of the incorporation of radiation technology [14]. New types of domestic accelerators (Fig. 4) with a beam power up to 50 kW have been developed [15, 16].

Evidently, one should consider the main trend in the development of the accelerators used in industrial radiation processes most recently to be the further increase of their power and reliability. The creation of facilities with a power of 1 MW and more under steady-state conditions already seems completely realistic [14, 17], although the solution of complex scientific and engineering problems is required for this to happen. However, the advisability of carrying out such developments is determined in the first place by the economic indicators of the corresponding energy-consuming radiation processes [10, 18] in comparison with the traditional methods of production of this or the other product. At the same time the electron beam power of 50-150 kW already achieved in many cases is sufficient due to the limited productivity of the other units of the technological process. Therefore an important direction in the refinement of industrial accelerators is also improvement of the engineering-economic indicators due to optimization of structural and layout solutions, reduction in the consumption of material, increase in reliability, and automation of control.

The introduction of activation analysis directly in industrial plants deserves attention. One of the effective uses in the analysis of the fluorine content in feed phosphates in the chemical industry with the use of an NG-150 neutron generator [19]. Along with neutron-activation analysis, analysis with the use of photonuclear reactions has received development. One can realize the high selectivity, rapidity, and representativeness

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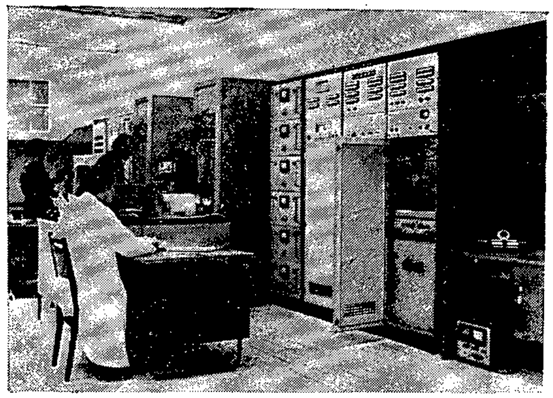

Fig. 5. Control console of a laboratory for rapid analysis at a mining reprocessing complex.

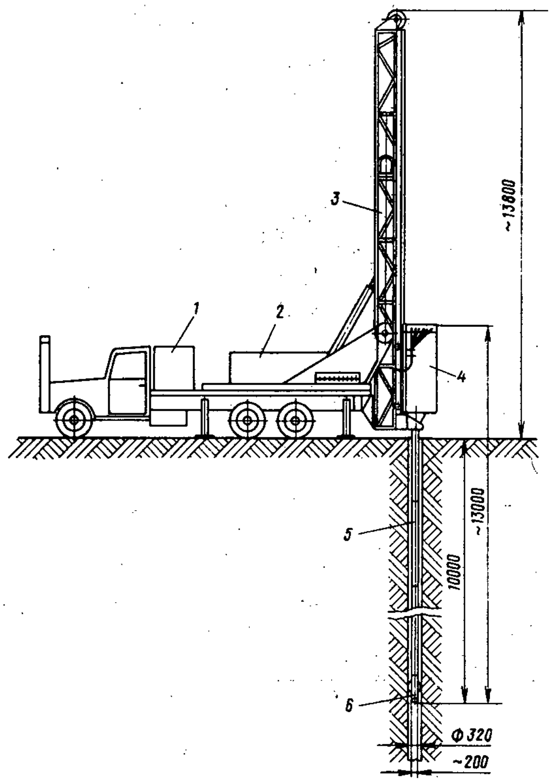

Fig. 6. Model of a transportable linear accelerator for activation analysis of ores in their natural state: 1) thermostatic control system; 2) shf generator modulator; 3) apparatus for shifting the accelerator; 4) electron accelerator; 5) electron conductor; and 6) target.

intrinsic to the method only with a sufficiently intense flux of bremsstrahlung, due to the relatively small interaction cross section of the radiation with matter. Such fluxes are produced by linear waveguide accelerators [20]. A laboratory for rapid analysis of ores for gold and accompanying elements has been created at a mining reprocessing complex with the use of the LUÉ-15A and LUÉ-8-5A developed at the D. V. Efremov NIIÉFA [21] (Fig. 5). The accelerators are steady-state, and the ore samples selected for analysis are delivered to the laboratory. But such laborious operations as sample selection, indexing, and wrapping are maintained. Therefore, the analysis of ores in their natural state deserves attention, since it would thus be possible not only to eliminate the operations listed but also to increase the sensitivity and rapidity of the analysis. A possible version of such an accelerator is shown in Fig. 6. The accelerator is rigidly attached to an electron conductor and a target apparatus; variation of the insertion depth of the target into the hole is achieved by variation of the lift height relative to the surface of the earth. The equipment is arranged in 2-3 automatic machines. The alternative of an accelerator inserted into the hole is possible. However, the creation of a special compact shf generator must precede its development.

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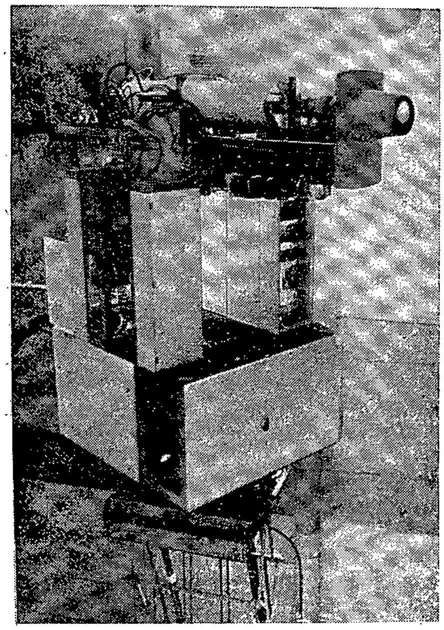

Fig. 7

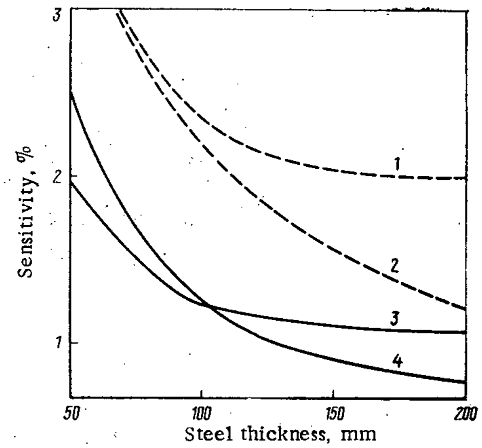

Fig. 8

Fig. 7. LUÉ-15-15000D accelerator–flaw detector in the process of assembly in the x-ray room.

Fig. 8. Sensitivity of the x-ray television method for an electron energy of 5 (1, 3) and 10 MeV (2, 4) without the memory unit (1, 2) and with it (3, 4).

Interest has grown in accelerator–flaw detectors in connection with the development in this country of atomic power machine construction. Just as abroad, flaw detectors based on linear waveguide accelerators have found wide application, which is due to the high increase in energy and the simplicity of beam entry and exit. The LUÉ-5-500D and LUÉ-15-15000D accelerator–flaw detectors have been developed in recent years [22], and the first batch of them has been prepared for the Izhorsk Factory Industrial Union and for Atommash (Fig. 7). The LUE-500D, which permits controlling a product made out of steel with a thickness up to 350 mm [23], will find the widest distribution, since the thickness of the majority of products does not exceed this value. Its parameters meet contemporary requirements. It is proposed to improve the flaw detector as a whole in the future. In particular, it is possible to increase the efficiency of some control operations of products of atomic power machine construction by virtue of the use of panoramic radioscopy. One should note that the possibilities of the developed accelerator–flaw detectors have not been completely realized, the results of radioscopy are still recorded on x-ray film, and the setting up and processing occupy up to half of the total control time. The recording and documentation of control results are being improved in several ways. Encouraging results have been obtained with the use of an x-ray television setup with a memory unit. The sensitivity achieved in the experiments (Fig. 8) satisfies the requirements for the control of the products of atomic power machine construction [24].

Electron accelerators, mainly the linear waveguide type, have received widespread distribution in medicine for radiation therapy. According to the data of [25], up to 600 such accelerators were counted in the world in 1978, and their number is increasing annually. In the Soviet Union their use for radiation therapy has not yet become widespread. About 10 accelerators are in use so far. Investigations of a working model of the LUÉ-15M (Fig. 9), in which it is possible to accelerate electrons to 20 MeV [26], have been completed. Based on this model, a pilot model of an accelerator, the LUÉ-15MÉ [27], with programmed control has been developed for radiation therapy. An integrated radiation head has been installed in it, which permits irradiating both with photons and with accelerated electrons. According to its parameters, it meets the requirements of the International Electrical Engineering Commission. It is possible with the aid of such an accelerator to treat more than 50% of the types of cancers. The cancer clinics of our country and the member-nations of the COMECON will be equipped with accelerators of this design. Development has begun on a medical accelerator for an energy of accelerated electrons up to 40 MeV for cancer research centers.

The use of protons beams in radiation therapy is known. According to the published data, more than 2500 patients worldwide have undergone such treatment. The localities being irradiated are now rather large, and

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Fig. 9. Working model of the LUÉ-15M accelerator.

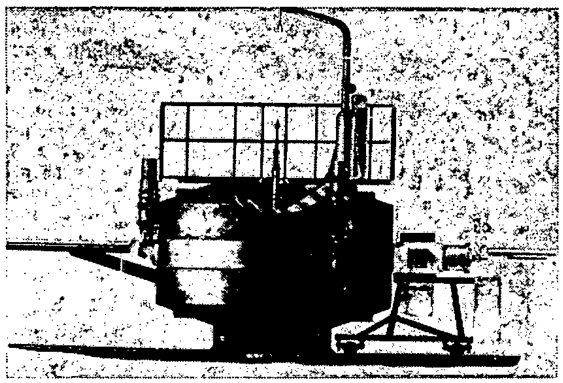

Fig. 10. External view of the cyclotron for production of short-lived isotopes.

they are constantly expanding. In the Soviet Union mediobiological investigations and clinical research with proton beams are carried out at the Joint Institute for Nuclear Research, the Institute of Theoretical and Experimental Physics, and the Leningrad Institute of Nuclear Research, but on accelerators intended for physics experiments and only adapted for medical purposes. NIIEFA, together with the Institute of Theoretical and Experimental Physics, the Cancer Research Center of the Academy of Medical Sciences of the USSR, and other organizations, has begun the development of a multibooth complex using a proton synchrotron with hard focusing for energy up to 220 MeV and a beam intensity up to 10^{12} protons/sec. The complex is intended for clinical use (100 radiation sessions per day), diagnostic purposes (proton radiography), physicoengineering and medicobiological investigations directed towards expansion of the use of a proton beam in medicine and biology, as well as for the production of radionuclides for diagnostic purposes. Eight channels of the external beam will be in the complex with entry into five procedure rooms.

A distinctive feature of medical accelerators consists of the fact that such radiation parameters as the size of the field, the strength of a dose, and the direction must be varied in the course of a radiation session according to a program specified by the physician. This fact leads to the necessity for the development of devices with whose help one can vary at a distance the radiation parameters and the program of the controlling computer. The accelerator should be equipped with a special platform for the patient with the necessary number of degrees of freedom in order to alter the position of the patient in the course of the irradiation. Thus the contemporary medical accelerator is a complicated device saturated with electronics.

One more promising trend in the application of charged-particle accelerators in medicine is the production of short-lived isotopes for the diagnosis of diseases. A special cyclotron has been developed for these purposes at the D. V. Efremov NIIÉFA (Fig. 10) [28, 29]. In order to provide a high productivity, the beam of protons accelerated to an energy of 25 MeV should reach 1.5 mA on the internal and 0.2 mA on the external target.

The information presented at the conference indicates further quantitative and qualitative expansion in the use of charged-particle accelerators in various areas of activity and their continuously increasing influence

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on the acceleration of the scientific-engineering process, the increase of labor productivity, and the improvement of the quality of manufacture output.

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ANALYSIS OF THE RELIABILITY OF PIPES AND PRESSURE VESSELS AT ATOMIC ELECTRIC POWER PLANTS

A. I. Klemin and E. A. Shiverskii

UDC 621.039.5.58:621.38.004.6

One of the most numerous and important components of the equipment of an atomic electric power plant are the pipes and pressure vessels. As domestic and foreign practice in the design and operation of atomic electric power plants (AEPP) shows, ruptures of large pipes, collectors, and vessels pose the greatest potential radiation hazard among all equipment failures. A quantitative analysis of their reliability becomes an obligatory part of evaluating the reliability of a plant as a whole at the design stage. The results of an investigation of the reliability of the pipes and vessels of an AEPP are presented in this article.

First let us formulate the concept of the failure of the elements under discussion. The failure of a pipe or pressure vessel is a loss in its efficiency due to depressurization to a level specified in the technical documentation or the appearance of such irreversible changes (cracks, thinning of a wall with dimensions and nature specified in the technical documentation, etc.), which can then lead to depressurization and disabling of the pipe (vessel).

The concepts of catastrophic and potential hazardous failure are often used for large pipes and vessels. Failures in which the rupture of a pipe (vessel) occurs in a short time (often practically instantaneously) are called catastrophic; the size of the defect for a pipe, e.g., is commensurable with its diameter, and rupture leads to significant damage. A catastrophic failure requires a rapid shutdown of the unit of the atomic electric power plant and the performance of extensive repair work. Potentially hazardous pipe and vessel failures are, as a rule, flaws in the basic metal and weld seams, looseness in mechanical joints, cracks of a definite size, local thinning of the walls (e.g., due to corrosion or erosion), etc. Potentially hazardous failures should be eliminated opportunely, since in the course of operation part of the potentially hazardous failures can become the cause of catastrophic failures.

Factors Which Determine the Reliability of Pipes and Vessels. The main factors which determine the reliability of the equipment items under discussion are given in Table 1, which is constructed on the basis of investigation and generalization of domestic and foreign testing of the maintenance of pipes and vessels at atomic and thermal electric power plants. The complexity of analysis of causes of failures of pipes and vessels lies in the fact that the factors listed often act together. In each case it is possible to speak of the domination of several factors and of an insignificant effect of the rest. Situations in which it is possible to identify a single factor as the cause of a failure are relatively rare.

Testing of the operation of pipes and vessels shows that failures which have occurred during the initial period are with great probability caused by factors of the first and second groups (see Table 1). Failures at the end of the term of service of an element are to a large extent related to factors of the third group; intermediate failures can be caused by factors of all groups. The relative contribution of individual groups of factors to the unreliability of pipes is [1]: structural – 16, engineering – 28, and operational – 56%.

Failures occur most often in weld seams, bends, etc. For example, according to the data of [2], 54% of the pipe failures at AEPP have occurred in weld seams, 40% in the basic metal, and 6% in the threaded joints of pipes.

Leaks precede the absolute majority of serious pipe and vessel failures. A practical conclusion follows from this fact: It is possible with the help of regular inspections and control of potentially hazardous sections of pipes and vessels (weld seams and others) to increase to a certain extent their reliability, and what is especially important, prevent catastrophic failures.

The factors listed in Table 1 which act on pipes and vessels can cause the following basic mechanisms of their rupture (in order of importance): fatigue (low-cycle, high-cycle, thermal), corrosion under a voltage potential, creep corrosion, brittle fracture, viscous fracture, erosion, and others.

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TABLE 1. Classification of the Main Factors Which Determine the Reliability of Pipes and Vessels

Structural	Engineering	Operational
Choice of material Presence of strain condensers (number and nature) Nature of geometrical shape of the hydraulic circuit of the coolant (presence of sharp bends, constrictions, and expansions result in the origin of large-scale eddies in the coolant flow and vibration) Provision of compensation for thermal expansions Provision of flexibility (exclusion of excessive liquid) of a pipe Nature of pipe mounting Adaptability to control and engineering inspection	Quality of material Quality of preparation of pipes, bends, and cast elements Quality of welding Quality of preparation of surfaces Effectiveness of outlet and intake control Quality of transport Quality of assembly	Loads: mechanical, thermal, and so on (steady-state and dynamical values and the nature of the loading) Action of contiguous medium (nature of the interaction: mechanical, chemical, and others) Action of specific factors, including unplanned ones (irradiation, vibration, sediment, and so on) Amount and nature of control and technical service Departure of operating conditions from normal ones Errors in operation and servicing

As a rule, the rupture mechanism is identified rather simply from the nature (external view) of the rupture. Such an analysis of failures with application to specific items and operating conditions is always necessary for reliable exposure of the dominant mechanisms and factors. Without this the development of effective measures for increasing the reliability of pipes and vessels and methods for predicting the level of their reliability is impossible.

Methods for Estimating the Reliability of Pipes and High-Pressure Vessels. One can isolate two groups of methods for determining the quantitative indicators of the reliability of these elements: From statistical data of operation or tests and under conditions of the absence (or insufficiency) of failure statistics.

With respect to operational failures of a noncatastrophic nature (potentially hazardous failures) a pipe is considered as a repairable system whose elements are straight sections, bends, weld seams, and cast elements (elbows, T-joints, etc.) [3]. Calculation of the reliability is based on the statistics of failures of the corresponding elements in operation. In estimating the reliability indicators the assumption is usually used that in normal operations the reliability law of a pipe (vessel) is approximated by the exponential function.

For each type of pipe (and vessel) it is suitable to calculate as the reliability indicators the following parameters of the flow of potentially hazardous failures:

a) for straight sections

$$\omega_1 = m_{\mathrm{r}}/Tl \quad \text{(per running meter)} \tag{1}$$

$$\omega_2 = m_{\mathrm{s}}/T\pi Dl \quad \text{(per unit surface)} \tag{2}$$

b) for weld seams

$$\omega_3 = m_{\mathrm{w}}/T\pi Dn_{\mathrm{w}} \quad \text{(per unit seam length)} \tag{3}$$

c) for bends

$$\omega_4 = m_{\mathrm{b}}/Tn_{\mathrm{b}} \quad \text{(per bend)} \tag{4}$$

Here m_r, m_w, m_b are the number of failures on straight sections, weld seams, and bends of the pipt, respectively; T, working lifetimes of the pipe; l and D, length and outside diameter of the pipe; n_b, number of bends in it; and n_w, number of weld seams.

In the general case, i.e., for a pipeline with pipes of several diameters,

$$\omega_2 = m_{\mathrm{r}}/T\pi \sum_j Dl_j; \quad \omega_3 = m_{\mathrm{w}}/T\pi \sum_j D_j, \tag{5}$$

where j is the number of the pipeline section of a single type or size.

For estimates of the failure flow parameters it is advisable to determine the confidence intervals of I_α: ω_l and ω_n (α is the confidence coefficient), which characterize the accuracy of the statistical estimates. One can obtain the upper and lower confidence limits for the failure flow parameters from the formulas

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$$\omega_{ui} = \omega_i/r_2; \ \omega_{li} = \omega_i/r_1, \tag{6}$$

where r_1 and r_2 are tabular coefficients which depend on m, i, and α [4, 5]; the subscript i denotes "r", "w", or "b", respectively.

In the absence of element failures during the working lifetime T the value of the confidence limit ω_m can be determined from the formula

$$\omega_{ui} = (1-\sqrt[n]{1-\alpha})/T, \tag{7}$$

where n is the number of scale units (running or square meters, bends). One can estimate from this same formula the upper confidence limit of a reliability indicator such as the intensity of the catastrophic failures (brittle fractures) of pipelines or vessels of an AEPP. Here the question is precisely the intensity and not the failure flow parameter, since one should consider the indicated elements as nonrestorable with respect to catastrophic failures.

When statistical information is absent or inadequate, the reliability of pipelines and pressure vessels can be estimated approximately with the aid of methods based on probability models describing the behavior of pipelines (vessels) under operating conditions and their depressurization or rupture.

The "load–strength" model [6] can serve as an example of the simplest model. With its help the probability of one random quantity (the load) exceeding another random quantity (the strength) of the material of a pipe or vessel is estimated. More complicated models are based on the application of methods for analysis of the stress state, the mechanics of rupture (taking the origin and development of cracks into account, and others. At present the majority of these methods of estimating reliability are in the development stage.

An Estimate of the Actual Reliability of Pipes and Pressure Vessels of an AEPP According to Operational Data. It is necessary to have representative statistics of failures in order to obtain reliable estimates of the reliability indicators. Since the failure of a pipe or vessel is a rather rare event, additional operation of these elements is necessary to obtain the required amount of original data. Up to the present time the operational period of only some domestic AEPP is sufficient for obtaining reliable statistical data on the reliability of pipes and pressure vessels.

Power units with a capacity of 1 million kilowatts with high-powered water-cooled channel (RBMK) and water-cooled–water moderated power (VVÉR) reactors are of the greatest interest for the development of nuclear power. A power unit with a VVÉR-1000 is in the construction stage. Power units with RBMK-1000 have up to 1978 been operated for about 11 reactor-years. Since there have been no failures of pipes and pressure vessels of the primary loop of these reactors, it is possible to make an upper estimate of the failure flow parameter for the pipe system of the primary loop of this type of reactor according to Eq. (6). It amounts to $\sim 10^{-5}$ h^{-1} per reactor.

The first two units of the Novovoronezh AEPP with a capacity of 210 and 365 kW, respectively, have the longest period of operation of domestic power units with VVÉR. It amounts to 17 reactor-years. From the moment it came into operation until 1975 two failures of pipes of the primary loop were recorded in all. In accordance with this, the failure flow parameter of the pipe system of a single circulation branch of these reactors is equal to $\sim 1.5 \cdot 10^{-6}$ h^{-1} per reactor.

In the interests of a more detailed analysis of the actual reliability of pipes and high-pressure vessels statistical information was collected at the first AEPP in the world at Obninsk, the Beloyarskaya AEPP, and an AEPP in Dimitrovgrad with a VK-50 reactor. These plants have been operated for an extended period of time (the Obninsk plant – from 1954, the VK-50 – from 1965, and the Beloyarskaya plant – from 1964). As a result it has proven possible to collect failure statistics which can be processed by the procedure outlined in the previous section. The data at the first AEPP in the world were collected from the start of its operation through 1975 inclusively and at the Beloyarskaya and VK-50 sites – from the start of operations through 1978.

The characteristics of the elements considered and the indicators obtained for their reliability are given in Tables 2-4. The overall failure flow parameter for the main pipes of the primary loop is given in Table 2. The failure flow parameters per weld seam and per bend of these pipes are approximately equal and amount to $5 \cdot 10^{-8}$ h^{-1}; the 80% confidence interval is $(1\text{-}8) \cdot 10^{-8}$ h^{-1}.

Analysis of the operational data of the pipes and vessels of the VK-50, which is an experimental industrial facility, has shown that practically no failures of the elements noted have been observed from the time of its start-up. It is possible under these conditions to calculate an upper confidence limit for the failure flow param-

Declassified and Approved For Release 2013/02/15 : CIA-RDP10-02196R000800020004-1

Declassified and Approved For Release 2013/02/15 : CIA-RDP10-02196R000800020004-1

TABLE 2. Characteristics and Failure Flow Parameters of Pipes and Pressure Vessels of the First AEPP in the World

Equipment	Material	Pressure, kgf/cm²	Temp., °C	ω, 10^{-6} h^{-1}
Main pipes of the first loop, D_y = 200	1Kh18N9T	100	100—300	20,8
Individual pipes, D_y = 200	1Kh18N9T	100	290	0
Volume compensator housing	1Kh18N9T	100	—	1,6
Economizer	1Kh18N9T	100 (14) *	200	3,9
Steam superheater	Carbon steel	100 (13) *	290	24,0
Evaporator	1Kh18N9T	100 (14) *	200	2,4

*The pressure inside the housing is given in parentheses.

TABLE 3. Characteristics and Upper 95% Confidence Limit for the Failure Flow Parameter of Vessels and Pipes of the VK-50

Equipment	Temp., °C	Pressure, kgf/cm²	ω_u, 10^{-5} h^{-1}
High-pressure separator	310	100	3,5
Low-pressure separator	241	35	3,5
Low-pressure preheater	—	15	7,5
Deaerator tank	104	1,2	3,5
Pipe assembly	100—300	1—100	7,5

TABLE 4. Technical Characteristics and Reliability Indicators of the Pipes and High-Pressure Vessels of the Beloyarskaya AEPP

Equipment	Diam. and thickness of wall, mm	Material	Pressure, kgf/cm²	Temp. °C	Failure flow parameter: overall ω; $I_{0,8}$, 10^{-6} h^{-1}	straight section ω_1; $I_{0,8}$, 10^{-8} $h^{-1}\cdot m$	weld seam ω_3; $I_{0,8}$, 10^{-8} $h^{-1}\cdot m$	bend ω; $I_{0,8}$, 10^{-8} h^{-1}
High-pressure (HP) pipe of loop I	279×14	Kh18N9T	160	340	—	1,5 0,8—2,1	44,3 35,7—52,1	9,2 6,0—12,1
HP pipe of saturated steam of loop I	219×16	Kh18N9T	110	316	—	0 0—1,6	13,0 5,4—19,4	4,2 0,9—6,8
Main steam pipe	279×15	12KhMF	95	510	—	0 0—0,9	27,8 19,4—35,6	59,0 46,1—71,1
HP supply pipe of loops I and II	245×18	St 20	135	300	—	5,4 2,2—8,1	30,0 19,5—39,5	49,6 38,3—59,8
Individual pipe	32×3	1Kh18M9T	145	330	—	0 0—0,04	0 0—1,2	0 0,02
Drum separator	1600×92	16GNM	150	340	5,8 2,4—8,7	—	—	—
Evaporator	2000×70	18TS	110	316	8,0 4,9—10,7	—	—	—
HP preheater	1500×12	22K	12	290	7,2 14,9—9,4	—	—	—
Hydraulic collector	273×28	1Kh18N9T	130	310	1,9 1,1—2,6	—	—	—
Collector of preheated steam	325×35	12KhMF	115	510	6 0—1,1	—	—	—

Declassified and Approved For Release 2013/02/15 : CIA-RDP10-02196R000800020004-1

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meter (see Table 3). The average value of the flow parameter and its lower confidence limit agree and are equal to zero. The characteristics of the pipes, pressure vessels, and collectors of the Beloyarskaya AEPP are presented in Table 4. There the 80% confidence interval $I_{0.8}$ for the failure flow parameter is given.

Analyzing the operational testing of the primary and secondary loops of the Beloyarskaya AEPP, as well as of the first AEPP in the world, one can note that the greatest part of the pipe and pressure vessel failures of these power units with channel reactors is related to the coolant flows in less important pipes of small diameters – flaws, breaks in drainage, surge, and air lines, and at the sites of welding of resistance thermometer cases. A similar conclusion has been drawn in [2] with respect to an AEPP with light-water reactor vessels. It is noted there that 70% of the rejected pipes at the power plant have a diameter of 150 mm. The majority of the failures in individual pipes of the primary and secondary units of the Beloyarskaya AEPP are associated with leaks from the attachment points of disconnected equipment.

Flaws and cracks in weld seams have arisen, as a rule, in the case of increased vibration of the pipes, their insufficient compensation ability, or poor quality in making the weld seams, as well as corrosion. A crack in the region of a weld seam is a dangerous kind of defect in collectors. Such defects can result in the complete breaking away of pipes from collectors. It is significant that no serious failures of large pipes and high-pressure vessels have been observed at all the operational domestic AEPP.

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OPTIMIZATION OF NUCLEAR POWER SYSTEM INTEGRATED WITHIN COMECON

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The development of nuclear power in the COMECON member-countries will proceed along the lines of growing integrating tendencies. Precisely in the case of nuclear power integration can yield maximum effect. Integration of the fuel cycle will make it possible to step up the rate at which fast reactors are introduced and this will be of a decisive importance in the future. The prospects of an integrated system for the COMECON member-countries have been shown in [1]. Calculations were carried out on an optimization model of a nuclear power system, integrated with respect to fuel connections, embracing the COMECON member-countries. A minimum demand for natural uranium in the entire forecast period was adopted as the optimization criterion. It was found that because of the integration the fraction of fast breeder reactors could be increased by 8-12% for COMECON as a whole by the end of the forecase period while the consumption of natural uranium would be reduced by 13-14%.

Whereas in the qualitative respect the conclusions of [1] are industputable, their quantitative aspect requires refinement. The point is that the use of natural uranium as an optimization criterion in nuclear power optimization does not take account of such important characteristics as capital investment, the cost of mining and processing nuclear fuel, and therefore distorts the quantitive characteristics of the structure of the nuclear power industry.

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In forecasting nuclear power development within national frameworks, as a rule use is made of the normalized expenditures. The use of normalized expenditures as the criterion within one country and minimum consumption of natural uranium as the criterion within COMECON leads to contradictory results. It is necessary, therefore, to develop methods taking account of the monetary costs of construction and operation of nuclear energy plants and to consider them within the framework of COMECON. This goal has been set in the present paper. In our considerations we do not go into the territorial location of particular capacities of atomic power plants with fast or thermal reactors. We only resolve the problem of optimization of the structure of the nuclear power industry, i.e., the quantitative relation between fast and thermal reactors.

In the development of a method of optimization according to a monetary criterion the basis adopted may be a nuclear power industry model consisting of system of linear balance equations reflecting the fuel connections among the atomic power plants in the COMECON member-countries and the links with the electric power systems of the countries. This model is given in [2] and for an integrated nuclear power system within the framework of COMECON it is written as

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

Let us note that the models of the nuclear power industry for each country within the national framework are analogous to Eq. (1) in form.

Let us consider the set S of countries and let us assume, as was done in [2], that the objective of each country is to minimize the total normalized expenditures for the development of a national power system for an assumed rate of growth. For each country these expenditures are calculated in the national currency. In accordance with this, we shall assume that the i-th country strives to develop its nuclear power system so that

$$C'_i x \to \min,\ i=1,\ 2,\ \ldots,\ S, \tag{2}$$

where C'_i is the vector of normalized expenditures of the i-th country (in the national currency) and **x** is the strategy for the development of a nuclear power system.

Actual accounting practice between COMECON member-countries employs a monetary unit called a conversion ruble. It is natural to use this unit in the present problem as well. If the factor for conversion from the currency of the i-th country to the conversion ruble is denoted by a_i, then the vector of specific normalized expenditures by the i-th country, expressed in conversion rubles, is calculated as $C_i = a_i C'_i$ and S functionals of the integrated nuclear power system can be written as

$$C_i x \to \min,\ i=1,\ 2,\ \ldots,\ S. \tag{3}$$

Thus, the problem of finding the optimal strategy for the development of an integrated nuclear power system is formulated as follows: On the set (1) find the set of optimal strategies which satisfy conditions (3).

This problem is one of optimizing a system of linear constraints with a vectorial linear criterion. The optimal set of strategies satisfying the conditions of the problem formulated is, as is known, a Pareto set. At the same time, in most known methods of solving problems with a vectorial criterion or, as they are also called, multicriteria problems, this set is not determined. Let us consider two such methods employed in practical problems: The method of a leading objective function and the step-by-step method.

The method of a leading objective function [3] envisages optimization of the problem according to an objective function. Each of the remaining objective functions, along with the limiting value given for it, are included as an additional inequality in the model of the system. Application of this method presumes that one of the criteria of the problem dominates over the others and, moreover, requires additional data about the possible range of values of the criteria incorporated into the model in the form of constraints. All of this precludes the use of the given method in the problem of optimizing the nuclear power industry within the framework of COMECON.

The step-by-step method [4] consists in the following. Let all the criteria be arranged in order of decreasing importance; $f_1, f_2, \ldots, f_S$. We shall assume that all of them should be reduced to their minimum value. The algorithm for finding a compromise solution is as follows. First, find the optimal value with respect to one criterion, f_1. Then, denote some "step" Δf_1 by which the value of the given criterion can be increased so as to obtain a better result for the next most important criterion, f_2. Next, we find the optimal solution for criterion f_2 with the subsidiary constraint

$$f_1(x)\geqslant f_1^0+\Delta f_1, \tag{4}$$

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where f_1^0 is the optimal value of criterion f_1. Then, we once again begin step Δf_2, which determines the deviation from the optimal value of the second criterion obtained in solving this problem; this is done to subsequently obtain the optimal solution for f_3, etc.

As is seen, this method also assumes that the criteria are of uneven importance and hence this makes the application of this method in the given concrete problem inappropriate as well.

How, then, should the formulated problem be solved? To begin with, let us find strategies which are Pareto-optimal. The strategies entering into the Pareto set are determined as follows. The strategy $\bar{x}_p$ does not belong to the Pareto set if in the set (1) there is a strategy $\tilde{x}_p$ such that the relation

$$C_k\tilde{x}_p < C_k\bar{x}_p,$$

holds for at least one functional of that strategy whereas for the other functionals

$$C_i\tilde{x}_p \leqslant C_i\bar{x}_p, \; i=1, \ldots, k-1, \; k+1, \ldots, S.$$

Clearly, all the strategies from region (1), apart from the set of strategies of the type $\bar{x}_p$, enter into the Pareto set and, therefore, are optimal. The economic sense of any Pareto-optimal strategy consists in the fact that this strategy can in no way be improved without at the same time increasing the total normalized expenditures for the development of the nuclear power industry, for at least one COMECON member-country.

There is a large number of Pareto-optimal strategies. Theoretically, without considering the essence and specifics of the given problem, one cannot determine precisely which strategy from the Pareto set should be preferred. However, some ideas concerning the properties of one Pareto-optimal strategy or other can be elucidated.

In particular, a Pareto-optimal strategy can be obtained for each country by minimizing the linear functional

$$f_i(x) = C_i x \to \min.$$

in set (1).

The pertinent strategy will be denoted $x^0{}_i$. For it we have

$$f_i^0 = C_i x^0 = \min_{\{x\,:\,Ax=b\}} C_i x \tag{5}$$

At least S such strategies exist. It is characteristic that each of them is optimal for some particular country but may not be optimal for the others. In our opinion, therefore, even though they do not enter into the Pareto set these strategies are less preferable than other strategies from the same set which possess better compromise properties, i.e., are more applicable for the entire collection of COMECON member-countries. It is proposed to find these strategies by using functionals of the type

$$\Phi_1 = [\alpha \min C_i x + (1-\alpha)\max C_i x] \to \min, \quad i=1, 2, \ldots, S, \tag{6}$$

where α is any parameter satisfying the condition $0 \le \alpha \le 1$ and

$$\Phi_2 = \sum_{i=1}^{S} p_i C_i x \to \min, \tag{7}$$

where $0 \le p \le 1$ and $\sum_i p_i = 1$. (For the given problem of optimizing the nuclear power system within the framework of COMECON it is expedient to take $p = 1/S$.)

Functionals (6) and (7) are minimized on the set (1). Functionals (7) are linear whereas functionals (6) are nonlinear but the method presented in [5] makes it possible to reduce the problem of minimizing (6) on set (1) to a set of problems in linear programming. Thus, the stage of looking for strategies which are Pareto-optimal can be carried out on the formalized level. Let us note that the solution of the problem with only one criterion (5) is equivalent to the well known generalized objective function method [6].

The proposed method thus allows a wider spectrum of optimal strategies to be determined than does the generalized objective function method. Since there can be any number of Pareto-optimal strategies, not all of them possess sufficiently good compromise properties. Therefore, it is desirable to isolate strategies with the best compromise properties. To this end it is necessary to formulate and supplement the model with special conditions in which the compromise properties would be expressed in explicit form.

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As is seen from Eq. (5), f_i^0 represent the minimum values of the expenditures of each country for its own nuclear power development program with an optimal ratio between atomic power plants with thermal and fast reactors and the condition that the other participants ($k = 1, 2, \ldots, i-1, 1+1, S$) of the integrated system carry out their own nuclear power development programs, without pursuing the objective of economy of expenditures, whereas the given, i-th country minimizes its expenditures.

In the integrated system, however, all the participating countries, striving to reduce their expenditures, cannot simultaneously attain f_i^0, $i = 1, 2, \ldots, S$.

A reasonable condition for compromise will be the condition of equality of relative deviations from f_i^0 for all countries. For any strategy x the relative loss for a given country is given by

$$\bar{f}_i(x) = [f_i(x) - f_i^0]/f_i^0.$$

In accordance with what has been said above, the Pareto-optimal strategy should be one such that the relative deviations $\bar{f}_i(x)$ of all countries would be equal, i.e.,

$$\bar{f}_i(x) = \bar{f}_j(x), \tag{8}$$

where $i = 1, 2, \ldots, S$; $j = 1, 2, \ldots, i-1, i+1, \ldots, S$. Then the problem of determining the appropriate compromise strategy (we shall call it the strategy of equal relative losses) is formulated as: in sets (1) and (8) find the strategy x^0 for $C_k x \to \min$ (k is any of $i = 1, 2, \ldots, S$).

This problem is solved by ordinary linear programming methods. The strategy of equal relative losses, of course, is not the only reasonable compromise strategy. The very principle of determining it (equal deviations from f_i^0, $i = 1, 2, \ldots, S$, for all countries) is not indisputable since f_i^0 reflect an unreal picture of the relations in the integrated system. Another approach to work out a compromise is closer to practice. Suppose that each country develops its planned nuclear power program independently (without integration) of the other countries. In this case the expenditures are φ_i, i.e., the results of optimization of the national nuclear power system without account for integration links according to the monetary criterion. Obviously, integration will produce a gain for all countries and it is reasonable to find such a compromise strategy for which the relative gain

$$\bar{\varphi}_i(x) = [f_i(x) - \varphi_i^0]/\varphi_i^0$$

would be identical for all countries. This condition is written as

$$\bar{\varphi}_i(x) = \bar{\varphi}_j(x), \tag{9}$$

where $i = 1, 2, \ldots, S$; $j = 1, 2, \ldots, i-1, i+1, \ldots, S$. To determine this strategy, which we shall call the strategy of equal relative gain, it is necessary to solve the following linear programming problem: In the sets (1) and (9) find the strategy x^0 for $C_k x \to \min$, where k is any of $i = 1, 2, \ldots, S$.

Thus, in the present paper a method is given for determining the Pareto-optimal strategies with implicitly expressed compromise properties and two strategies of equal relative losses and gains in which the sense of the compromises is obvious. Let us note in conclusion that the strategies of equal relative losses and gains are invariant with respect to the currency conversion factors.

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FUEL CONTRIBUTION TO THE COST OF NUCLEAR POWER

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The fuel used in a nuclear power station has some major effects on the calculation of costs in the production of electrical power. The fuel spends a considerable time in the reactor, and there is therefore a gradual transfer of the cost of the fuel (not only rods but also the complete heat-producing assembly) to electrical energy. Therefore, the partially exhausted fuel in the reactor always has some residual (actual) cost, which is proportional to the remaining available energy, i.e., the difference between the design burnups and the value actually reached.

Other features of the power production are the need for an initial fuel load at the start of operation, while the fuel enrichment at the start may be small throughout the volume by comparison with the steady-state mode of operation, and the effective transient period of operation is fairly long. During this period the design burnup for the fuel is gradually increased to values corresponding to the steady-state.

The fuel component of the cost of nuclear energy is therefore determined as the ratio of the actual cost of the reactor core to the energy reserve at a given time t:

$$C_f(t) = K(t)/\mathscr{E}(t). \tag{1}$$

In the general case of a multizone reactor (with zones having cassettes differing in initial fuel enrichment, uranium load, and cost), the expanded form of the expression becomes

$$C_f(t) = \frac{100}{24} \frac{\sum_{j=1}^{l} n_j(t) C_j \{[P_j(t) - \widetilde{P}_j(t)]/P_j(t)\}}{\eta \sum_{j=1}^{l} g_j n_j(t) [P_j(t) - \widetilde{P}_j(t)]} [\text{kopecks/kW} \cdot \text{h}], \tag{2}$$

where j is the zone index and l is the number of zones, i.e., the number of types of fuel-rod assembly differing in the above features; $n_j(t)$, number of rod assemblies of type j at time t; C_j, initial cost of an assembly (the cost of a fresh assembly) of type j in rubles; $P_j(t)$, design burnup averaged over all assemblies of type j at time t in kW·day/kg of U; $\widetilde{P}_j(t)$, average actual burnup attained at time t for all assemblies of type j actually in the reactor in kW·day/kg of U; g_j, uranium load in an assembly of type j in kG; and η, net efficiency.

In a one-zone reactor, all the assemblies have identifal initial engineering and economic parameters, in which case the expression takes the traditional form

$$C_f(t) = \frac{100}{24} \frac{C}{\eta g P(t)}. \tag{3}$$

In the case of a power station with RBMK channel reactors, additional absorbers are used to suppress the excess reactivity at the start of the transient period. As the reactor produces power and fission products accumulate in the fuel, these additional absorbers are replaced by fuel-rod assemblies. The cost of the additional absorbers is not incorporated into (2) and (3). The increase in the fuel component when the cost of the additional absorbers is incorporated is $\Delta C_f^{a a}$, which can be derived by referring the cost of all the additional absorbers in the reactor at the start of the run to the amount of electrical energy produced during the period T_{aa} during which the additional absorbers are in the reactor. If some of the additional absorbers remain in the reactor after the start of fuel recharging, one can take as T_{aa} the time up to the unloading of the last of the additional absorbers initially present.

The usual period T for calculation purposes is a calendar year, and the mean over this period $\bar{C}_f$ is given by

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$$\overline{C}_f = \int_{t_1}^{t_2} C_f(t)\, dt/T, \tag{4}$$

where $T = t_2 - t_1$.

In what follows the fuel contribution to the cost of power is understood as the mean over the calendar year. The fuel component calculated in this way allows one to incorporate all the power station costs for fuel apart from the fraction corresponding to the residual cost of an assembly actually present in the reactor at a given time. Clearly, the residual cost at the start of operation is the cost of the entire original fuel load. At the end of the transient period and subsequently, this is approximately half of the cost of the installed fuel load. This situation at the end of the transient period is characteristic of the RBMK reactors, which work with continuous reloading.

In the case of the VVÉR reactors, a large fraction of the cassettes may be changed at a single operation, and then the residual cost of the fuel load after the end of the transient period fluctuates between values equal to about 1/3 of the cost of the full installed load before the next reloading and 2/3 of that after reloading. The mean value of the residual cost for the fuel load in the VVÉR after a reasonably long period resembles that for the RBMK in being close to half the clost of the actual fuel load.

The residual cost of the fuel load remaining in the reactor corresponds to the part of the cost of the fuel not yet transferred to electrical power. This is part of the balance of the power station and is reckoned along with the stock of fuel, which is usually a certain percentage of the annual consumption, and this forms part of the circulating funds. Therefore, the balance for the power station always contains very considerable contributions from the fuel in the reactor as yet only partially utilized. This feature is characteristic of nuclear power and has no analog in ordinary power production. However, in financial and practical respects the feature has some major disadvantages, since it involves withdrawing from the economy some substantial and continually increasing capital resources, in part on account of the increasing scale of nuclear power.

In fossil-fuel engineering, the standard circulating funds are designed to cover fuel stocks and in character constitute emergency reserve funds [1]. In nuclear power, this position is maintained, as should be the case, by additional transfer to the cost of nuclear power of some part of the cost of the fuel as defined by the residual cost. This approach has been considered in [2], where it was suggested that 2/3 of the cost of the installed fuel load in the VVÉR reactor should be assigned to the cost of the electrical energy during the standard period of return.* The period within which the residual cost of the fuel load is to be written off may be taken as the calculated working life of the nuclear power station for discussion purposes (30 years).

It is preferable to write down this cost over the standard period for return on investments; this approach correctly reflects the need for efficient use of the working funds associated with the fuel load. This requirement is met by minimizing the fuel component of the calculated costs for electrical power in the power-station design [3]. The standard return period is also close to the length of the transient period, during which there are ongoing increases in the economic fuel parameters for various technical reasons. The period is one of gradual improvement and stabilization of the technical and economic parameters of the station.

The writing-down period for the residual cost of the fuel load may be divided in various ways over the years throughout the total return period, or else one can use a linearly decreasing proportion (accelerated writeoff method) while retaining the same writeoff period. In the latter case, there is an increase in the rate of return on the investment in the fuel load in the power station, while the working funds are reduced to the value corresponding to the fuel stock at the station.

The increase (ΔC_f) in the fuel component by comparison with the values given by (2) and (3), viz., C_f takes the following form by years for the above methods of referring the actual cost of the fuel load to the electrical energy production:

$$\Delta C_{fk} = 100\alpha m_{st}\, C/TW_k; \tag{5}$$

$$\Delta C_{fk} = \frac{100\alpha m_{st}\, C}{TW_k} 2\left(1 - \frac{k-0{,}5}{T}\right), \tag{6}$$

*In [2], this component of the cost was termed the constant (capital) component of the fuel component of the cost of electrical energy over the standard return period for capital investment. The division of the fuel component into two parts corresponds to the approach taken here, but the method suggested there for calculating the fuel component appears incorrect to us. The exact mode of variation of the circulating funds for a nuclear power station was not considered in [2].

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Declassified and Approved For Release 2013/02/15 : CIA-RDP10-02196R000800020004-1

where k is the year of operation of the nuclear power station within the standard return period for capital investments; T, that period; M_{st}, number of fuel assemblies in the working state (in the steady state of operation); C, cost of one fuel assembly FA in rubles; W_k, output of electrical energy in year k during the standard period for return on capital investments in kW·h/yr; and α, a coefficient equal to the ratio of the residual cost of the fuel load to the cost of the steady-state fuel load when the latter consists of fresh fuel assemblies, i.e., the ratio of the residual (actual) energy reserve to the design energy reserve in the core, or the ratio of the difference between the burnup of the loaded fuel assemblies and the mean burnup attained over the reactor as a ratio to the burnup of the loaded fuel assemblies in the steady state. The value of α must be determined by physical calculation. The value is close to 1/2 for reactors of RBMK type.

In the case of a multizone reactor (when there are fuel assemblies of different types in the steady-state load), one can calculate ΔC_f from the following formulas:

$$\Delta C_f = \frac{100}{TW_k} \sum_{i=1}^{l} \alpha_j m_{st\ j} C_j; \tag{7}$$

$$\Delta C_f = \frac{100}{TW_k} 2 \left(1 - \frac{k-0{,}5}{T}\right) \sum_{j=1}^{l} \alpha_j m_{st\ j} C_j. \tag{8}$$

Then the fuel component of the power cost in the initial period (C_f^*), which is equal to the standard return period for capital investments, has two components: The theoretical fuel component of the cost (as given by (2) or (3) and an additional fuel component given by (5) and (6) or (7) and (8), with the latter due to the residual cost of the fuel load as assigned to the power output when the reactor has reached the steady state:

$$C_f^* = C_f + \Delta C_f. \tag{9}$$

If the length of the transient state is less than the standard period for return on capital investments, as should be the case for a power station working to its design parameters, then one can calculate the fuel component of the power cost from (9) while incorporating all the costs for fuel from the instant when the reactor was commissioned to the end of the return period, i.e., in year T + 1 and subsequently the balance will contain only the cost of the fuel assemblies that constitute the fuel stock, which is a standardized quantity essential to ensure ongoing station operation.

When the fuel component is calculated from (9), the working funds of the station for the standard return period are reduced from the value equal to the cost of the initial fuel load plus the reserve stock of fuel elements at the station in the first year to a value equal to the cost of the stock of fuel assemblies at the station in the year following the end of the return period.

We now consider how the working funds of the station vary within this period for the case of a channel reactor working with continuous reloading and having in stock a set of assemblies constituting a fraction β of the planned consumption of fuel assemblies for the current year. In this case, the working funds in rubles for the first year of operation should be

$$\Phi_1 = Cm_0 + \beta n_1 W_1. \tag{10}$$

In the second year of operation the component Cm_0 of this sum relating to the fuel load should be reduced by the amount equal to the difference between all the electrical energy produced in the first year and the cost for fuel assemblies additional to the original load. Then the working funds of the power station in the second year are

$$\Phi_2 = Cm_0 + Cn_1 W_1 - \frac{C_{f_1}^* W_1}{100} + \beta C n_2 W_2; \tag{11}$$

and in the third year

$$\Phi_3 = Cm_0 + Cn_1 W_1 - \frac{C_{f_1}^* W_1}{100} + Cn_2 W_2 - \frac{C_{f_2}^* W_2}{100} + \beta C n_3 W_3; \tag{12}$$

and in year k within the capital-investment return period

$$\Phi_k = Cm_0 + C \sum_{i=1}^{k-1} n_i W_i - \frac{1}{100} \sum_{i=1}^{k-1} C_{f_i}^* W_i + \beta C n_k W_k; \tag{13}$$

and in year k = T + 1:

$$Cm_0 + C \sum_{i=1}^{T} n_i W_i = \frac{1}{100} \sum_{i=1}^{T} C_{f_i}^* W_i; \tag{14}$$

$$\Phi_{T+1} = \beta C n_{T+1} W_{T+1}, \tag{15}$$

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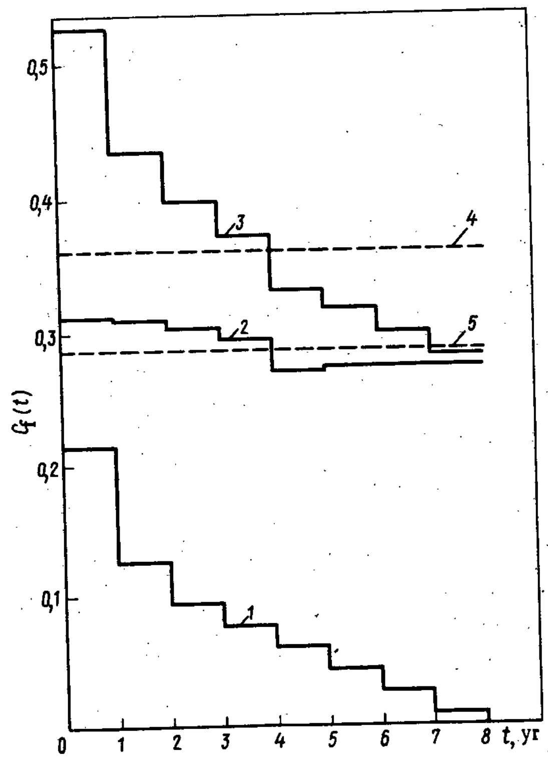

Fig. 1

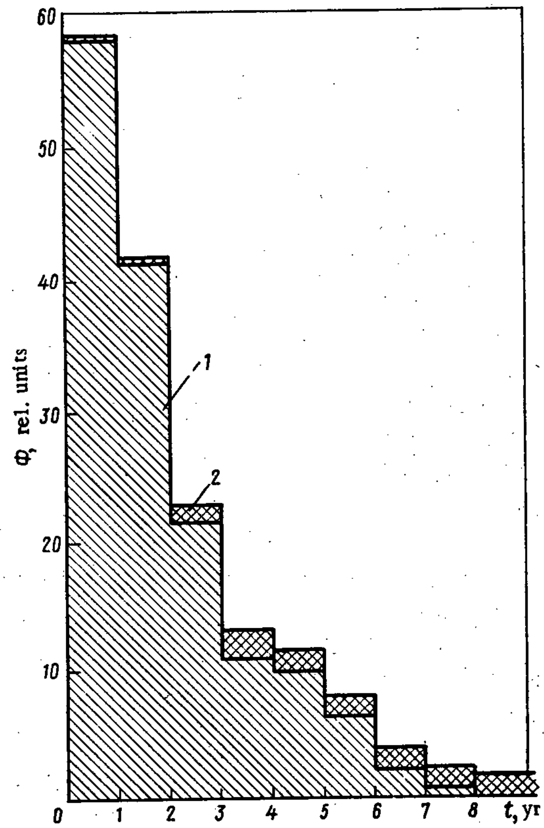

Fig. 2

Fig. 1. Effects of working time on: 1) additional fuel component; 2) theoretical fuel component; 3) total for the cost of electrical power (curves 4 and 5) are the means with and without allowance for the additional component, respectively).

Fig. 2. Effects of working time on the working funds associated with: 1) the fuel load in the reactor; 2) the fuel stock in the store.

where m_0 is the number of fuel assemblies in the initial load, n_i is the rate of reloading in year i per kW·h, and W_i is the output of electrical energy in year i kW·h/yr.

$$\sum_j^l C_j m_{0j};\quad \sum_j^l C_j n_{ij} W_i;\quad \sum_j^l \beta_j n_{ij} W_j. \tag{16}$$

This scheme is also applicable to nuclear power stations with VVÉR reactors. The number of fuel assemblies at the start of operation corresponds to the steady-state fuel load, while the number of fuel assemblies changed in one reloading cycle is about 1/3 of the load in the core:

$$m_0 = m_{\rm st};\quad n_i W_i = 1/3 m_{\rm st}. \tag{17}$$

The fuel assemblies have several different degrees of enrichment in the transient state and in the steady state and therefore the VVÉR is to be considered as a multizone reactor, and the parameters of formula (16) are to be used in calculating the working funds.

The particular question of the standards for fuel stock must be based on experience with operation and reloading, as well as the dynamics of financing and fabricators' supplies. As an example, Figs. 1 and 2 show the variation in the planned fuel component and in the working funds for a power station with a RBMK-1000 channel reactor. The period for writing down the residual cost of the fuel load is taken as the standard period for return on capital investments, viz., 8 years. The writing down is performed by the accelerated method of (6) and (8). The stock of fuel rods in the store is taken as 10% of the annual consumption of fuel assemblies for the current year.

If the actual performance of the reactor during running deviates from the planned level, it is necessary to correct the previously calculated economic parameters on the basis of the actual values, and also to perform a new physical calculation on the basis of the consequences of the deviation. This recalculation affects

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the fuel component of the power cost calculated in the previous year as well as the working funds, since the newly calculated values deviate from those expected from the previous year, which results in an additional quantity (positive or negative), which arises from the deviation from the previous year.

Availability of planned and actual values for the product costs and for the working funds is a feature common to nuclear engineering and to ordinary power engineering, but a specific feature of nuclear power production is associated with the prolonged time spent by the fuel in the reactor, which complicates matching planned and actual economic parameters. Nevertheless, it can be said with confidence that such calculations can be performed with certainty as experience in operating nuclear power stations accumulates.

The current scales of advance in nuclear power engineering require the solution of many practical problems in the actual economic relationships between nuclear power stations and their supporting organizations, as well as users of electrical energy. This method of calculating the fuel component of the cost not only has an advantage over the previous approaches but also provides for flexible response to any changes in the fuel cycle arising from internal factors and from national economic demands.

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MATHEMATICAL MODEL OF THE OPTIMIZATION OF THE STRUCTURE OF NUCLEAR HEAT SOURCES

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In a number of papers on the use of nuclear fuel for supplying heat the efficiency of nuclear heat sources (NHS) was determined by the technioeconomic component of various versions of local power supply systems with nuclear and fossil-fuel heat sources. The excessive consumption of fossil fuel in these versions was taken into account by closing costs, and nuclear fuel by prime costs. We note that the rate of growth, the nuclear power structure, and external fuel cycle indices affect the fuel component of NHS costs. In our opinion the solutions for the development of a long-term program for the use of nuclear energy to supply heat which were obtained by alternate calculations for individual heat-supply systems should be corrected by taking account of the growth of nuclear power and external fuel cycle plants. Thus, to optimize the structure of NHS it is necessary to take account not only of such local conditions as the location of the area, the conditions of the industrial water supply, the cost of heat transport, the dynamics of growth, the structure of the heat loads, etc., but also to consider a number of system factors: The structure and rate of introduction of nuclear power sources, the effect on plutonium of the accrued operating time of NHS and nuclear power plants (NPP), the change in cost of natural uranium as a function of its resources and rate of consumption, indices of the enrichment unit of the fuel cycle, the possibility of a future changeover of NHS to a new form of fuel (uranium metal or uranium oxide of a different enrichment, plutonium) and also the limited resources of nuclear engineering and the conditions for concurrent operation of NPP and ATÉTs in the power system. Such an approach is necessitated by the fact that NHS on the one hand operate in local heat-supply systems, and on the other hand are a component part of the nuclear power system, share a common fuel cycle with NPP, and together with them carry the electrical load of the power system.

The present article is devoted to the development of methods of complex optimization of the structure of NHS, taking account of system factors (primarily external fuel cycle indices). One efficient method for solving the problem posed makes use of economic-mathematical modeling with a computer. A number of mathematical

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models have been constructed for prognostic studies of the development of nuclear power. These models can be used to optimize the structure of NHS, to determine the role of various types of reactors in power systems, the value of plutonium etc.

The model presented in [1] uses a systems approach to optimize the structure of centralized heat-supply sources using fossil fuel. The model developed at the High-Temperature Institute of the Academy of Sciences of the USSR permits taking account of the use of nuclear fuel at ATÉTs whose structure is not optimized. Calculations were performed in [2] with a model for optimizing an NPP system under development when the NPP is supplemented by NHS. The NHS system was represented by a simplified model which was related to the optimization model of an NPP system through balance equations for natural uranium and plutonium. Calculations with these models showed that they are strongly coupled, principally through Pu.

This is shown by the large return flow of plutonium from NHS to NPP whose economic effect (economy in the NPP system) comprises 7-10% of the NHS costs [3]. This effect can be used for local technicoeconomic calculations.

These studies resulted in the development of an optimization model of an NPP—NHS system which takes account of a more detailed set of factors by using certain principles introduced in [2-4].

Basic Principles of the Mathematical Model. 1. The growth of NPP power is specified for base load and semipeak conditions on the basis of prognostic data. The type of electrical generating systems (NPP, ATÉTs), the type of reactors (VVÉR, RBMK, VK, BN), the unit power, and other parameters are not specified, but are to be determined. Increases in AKÉS and NHS power during intervals of the design period are optimized.

2. In the immediate future (up to 1990) NHS are considered as separate power sources sited and operating locally in heat-supply systems of specific industrial-residential agglomerates (IRA). Technicoeconomic calculations showed the expediency of replacing fossil-fuel heat sources by nuclear sources. A list of these IRA was produced as a result of developments at the VNIPI Energy Industry Institute. Different versions of the development of a heat-supply system up to 1990 using NHS are specified for each IRA, and for each version a technicoeconomic calculation is performed to determine indices used in the model. Up to 1990 the structure of NHS will be optimized by a choice (as a rule close to the local optimum) of one of these versions for each IRA, taking account of the overall optimum of the growth of nuclear power. This approach permits a sufficiently detailed account of the effect of local conditions and system factors on the optimal structure of NHS.

3. In view of the lack of research on the efficiency of NHS in heat-supply systems of specific IRA, the structure of NHS after 1990 will be optimized in the more distant future without taking account of their tie-in to specific IRA.

4. NHS which generate electrical power (ATÉTs and AKÉS-T*) are represented in the model by two variables: installed electric capacity and heat load, which are related by a special equation. The variables must be separated for NHS because situations can arise when the central-heating load of some of the turbogenerators may not reach maximum at once, but lag the startup of the main equipment, particularly in AKÉS.

5. Optimum dates of changeover of NHS and AKÉS from one form of fuel to another are determined in the model. To this end auxiliary variables characterizing the plant capacity are introduced into the matrix of the model, which shift to the use of the new form of fuel. These variables are related to the capacity of plants started up earlier on the old form of fuel by a special equation.

6. The balance equations for natural uranium and plutonium are similar to those in [3] except that these take account of the presence of NHS and the change in the consumption of nuclear fuel resulting from a changeover of a part of the nuclear power sources to a new form of fuel. The uranium resources are represented by separate categories of deposits which differ in extraction costs.

7. In contrast with the model in [2], enrichment costs are removed from the fuel component cost for AKÉS and NHS. They are taken into account in the system as a whole by introducing a special condition on the balance of the separative work. This provides a more convenient way to vary the enrichment costs and to provide a changeover from one form of fuel to another in nuclear power sources by changing the coefficients in the balance equations. The fuel component of the costs of AKÉS and NHS takes account of the cost of fabri-

*AKÉS-T are condensing NPP which can be used for central heating by means of auxiliary space-heating pipes or the use of unregulated sets of condensing pipes for heat distribution.

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cating and transporting fuel elements, and part of the cost of chemical processing of spent fuel, which is not related to enrichment. Thus, the changeover to a new form of fuel does not lead to a significant change in the fuel component of the plant cost, but this fact must be taken into account in the balance equations.

8. The problem of optimizing the external fuel cycle is not considered in this model, but an attempt is made to take account of its effect on the shaping of the optimal NHS structure.

9. The model takes account of the limited possibilities of nuclear engineering. This refers mainly to the manufacture of VVÉR reactor vessels. The restrictions on the production of RBMK are not as rigorous as those on VVÉR from the point of view of reactor engineering. In view of this it is assumed that nuclear power from both VVÉR and RBMK reactors will increase in the near future. This is reflected in the model in the form of a restriction of the production of VVÉR reactor vessels, leading to competition of AKÉS with vessel-type reactors and ATÉTs. Within the next few years nuclear engineering plants must produce products for the projected development of AKÉS. Therefore the use of ATÉTs leads to their competition with AKÉS not only in covering the schedule of electrical load, but also in the distribution of nuclear engineering products.

10. The length of the design period is assumed to be 40 years (1981-2020). Extension of the design period beyond the year 2020 is prevented by the lack of reliable long-term predictions of the increase of heat loads.

In accord with the above the model can be presented as a mixed integer problem of linear programming.

Mathematical Formulation of the Problem. It is required to find

$$\min \Phi = \sum_{v=v_j}^{V} \sum_{j=1}^{J} \Big(\sum_{r_h=1}^{3} X_{r_h jv} c_{r_h jv} + \sum_{r_\alpha=0}^{4} X_{r_\alpha jv} c_{r_\alpha jv} \Big) +$$

$$+ \sum_{p=1}^{P} \sum_{i=1}^{I} Y_{ip} C_{ip} + \sum_{v=1}^{V} \big(X_v^{(o)} c_v^{(o)} + X_v^{(S)} c_v^{(S)} \big) + \sum_{s=1}^{S} X_{sv} c_{sv} - X_V^{(Pu)} c_\infty^{(Pu)}.$$

Integer variables: Y_{ip} is a Boolean variable ($Y_{ip} = 0$ or 1); if $Y_{ip} = 1$, the i-th version of the heat supply enters the optimal solution for the p-th IRA. The integer nature of this quantity is determined by the initial assumption that only one of the considered versions of the development of a heat-supply system using an NHS can be realized.

Continuous variables: $X_{r_h jv}$ is the increase in installed (condensing) capacity of NPP (AKÉS and ATÉTs), GW (el); $x_{r_\alpha jv}$ is the increase in heat load of ATÉTs and the part of AKÉS transferred into the central-heating regime, and also the increase in power of single-purpose nuclear heat-supply installations, nuclear heat-supply plant (NHP), Gcal/h; X_v^o and X_v^s are the separative works in the interval v of the design period for operating enrichment plants and those started up in the same interval, assuming that all the uranium enriched in the interval v is completely consumed, 10^6 separative work units/interval v; X_{sv} is the amount of natural uranium extracted in the interval v from a source s, assuming that all the natural uranium extracted in the interval is consumed in that same period, 10^3 tons of U/interval v.

Subscripts: v is an interval in the design period ($v = 1, 2, \ldots, V$); j, type of plant (AKÉS, ATÉTs, NHP), the type of reactor (VVÉR, RBMK, VK, BN), the unit power of the block, the kind of fuel (oxide or uranium metal, plutonium), and the degree of enrichment; v_j, interval in which a plant of the j-th type can be put into operation; r_h, r_α, times the plant supplies electric, thermal power ($r_h = 1, 2, 3$: $r_\alpha = 0, 1, 2, 3, 4$; $r_h = h - 4$; $r_\alpha = 4\alpha$; h is the number of h/yr the station supplies electric power, $h = 5, 6, 7 \times 10^3$ h/yr); α, ratio of the central-heating power of ATÉTs and AKÉS-T turbo-units, or the ratio of the thermal power of NHP reactors to the heat load of the plant; p, number of IRA in the heat-supply system which are schedule to use NHS from 1980 ($p = 1, 2, \ldots, P$); i, number of the version of the development of the heat-supply system of the p th IRA using NHS during 1980-1990 ($i = 1, 2, \ldots, I_p$); s, number of the natural uranium source, characterized by the cost of extraction c_s and uranium resources G_s^u ($s = 1, 2, \ldots, S$);

Coefficients of Unknowns in Objective Function. The $c_{r_h jv}$ are the reduced costs per unit of installed capacity of an NHS of the j-th type put into operation in the interval v and operating in the r_h mode, without taking account of the cost of enriched fuel (rubles/kW·yr); the $c_{r_\alpha jv}$ are the reduced costs per unit of thermal power of an NHP put into operation in the interval v, or supplementary costs related to the central-heating load of selections of ATÉTs or AKÉS-T in the interval v. In the latter case these costs include expenditures for heat transport, heat-exchange equipment, peak-reserve boilers and the related consumption of fossil fuel. They also include expenditures for the electrical energy necessary to compensate for the decrease in electric power generated by NPP during a heating period for a space-heating load of groups of turbines with the space-heating coefficient α, without taking account of the cost of enriched uranium, rubles/Gcal/h·yr: The C_{ip} are the reduced costs connected with the i-th version of the development of the heat-supply system of the p-th IRA

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during 1980–1990 (10^6 rubles/yr) without taking account of the cost of uranium enrichment; c_v^o, c_v^s, reduced separative work costs in the interval v for operating and newly started up enrichment plants without taking account of the cost of the electrical energy consumed, rubles/unit of separative work; c_∞^{Pu}, value of the plutonium consumed yearly beyond the end of the design period, 10^6 rubles/ton of Pu; c_{sv}, reduced cost of the extraction of a ton of natural uranium from source s in the interval v of the design period, rubles/kg U.

The minimum of the objective function is determined by satisfying the following conditions expressed by equalities or inequalities. The electric power balance takes account of the decrease in the generation of electric power with the ATÉTs and AKÉS heat load, and also the electrical energy consumed in the enriching unit of the fuel cycle. Up to 1990 only one of the considered versions of the development of a heat-supply system can be realized for each IRA. The heat balance (low potential for domestic and technological needs) is taken into account starting in 1990.

The thermal and electric powers of NPP (ATÉTs, AKÉS) are related, since their heat load is available only after startup of the main plant, and must not exceed the maximum value of the corresponding electric power introduced. The heat load limit of an AKÉS for each interval of the design period is related to the restricted number of areas for an AKÉÉ which can be used for a heat supply. The production and consumption balances of nuclear fuel (natural uranium and plutonium) are similar to those described in the model for a growing NPP system except that the supplementary variables $X_{r\alpha jv}^{U-Pu}$, $X_{\eta_h jv}^{U-Pu}$, and X_{ipv}^{U-Pu} are introduced in order to optimize the changeover of nuclear energy sources to a new form of fuel ($U \rightleftharpoons Pu$).

The balance of the separative work necessary for the enrichment of the fuel consumed by NHS and AKÉS started up in an interval of the design period and earlier takes account of the decrease in the separative work in the changeover of part of the plant from uranium to plutonium.

Restrictions are introduced on the capacities of existing enriching plants and nuclear engineering plants.

Plants with thermal reactors can be changed from uranium to plutonium fuel no earlier than in the next interval after they were put into operation. The total power of these plants operating in the interval w must not exceed the power of plants of the same type put into operation in preceding stages ($v = 1, 2, \ldots, W-1$).

On the basis of a systems approach, and taking account of the dynamics of growth and the interaction between various types of nuclear energy sources, the model developed permits an optimization of the structure of NHS with respect to types of plants (NHP, ATÉTs, AKÉS-T) and installed equipment, an estimate of the relative efficiency of various types of NHS and the efficiency of the management of the chemical reprocessing of spent fuel, the optimization of the changeover of plants from one form of fuel to another, and a study of the effect of various factors (indices) on the optimal solution.

Using this model, calculations were performed with preliminary input data. The results obtained showed that the cost and available resources of natural uranium and the management of the chemical processing of nuclear fuel strongly affect the structure of NHS, the specific capital investments in plants of various types, and local siting conditions.

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OPERATING EXPERIENCE WITH AUTOMATIC REACTOR-POWER CONTROL SYSTEM AT OBNINSK ATOMIC POWER PLANT EMPLOYING SIGNALS FROM IN-CORE SELF-POWERED DETECTORS

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The possibility of using rhodium self-powered detectors to control reactor power has been demonstrated in a number of Soviet papers [1-4]. Thus, in experiments on the IRT-2000 research reactor [3, 4] tests were made with various circuits of in-core control and algorithms of direct digital control. The necessity to develop an in-core control system for the spatial energy distribution arises from the tendency towards higher and higher unit power and size of reactors, resulting in various kinds of oscillations and fluctuations of power in the reactor core. Such a control system for the total power of a reactor should not be inferior to the traditional system in respect of quality and reliability and should also satisfy the requirements of nuclear safety.

In-reactor monitoring systems (IRM) in which Compton and activation self-powered detectors, as a rule, serves as secondary measuring transducers for information about the neutron flux density are an integral part of nuclear power reactors. In the practice in other countries [5, 6] Compton detectors with a platinum or cobalt emitter are used for in-core control and activation self-powered detectors with a rhodium or vanadium emitter are used for monitoring the energy distribution. The IRM system [6] forms the initial body of data of the program for reproducing the neutron field; this is used to calibrate platinum detectors, information from which is converted by a computer into control signals for local power regulators. However, at low power the neutron flux density is measured by ionization chambers. Emel'yanov et al., [7] described an independent system developed for the Leningrad Atomic Power Plant, a system consisting of five local automatic regulators, but they did not give the operating characteristics of the system or the type of detector.

An extremely promising IRM design is one for a system, in which signals from the same in-reactor detectors are used both to minitor the distribution of the neutron flux density (energy distribution) over the volume of the reactor core and to carry out automatic control of the reactor power. Use of a system of this kind makes it possible. To reduce the number of neutron detectors and to cut the cost of the system as a whole; to obtain information about the spatial distribution of the neutron flux density and reliable information about the total reactor power, information which is necessary for optimal reactor operation, and also to effect automatic regulation of the reactor power; and to create an in-core system of automatic regulation. Validation of the choice of in-reactor neutron-flux detectors that would be reliable and have acceptable metrological characteristics is highly important in the development of these systems.

Compton self-powered detectors are inertialess with respect to the main current-producing process. However, they are considerably inferior to activation self-powered detectors in respect of such metrological characteristics as sensitivity, linearity limit, signal-to-noise ratio, and contribution from the reactor γ-ray background [8]. For example, rhodium activation self-powered detectors are linear in a range of neutron-flux variation extending four orders of magnitude whereas Compton detectors are linear in a range of only two orders of magnitude.

A system in use since 1975 in the reactor of the Obninsk Atomic Power Plant can, in accordance with the standard GOST 21935-76, be classified as an IRM system providing primary measured information and connected into the automatic regulating system of the reactor (henceforth such systems will be denoted as IRM and AR). The primary measuring transducers in this system are 12 integrated detectors of the DPZ-7 type with a rho dium emitter whereas the active elements of the functional devices of the apparatus are series-841 hybrid integrated microcircuits.

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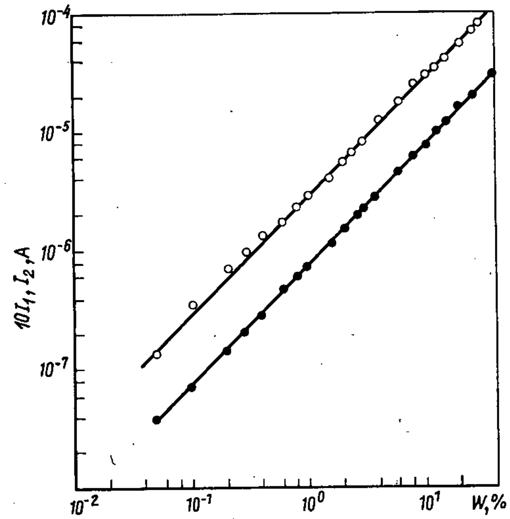

Fig. 1. Current from detectors of AR-1 (I_2) and AR-2 (I_1) vs thermal power W of reactor: ○) AR-1; ●) AR-2.

The IRM and AR system has the following parameters: Nonidentity of detectors, no more than ±2%; confidence coefficient for fault-free operation of detectors for 10,000 h, 0.85; range of measurement of detector current, $5 \cdot 10^{-9}$–$1 \cdot 10^{-5}$ A. range of regulation of total reactor power from nominal value, 0.2-125%; error of measurement of detector current by apparatus, no greater than ±0.5%; and range of variation of power-controller sensitivity by 1% of reactor-power deviation, 5-10 mV.

The system of automatic regulation of the reactor power at the Obninsk Atomic Power Plant consists of two regulators AR-1 (from ionization chambers) and AR-2 (from in-reactor detector of DPZ-7 type of IRM system). The automatic regulators operate according to the schedule: 15 days for regulator AR-1 and 15 days for regulator AR-2. Signals about the neutron flux distribution are recorded continuously and independently of the operating mode of the regulators.

Systems employing DPZ-7 detectors and operating in an information-collecting mode or as "advisers to the operator" are used in the reactors of the Beloyarsk Atomic Power Plant and the Bilibinsk Atomic Heat and Power Plant. The results of investigations on their characteristics are given in [8] and, therefore, in the present paper we report only on the results of investigations on the use of signals from in-core reactors for automatic regulation of reactor power.

During the studies on dynamic and static characteristics of the IRM and AR system we recorded: The current of each in-core detector, the mean total current of the detectors and the unbalance signal at the output of the power controller of regulator AR-2, by the IRM and AR system with outputs to an M-1633 galvanometer; the current of the regular ionization chambers and the reactivity, by a 3RTA-01 reactimeter with recording on an N-107 loop oscillographs; and coolant heating in the reactor (Δt), by resistance thermometers with output to a KSP recorder with displaced zero and extended scale. During recording on the loop oscillograph, each parameter studied according to the mean value of the constant component.

Figure 1 shows the plot of the mean total current from the detectors of AR-2 and the current from the ionization chambers of AR-1 vs the thermal power of the reactor. At each power level the reactor parameters were stabilized for no less than 2 h and the detector and chamber currents studied were averaged over five readings. It is seen from Fig. 1 that in the range from 0.05 to 40% of the nominal power the mean current of the system is linearly related to the thermal power of the reactor. The scatter of the measured currents from the detectors of AR-1 and AR-2 was ±5% and ±2.5%, respectively.

During operation the sensitivity W/I_1 of the system practically did not change with the operating time of the reactor, to within the limits of error which did not exceed ±5% (Fig. 2). No malfunctions of the detectors and functional devices of the secondary apparatus were detected.

The dynamic characteristics of the IRM and AR system were studied during regulation of the reactor power by AR-2 over the range from 0.2 to 40% of the nominal value. Perturbations in the reactivity by ±0.15 β_{eff} were created by moving the rods of AR-1; the rate at which perturbation was introduced was up to 0.05 β_{eff} per second. Figure 3 gives oscillograms of the ionization chamber, the reactivity, the mean current of the detectors of the system, and the AR-2 unbalance signal for various levels of reactor power. Analysis of

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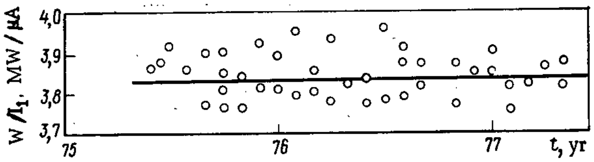

Fig. 2. Sensitivity of IRM and AR system.

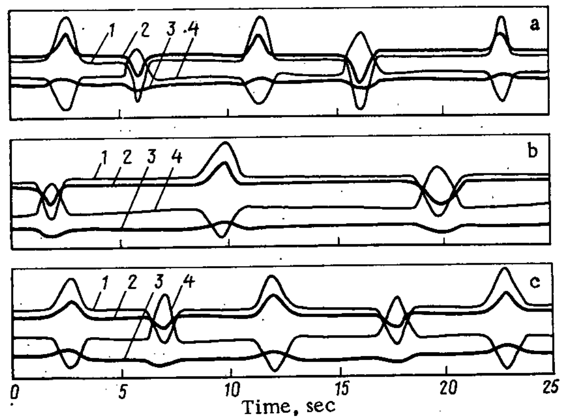

Fig. 3

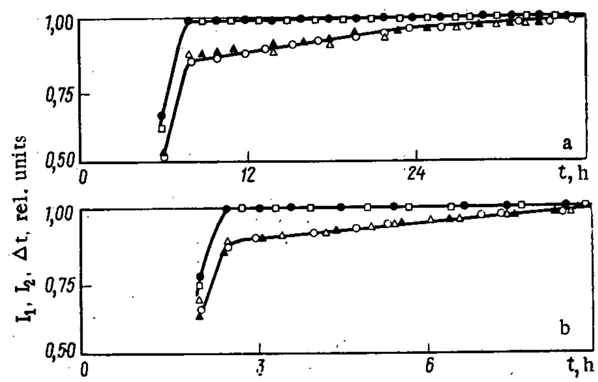

Fig. 4

Fig. 3. Oscillograms of ionization chamber current (1), mean current from detectors of AR-2 (2), unbalance signal from AR-2 (3), and reactivity of reverse polarity (4) during operation of reactor with AR-2 at power of a) 30, b) 5, and c) 0.5% of the nominal value.

Fig. 4. Variation of coolant heating (□), mean current of in-core detectors (●), current of ionization chambers IC_1 (△), IC_2 (▲), and IC_3 (○) when reactor is brought up to given power level by means of AR-1 (a) and AR-2 (b).

the parameters studied shows that the inertia of the AR-2 channel is no worse than that of the AR-1 channel; the time it took regulator AR-2 to produce reactivity perturbations of $\pm 0.15\beta_{eff}$ is 2 sec; when AR-2 generates reactivity perturbations the reactor power remains constant over the range from 0.2 to 40% of the nominal value; the character of the perturbation generation does not depend on the reactor power. Analysis of the data on coolant heating in the reactor during operation of AR-1 and AR-2 showed that the regulators sustain the thermal power of the reactor with the same quality.

Figure 4 presents plots of the currents from the ionization chambers of the reactimeter, the coolant heating Δt, and the mean current of the detectors against the operating time of the reactor as it was brought up from the minimum controlled level to a steady-state level. Along the ordinate axis the parameters have been normalized to the maximum value. With the AR-1 in operation at a steady-state power level the currents of KNT-53 ionization chambers increase by 20% in 30 h and the current of the chamber of the reactimeter (KNK-56), by 16%, whereas coolant heating and the mean current of the detectors in the IRM and AR system remain constant within the limits of 1% error. As the reactor is brought up to a steady-state power level by means of regulator AR-2 the ionization chamber currents increase by 11% in 6 h while the coolant heating and the mean detector current do not change within the limits of error, ±1%. Comparison of the curves (see Fig. 4) reveals that it is preferable to use AR-2 to regulate the total power of the reactor and to bring the reactor up to a given power level from the minimum controlled level.

For a fuller picture of the possibilities of local regulation we carried out experiments on maintaining the neutron flux density at the sites of experimental devices in the reactor core, the input characteristics of these devices being critical to the neutron flux density. The output parameters of each device were recorded with compensation for the mean value of constant component during recording on an analog recorder.

The neutron flux density was maintained at the sites at which the experimental devices are installed by regulators AR-1 and AR-2 from all self-powered detectors and from one detector set up in the direct proximity of each of the experimental devices in the diametrically opposite cell. Altogether we studied five different arrays of experimental devices, regulating rods, and detectors. The reactivity perturbations were $\pm 0.1\beta_{eff}$.

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The error of local regulation of the neutron flux density in the experiment did not exceed ±1%. Our investigations showed that regulation with the aid of AR-2 is most applicable.

Operation of the IRM and AR system based on DPZ-7 detectors showed that:

1) the mean total current of the detectors in the power range from 0.05 to 40% of the nominal level is linearly related to the thermal power of the reactor;

2) burning of the neutron-sensitive element of the detectors does not affect the sensitivity of the IRM and AR system at least during 3 years of operation within the limits of error of determination of the thermal power of the reactor;

3) the main characteristics do not differ from those of a regular system with in-core chambers: The range of power regulation is 0.2-40% of the nominal value; the time required to generate a reactivity perturbation of $\pm 0.15\beta_{eff}$ is no more than 2 sec and the error of maintenance of total power is ±1%;

4) in 2 years of operation of the system in the AR-2 version no malfunctions of detectors and functional devices of the apparatus were observed; this attests to the possibility of constructing a reliable system.

Thus, according to the principal metrological parameters, activation detectors with inertia correctors can be used successfully in the control system of a high-temperature reactor with an energy load. Combination of the functions of obtaining primary measured information and generating signals for automatic regulation in one system makes it possible to use a minimum number of detectors with an adequate volume of information.

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FISSION PRODUCTS AS γ-RAY SOURCES

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The great success in investigation of applied nuclear physics and in the development of the practical use of nuclear energy has not only led to rapid development of nuclear energy and industrial production of isotope sources of ionizing radiation, but has also fostered the development of new branches of science: radiation biology, study of radiation materials, etc. One of these sciences is radiation chemistry. Radiation chemistry technology is presently in the stage of its development where a transition is occurring from laboratory investigations to pilot-industrial and industrial scale production for a number of processes, and some radiation chemical processes have already been set up in industry, both in the USSR and abroad [1]. It is clear that high-power and low-cost sources of ionizing irradiation, particularly γ radiation* are required for large-scale radiation production.

Industrial γ facilities have become available because of the considerable successes in the development of nuclear power, which, in the words of A. P. Aleksandrov, "... in the future ... it stands out as power for multipurpose complex production of electrical energy and other forms of production... ." Experimental and design studies show that the use of radiation-chemical processes makes it possible to obtain chemical production at an atomic power station by using only γ radiation emitted from the reactor, the cost being fully comparable with the cost of electrical energy produced at the AES" [2]. The sources may be fission products (FP) from nuclear fuel, where the γ-ray energy increases as the total power of nuclear reactors (NR) increases. The present review addresses the problem of using FP as γ-ray sources in radiation facilities.

The average power W_γ of a radiation facility using γ-ray FP is related to the thermal power W_T accompanying the nuclear fuel transferred to the radiator during its operating time in the *a* nuclear reactor core, by the formula

$$W_\gamma = k\eta_\gamma W_T, \tag{1}$$

where k is the fraction of γ-ray FP energy from fission energy; and η_γ is the transfer coefficient for FP γ-rays transferred to the radiator of the radiation facility. By definition [3]:

$$\eta_\gamma = \int_{t_f} q_\gamma (t_r, t)\, dt/kt_r, \tag{2}$$

where $q_\gamma (t_r, t)$ is the specific power of FP γ-rays, per unit thermal power, which depends on the dwell time of the nuclear fuel in the reactor core t_r and the time after fissions were stopped (the storage time) t; t_f is the exposure time of the nuclear fuel in the radiation facility. Thus, Eqs. (1) and (2) indicate that for a given reactor power W_γ the radiation characteristics of the FP radiation are determined by the exposure time of the nuclear fuel in the reactor core and in the radiator and by the time to transport the material from the core to the radiator.

Many investigators have determined the total γ-ray FP energy E_γ for a single fission [4]. These data mainly show good interagreement. The recommended values [5] are $E_\gamma = 7.19 \pm 1.3$ MeV/fission. Therefore, for a fission energy of 200 meV [5], we have $k = 0.036 \pm 0.0065$.

At present the information on the function $q_\gamma (t_r, t)$ is quite vast, obtained both theoretically and experimentally, over a wide range of time t_r, ranging from $t_r = 0$ (instantaneous fission) up to hundreds of days or even years (nuclear fuel withdrawn when spent) and for $t < 10^8$. Analysis of the published information shows that the most reliable data are those of [8], obtained by "matching" the results of [6, 7] for instantaneous fission ($t_r = 0$), and the experimental data of [9] in the region $10\,\text{sec} \le t_r \le 10$ h (Fig. 1). For $t_r \gg 10$ h the results

*The use of other types of ionizing radiation is either difficult (the irradiated medium is contaminated by fission products for chemonuclear processes, or activation of objects when neutron irradiation is used), or only small items (tape, sheets or tubes) can be irradiated by charged particles because of low penetrating power.

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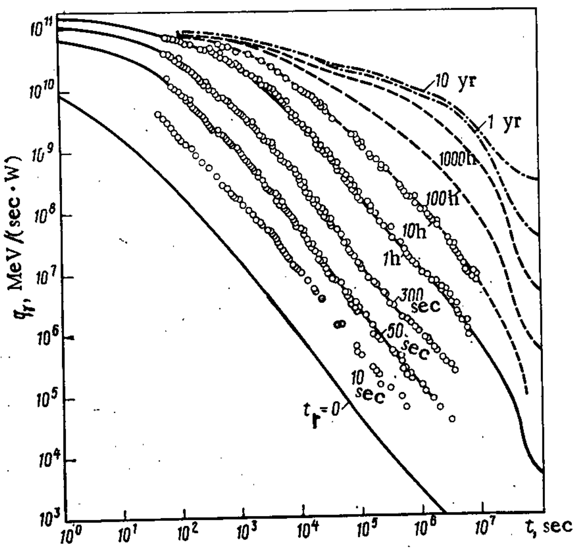

Fig. 1. Dependence of q_γ on t and t_r: ○) (the data of [9]; ——) [8]; – – –) [7]; – · –) [11].

of all investigations show quite good interagreement, and one can use, e.g., the readily available and detailed tables [10] or the data of [11] (see Fig. 1).

To calculate the absorbed dose field in an irradiated environment and to design facility shielding one must have information on the spectra of FP γ radiation, which is a source of very complex composition (up to 500 isotopes emit more than 5000 lines [12]). The published information is very extensive, but theoretical and experimental results from individual sources often show unsatisfactory agreement. Therefore, the use of different data leads to substantial differences in the design of radiation facilities with FP as a γ-ray source. In our opinion, the most reliable and satisfactory agreement was found for the data of the sources listed below.

Of special interest is the spectrum of instantaneous fission, since in the limiting case we may use it to calculate the spectrum for any t_r and t (of course, without allowing for FP burnup). Analysis of published data shows that at present it makes sense to use the results of [8] for energy distribution for $t < 2 \cdot 10^5$ sec and the most recent data [11, 13] for t up to 100 years. In the region $t_r \leq 80$ sec the preferred information is [14], and in the region $t_r = 50$ sec–10 h it is [8, 11, 15]. For design of facilities with $t_r > 10$ h it is quite accurate to use the data of [10, 11]. Preliminary design of shielding may be done for an average spectral energy $\bar{E}_\gamma$ [10]. In addition, $\bar{E}_\gamma$ can be used to compare the spectral characteristics of sources. Figure 2 shows calculated $\bar{E}_\gamma$ as a function of the exposure time $\bar{E}_\gamma$ for t_r equals 0–10 years, from the data of [8, 11, 15].

It is clear that with increase of t_r we find an increase in the fraction of energy of FP γ-rays liberated in the reactor core. Naturally, the question arises as to which E_γ to use for given values of t_r and the time to transport the nuclear fuel from the reactor core to the irradiator of the radiation facility $t_{r.f}$. For this one must know the fraction of nonfission energy in the FP γ radiation (t_r, t) up to a given exposure time t for a specific t_r, i.e., one must know the relation

$$\Pi(t_r, t) = \int_t^\infty q_\gamma(t_r, t)\, dt/kt_r. \tag{3}$$

The dependence of Eq. (3) is shown in Fig. 3, constructed by numerical integration of the relation $q_\gamma(t_r, t)$ (see Fig. 1), and it also shows that, to increase the fraction of γ-radiation energy in the FP released in the radiator, one must decrease t_r and $t_{r.f}$ and increase t_f. However, as can be seen from Fig. 3, it makes sense within specific limits to reduce $t_{r.f}$ and increase t_f. In fact, with an increase of t_r there is a drop in the fraction of γ radiation from short-life FP, and excessively fast transport of nuclear fuel to the radiation facility does not lead to an appreciable increase in the radiator power. For example, for $t_r = 600$ sec it is sufficient that $t_{r.f} \approx 100$ sec, and for $t_r = 10$ h it is sufficient to have $t_{r.f} \approx 10^3$ sec (see Fig. 3). It is desirable to increase t_f only to values for which the γ-ray power of the radiator satisfies the requirements of radiation production technology in an optimal economic fashion.

The data on the function (t_r, t) enable us to estimate the energy capability of known methods [16] of using FP γ-rays in radiation facilities, where FP fuel elements are used which are spent fuel elements with in–

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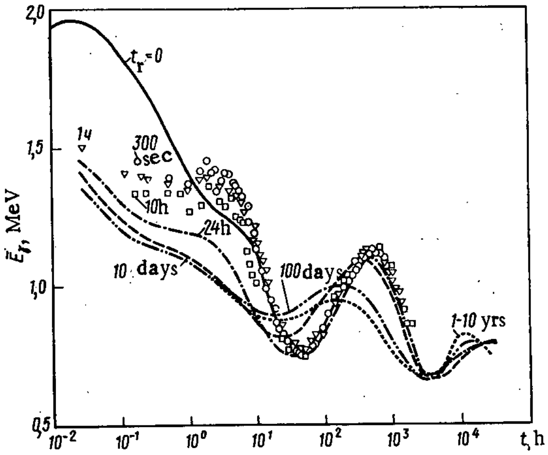

Fig. 2. The quantity $\bar{E}_\gamma$ as a function of t and t_r, constructed from the results of [15]: ∇, ○, □ for $t_r = 0$ [8]; — · —, — — —, — ·· — — for the other values of t_r [11].

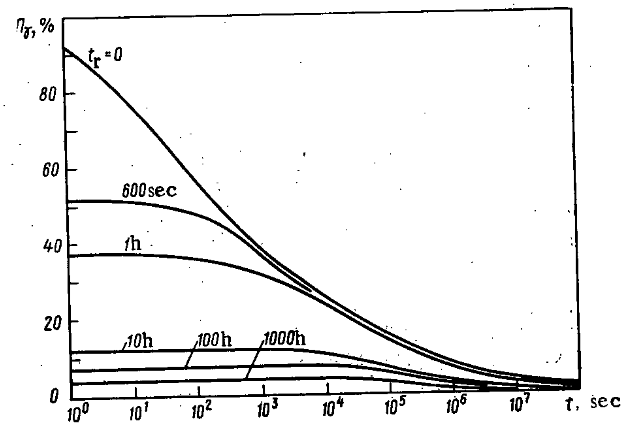

Fig. 3. Quantity Π_γ as a function of t and t_r.

complete burnup, fuel elements which have been taken repeatedly from the reactor to the radiator and back (discrete circulation); nuclear fuel which has been continuously circulated in a closed reactor—radiator system, and the uranium radiation circuit (URC).

Radiation facilities with spent fuel elements were built earlier than the other types and find the most use at present [17]. This is natural, since the introduction of a large number of nuclear reactors has provided an apparently "free" source of γ radiation. But a very small fraction of FP γ radiation is used in these facilities [16]. The reason is that the operational lifetime of must nuclear reactors is many months, and frequently 1-2 years, and, as is seen from Fig. 3, even with an operational lifetime of 1000 h, a maximum of 3-4% of the FP γ radiation can be used. Therefore, in facilities conducting FP investigations W_γ may be $\sim 10^{-4} W_T$, at the most. Since the fuel elements of nuclear power reactors may be placed in the facility ordinarily several months after the reactor is halted for recharging, the actual power of the radiators in these facilities may only be $\sim 10^{-5}$ of the thermal power available in these elements during their exposure in the reactor. The time-average power of a radiator may easily be determined from nomograms [18].

Gamma facilities with spent fuel elements, intended for research purposes and tests in large-scale and industrail-scale equipment, have been constructed both in the USSR and abroad. Quite a complete survey of the radiation parameters and the structural features of most of these has been given in [17, 19].

The universal facility in the USSR [18] was designed to use up to 100 fuel elements from research reactors, with a total γ-radiation power of up to 10 kW. This facility was used to obtain a first batch of radiation-vulcanized automobile tires [16].

Although spent fuel elements are a cheap source of γ radiation, to equip a facility with these involves large expenditure on the basic and auxiliary equipment and on remote control. With a low coefficient of γ-radiation power released to the radiator, all of this contributes greatly to the cost of the radiation, which is comparable with that of radiation from ^{60}Co [20]. In addition, the power of the γ radiator is very unstable with time. Thus, from the energy and economy viewpoint facilities using spent fuel elements have low efficiency, although they may be justified in certain specific cases.

One should note the possibility of using γ-radiation from high-activity residues from nuclear fuel processing. Studies [21] have shown that in the USSR, allowing for the increase in nuclear power level, particularly at the AES with fast reactors, in future there will be a considerable reserve in the production of residue in regenerating nuclear fuel. However, although one may expect a lower cost for γ-ray residues than for ^{60}Co [21], the solution to the problem of wide use in industry must await a detailed development in the technology for commercial production of these sources.

At the beginning of the 1960s it was shown to be desirable to periodically exchange incompletely burned fuel elements between a nuclear reactor and a radiation facility [22] (discrete circularization), the essence of which being as follows: Fuel elements which have remained in the core of an operational reactor for a time t_r, are loaded after a time $t_{r.f}$ into the radiator for time t_f, and thereafter returned to the reactor. The process is repeated many times until the fuel elements are completely burned. Here t_r and t_f may vary from

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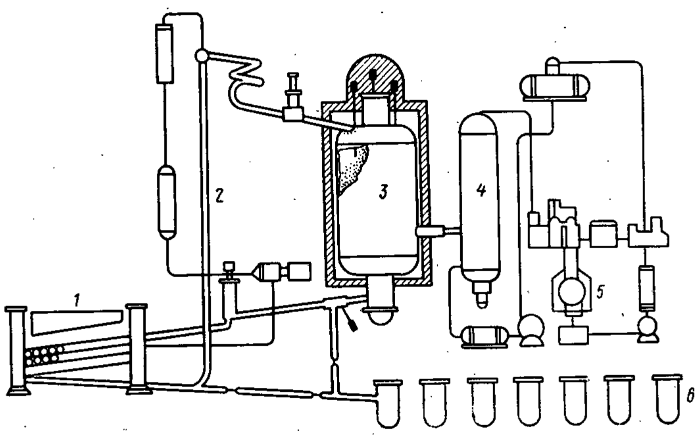

Fig. 4. Layout of the VGR-50 facility: 1) radiator; 2) URC channel; 3) reactor; 4) steam generator; 5) power equipment; 6) fuel element store.

cycle to cycle. The operational cycle for discrete circularization is usually weeks [23], and may be as low as several hours [24], if fast transport of fuel element is available (fractions of an hour).

Calculations [22] and values of the average γ-radiator power of existing facilities indicate that discrete circularization of fuel elements leads to an increase of η_γ by several times, compared with facilities using spent fuel elements (see Fig. 3). Thus, the γ-radiation power of a radiator may be up to 10^{-3} of the thermal power available in the fuel elements loaded from the reactor to the facility. On the basis of the general theory of radiation profiles [16] methods have been developed for calculating the radiative-engineering parameters of facilities with discrete circularization of fuel elements [22, 25], methods for equalizing the dose fields in irradiated objects have been suggested, and in particular it has been shown to be effective to combine fuel element with cobalt sources for these purposes [26].

The literature contains descriptions of two foreign facilities with discrete circularization of fuel elements in the research reactors BSR (U.S.A) [24] and BBP-C (the GDR) [23], which are used to sterilize food products and medical instruments, to conduct materials radiation hardening tests, and in a Soviet facility in the VVR-Ts reactor [26]. The Soviet facility operated in a one or two-week cycle. The loaded elements contained up to five fuel elements each. The γ-radiative power was 125 W. At present it has been reconstructed to provide for pilot production of articles made of wood and concrete, impregnated with vinyl monomers. The average power from a planar radiator is ~ 1 kW. It is proposed to use ^{60}Co sources to equalize the dose field.

It is more efficient to use γ-radiation FP, as is shown by investigations [3, 27-29] in the URC radiation engineering. In fact, the maximum values of η_γ, and therefore also of W_γ, can be obtained primarily by reducing t_r and $t_{r.f}$ (see Fig. 3). In an operating reactor this is possible only when nuclear fuel is circulated in a closed reactor—radiator cycle, i.e., by creating a URC.

Soviet scientists have suggested using URC as a source of γ-radiation even in the 1950s [30], but until now there was no information on its being accomplished. This is due to the fact that only now, for the first time, have the required high-power γ-ray sources (hundreds of kilowatts) been prepared for inclusion into industrial process; secondly, it has been shown experimentally that one may accomplish a URC, in particular, with spherical fuel elements [27]. In addition, in recent years the economic efficiency has been calculated [28, 29]. In fact, the parameters of circulating nuclear fuel in a URC channel depend on the type of reactor and the aggregate condition of the circulating working substance (uranium hexafluoride, fused salts of fission isotopes, spherical fuel elements, etc.).

At present the most promising for feasible use are the URC with VTGR and spherical fuel elements (the AVP type [31]): On the one hand, a test of operation of the AVR reactor has shown that such systems are reliable, and on the other hand, it has been shown experimentally that the circulation of spherical fuel elements may be arranged comparatively simple with $t_r = 1\text{-}3$ hr, $t_{r.f} = 10$ min, $t_f \sim t_r$.* Here W_γ is $(4\text{-}6) \cdot 10^{-3}\, W_T$ [32].

Soviet scientists are developing a complex test power-chemical facility with the VGR-50 reactor [33, 34], with spherical fuel elements (Fig. 4). The power of the FP γ-radiation in the radiator, allowing for self-absorp-

*In each specific case $t_{r.f}$ is determined by the condition that there be practically no activation of irradiated objects by delayed neutrons "carried out" into the radiator.

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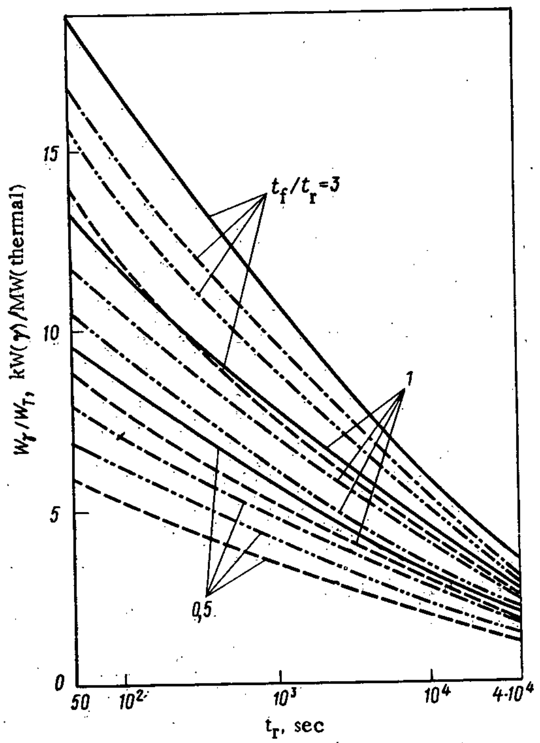

Fig. 5

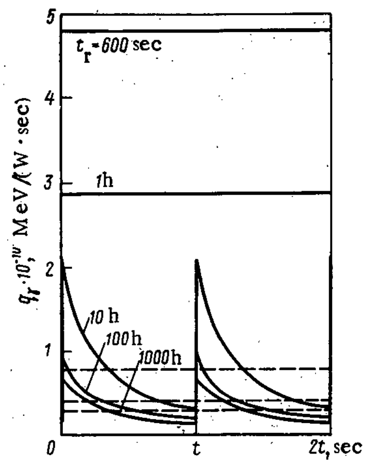

Fig. 6

Fig. 5. The specific γ-radiation power in the URC radiator W_γ/W_T as a function of t_r and the ratio t_f/t_r, for $t_{r.f}/t_r = 0$ (——); 0.05 (– · –); 0.1 (– ·· –) and 0.2 (– – –).

Fig. 6. The specific γ-radiation power (FP as a function of time in URC radiators ($t_r = 600$ sec and 1 h) and in facilities with discrete circulation of fuel elements ($t_r = 10$, 100 and 1000 h); – – – – is the average power).

tion in the fuel elements, is ~400 kW. The facility is designed for radiation processing of materials and articles to obtain improved or new physical and engineering properties. For example, it may be used to prepare radiation-modified polythene tubes, whose thermal stability is increased so much as a result of γ radiation that it becomes possible to use them in systems for supplying hot water or heating, instead of expensive steel tubes [34]. The search for methods to reduce the cost of the fuel cycle have led to the development of a reactor with a flowing nuclear fuel: liquid and gas. In the main this reactor uses fused fluorides (the MSE and MSRE [35] reactors in the USA) and gaseous fuel-UF_6 (the possibility of creating a reactor using UF_6 was shown in critical tests [36]).

Because of the high speed of transfer of flowing fuel it appears possible in principle to increase the efficiency of using fission product γ radiation in the radiator of the URC by reducing t_r and $t_{r.f}$, in other words, by a sharp increase in the fraction of γ radiation from short-lived nuclides in the radiator. According to this principle, in the URC with flowing nuclear fuel the γ-radiation coefficient in the radiator may be as high as 60% [29] (0.02 of the thermal power). However, as was pointed out earlier, this quantity is determined by the allowable minimum value of $t_{r.f}$.

The URC with flowing nuclear fuel is exploring in principle a new possibility for increasing η_γ, i.e., introducing part of the FP from the circulating nuclear fuel into the operational circuit process and concentrating it in the radiator [37]. By a flow of inert gas through the melt one can release the nuclides of krypton and xenon formed in the fission, and it is not a technically complex problem to concentrate these, together with their nonvolatile descendants. A still more important increase in η_γ can be achieved in practice by a complete release of FP in the radiator from the flow of circulating gaseous nuclear fuel, in particular UF_6 [38].

When UF_6 is used the operation of the circuit is as follows. The fission occurs in the UF_6 medium with addition of F_2, required to regenerate the UF_6 molecules broken down in the ionizing radiation field. With high temperature and power the dose of mixed radiation from most of the FP forms fluorides [38], which are carried away by the gas stream to the radiator-filter. The purified gas is returned to the reactor. The maximum possible γ-radiation power of FP deposited in the radiator-filter when no FP is deposited in the communication lines and when the FP are completely leached out by the filter, is 87% of the total γ-radiation power of the FP

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Declassified and Approved For Release 2013/02/15 : CIA-RDP10-02196R000800020004-1

TABLE 1. Order of Magnitude of Values of Governing Parameters of Radiation Facilities Using Fission Products as γ-Radiation Source

FP used	Exposure time in the reactor and radiator	Transfer time from the reactor to the radiator and back	Coeff. for removal of γ-radiation in the radiator, %	Energy flux, W/cm^2	Instability of radiator power	Cost of γ-radiator energy, rubles/kW·h)	Unit power of facility, kW;
Fuel elements processed during operating cycle	Months-years	Weeks-months	0.5	10^{-3}	tens of percent	5	50
Discrete circulation of in completely burned elements	Days-weeks	h-days	5	10^{-2}	Varies by a factor of several	3	50
URC with spherical elements	Min-h	Min	20	10^{-1}	80% of the steady-state level up to cycle 20	1	1000
URC with flowing fuel	Min	Sec-min	up to 60	1	Practically stable for constant reactor power	-	1000

in the channel, as is shown by calculations [39]. From the investigations of [39] one can conclude that the γ-radiation power in the radiator of a gaseous URC can be $\sim 2 \cdot 10^{-2}$ of the reactor thermal power.

The radiation characteristics of the URC were studied in two stages. First methods were developed for calculating the radiator γ-radiation power [22, 30], and these had low accuracy because of the large error in the initial data. These methods can be used to estimate the possibility of making a URC in industry [16], as a stimulus to developing operations at a higher level. Later a method was suggested for calculating the power and its distribution in the radiator [3], based on reliable data [8]. Calculations [27], carried out with this method, have shown that it is desirable to use a URC as a γ-radiation source. Therefore, the radiation characteristics of the radiator of a URC [40-45] were investigated in detail, investigations of reliability were commenced [46, 47] for complex power-chemical installations, and calculations of the economic efficiency were performed [29, 48]. Work was carried out to solve a number of specific problems in the physics and technology of reactors with circulation of nuclear fuel. The VVR-Ts reactor was used with a cyclic irradiation of uranium specimens to obtain information on the output power dynamics of a perfect radiator ($t_{r.f} = 0$) in a URC at a steady-state level [49], and on the steady-state level of an actual URC as a function of the cycle time parameters [32] (Fig. 5). A mathematical model was developed for computer computation of the radiator characteristics of the URC. It was suggested that data be obtained on radiation characteristics of a URC from the VGR-50 facility [51], required for calculating the productivity and for circuit protection. The basic technological requirements were formulated for the instrumentation of the VCGR with spherical fuel elements and for it to be used as the basis of a complex power-radiation-chemistry facility [32].

Comparison of the predictions for facilities with the schemes examined for using FP γ-ray sources (Table 1, Fig. 6) reveals the appreciable advantages of URC: A high coefficient for radiation going to the radiator, temporal stability of the γ-radiation power, high economic efficiency and large unit power of the facility.

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PARTICLE LOSS IN A LINEAR PROTON ACCELERATOR DUE TO RANDOM ERRORS IN THE CHANNEL FOCUSING PARAMETERS

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UDC 621.384.64.01

It is well known [1] that with the help of the Einstein–Fokker equation, one can analytically investigate particle loss due to random perturbations of the motion in circular accelerators. This article shows that an analogous method can be used in linear proton accelerators for calculating particle loss caused by random errors in the apparatus and in the parameters of the focusing element.*

We will solve the problem under the assumption that the space-charge density in the beam is negligibly small. In this case the equation for the transverse oscillation of particles in the focusing channel of a linear accelerator has the form [2]

$$d^2x/d\tau^2+Q(\tau)x=0, \tag{1}$$

where Q is a periodic function of τ; $Q(\tau)=Q(\tau+1)$. The solution may then be expressed through the phase Φ and modulus ρ of the Floquet function:

$$x(\tau)=A\rho(\tau)\cos[\Phi(\tau)+\alpha]. \tag{2}$$

The quantities A and α depend only upon the initial conditions and are constants of the motion. In particular

$$A^2=\left(\rho\frac{dx}{d\tau}-\frac{d\rho}{d\tau}x\right)^2\left(\frac{x}{\rho}\right)^2. \tag{3}$$

Let us assume that there are random errors in the placement of the quadrupole lenses of the focusing channel and in the gradient of the focusing fields. These errors will perturb the transverse oscillations and lead to an increase of the effective phase volume of the beam. In this case the amplitude of the oscillations will exceed the radius of the channel aperture, R, and the particles will be lost.

Let $P(u, \tau)$ be the probability that a particle has oscillation amplitude $A^2=u$ at "time" τ. The Einstein–Fokker equation for $P(u, \tau)$, neglecting damping of the oscillations, is

$$\frac{\partial P(u,\tau)}{\partial\tau}=-\frac{\partial}{\partial u}(\overline{\Delta u}P)+\frac{1}{2}\frac{\partial^2}{\partial u^2}(\overline{\Delta u^2}P), \tag{4}$$

where $\overline{(\Delta u)^n}=\lim\limits_{\Delta\tau\to 0}\frac{1}{\Delta\tau}\int\limits_{-\infty}^{\infty}Q(u,\Delta u,\Delta\tau)(\Delta u)^n\,d(\Delta u)$; and $Q(u, \Delta u, \Delta\tau)$ is the probability that the parameter u changes by an amount Δu during the time interval $\Delta\tau\to 0$. In our case, Q is determined by the error distribution in the lens arrangement and in the field gradients.

Let us evaluate $\bar{\Delta}u$ and $(\bar{\Delta}u)^2$. The change in the function $A^2=u(x, x')$ can be expanded in a power series and averaged over the initial phase of the oscillations, assuming all values of the latter are equally probable. Then if the pertubations Δx, $\Delta x'$ are independent of the phase (as for coherent pertubations),

$$\overline{\Delta u}=\left(\rho'^2+\frac{1}{\rho^2}\right)\overline{(\Delta x)^2}+\rho^2\overline{(\Delta x')^2}-2\rho\rho'\overline{(\Delta x\Delta x')}. \tag{5}$$

After averaging $(\Delta u)^2$ is (under the same assumptions)

$$\overline{(\Delta u)^2}=2A^2\left[\left(\rho'^2+\frac{1}{\rho^2}\right)(\Delta x)^2+\rho^2\overline{(\Delta x')^2}-2\rho\rho'\Delta x\Delta x'\right]=2u\overline{\Delta u}. \tag{6}$$

For "parametric" pertubations, when Δx, $\Delta x'$ depend upon the particles phase,

$$\overline{(\Delta u)^2}=u\overline{\Delta u}. \tag{7}$$

*It is possible to evaluate particle loss (e.g., at the end of an accelerator) with the help of a method formulated by A. D. Vlasov: "Influence of parametric errors in linear accelerator," Zh. Tekh. Fiz., 46, No. 6, 1295 (1976).

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Hereafter the averaging due to the errors is indicated by a bar, while averaging over the initial phase of oscillation is implicit.

If the particle's trajectory has a harmonic form (or quasiharmonic, as in our case), then the mean value $\overline{(\Delta u)^2}$ can be related to the mean value Δu using (6) or (7), depending upon the character of the pertubations. For coherent pertubations and those pertubations connected with the rotation of the lens axis (see below), expression (6) is correct, while for the parametric pertubations, expression (7) is appropriate.

According to [3], averaging the phases of the pertubations is equivalent to assuming isotropic random pertubations. This assumption is physically justified since a particle will experience pertubations with practically all phases before being lost. In other words, since the process is repeated (there is a large number of pertubations), particles having the same value of the parameter u in the mean are found in various conditions, independent of the initial phase. Although the "isotropic model" underestimates the result of the pertubations, we will use it as long as it allows an analytical expression to be obtained for particle loss in the cases of both coherent and parametric pertubations.

If we substitute (6) into (4) we obtain

$$\frac{\partial P}{\partial \tau} = \frac{\partial}{\partial u}\left[u\frac{\partial(\overline{\Delta u}P)}{\partial u}\right] \tag{8}$$

with an obvious boundary condition

$$P(u_{\max}, \tau) = 0, \tag{9}$$

where $u_{max} = x^2_{max}/\rho^2_{max} = R^2/\rho_0^2$. This condition means that particles with parameters $u \geq u_{max}$ do not exist; they leave the accelerator system. The initial condition is determined by the particle distribution in phase space at the initial time:

$$P(u, 0) = P_0(u), \ 0 \leqslant u < u \tag{10}$$

For example, if particles at $\tau = 0$ uniformly fill the phase volume, then $P_0 = \text{const}$.

The Einstein–Fokker equation (4) has a different form for parametric pertubations:

$$\frac{\partial P}{\partial \tau} = \frac{1}{2}u\frac{\partial}{\partial u^2}(\overline{\Delta u}P), \tag{11}$$

while P satisfies the boundary and initial conditions (9) and (10), as for the coherent pertubations.

We will express $\overline{\Delta u}$ in terms of the errors in the lens apparatus and their parameters. If the errors are relative to the middle of the focusing section, where $\rho' = 0$, $\rho = \rho_0$, then [see (5)]

$$\overline{\Delta u} = \frac{1}{\rho_0^2}\left[\overline{(\delta x_0)^2} + \frac{1}{\nu_0^2}\overline{(\delta x_0')^2}\right], \ \nu_0 = \frac{1}{\rho_0^2},$$

where δx_0 and $\delta x'_0 = \delta\,(dx_0/d\tau)$ are pertubations to the particles' trajectory at the point of the period considered.

Let us assume that in each period there are several types of errors. Then

$$\delta x = \Sigma\Delta x, \ \delta x = \Sigma\Delta x',$$

where the sum is over all forms of errors. Assuming that the errors are independent, we obtain

$$\overline{\Delta u} = \frac{1}{\rho_0^2}\left[\Sigma\,\overline{(\Delta x_0)^2} + \frac{1}{\nu_0^2}\Sigma\,\overline{(\Delta x_0')^2}\right]. \tag{12}$$

Typically, we consider the following types of errors: Displacement of the magnetic axis of the lens relative to the channel axis, inclination of the lens magnetic axis relative to the channel axis, rotation of the lens about the channel axis, or deviation of the field gradient from its calculated value. These errors can be related to the corresponding quantities Δx_0 and $\Delta x'_0$ (see, e.g., [12]). Omitting the lengthy computations, we obtain the following formulas:

1) if the axis of every lens is randomly displaced relative to the channel axis so that the root-mean-square displacement is $\sqrt{\overline{(\Delta r_0)^2}} = \langle \Delta r_0 \rangle$, then

$$\langle \Delta x_0 \rangle = a_2 K^2 \langle \Delta r_0 \rangle; \ \langle \Delta x_0' \rangle = b_2 \frac{K^2}{\varepsilon}\langle \Delta r_0 \rangle, \tag{13}$$

where $K^2 = D^2(300G/p)$; $\varepsilon = D/S$; D is the lens length; S is the length of the focusing period; G is the gradient, Oe/cm; and ρ is the particle momentum in eV/sec;

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2) if an accidental inclination of the lens axis takes place, characterized by the root-mean-square value of the displacement of every lens end $\langle \Delta r_k \rangle$, then

$$\langle \Delta x_0 \rangle = a_1 K^2 \langle \Delta r_K \rangle; \quad \langle \Delta x_0' \rangle = b_1 K^2 \langle \Delta r_K \rangle; \tag{14}$$

3) if the lens is rotated about the channel axis and the root-mean-square value of the rotation of the corner is $\langle \Delta \psi \rangle$, then

$$\langle \Delta x_0 \rangle = \sqrt{2}\, a_2 K^2 A \langle \Delta \psi \rangle; \quad \langle \Delta x_0' \rangle = \sqrt{2}\, b_2 K^2 A \langle \Delta \psi \rangle; \tag{15}$$

4) given an error in the gradient ΔG, we have

$$\langle \Delta x_0 \rangle = \frac{a_3}{\sqrt{2}} K^2 A \left\langle \frac{\Delta G}{G} \right\rangle; \quad \langle \Delta x_0' \rangle = \frac{b_3}{\varepsilon \sqrt{2}} K^2 A \left\langle \frac{\Delta G}{G} \right\rangle. \tag{16}$$

The coefficients a_i, b_i have a complex form and are dependent upon the structure of the focusing channel. They are discussed in [2]. It is evident from (13) and (14) that displacement and inclination of the lens magnetic axis lead to beam pertubation as a whole and consequently induce coherent particle oscillations. Although the averaged dimensions of the beam do not change during these oscillations, the effective aperture of the channel diminishes to the value of their amplitude. Therefore, from the viewpoint of particle loss, the pertubation of the beam as a whole is equivalent to the excitation of usual (incoherent) oscillations.

The quantity $\bar{\Delta} u$, caused by coherent pertubations (13) and (14), is not dependent upon u. In this case, transforming to the variables $d\xi = (\overline{\Delta u}/u_{max})\, d\tau$, $y = u/u_{max}$, (3) takes the form

$$\frac{\partial P(y,\xi)}{\partial \xi} = \frac{\partial}{\partial y}\left[y \frac{\partial P(y,\xi)}{\partial y} \right]. \tag{17}$$

The solution of (17), corresponding to the boundary conditions $P(1, \xi) = 0$, is

$$P(y,\xi) = \sum_{s=1}^{\infty} C_s(\lambda_s) J_0(\lambda_s \sqrt{y}) \exp\left(-\frac{\lambda_s^2}{4}\xi\right), \tag{18}$$

where (see, e.g., [4])

$$C_s(\lambda_s) = \frac{1}{J_1^2(\lambda_s)} \int_0^1 P(y, 0) J_0(\lambda_s \sqrt{y})\, dy, \tag{19}$$

J_0, J_1 are Bessel functions, and the λ_s are roots of the equation $J_0(x) = 0$.

If the lens is rotated about the channel axis, then $\overline{\Delta u} \sim u$. If we assume $\overline{\Delta u} = u\delta$ in Eq. (8) and change to the variables $y = u/u_{max}$, $d\xi = \delta d\tau$, where δ is a constant determined by (15), we obtain*

$$\frac{\partial P(y,\xi)}{\partial \xi} = \frac{\partial}{\partial y}\left(y \frac{\partial (yP)}{\partial y} \right). \tag{20}$$

The solution of this equation is

$$P(y,\xi) = \frac{1}{2y\sqrt{\pi\xi}} \int_0^1 P_0(y')\left\{\left[\exp\left[-\frac{(\ln y - \ln y')^2}{4\xi}\right] - \exp\left[-\frac{(\ln y + \ln y')^2}{4\xi}\right]\right\} dy'. \tag{21}$$

We will now consider (11). If $\overline{\Delta u} = u\delta$, where δ is determined by (16) and we define the variables $y = u/u_{max}$, $d\xi = (\delta/2) d\tau$, then the equation is

$$\frac{\partial P}{\partial \xi} = y \frac{\partial^2 (yP)}{\partial y^2}, \tag{22}$$

whose solution is

$$P(y,\xi) = \frac{\exp(-\xi/4)}{2\sqrt{\pi y \xi}} \int_0^1 \frac{P_0(y')}{\sqrt{y'}} \left\{\exp\left[-\frac{(\ln y - \ln y')^2}{4\xi}\right] \exp\left[-\frac{(\ln y + \ln y')^2}{4\xi}\right]\right\} dy' \tag{23}$$

[compare with (21)].

Given the expression for $P(y, \xi)$ it is easy to find the number (or fraction) of the particles lost, $N(\xi)$, at any moment of time. If

$$N(u, \xi)\, du = N_0 P(u, \xi)\, du$$

*Equation (8) for pertubations connected with the rotation of the lens axis is justified if x, x' and Δx, $\Delta x'$ are averaged on the phase independently (motion is considered in the horizontal plane x, x', but the increase Δx, $\Delta x'$ is due to pertubations in the vertical plane y, y').

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is the number of particles having, at the moment ξ, a parameter u between u and u +Δu, then the number of accelerated particles (particles with parameters $0 \leq u \leq_{max}$ is

$$N(\xi) = \int_0^{u_{max}} N(u, \xi)\, du = N_0 \int_0^{u_{max}} P(u, \xi)\, du.$$

According to our definition, particles are considered to be lost when the amplitude of oscillation exceeds the channel aperture (specifically, $u > u_{max}$). Their number is

$$\tilde{N}(\xi) = N_0 - N(\xi) = N_0 \left[1| - \int_0^{u_{max}} P(u, \xi)\, du\right], \quad (24)$$

where N_0 is the number of particles at the initial time $\xi = 0$.

For coherent pertubations, (24) can be integrated exactly. By substituting (18) for P and realizing that $P(u, \xi)du = P(y, \xi)dy$, we obtain

$$n(\xi) = \frac{\tilde{N}(\xi)}{N_0} = 1 - \sum_{s=1}^{\infty} C_s \frac{J_1(\lambda_s)}{\lambda_{s_j}} \exp\left(-\frac{\lambda_s^2}{4}\xi\right). \quad (25)$$

If $P(u, \xi)$ is given by (21), the fraction of particles lost is

$$n(\xi) = 1 + \int_0^1 P_0(y')\, \Phi\left(\frac{\ln y'}{\sqrt{2\xi}}\right) dy', \quad (26)$$

where $\Phi(x) = \sqrt{\frac{2}{\pi}} \int_0^x e^{-t^2/2}\, dt$ is the probability integral.

For the case described by the distribution function (23),

$$n(\xi) = 1 - \frac{1}{2} \int_0^1 P_0(y') \left\{1 - \Phi\left(\frac{\ln y'}{\sqrt{2\xi}} + \sqrt{\frac{\xi}{2}}\right) \frac{1}{y'}\left[1 + \Phi\left(\frac{\ln y'}{\sqrt{2\xi}} - \sqrt{\frac{\xi}{2}}\right)\right]\right\} dy'. \quad (27)$$

To calculate ξ we will assume that the errors (and consequently $\overline{\Delta u}$) are independent and equally probable at various periods. In this case taking into account that $\Delta\tau = 1$, we obtain

$$\xi = \int_0^{\tau} \frac{\overline{\Delta u}}{u_{max}}\, d\tau \approx \frac{\overline{\Delta u}}{u_{max}} \sum_{i=1}^{N_F} \Delta\tau_i = \frac{\overline{\Delta u}}{u_{max}} N_F,$$

where N_F is the number of focusing periods. It is then evident that ξ is completely defined if the mean increase per period of the square of the amplitude, $\overline{\Delta u}$, is known.

Thus, knowing the dependence of ξ on the error in the focusing channel and the initial amplitude distribution of the particles, we can use (25)-(27) to determine the beam loss over the entire length of the accelerator. Furthermore, with the help of (18), (21) and (23) we can calculate how the particle amplitude distribution changes. In particular, it is clear that the beam will be well spread in the process of acceleration, and the so-called halo will appear, leading to a particle loss.

Figure 1 shows the numerically calculated particle loss in the linear accelerator at the Institute for Nuclear Research, Academy of Science of the USSR, and at Los Alamos (LAMPF, U.S.A.). Curves 1-3 were calculated according to (21) assuming the presence of errors due to the random rotation of the axis of the quadrupole lenses. We further assume that the geometrical parameters of the channel (aperture, relative length of the lens, etc.) do not change along the accelerator. It is clear from the curves that the requirement of minimal particle loss (at the 10^{-5} level) leads to additional limitations on the parameters of the injected beam (specifically, on the initial distribution of particles in the phase volume). By comparing the focusing structure of the accelerators at the Institute for Nuclear Research and at Los Alamos, we can conclude that with a suitable choice of channel parameters and alignment tolerance, it is possible to decrease the loss. Curve 4 in Fig. 1 was obtained by using (25), so that only the coherent oscillations were taken into account. It is evident that in the absence of corrections to the transverse position, the fraction of the particles lost may exceed 10^{-4}. By comparing curves 1 and 4 we can see that loss due to coherent pertubation is substantially greater than loss caused by the random rotation of the lens axis. The level of "coherent loss" is 10^{-3} at the end of the accelerator, although in principle these losses can be eliminated, since the coherent errors are easily corrected. In the presence of systems correcting the beam position, the major contribution to particle loss will be associated with errors in the rotation of the lens axis, which are difficult or impossible to eliminate (or correct) in practice. Therefore their calculation is essential, particularly in the case of poor beam quality at the accelerator entrance.

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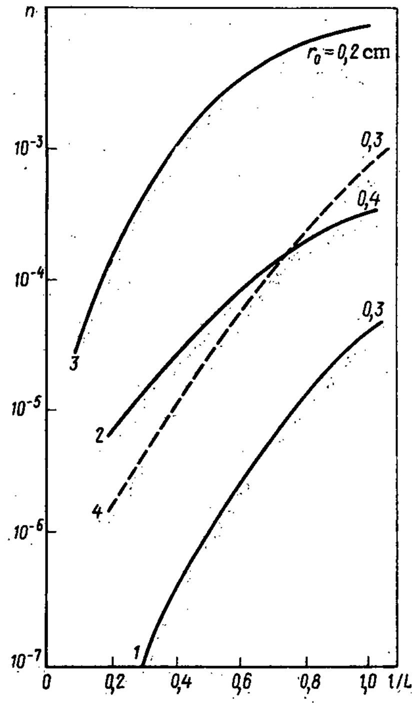

Fig. 1

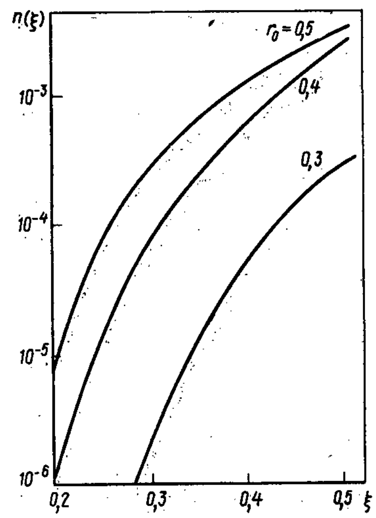

Fig. 2

Fig. 1. Particle loss in the linear proton accelerator at the Institute for Nuclear Research (1, 2, 4) and at Los Alamos (3) for various beam radii at the accelerator entrance (radii shown beside curves).

Fig. 2. Particle loss as a function of ξ for various beam radii at the accelerator entrance (radii shown beside curves).

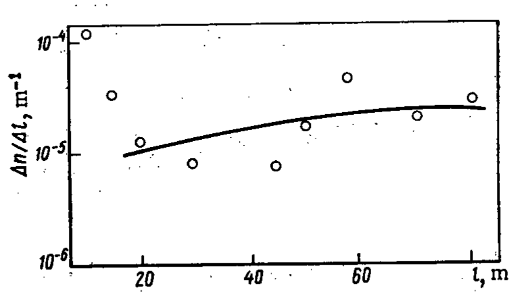

Fig. 3. Linear particle loss rate in the I-100 accelerator (O – experiment [6]).

The curve in Fig. 2 was calculated from (27) and corresponds to the errors of the lens gradient. For the accelerator considered, $\xi < 0.1$ and the losses are insignificant. In Fig. 3 the dependence of particle loss rate in the I-100 accelerator [5] is shown, calculated with the help of our formulas. The initial distribution is gaussian: $P_0(s) \sim \exp(-s/s_0)$, with a parameter $s_0 = 0.06$ (a normalized emittance contains 86% of the particles). For comparison, experimental points from [6] are shown. The agreement with experiment can be considered satisfactory if one considers that the basic contribution to the particle loss is the random pertubation of the transverse oscillations. Our results may be confirmed by measurements on the 200-MeV linear injector-accelerator at the Brookhaven National Laboratory [7]. In this accelerator, at low energies the growth of the beam occurs mainly at the expense of the peripheral particles (compare with Fig. 3). This may be explained as the motion of space charge. At energies exceeding 10 MeV, the emittance grows at the expense of changing particle distribution in the central region of the beam, in agreement with our assertions.

Up to now we have discussed the determination of particle loss in the accelerator given certain errors (tolerances). One may, however, solve the inverse problem: Given an acceptable level of particle loss, determine the permissible tolerances in the parameters of the focusing channel. This problem is of special interest for strong-focusing linear accelerators (meson producers), in which beam loss cannot exceed 10^{-4}–

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10^{-5} because of the radiation danger. Requiring loss at the 10^{-5} level may necessitate high tolerances and the correction of the particle amplitude distribution along the accelerator (with the help, for example, of "scrapers").

The authors express their sincere thanks to S. K. Esin for interest in the work and valuable observations.

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NEW BOOKS

S. M. Gorodinskii

METHODS OF INDIVIDUAL PROTECTION OF WORKERS HANDLING RADIOACTIVE MATERIAL*

Reviewed by Yu. V. Sivintsev

The development of the nuclear power industry and nuclear energy is impossible without exposing personnel to radiation danger. The development of individual protection for respiratory organs and skin of workers engaged in operations with open radioactive materials played an important role in the solution of this problem. Such protection is most essential in uranium mines, where the major sources of radiation danger are gaseous radon and the radioactive aerosol products of its decay. Another exceptionally important area of the use of such means is the clean-up of the aftereffects of a radiation accident. A series of investigations begun in the 1950s is successfully continuing in our country under the directorship of S. M. Gorodinskii. This serves as the basis for the publication of a new edition of the book reviewed here.

After a short initial chapter introducing the problem, a section follows which is illustrated with investigations of materials designed to be used for constructing individual methods of radiation protection. A successful choice of a model solution containing ^{90}Sr, ^{144}Ce, ^{134}Cs, and ^{106}Ru (the most diffusive elements among fission products) allowed the evaluation of the shielding effectiveness of polymeric materials (in particular films) and rubber-based materials. Due to the nonreproducibility of results for ^{95}Zr + ^{95}Nb and ^{60}Co, these were excluded from the model solution. The gist of this section is the new principle for improving the shielding properties of polymeric materials by introducing a "sweated out" addition to such materials, formed on the surface of a self-renewing layer. The theme of Chap. 3 and 4 is the methods of physiologic-hygienic evaluation and the investigation of effective means for individual protection. The appropriateness of including these sections in the book is convincingly demonstrated by the author. As a rule, the use of such methods, especially insulated suits, will cause the accumulation or loss of heat in the body and, as a consequence, considerably physiological changes which lead to a reduction of physical and mental efficiency. Investigations done under the guidance of S. M. Gorodinskii allowed a determination of the optimal parameters of the microclimate in the space beneath the suit. In the last two sections the reader will find information on the design and investigation of insulated suits for use in remodeling and accident work in an environment of radioactive pollution, as well as information about systems for protecting respiratory organs during work with radioactive substances. Chapters 7 through 9 concentrate on information about working clothes, gloves, and working shoes. The book concludes with a section about shielding as a means of protection, appropriate tabular supplements, and a large list of references. A small subject index makes the book easy to use.

In every chapter the author describes the investigative methods and generalizes the results of the comprehensive and lengthy experimental work (as a rule, this information is presented in compact tables and histograms). Especially interesting is the section on protecting the respiratory organs, a problem which has caused many difficulties. During the 1930s the Hungarian scientist Brezina aphoristically formulated the essence of the problem: "A good respirator does not allow breathing, but a respirator which allows breathing affords poor protection." In 1955-1956, I. V. Petryanov, S. M. Gorodinskii, S. N. Shatskii and P. N. Basamov worked out a new scheme for a valveless respirator on the basis of calculations and experimental models. This served as the basis for the creation of the "Petal", in which a unique natural filter, FP, was used. Today this respirator is manufactured by the millions every year (p. 185). No less essential results are seen in other areas of individual protection. We see, for example, that the protective effectiveness is 99.998% for the pneumomask LIZ-Z in uranium mines. Unfortunately, in one of the most important places in the text there is an annoying misprint: In the final Table 23 (p. 163) the protective effectiveness of the complete set LG-5 is given as three orders of magnitude too small.

Basically the book is lively and interesting; most of the important sections are well illustrated. It shows the frontiers which Soviet scientists study in the area of radiation hygiene and physiology. However, several

*Third edition, revised and enlarged, Atomizdat, Moscow (1979), 296 pp., 3 rubles 40 kopecks.

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sections of the monograph are not free from a careless style and terminology: On pages 161 and 239 we see special medical terms without the necessary interpretation of their meanings; Chap. 10 is written in a less lively style than others.

In conclusion we will mention the emblem illustration on the jacket of the book. A knight in "armor" with a shield wards off the blow" from a beam of radiation. The illustration symbolizes the main idea of the monograph: Personnel of the nuclear power industry must be protected from danger. The success of the book is due largely to the author — S. M. Gorodinskii, recipient of the Lenin prize, Doctor of Medical Sciences.

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EFFECT OF γ RADIATION ON THE DETECTING PROPERTIES OF LAVSAN FILM

S. P. Tret'yakova and T. I. Mamonova

UDC 535.1.074

Since polyethylene terephthalate (Lavsan) can record the passage of heavy charged particles, it can be used as a detector in physical and cosmic experiments, and to produce nuclear filters [1]. After bombardment with ions of various kinds the detecting properties can be changed by electromagnetic radiation [2, 3]. We have investigated the effect of γ radiation on the properties of Lavsan film of domestic manufacture under various exposure conditions.

Films 10 and 50 μm in thickness were bombarded with a flux of 10^5-10^8 xenon and argon ions per cm^2 at the U-300 cyclotron of the JINR Laboratory of Nuclear Reactions. The energies of the xenon and argon ions were 0.75 and 1.5-7.6 MeV/nucleon, and the specific energy losses were 80 and 26.0-12.5 MeV$\cdot$cm^2/mg, respectively. Bombardment at angles of 30, 45, and 90° permitted a study not only of the etching rate of the polymer damaged by an ion, but also the shape of the channels of the particle tracks. Films were exposed to doses of $5 \cdot 10^3$-$2 \cdot 10^8$ rad of γ radiation from ^{137}Cs. Film samples which had been bombarded with ions, and samples which had not, were divided into three groups: The first and second groups were exposed to γ radiation in air and in a vacuum (10^{-3}-10^{-4} mm Hg); the third group was not exposed to γ radiation. The spectrometric characteristics were photographed, and the kinetics and selectivity of the etching of the film samples were studied.

A study of the spectral characteristics of Lavsan showed that the exposure of Lavsan film to γ radiation in air increased the absorption of light in the wavelength range from 3150 to 3400 Å, with the increase being appreciable for doses of more than 10^8 rad. The irradiation of film in a vacuum does not change the absorption spectrum. The participation of oxygen in polymer damage from γ radiation is confirmed by experiments on the etching of ion track channels. It was established that the etching rate along the diameter of a track channel of a xenon ion which entered the film at an angle of 90° was 1.3-1.4 times as large when irradiated with γ rays in air as when irradiated in a vacuum. Figure 1 shows microphotographs of tracks of xenon and argon ions which entered the film at an angle of 30° with the surface. It is clear that irradiation with γ rays is effective only in the presence of oxygen, and therefore the selectivity of the etching of ion tracks was studied for irradiation in air.

The results of the study of the etching rate along a track V_{tr}, the etching rate of unirradiated film V_{un}, and the selectivity of the etching process V_{tr}/V_{un} as functions of the γ dose are shown in Fig. 2. The electrolytic method [4] was used to determine the etching rate along a track by the time to etch through an ion track channel. It was established that the etching rate of an unirradiated film increases for doses of more than 10^8 rad, which is in good agreement with the spectral data. The etching rate of a polymer track is appreciably increased even for a dose of $5 \cdot 10^5$ rad, and therefore the selectivity of the process also increases. This indicates that a polymer damaged by ions is more sensitive to damage by γ rays. It is interesting to note that the increase in the etching rate of an ion track in the polymer stops at a definite dose which depends on the ion type and energy. This can probably be accounted for by the extent of the polymer damage in the region of the track. The maximum etching rate and the selectivity of the process occur at a γ dose of the order of $5 \cdot 10^6$ rad for xenon ions, and at $4 \cdot 10^7$ rad for argon ions. The energy of the xenon ions was 0.75, and that of the argon ions 5.6 MeV/nucleon.

Our investigation showed that the effect of γ radiation on increasing the etching rate of Lavsan tracks of argon ions with energies of 1.5 and 5.6 MeV/nucleon was greater for the higher energy ions. This is probably related to the energy distribution of secondary delta electrons which are produced in the passage of ions through Lavsan, and in some way interact with it. The region of the polymer through which these electrons passed undergoes additional damage from γ radiation, and the etching rate is greater than in a control sample.

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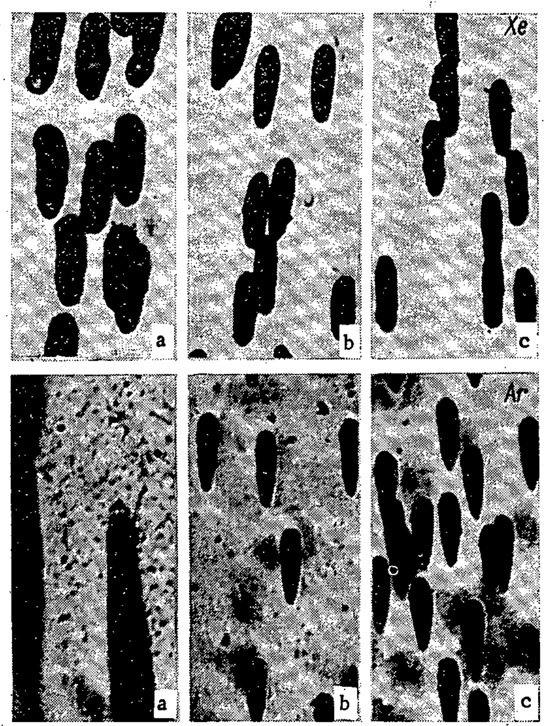

Fig. 1. Microphotographs of xenon and argon tracks in Lavsan films exposed to a γ dose of $6.7 \cdot 10^7$ rad a) in air; b) in vacuum; c) in control films. Films were etched in a 20% solution of NaOH at 20°C for 24 h (× 1000).

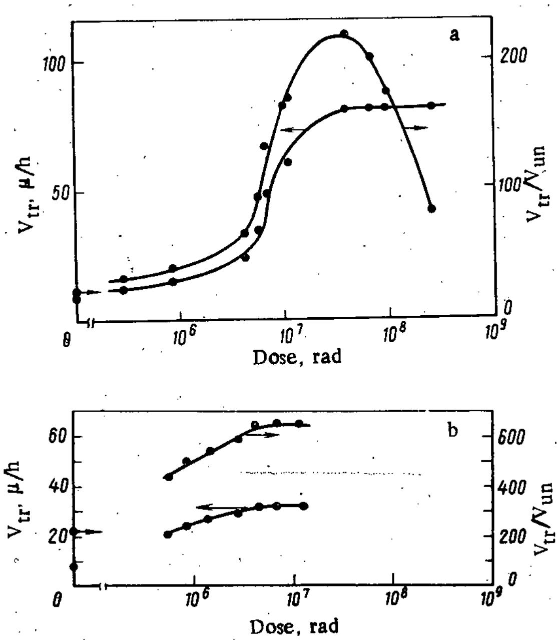

Fig. 2

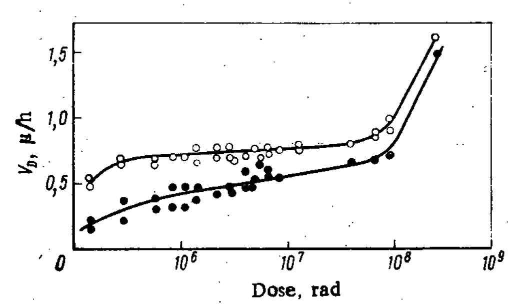

Fig. 3

Fig. 2. Rate and selectivity of the etching of tracks of a) argon and b) xenon ions in a 20% alkali solution as functions of the γ dose at a) 50°C; b) 30°C: ●) experiment.

Fig. 3. Etching rate of argon (●) and xenon (○) ion track channels in a 20% alkali solution at 50°C as a function of γ dose.

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Since the selectivity of the etching process affects the shape and size of a track channel, it is interesting to trace the variation of a channel diameters with γ dose for etching prolonged until breakthrough occurs. The channel diameters were measured by the "bubble" method [4], and the shape was observed with an optical microscope. Curves were obtained for the dependence of the etching rate of track channels along a diameter V_D on the γ dose in air (Fig. 3). The sharp increase in the etching rate for doses above 10^8 rad for both kinds of ions is due to the damage of the material. The variation of the selectivity of the etching process with dose is most noticeable for argon ion tracks (Fig. 2a), and is reflected in the behavior of the curve for the etching rate of channels (cf. Fig. 3). The shape of the ion tracks after prolonged etching of the film can be seen in Fig. 1. The increase in etching rate resulting from γ radiation is larger for argon than for xenon ions. In this experiment the argon ions had an energy of 7.6 and the xenon ions 0.75 MeV/nucleon.

The results presented show that exposure to a certain γ dose in air increases the etching rate of the polymer along an ion track and the selectivity of the process. This is related to the difference in sensitivity of the polymer to the action of γ radiation before and after ionization damage in the region of an ion track. The decrease in selectivity for doses of more than 10^8 rad is explained by the sharp increase in the etching rate of polymer which has not been subjected to ion bombardment. Irradiation in a vacuum does not affect the etching rate of the polymer or the selectivity of the process, and therefore to increase the etching rate of an ion track in the polymer γ radiation should be used only in the presence of oxygen.

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USE OF CALIFORNIUM NEUTRON SOURCES TO DETERMINE BASIC ELEMENT-SALT COMPOSITION OF SEAWATER UNDER NATURAL CONDITIONS

E. M. Filippov

UDC 551.464.621.039.8

Californium sources have a yield of up to 10^{10} neutrons/sec. They can be employed to study changes in the element-salt composition of seawater under natural conditions [1-3]. We shall consider the application, for this purpose, of the neutron γ-ray method (NGM) based on the (n, γ) reaction and the neutron activation method (NA).

In the case of the NGM for a point source and a detector separated by a distance l the equation for the counting rate is

$$N = \frac{Q\varepsilon s i \Sigma}{8\pi\Sigma_a (L_s^2 - L^2)} B(L_s, L) = Q_0 B(L_s, L) = Q_0 [B(L_s) - B(L)]. \tag{1}$$

Here Q is the neutron yield from the source; ε, detector efficiency; s, area of the detector; i, number of quanta formed during the capture of neutrons by chlorine; Σ and Σ_a, macroscopic cross sections for the (n, γ) reaction and neutron absorption in water, respectively; L_s and L, respectively, neutron moderation and diffusion lengths in seawater, and

$$B(L_s) = \frac{L_s}{l}\left\{ e^{-l/L_s}\left[\mathrm{Ei}\,[-l(\tau - L_s^{-1})] + \ln\frac{1+\tau L_s}{|1-\tau L_s|}\right] - e^{l/L_s}\,\mathrm{Ei}\,[-l(\tau + L_s^{-1})]\right\}, \tag{2}$$

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TABLE 1. Values of Functions $B(L_S, L)$ at Various Salinities of Seawater

l, cm	E_γ, MeV	Salt content in water, g/liter					
		0	17,5	35	0	17,5	35
		Point source and detector			Point source and detector of finite size		
10	0,5	0,2233	0,2440	0,2535	0,07440	0,08284	0,08706
	3,0	0,3232	0,3537	0,3696	0,1397	0,1548	0,1630
	6,1	0,3535	0,3871	0,4051	0,1608	0,1781	0,1874
30	0,5	0,007135	0,007361	0,007309	0,005035	0,005231	0,005182
	3,0	0,02199	0,02314	0,02351	0,01819	0,01916	0,01944
	6,1	0,02922	0,03093	0,03162	0,02456	0,02601	0,02656
50	0,5	0,0003350	0,0003434	0,0003319	0,0002869	0,0002923	0,0002811
	3,0	0,003164	0,003317	0,003332	0,002730	0,002842	0,002842
	6,1	0,005433	0,005749	0,005837	0,004639	0,004874	0,004927

where τ is the linear attenuation coefficient for rays. The function $B(L)$ has much the same form when L_S is replaced by L.

If the detector and source are enclosed in casings of finite sizes with radii r_1 and r_2 and distances h_1 and h_2 from their centers to the ends of the casings aligned on one axis, then Eq. (1) can be recast as

$$N=Q_0\,[B(L_s)+\Delta B(L_s)-B(L)-\Delta B(L)]=Q_0B(L_s,L). \tag{3}$$

Here the functions $\Delta B(L_S)$ and $\Delta B(L)$ take account of the interaction in external space, between the ends of the source and detector casings, and $B(L_S)$ and $B(L)$ take account of the interaction in the inner space. Let us calculate these functions. If the radii of the source and detector casings are equal, $r_1=r_2=r_0$, then

$$B(L_s)=-\int\limits_{-1}^{1}\varepsilon^{yl/L_s}\,\mathrm{Ei}\left[-\frac{l}{2}(\tau+L_s^{-1})(u+y)\right]dy=-\int\limits_{-1}^{1}f(L_s,\,u+y)\,dy; \tag{4}$$

$$\Delta B(L_s)=\int\limits_{\alpha_2}^{\alpha_1}f(L_s,\,u+y)\,dy-\int\limits_{\alpha_2}^{-1}f(L_s,\,1+y)\,dy, \tag{5}$$

where $u=\sqrt{1+(2r_0/l)^2/(1-y^2)}$, $\alpha_1=1-2h_1/l$, and $\alpha_2=-1+2h_2/l$, $f(L_s,\,1+y)$ is the same as $f(L_S,\,u+y)$ when u is replaced by unity.

In the case $r_4\neq r_2$, the integrals under consideration become

$$B(L_s)=-\int\limits_{0}^{1}f(L_s,\,u_1+y)\,dy+\int\limits_{0}^{-1}f(L_s,\,u_2+y)\,dy; \tag{6}$$

$$\Delta B(L_s)=\int\limits_{0}^{\alpha_1}f(L_s,\,u_1+y)\,dy+\int\limits_{0}^{\alpha_1}f(L_s,\,1+y)\,dy-\int\limits_{0}^{\alpha_2}f(L_s,\,u_2+y)\,dy+\int\limits_{0}^{\alpha_2}f(L_s,\,1+y)\,dy. \tag{7}$$

The functions u_1 and u_2 are the same as u when r_0 is replaced by r_1 and r, respectively.

Californium sources, as is known, are extremely small in size and, therefore, in our calculations can be assumed to be point sources. In this case $r_1=h_1=0$ and, therefore, $u_1=\alpha_1=1$. Equations (6) and (7) simplify here to

$$B(L_s)=-\int\limits_{0}^{1}f(L_s,\,1+y)\,dy+\int\limits_{0}^{-1}f(L_s,\,u_2+y)\,dy; \tag{8}$$

$$\Delta B(L_s)=-\int\limits_{0}^{\alpha_2}f(L_s,\,u_2+y)\,dy+\int\limits_{0}^{\alpha_2}f(L_s,\,1+y)\,dy. \tag{9}$$

All the integrals must be calculated numerically. In all cases the functions $B(L)$ and $\Delta B(L)$ are analogous to the functions $B(L_S)$ and $\Delta B(L_S)$ when L_S is replaced by L.

Calculations of these functions for a point source and a detector of Eq. (1) and for a point source and a "Limon" NaI(Tl) detector of standard size (15×10 cm, $d=2$, $r_2=15$ cm, $h_2=5$ cm) by Eqs. (8) and (9) are given in Table 1. A salinity of 17.5 g/liter corresponds to the waters of the Black Sea and 35 g/liter, to ocean water. It is seen from Table 1 that with an increase in l the values of the functions $B(L_S, L)$ decrease sharply whereas with an increase in the γ-ray energy and the salinity of the water they increase slightly. Taking the detector

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TABLE 2. Expected Errors of Determination of Chemical Elements of Salt Composition of Black Seawater

Element	C, g/liter	T	E_γ, MeV	i	Σ_i, cm^{-1}	δC, %
Sodium	5,25	15 h	2,76	1	$7{,}42 \cdot 10^{-5}$	0,805
			1,38	1		
Bromine	0,0325	17,55 min	0,62	0,13	$1{,}07 \cdot 10^{-6}$	8
Magnes.	0,675	10 "	1,013	0,3	$4{,}8 \cdot 10^{-8}$	23
Calcium	0,2	8,75 "	4,68	0,03	$5{,}2 \cdot 10^{-9}$	23,5
			4,05	0,08		
			3,1	0,89		

geometry into account results a decrease in the values of the functions $B(L_S, L)$. The most pronounced change in the geometry of measurements turns out to occur for small values of l. This is due to the fact that the gap between the casings of the source and the detector when their concrete dimensions are introduced is small in comparison with l.

With NGM, in the case of neutron capture by chlorine γ rays with an energy of 6.11 MeV ($i = 0.1578$) are formed with the highest probability. Therefore, all NGM calculations were carried out for that radiation. In interaction with the material of the detector this radiation splits because one of two photons with an energy of 0.51 MeV is carried off. As a result, there is a triad of γ-rays quanta with energies of 6.11, 5.60, and 5.09 MeV. In measurements at the optimal distance of 20 cm for 1 sec we recorded 1403, 2022, and 2265 counts/sec, respectively, in these photopeaks (with $Q = 10^8$ neutrons/sec). From the total counting rate the chlorine content in Black Seawater (9.5 g/liter) is determined with a relative error of $\delta C = 1.325\%$ and with a measuring time of 2 sec, with a relative error of 0.94%.

The equation for the counting rate of γ rays from radionuclides formed during NA will be of a form analogous to Eqs. (1) and (3) with Q_0 replaced by $Q_1 = Q\varepsilon_s i \Sigma_i f(t)/4\pi\Sigma_a (L_s^2 - L^2)$. Here, $f(t) = \frac{T}{0{,}693}\left[1 - \exp\left(-\frac{0{,}693}{T} t_i\right)\right] \times \left[\exp\left(-\frac{0.693}{T} t_p\right) - \exp\left(-\frac{0.693}{T} t\right)\right]$ is a time factor in which t_i, t_p, and t are, respectively, the irradiation time, the pause, and the decay time of the induced activity after the irradiation of the medium under study has ended, and T is the half-life.

The chemical elements which are best activated by the NA technique are those given in Table 2 for a source with a yield of 10^8 neutrons/sec. The functions $B(L_S, L)$ for γ rays from the radionuclides formed can be easily estimated from the data of Table 1. In calculations of the counting rate by the NA technique, we have $t_p = 0$, $t_i = t = 30$ min. When the activity of the source is increased by an order of magnitude there errors decrease by roughly a factor of three. In these analyses potassium can be determined from the natural radioactivity of ^{40}K ($E_\gamma = 1.46$ MeV). With a measuring time of 1 h the potassium content in Black Seawater (0.19 g/l) is determined with a relative error of 1.74% (data of the experiment).

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EFFICIENCY OF NUCLEAR-FUEL UTILIZATION BY MOLTEN-SALT CONVERTER REACTORS

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UDC 621.039.542.4

The molten-salt reactor (MSR) is usually considered as one possible system capable of ensuring expanded production of fissionable materials in the uranium–thorium fuel cycle [1, 2] if continuous extraction of fission products is organized right at the reactor. However, in the first stage of development such reactors can apparently operate without continuous fuel reprocessing [2]; it therefore makes sense to estimate the efficiency of a MSR operating without continuous extraction of fission products. In this case the reactor operates in the breeder mode (with a conversion ratio of less than one) and should be continuously supplied with make-up fissionable fuel. Not only a uranium–thorium fuel cycle but also a uranium–plutonium cycle can be realized in a molten salt breeder reactor.

The maximum possible concentration of fission products in the fuel salt is limited by their solubility which reaches 5% at the operating temperature (700°C). Such a concentration in a reactor of the MSBR-1000 [1] type in the absence of fuel reprocessing is attained after 12 years of operation. The molten-salt breeder reactor can, therefore, operate with one charge of salt-carrier for 10-12 years with a continuous supply of make-up enriched uranium to compensate for the burn-up and poisoning of the reactor with fission products. At the end of a run the fuel salt is replaced by fresh salt (open cycle) and the fissionable material contained in the spent salt can be extracted from it by the fluorination method, whose technological feasibility was demonstrated on the 8-MW experimental reactor MSRE [4], and returned to the fuel cycle (closed cycle). The fuel component of the cost of electrical energy generated by an MSR will be several times lower than for present-day light-water reactors (LWR) [5]. This estimate, however, is associated with the uncertainty of the prices of the fuel itself as well as other components of the fuel composition. At the present time, therefore, it is more objective to estimate the efficiency of utilization of natural resources of nuclear fuel; quantitatively, this can be characterized by the utilization factor for natural fuel, defined as

$$\alpha(t) = Q(t)/G(t), \tag{1}$$

where Q(t) is the quantity of electrical energy produced by the energy system in the time t and G(t) is the integrated consumption of natural fuel (natural uranium or thorium) during that time.

With the continuous growth of nuclear power the fuel consumption G(t) can be written as [6]

$$G(t) = W(0)\int_0^t y(t')\,dt' + \int_0^t dt_1 \int_0^{t_1} y(t_2 - t_1)\frac{dW(t_2)}{dt_2}\,dt_2, \tag{2}$$

where W(t) is the total nuclear energy at the time t and y(t) is the rate of natural-fuel consumption per unit nuclear power. If the duration τ_1 of the reactor run is taken to be a fraction of the reactor lifetime τ_c, i.e., $\tau_c = n\tau_1$, where n is an integer, then the function y(t) is of the form

$$y(t) = \begin{cases} y_0\delta(t) + y_1(t) - y_2\delta(t-\tau_1), & 0 \leqslant t \leqslant \tau_1; \\ y\,[t - Ent\,(t/\tau_1)], & \tau_1 \leqslant t, \end{cases} \tag{3}$$

where Ent(x) is the integral part of x taken with a deficiency, y_0 is the quantity of natural fuel used for the initial loading of the reactor, $y_1(t)$ is the consumption with one fuel for a continuous make-up supply to the reactor, and y_2 is the quantity of natural uranium saved by reuse of fissionable materials extracted from the spent fuel composition at the end of the run.

With an exponential growth of the power of the energy system with a fuel-doubling time $T_2 = \ln 2/\omega$, when $W(t) = W_0 \exp(\omega t)$,

$$Q(t) = \varphi \int_0^t W(t')\,dt' = \frac{\varphi}{\omega} W_0 e^{\omega t} - 1, \tag{4}$$

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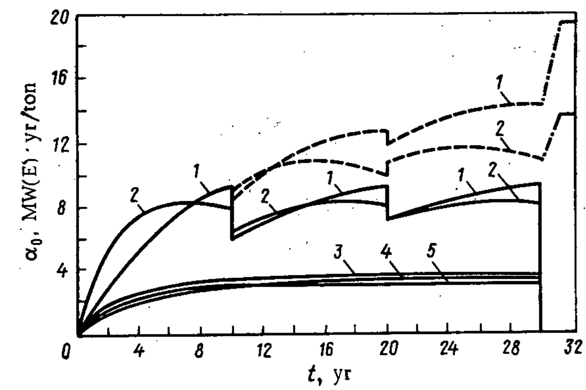

Fig. 1

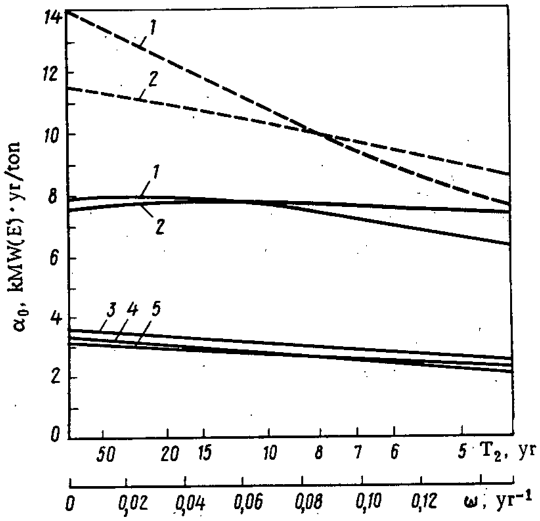

Fig. 2

Fig. 1. Natural-fuel utilization factor α_0 vs operating time t of reactor at constant power of the energy system and load factor $\varphi = 0.8$; 1) MSR (U−Th); 2) MSR (U−Pu); 3) LWR (UO_2 +3.2% ^{235}U) [1]; 4) RBMK-1000 (UO_2 +1.8% ^{235}U) [2]; 5) VVÉR-1000 (UO_2 +4.4% ^{235}U [3]; ——) and – – –) open and closed cycles, respectively; – · – · –) coefficient α reaching asymptotic value as $t \to \infty$.

Fig. 2. Natural-fuel utilization factor α vs growth rates $\omega = \ln 2/T_2$ of power of energy system with development of nuclear power engineering over 30 years: 1) MSR (U−Th); 2) MSR (U−Pu); 3) LWR (UO_2 +3.2% ^{235}U [6]; 4) RBMK-1000 (UO_2 +1.8% ^{235}U) [7]; 5) VVÉR-1000 (UO_2 +4.4% ^{235}U) [8]; ——) and – – –) open and closed cycles, respectively.

where φ is the reactor load factor, and the expression for $\alpha(t, \omega)$ becomes

$$\alpha(t, \omega) = [\varphi/g(\omega, t)]\,[1 - \exp(-\omega t)], \tag{5}$$

where

$$g(\omega, t) = \omega \int_0^t y(\tau)\, e^{-\omega\tau}\, d\tau. \tag{6}$$

In the calculations of the coefficient α for an energy system based on a MSR we considered reactors with a reactor core of 50 m^3, surrounded by a graphite reflector 0.8 m thick. The core of the MSR was constructed of graphite rods between which flows a stream of molten salt with the molar composition (72−x)% ^7LiF +16% BeF_2 +12% $^{238}UF_4$ +x% $^{235}UF_4$ for the U−Pu fuel cycle or (72−x)% ^7LiF +16% BeF_2 +12% ThF_4 +x% $^{235}UF_4$ for the U−Th fuel cycle, where x is the initial critical concentration of ^{235}U. The composition of the reactor core and the breeding region of the U−Pu reactor was optimized for maximum α. As a result of these calculations the choice was a reactor without a specially separated breeding region and with a specific charge of 1.12 kg ^{235}U/MW (e) and a power of 1000 MW (e), in whose core 7% of the volume is occupied by molten salt. The uranium−thorium MSR was considered with a composition much like that of the MSBR-1000 (the fuel salt occupies 13% of the volume of the core and 37% of the volume of the breeding region). The data for the calculation of α for the uranium−thorium MSR were taken from [5].

Figure 1 gives the values of $\alpha_0(t) = \alpha(t, \omega = 0)$ for MSR with open and closed fuel cycles as well as for present-day reactors (LWR). For the MSR the coefficient α_0 in the case of the open fuel cycle is 2.5 times higher with the U−Pu cycle and 2.9 times higher with the U−Th cycle than for uranium−plutonium LWR. In the case of the closed cycle these figures rise to 4 and 6 times, respectively.

A comparison of $\alpha(t, \omega)$ for energy systems under development on the basis of reactors of various types is made in Fig. 2. It is seen that the efficiency of utilization of fuel in MSR decreases with the doubling period of the power of the energy system. Thus, MSR can use more expensive fuel without surpassing the cost of energy from LWR.

Indeed, suppose that the competitiveness of a nuclear energy system based on reactors of the i-th type is ensured at a natural-fuel cost no greater than Z_i. The total quantity of electrical energy which can be produced by reactors of the i-th type is given by Eq. (1)

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$$Q_i = \alpha_i G(Z_i), \tag{7}$$

where $G(Z_i)$ is the quantity of natural fuel procured at a cost not exceeding Z_i.

According to estimates [1], the capital component of the electrical energy cost for MSR is roughly the same as for LWR. Thus, the competitiveness of MSR in comparison with LWR will be ensured if $C_2 \le C_1$, where C_1 and C_2 are the fuel components of the electrical energy cost for LWR and MSR, respectively.

The quantity C_i is related by Z_i by

$$C_i = Z_i/(\alpha_i + C_{0i}), \tag{8}$$

where C_{0i} is the contribution to the fuel component of the cost of fabricating fuel elements or producing the fuel composition and reprocessing the fuel.

Thus we have

$$Z_2 \leqslant (\alpha_2/\alpha_1) Z_1 + \alpha_2 (C_{01} - C_{02}). \tag{9}$$

Since $C_{01} > C_{02}$ [4, 9], from Eq. (7) and (11) we get

$$Q_2/Q_1 \geqslant (\alpha_2/\alpha_1) \left[G\left(\frac{\alpha_2}{\alpha_1} Z_1\right) \Big/ G(Z_1) \right]. \tag{10}$$

Increasing the electrical production extends the period of competitive development of the energy system with

$$\Delta T \approx (T_2/\ln 2) \ln (Q_2/Q_1). \tag{11}$$

With $Z_1 = 22$ dollars/kg U_3O_8 and with the G(Z) relation given in [10], for an energy system with $T_2 = 10$ yr we have:

for open-cycle breeder MSR

$\alpha_2/\alpha_1 = 2.9$, $Q_2/Q_1 \ge 15$, $\Delta T \approx 39$ yr for U–Pu cycle;

$\alpha_2/\alpha_1 = 2.8$, $Q_2/Q_1 \ge 14$, $\Delta T \approx 38$ yr for U–Th cycle.

for closed-cycle breeder MSR

$\alpha_2/\alpha_1 = 3.8$, $Q_2/Q_1 \ge 27$, $\Delta T \approx 48$ yr for U–Pu cycle;

$\alpha_2/\alpha_1 = 4.9$, $Q_2/Q_1 \ge 30$, $\Delta T \approx 49$ yr for U–Th cycle.

Thus, the use of MSR makes it possible to extend the period of competitive development of nuclear power in comparison with LWR reactors (with open cycle) by roughly 40-50 years.

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GAS-CHROMATOGRAPHIC EXAMINATION OF THE ACCUMULATION OF ^3H, ^{85}Kr, AND ^{133}Xe IN THE PROTECTIVE GAS, SODIUM COOLANT, AND CONSTRUCTIONAL MATERIALS OF THE BR-10

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UDC 621.039.526

Increased importance has been attached to the elimination of ^3H and ^{85}Kr from reactor gas releases in recent years on account of the expanding nuclear power network. Further, tritium is produced by triple fission in nuclear fuel and may influence the properties of fuel-rod sheaths [3]. It is therefore necessary to examine the formation and migration of gaseous fission products in nuclear fuel, fuel-rod sheaths, coolants, and protective gases in nuclear power stations, as well as the effects of fision products on the parameters of reactor materials.

Little has been published on the accumulation and behavior of tritium in fissile assemblies and fuel-rod sheaths. About 80% of the tritium produced in nuclear fuel enters the fuel-rod sheath (stainless steel) and diffuses into the coolant [4]. Within a day of operation at full power, the rate of leakage of tritium from the fuel rods is equal to the rate of formation. Further, less than 1% of the tritium is retained in the fuel [5] in a fast reactor when there is 5-8% burnup. The rates of formation of tritium from all sources may be about 30,000 Ci/yr in the EBR-II [6, 7] and LMFBR [8] reactors, and about 5% of this amount escapes into the atmosphere or enters the condenser water. Existing methods of calculating the yields of tritium from nuclear fuel are inadequate, as are means of calculating the transfer to the coolant, the construction materials, and the protective gas.

Sample Preparation and Examination. The gas samples were taken directly from the gas system in the BR-10 [9], and also from the sodium coolant in the first and second loops by means of a tubular sampler [10], while specimens of fuel-rod sheaths were cut with a lathe.

The ^3H, ^{85}Kr, and ^{133}Xe were released as gases from the sodium and steel specimens: In the first case by vacuum distillation of the sodium at 550°C [11] and in the second by melting the steel in a current of inert gas [12]. Before melting, the sheaths were washed in nitric acid and then in water before being dried for about 30 min at about 100°C.

The levels of ^3H, ^{85}Kr, and ^{133}Xe were examined with a radiochromatograph with a flow-type proportional counter [13]. The identification was further confirmed by calibrating the radiochromatograph on standards of tritium, krypton and xenon, as well as by measuring the γ-ray spectra of ^{85}Kr and ^{133}Xe with a Ge(Li) detector. The lower limits to the detection of ^3H, ^{85}Kr, and ^{133}Xe with the radiochromatograph were $2 \cdot 10^{-10}$ Ci/cm^3 at the ±8% level.

Levels of ^3H, ^{85}Kr, and ^{133}Xe in BR-10 Gas. The protective gas taken from the EMN pump tanks was examined only for ^3H, ^{85}Kr, and ^{133}Xe. Other gaseous radionuclides were not examined. The gas samples were taken from the reactor over a period of 12 months. Table 1 gives the mean levels of the gaseous fission products found in 3-7 parallel measurements. Gas specimens 5-7 were taken from the reactor when the cold trap for sodium oxides in the sodium loop [14] had been working continuously.

The level of ^3H fell as the run time increased, which is evidently related to the operation of the oxide cold trap, while the levels of ^{85}Kr and ^{133}Xe increased with the reactor power, evidently due to leakage in some of the fuel-rod sheaths and possible contamination of the first loop by nuclear fuel.

Levels of ^3H, ^{85}Kr, and ^{133}Xe in the Sodium Coolant in the BR-10. The measurements on the gaseous fission products in the sodium when the reactor had operated from the start of a run at various power levels

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TABLE 1. Levels of ^{3}H, ^{85}Kr, and ^{133}Xe in Protective Gas, Ci/cm^3

Specimen	Sampling point	Reactor production, million kWh	^{3}H	^{85}Kr	^{133}Xe
1	EMN-1	21,3	$6{,}6 \cdot 10^{-9}$	$< 2{,}0 \cdot 10^{-10}$	$8{,}8 \cdot 10^{-9}$
2	EMN-1	23,9	$2{,}1 \cdot 10^{-8}$	$< 2{,}0 \cdot 10^{-10}$	$2{,}7 \cdot 10^{-8}$
3	EMN-2	23,9	$9{,}0 \cdot 10^{-10}$	$3{,}0 \cdot 10^{-10}$	$3{,}5 \cdot 10^{-7}$
4	EMN-1	29,1	$2{,}5 \cdot 10^{-10}$	$2{,}0 \cdot 10^{-10}$	$6{,}0 \cdot 10^{-8}$
5	EMN-1	43,5	$< 2{,}0 \cdot 10^{-10}$	$1{,}4 \cdot 10^{-8}$	$1{,}6 \cdot 10^{-6}$
6	EMN-2	29,1	$< 2{,}0 \cdot 10^{-10}$	$1{,}8 \cdot 10^{-8}$	$8{,}0 \cdot 10^{-6}$
7	EMN-2	43,5	$2{,}0 \cdot 10^{-10}$	$2{,}9 \cdot 10^{-8}$	$3{,}7 \cdot 10^{-7}$

TABLE 2. Tritium Levels in the Sodium Coolant in Br-10 Loops

Specimen	Reactor production million kWh	Amount of tritium in core (cal.), Ci	Specific, Ci/liter		Total, Ci	
			first loop	second loop	first loop	second loop
1	21,3	23,0	$1{,}3 \cdot 10^{-4}$	—	0,268	—
2	23,9	26,0	$3{,}0 \cdot 10^{-4}$	$2{,}3 \cdot 10^{-5}$	0,62	0,124
3	29,1	31,6	$6{,}0 \cdot 10^{-4}$	$3{,}5 \cdot 10^{-5}$	1,24	0,189
4	43,5	47,1	$9{,}2 \cdot 10^{-6}$	—	0,019	—

TABLE 3. Levels of H, ^{3}H, and ^{85}Kr in Sheaths of BR-10 Fuel Rods

Rod No.	Specimen No.	H, 10^{-2} at %	^{3}H, 10^{-10} at %	^{85}Kr, 10^{-9} at.%	^{3}H, Ci/h	^{85}Kr, Ci/h 10^{-6}
279	1	6,1	—	6,1	—	6,4
	2	6,1	1,8	5,8	$9{,}5 \cdot 10^{-9}$	6,3
	3	5,5	3,1	5,3	$1{,}7 \cdot 10^{-8}$	5,8
277	4	3,3	1,5	3,4	$7{,}8 \cdot 10^{-9}$	3,7
	5	3,4	1,7	2,9	$9{,}0 \cdot 10^{-9}$	3,2
	6	5,0	1,8	4,1	$9{,}8 \cdot 10^{-9}$	4,4

and had produced about 21 million kWh (the modes of operation of the sodium loops corresponded to steady-state technology [9]). The oxide cold trop in the first loop [15] had been operating for about 3800 h (about 16% of the time spent by the reactor at full power). Table 2 collects the mean values for the tritium contents in sodium specimens taken from the first and second loops during a period of 13 months of continuous operation. Samples 1-3 were taken in the period when the oxide cold trap was not working.

Table 2 shows that the sodium coolant contained much tritium, with the levels varying in specimens from the first loop between $9.2 \cdot 10^{-6}$ and $1.3 \cdot 10^{-4}$ Ci/liter of sodium, the exact level being dependent on the length of the run and the time for which the cold trap had operated. The cold trap reduced the tritium levels in the sodium by a very substantial factor (specimen 4). This effect is due to the accumulation of sodium hydride and hydroxide dissolved in the sodium. The second loop also contained tritium at a level of about $3 \cdot 10^{-5}$ Ci/liter. The tritium entered the second loop from the first by diffusion through the wall of the steam generator, because it is known [16] that steel is permeable to hydrogen.

No ^{85}Kr or ^{133}Xe was observed in the sodium coolant. The inert gases are virtually insoluble in sodium [17], and ^{85}Kr and ^{133}Xe can be present in the sodium coolant only as free gases, which are transported mechanically by the sodium and which are not recorded in the method of sampling the latter [10].

Levels of ^{3}H and ^{85}Kr in BR-10 Fuel-Rod Sheaths. We took 7 rods with a burnup of 1.3% that contained monocarbide fuel which had sheaths of OKh18N9T steel. These rods had been exposed to a neutron fluence of $9.1 \cdot 10^{21}$ neutrons/cm^2 which gave a maximum energy production rate of about 464 W/cm^2, with the temperature of the middle layer of the sheath at the hottest point about 644°C. Table 3 gives the tritium levels in the sheaths for two rods, and levels for the other five rods were analogous. Specimens 1 and 4 were taken from the top, 2 and 5 from the middle, and 3 and 6 from the bottom.

It has previously been reported [18] that ^{85}Kr occurs in the steel sheaths of fuel rods from fast reactors; this is due to the recoil nuclei produced by fission-fragment bromine, from which ^{85}Kr is produced by radioactive transformation. We consider that additional studies are needed to determine how the krypton enters the steel sheaths.

Conclusions. The tritium level in the sodium coolant in the first loop on the BR-10 attained $6 \cdot 10^{-4}$ Ci/liter, but this fell to $9 \cdot 10^{-6}$ Ci/liter when the oxide trap was working. The second loop accumulates tritium up to $3.5 \cdot 10^{-5}$ Ci/liter with the cold trap switched out. The tritium level in the protective gas in the first loop is low (10^{-10}–10^{-8} Ci/cm^3). Fuel-rod sheaths contain ^{3}H and ^{85}Kr. No tritium was detected in the air vented to the stack in the BR-10.

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CRITICAL TEMPERATURE RISE IN A COOLANT IN AN ANNULAR CHANNEL

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UDC 624.07

Active zones (cassettes) containing fuel rods show bending of the latter in response to temperature gradients; these deviations in geometry and nonuniformities in heat production cause the rods to deflect within the gaps provided in the assembly and also relative to the supports. The effects are accentuated by the marked changes in flow rate and rod temperature, because the rods come into contact and tend to move as groups. In early studies [1-3] and in a later study [4] it has been emphasized that a self-consistent calculation is required for the speed of the coolant, the temperature, and the deformation.

Consider a cylindrical rod in an annular channel; in practice, such a rod is neither coaxial nor concentric. The variations in the gap along the length and in aximuth alter the coolant flow pattern, and this results in an aximuthal variation in the temperature. This in turn causes the rod to deflect, which results in secondary alterations in the gap, temperature pattern, velocity, etc. The process has been considered [5, 6] for a system of rods in a cylindrical reactor. It was shown that there is a certain critical temperature rise at which the rods always deflect into contact, no matter what the original eccentricity.

We now consider the analogous case for an annular rod. The shape of the elastic axis of the rod with hinged mounting of the ends is described by a cubic parabola for planar bending under conditions of linearly increasing temperature [6]:

$$f=\frac{\alpha_w \Delta \overline{T}_w L^2}{3d}\left[\frac{z}{L}\left(1-\frac{z^2}{L^2}\right)\right], \tag{1}$$

where f is the current value of the deflection; L and d, length and diameter of the fuel rod; α_W, coefficient of linear expansion for the sheath; and ΔT_W, mean temperature difference between the hot and cold parts of the perimeter taken over the length.

The maximum deflection is

$$f_{\max}=\alpha_w \Delta \overline{T}_w L^2/7.8d. \tag{2}$$

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The relative change in the gap width and in the hydraulic diameter can be expressed approximately if the width of the slot is substantially less than channel diameter and the deflection is small:

$$\frac{\delta}{\delta_0} \approx \frac{d_h}{d_{h0}} \approx 1 - \left[\frac{e}{\delta_0} + 2.6\,\frac{f_{max}}{\delta_0}\,\frac{z}{L}\left(1 - \frac{z^2}{L^2}\right)\right]\cos(\varphi - \varphi_0), \tag{3}$$

where e is the eccentricity of the fuel rod in the annular channel and φ_0 is the angular position of the plane of eccentricity and deflection.

We adopt the isobaric hypothesis, in which the pressure of the coolant is constant in a cross-section, and get [6]

$$\frac{W_z\delta}{W_{z0}\delta_0} \approx 1 - \frac{3}{2-n}\left(1 - \frac{d_h}{d_{h0}}\right) = 1 - \frac{3}{2-n}\left[\frac{e}{\delta_0} + 2.6\,\frac{f_{max}}{\delta_0}\,\frac{z}{L}\left(1 - \frac{z^2}{L^2}\right)\right]\cos(\varphi - \varphi_0), \tag{4}$$

where n is the parameter appearing in the law of friction $\lambda \sim Re^{-n}$; we average (4) with respect to φ for the hot and cold halves of the perimeter:

$$\frac{W_h\delta_h}{W_{z0}\delta_0} \approx 1 - \frac{3}{2-n}\,\frac{2}{\pi}\left[\frac{e}{\delta_0} + 2.6\,\frac{f_{max}}{\delta_0}\,\frac{z}{L}\left(1 - \frac{z^2}{L^2}\right)\right];$$
$$\frac{W_c\delta_c}{W_{z0}\delta_0} \approx 1 + \frac{3}{2-n}\,\frac{2}{\pi}\left[\frac{e}{\delta_0} + 2.6\,\frac{f_{max}}{\delta_0}\,\frac{z}{L}\left(1 - \frac{z^2}{L^2}\right)\right]. \tag{5}$$

The heat-transfer equations for the hot and cold parts of the ring channels are written in a form that applied for q_F = const:

$$C_p\gamma W_{z0}\delta_0\left\{1 - \frac{3}{2-n}\,\frac{2}{\pi}\left[\frac{e}{\delta_0} + 2{,}6\,\frac{f_{max}}{\delta_0}\left(\frac{z}{L} - \frac{z^3}{L^3}\right)\right]\right\}\frac{dT_{fh}}{dz} = q_F - \frac{(T_{fh} - T_{fc})}{(\pi R)^2}(\lambda_f\delta_0 + \lambda_w\delta_w);$$
$$C_p\gamma W_{z0}\delta_0\,\frac{W_c\delta_c}{W_{z0}\delta_0}\,\frac{dT_{fc}}{dz} = q_F + \frac{(T_{fh} - T_{fc})}{(\pi R)^2}(\lambda_f\delta_0 + \lambda_w\delta_w). \tag{6}$$

The latter terms in these equations characterize the heat leak from the hot part of the perimeter to the cold part. This effect is comparatively small for reasonably high coolant flow rates, so we solve (6) by successive approximation:

$$\frac{T_{fhc}}{\Delta T_{f0}} = \frac{z}{L} \pm \frac{3}{2-n}\,\frac{2}{\pi}\left[\frac{e}{\delta_0}\,\frac{z}{L} + 2{,}6\,\frac{f_{max}}{\delta_0} \times \left(\frac{z^2}{2L^2} - \frac{z^4}{4L^4}\right)\right] - \frac{(1 + \lambda_w\delta_w/\lambda_f\delta_0)}{Pe_0}\,\frac{2\delta_0 L}{(\pi R)^2}\,\frac{3}{2-n}\,\frac{4}{\pi} \times \left[\frac{e}{\delta_0}\,\frac{z^2}{2L^2} + 2.6\,\frac{f_{max}}{\delta_0}\,(z^3/6L^3 - z^5/20L^5)\right], \tag{7}$$

where $\Delta T_{f0} = q_F L / C_p\gamma W_{z0}\delta_0$ is the mean temperature rise in the coolant and $Pe_0 = C_p W_{z0} 2\delta_0/\lambda_f$ is the Péclet number.

The wall temperatures for the hot and cold parts of the perimeter are put as

$$\frac{\Delta T_{wh.c}}{\Delta T_{f0}} = \frac{T_{fh.c}}{\Delta T_{f0}} + \frac{\Delta T_{\alpha 0}}{\Delta T_{f0}}\,\frac{\alpha_0}{\alpha_{h.c}}, \tag{8}$$

where $\Delta T_{\alpha 0}$ is the mean temperature difference between the wall and the fluid. We assume that the heat transfer is stable and average (4) over the half-perimeters to get from $Nu \sim Pe^p$ that

$$\frac{\alpha_{h\,c}}{\alpha_0} \approx 1 \pm \frac{2}{\pi}\left[\frac{e}{\delta_0} + 2.6\,\frac{f_{max}}{\delta_0}\left(\frac{z}{L} - \frac{z^3}{L^3}\right)\right]\left(1 - \frac{3p}{2-n}\right). \tag{9}$$

The mean temperature difference between the hot and cold half-perimeters is given by

$$\frac{\Delta\bar{T}_w}{\Delta T_{f0}} = \frac{6}{\pi(2-n)}\left[1 - \frac{4}{3}\,\frac{\delta_0 L}{(\pi R)^2}\,\frac{1 + \lambda_w\delta_w/\lambda_f\delta_0}{Pe_0} + \frac{2}{3}(3p - 2 + n)\frac{\Delta T_{\alpha 0}}{\Delta T_{f0}}\right]\left(\frac{e}{\delta_0} + 0.61\,\frac{f_{max}}{\delta_0}\right). \tag{10}$$

System (2)-(10) provides the solution, since the geometrical parameters (δ_0, L, R, δ_w, e), the thermophysical parameters (λ_w, λ_f, α_w), and the working parameters (n, Pe_0, Nu_0, p) may be specified to define the sagitta of the deflection f_{max}/δ_0 and the mean aximuthal temperature nonuniformity $\Delta\bar{T}_w/\Delta T_{f0}$ in relation to the output from the fuel rod, viz., the temperature rise ΔT_{f0}. We eliminate $\Delta\bar{T}_w/\Delta T_f^0$ from (2)-(10) to get

$$\frac{f_{max}}{\delta_0} = \frac{e}{\delta_0}\Big/\frac{\pi(2-n)}{6}\left[\frac{15{,}6\delta_0 R}{\alpha_w L^2 \Delta T_{f0}}\right] \times \left[1 + \frac{4}{3}\,\frac{\delta_0 L}{(\pi R)^2}\,\frac{1 + \lambda_w\delta_w/\lambda_f\delta_0}{Pe_0} - \frac{2}{3}(3p - 2 + n)\frac{\delta_0}{L}\,\frac{Pe_0}{Nu_0}\right] - 0.61. \tag{11}$$

This shows that the sagitta is proportional to the initial eccentricity and increases linearly with ΔT_{f0} at first but then hyperbolically. The deflection can become arbitrally large for the critical difference $\Delta T_{f0\,cr}$

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no matter what the initial eccentricity, since the denominator in (11) tends to zero. At this critical temperature rise or above, a fuel rod in an annular channel always bends to contact with the outer tube or with the spacing fins:

$$\Delta T_{0\,\mathrm{cr}}=\frac{13{,}4\,(2-n)\,\delta_0 R}{\alpha_w L^2}\left[1+\frac{4}{3}\,\frac{\delta_0 L}{(\pi R)^2}\,\frac{1+\lambda_w\delta_w/\lambda_f\delta_0}{\mathrm{Pe}_0}-\frac{2}{3}\,(3p-2+n)\,\frac{\delta_0}{L}\,\frac{\mathrm{Pe}_0}{\mathrm{Nu}_0}\right]. \quad (12)$$

This critical temperature rise is dependent primarily on the geometrical parameters of the fuel rods, since higher stability occurs in short rods of large diameter in wide gaps. If $\delta_0 = 2$ mm, R = 10 mm, L = 1 m, $\alpha_w = 14 \cdot 10^{-6}$ deg^{-1}, δ_w = 2mm, λ_w kcal/m · h · deg, λ_f = 20 kcal/m · h · deg, Pr ≪ 1, p = 0.8, n = 0.25 and Pe_0 = 10-1000 we get $\Delta T_{f_0\,\mathrm{cr}}$ = 20-30°C.

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MEASUREMENT OF THE TOTAL NEUTRON CROSS SECTION OF ^{145}Nd

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UDC 621.039.556

We have measured the total neutron cross section of ^{145}Nd in the energy range from 0.02 to 350 eV. Neodymium is one of the most important fission products; it has a relatively large resonance integral (I_γ = 240 b) and thermal cross section (σ_γ = 42 b [1]); it has a high yield in thermal and fast neutron fission (3.8 and 3.5% respectively). Therefore, the values of the resonance parameters and the energy dependence of the total cross section are important for the design and description of processes in thermal and fast reactors [2].

Before our measurements there was information on the resonance parameters of ^{145}Nd levels in the energy range from 4.3 to 4600 eV [3-6] obtained at linear accelerators. The only values reported for the neutron cross section below 4.3 eV were obtained [1] on a neutron spectrometer in the range from 0.02 to 9 eV. The values of the 2200 m/sec absorption cross section obtained by activation methods vary from 37 to 52 b. A value of 240 ± 35 b for the resonance capture integral was determined from activation measurements [7].

We have measured the neutron cross section by the time-of-flight method on the 91.7 m flight path of the neutron spectrometer at the SM-2 reactor [8]. The best resolution of the spectrometer was 70 nsec/m. The neutrons were recorded by an array of 120 SNM-17 helium counters and an AI-4096 time analyzer.

The transmission measurements were made with a sample of Nd_2O_3 (Table 1). The sample also contained small admixtures of the following elements: Er < 0.03%; Sm < 0.05%; Pr < 0.3%; Ce < 0.1%; Eu < 0.03%. The transmission of the sample was studied at energies up to 350 eV. In the thermal energy range from 0.02 to 1 eV the total cross section of ^{145}Nd was measured with a resolution of 1500 nsec/m by using two detectors placed at different flight distances to take account of small-angle scattering. To eliminate possible systematic errors in the determination of the total neutron cross section in the thermal energy region additional measurements were made of the transmission of gold leaf $8 \cdot 10^{-3}$ atoms/b in thickness.

Results and Discussion. In the energy range up to 350 eV 21 ^{145}Nd levels were identified, and resonance parameters for 19 of them were determined. For levels up to 103 eV the resonance parameters were calculated by the "shape" method, and for levels above 103 eV by the "area" method. Table 2 lists the values of the ^{145}Nd resonance parameters obtained from our measurements, together with results from [1].

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TABLE 1. Composition of Neodymium Sample

Character-istics of sample	^{142}Nd	^{143}Nd	^{144}Nd	^{145}Nd	^{146}Nd	^{148}Nd	^{150}Nd
Isotopic composition, %	0,60	0,50	2,99	84,80	10,72	0,28	0,11
Thickness, atoms/b	$6{,}3\times10^{-5}$	$6{,}2\times10^{-5}$	$3{,}11\times10^{-4}$	$8{,}76\times10^{-3}$	$1{,}1\times10^{-3}$	$2{,}8\times10^{-5}$	$1{,}1\times10^{-5}$

TABLE 2. Parameters of ^{145}Nd Neutron Levels

E_0, eV	Γ_γ, MeV	$2g\Gamma_n^0$, MeV	$2g\Gamma_n^0$, MeV[1]
−2,5	(60)	0,95	—
4,35±0,02	57±4	0,54±0,03	0,57±0,04
18,9±0,1	(60)	0,005±0,001	—
42,5±0,3	65±8	52,2±3,0	47,6±2,3
85,6±0,6	58±10	1,52±0,15	1,33±0,11
95,9±1,1	(60)	0,47±0,03	0,52±0,02
102,6±1,2	68±15	1,11±1,0	11,3±0,5
103,2±1,2	(60)	4,1±0,3	3,83±0,12
146,6±1,4	—	1,4±0,2	1,73±0,12
151,9±1,4	—	1,31±0,06	1,20±0,04
169,8±1,5	—	0,20±0,02	0,23±0,02
188,8±1,7	—	2,25±0,18	2,4±0,2
232,5±2,0	—	0,24±0,03	0,43±0,03
241,5±2,2	—	4,8±0,3	4,5±0,2
247,0±2,2	—	0,3	0,24±0,03
259,2±2,3	—	5,3±0,6	6,2±0,3
274,0±2,5	—	7,4±0,7	8,9±0,9
305±3	—	—	3,3±0,7
310±3	—	20±1	17,2±0,9
320±4	—	—	0,67±0,06
342±4	—	3,0±0,2	3,1±0,2

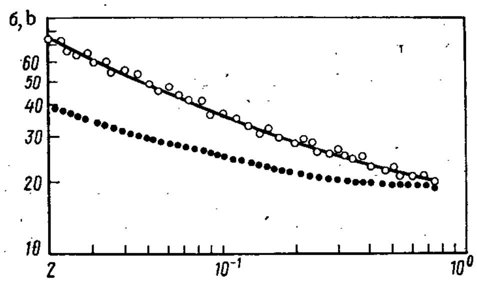

Fig. 1. Total neutron cross section in the thermal region: o) experiment; • • •) calculated from positive levels; ——) calculated by taking account of negative level.

The parameters $2g\Gamma_n^0$ at resonance energies $E_0 = 305$ and 320 eV are not listed because of inadequate resolution of the spectrometer. We found a weak level at $E_0 = 18.9$ eV which was not reported in [3, 5, 6] because of the small thickness of the samples ($\sim 10^{-4}$ atoms/b).

The measured values of the total neutron cross section in the thermal region are shown in Fig. 1. Previous measurements of the total neutron cross section of gold showed no systematic deviations, and within the limits of error of $\sim 2\%$ the measured values agreed with the precision measurements of $\sigma_t(E)$ reported in [9]. The effects of paramagnetic small-angle scattering on Nd_2O_3 powder were taken into account in the ^{145}Nd measurements. The magnitude and energy dependence of paramagnetic scattering in the 0.02-1 eV range were based on the results in [10]. This correction introduced a basic error into the measured values of $\sigma_t(E)$. Figure 1 shows that the ^{145}Nd total neutron cross section calculated from the resonance parameters of positive levels (Table 2) does not satisfactorily describe the experimental value of the total cross section. Fitting the calculated cross section to the experimental by taking account of a negative level gave the following parameters: $E_0 = -2.5$ eV, $2g\Gamma_n^0 = 0.95$ MeV with the assumed value $\Gamma_\gamma = 60$ MeV. These results differ from those in [10], where the following parameters were given: $E_0 = -20$ eV, $2g\Gamma_n^0 = 140$ MeV ($\Gamma_\gamma = 70$ MeV). In [11] the authors recommend a negative level at $E_0 = -6$ eV.

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The total neutron cross section of ^{145}Nd at $v_0 = 2200$ m/sec is 64 ± 4 b. Based on the value $\sigma_s = 17 \pm 2$ b obtained from an analysis of the interresonance region of the neodymium sample, the absorption cross section of ^{145}Nd is $\sigma_a = 47 \pm 3$ b. It should be noted that the negative level contributes 85% to the absorption cross section at $v_0 = 2200$ m/sec.

From the resonance parameters obtained the resonance capture integral was calculated to be 245 ± 30 b. The 6 b contribution of the negative level was found from the relation given in [11].

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WIDE-RANGE FISSION CHAMBER FOR CONTROL AND SAFETY SYSTEMS OF NUCLEAR REACTORS

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UDC 621.387:539.1.074

During the operation of nuclear reactors it is necessary to conduct steady monitoring roughly from 10^{-10} to 100% of the nominal power. As a rule, such monitoring is done by measuring the neutron flux density with neutron ionization chambers. The most appropriate detector is an ionization fission chamber with a radiator containing ^{235}U; this chamber is used successively in modes of recording individual pulses and measuring the mean ionization current. In order to ensure highly reliable monitoring it is important that the boundaries of the two operating modes of the chamber overlap within the limits of approximately of one-tenth of the order of magnitude of the change in the neutron flux. The commercially available fission chambers of the type of KNT-31, KNT-54-1, and KNT-15 [1] do not meet this requirement since the lower bound of the current mode of these chambers is determined by the intrinsic γ-ray background of the radiator, which reaches $1 \cdot 10^{-8}$ A. The upper bound of the pulse mode is limited by the speed of the chamber itself as well as of the electronic equipment used and in a number of cases, also by the presence of spurious signals due to the superposition of a large number of individual pulses induced by the γ rays of the reactor [2]. In view of this, the overlapping of two operating modes of the fission chamber requires the application of different types of detectors at the same time in the control and safety systems of reactors. As a rule, fission chambers are employed only in the pulse mode over the range 10^{-10}-10% of the nominal power of the reactor; in other cases, use is made of current chambers with a boron or helium radiator with a supplementary working volume for compensating the γ-ray backgrouns (KNK-53M, KNK-3, and KNK-4 chambers [1]). The disadvantages of such systems include the use of a large number of detectors and, hence, of transmission lines, as well as a more complicated auxiliary apparatus. The use of one wide-range detector would facilitate simplification of the control and safety system while at the same time enhancing its reliability and economy.

At the present time for the first time in the USSR a wide-range fission chamber, the KNK-15-1, with an increased degree of compensation of the γ-ray background has been developed and put into commercial

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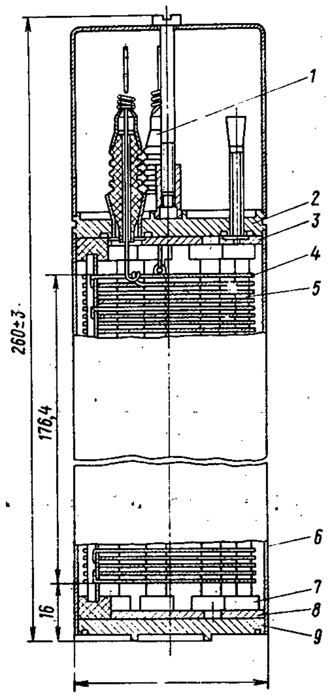

Fig. 1

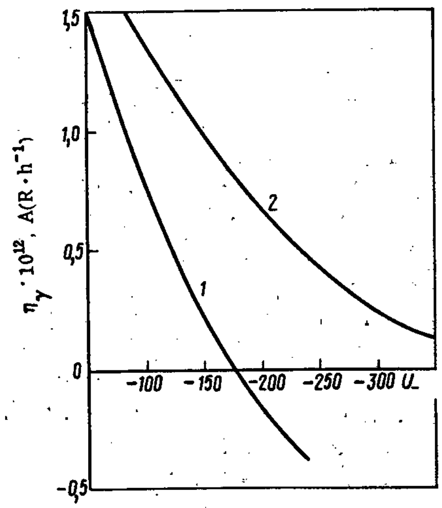

Fig. 2

Fig. 1. Construction of KNK-15-1 ionization chamber: 1) cermet terminals; 2, 3, 8, 9) flanges; 4) plates of electrode system; 5) struts; 6) case; 7) supporting insulators.

Fig. 2. γ-Ray sensitivity of chamber vs voltage U_- of compensation section at $P_\gamma = 1.6 \cdot 10^5$ R/h: 1) for KNK-15-1; 2) for KNK-15.

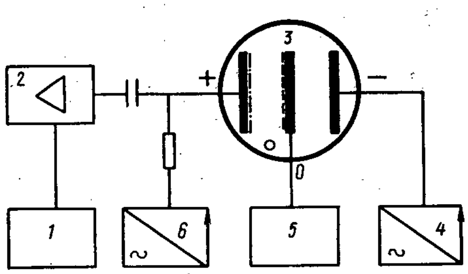

Fig. 3

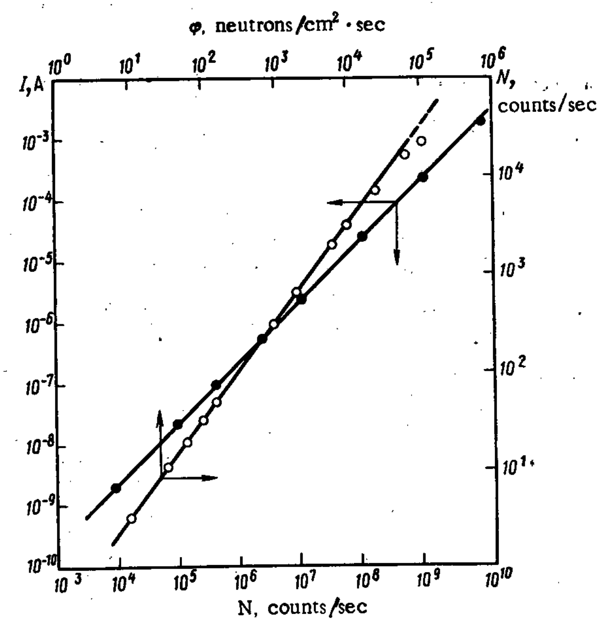

Fig. 4

Fig. 3. Connection diagram of chamber: 1) pulse recorder; 2) amplifier; 3) KNK-15-1 chamber; 4, 6) constant-voltage source; 5) dc meter.

Fig. 4. Operating characteristics of KNK-15-1 chamber for pulse (○) and current (●) modes.

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production. Figure 1 shows the construction of the chamber. The electrode system consists of a set of 89 plates with a diameter of 44 mm, connected in two sections: A neutron section sensitive to neutrons and γ rays and a compensation section sensitive only to γ rays. The plates of the neutron section are coated with a 1 mg/cm^2 layer of U_3O_8. Each plate has three projections which fit into openings in the struts mounted in the supporting insulators (3, 8). The positioning of the openings in the struts ensures an interelectrode gap in 1.6 mm. The electrical contact of the plates with the struts is fashioned by spot welding. The electrode system is enclosed in a cylindrical case with flanges 2 and 9, one of which bears three cermet terminals serving as the electrical leads of the chamber. Argon-arc welding makes the chamber airtight. The principal metal parts are made of 12Kh18N10T stainless steel while the insulators are made of high-alumina ceramic. The chamber is filled with a mixture of 96% Ar +2% He +2% N_2 at a pressure of 3.5 atm. The addition of helium serves to monitor the hermiticity of the chamber after it has been built. The construction ensures mechanical resistance to vibrations and shocks in accordance with GOST 16962-71. The chamber diameter is 50 mm, its length is 260 mm, and its weight is 950 g. The capacitance of each section is 360 pF. At an operating voltage of 250 to 400 V the collection time for ionization electrons is 50 to 100 nsec. The chamber can be operated at 315°C for 25,000 h. The operating lifetime is 5 years.

Unlike the case in the similar KNK-15 chamber, the radiator of the KHK-15-1 contains less ^{234}U which determines the α activity of the radiator. Because of this, the intrinsic background of the chamber does not exceed $5 \cdot 10^{-10}$ A, thus making it possible to the lower bound of the current mode by more than an order of magnitude. The sensitivity to thermal neutrons is $(2.6 \pm 0.3) \cdot 10^{-13}$ A/neutron/cm$^2 \cdot$ sec.

To balance the γ-ray sensitivity, part of the plates of the compensating section are made of tantalum. This permitted a substantial reduction of possible chamber decompensation during variations of the γ-ray spectrum in the course of the reactor run. The design adopted makes it also possible to vary the current distribution over the sections between broad limits, while preserving the linearity of the operating characteristic of the chamber. Figure 2 shows the dependence of the sensitivity to 1.25 MeV γ-rays on the negative voltage across the electrodes of the compensation section with a constant voltage of 250 V in the neutron section. At a dose rate of up to $1 \cdot 10^5$ A/R $\cdot$ h^{-1} and voltages of +250 and −175 V across the chamber electrodes the γ-ray sensitivity does not exceed $1 \cdot 10^{-13}$ A/R $\cdot$ h^{-1}. For comparison, Fig. 2 gives the analogous curve for the KNK-15 chamber in which the material of all the electrodes was the same. With the same supply voltages the γ-ray sensitivity of the latter chamber is substantially higher and complete compensation is not achieved.

In the mode without compensation the γ-ray sensitivity of the KNK-15-1 chamber is $2 \cdot 10^{-10}$ A/R $\cdot$ h^{-1}. The chamber retains the linearity of its operating characteristic at a current of $5 \cdot 10^{-3}$ A and a voltage of +350 V in the neutron section and −235 V in the compensation voltage.

In operation in the pulse mode the mean charge of ionization electrons is $1.0 \cdot 10^{-13}$ C ± 20%. Depending on the discriminator threshold, the sensitivity of the chamber in the pulse mode is 0.1-0.5 pulse/neutrons $\cdot$ cm^2. Figure 3 shows the connection diagram of the chamber which allows it to be used simultanesouly in two modes: pulse and current, with the grounded case acting as a protective electrode in the current mode. The circuit was tested in the IRT-2000 reactor at the Moscow Engineering Physics Institute (MIFI) and came through with flying colors. Figure 4 shows the plots of the pulse counting rate N and the chamber current I against the neutron flux density. It shows that the chamber ensures the necessary overlap of ranges monitored in the different modes.

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RADIOCHEMICAL DETECTOR OF LOW-INTENSITY FAST NEUTRONS

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UDC 539.1.074

For some problems of experimental physics it is necessary to record extremely flow fluxes of fast neutrons. These problems include that of detecting the flux of fast neutrons from the earth, which is very important for radiochemical detectors of solar neutrinos [1]. The expected flux density of fast neutrons arising as the result of spontaneous fission of ^{238}U and the (α, n) reaction from α-active isotopes in minerals is about 1 neutron/cm^2·day [2]. The present paper describes the construction of a radiochemical detector capable of recording such a low flux.

The $^{40}Ca(n, \alpha)^{37}Ar$ reaction is used to detect fast neutrons. The ^{37}Ar formed is a radioactive K-capture isotope with a half-life of 37 days. The cross section for this reaction is shown in Fig. 1 [3] from which it follows that a detector based on ^{40}Ca detects fast neutrons with an energy $E \geq 2$ MeV. A simple estimate shows that about 100 kg of calcium is required to detect such a low neutron flux, even with allowance for the application of the technique for detecting the very low activity of ^{37}Ar formed. If the well developed technique of extracting inert gases from a liquid [3, 4] is employed, the ^{37}Ar formed can be extracted from aqueous solutions of calcium salts. All existing calcium salts, however, are poorly soluble in water and this results in an excessive increase in the size of the detector. Moreover, the large quantity of water in the detector acts as a moderator of fast neutrons, i.e., reduced the registration efficiency. In constructing the detector described here, therefore, use was made of the capability of ^{37}Ar to be easily extracted from certain solid organic compounds [6]. A study was made of the possibility of extracting ^{37}Ar from a number of powdered calcium compounds. The gist of the method was as follows [7]: The substance under study was placed in a hermetically sealed flask with a volume of about 50 cm^3 which was then pumped down to a pressure of $\sim 10^{-4}$ mm Hg and irradiated with neutrons from a Pu–Be source. The flask was then connected to a trap containing activated carbon at −196°C. The ^{37}Ar released from the crystals of the substance was sorbed by the carbon in the trap. The extracted gas was purified and pumped into a miniature proportional counter with a volume of 0.5 cm^3 which measured the ^{37}Ar activity. The apparatus for purifying and pumping the ^{37}Ar was described in detial in [8]. For comparison in the same way we irradiated an aqueous solution of calcium chloride from which ^{37}Ar was extracted by blowing through helium. It was assumed that 100% of the ^{37}Ar is extracted from the solution. The results of the investigations are presented in greater detail in [7]; here, we merely point out that nearly 100% extraction (to within ±3%) of the ^{37}Ar was attained for calcium oxalate (CaC_2O_4) which has a high (31 wt. %) calcium content and does not contain hydrogen, a moderator of fast neutrons. The results obtained served as a basis for building a fast-neutron detector.

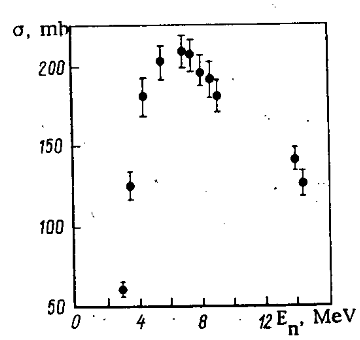

Fig. 1. Cross section for $^{40}Ca(n, \alpha)^{37}Ar$ reaction.

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The detector consists of a hermetically sealed cylinder of diameter 60 cm and height 120 cm, filled with 185 Kg Ca_2O_4. Before being filled with the CaC_2O_4 the cylinder was dried at 200°C to remove chemically bound water which substantially reduces the ^{37}Ar yield. A valve with a 20-mm opening mounted on the upper end of the cylinder for pumping down the detector and withdrawing the ^{37}Ar formed. A PTM-4M manometric tube in the bottom part of the cylinder is used to monitor the pressure.

In investigating the rate at which ^{37}Ar was withdrawn we pumped the detector down to a pressure of 10^{-3} mm Hg and then irradiated it with neutrons from a Pu–Be source. The ^{37}Ar formed was sorbed in a glass trap containing 5 g activated carbon at −196°C. The quantity of ^{37}Ar sorbed was measured as a function of the extraction time. It was established that the process of ^{37}Ar extraction obeys the law $Q/Q_0 = [1 - \exp(t/t_0)]$, where Q/Q_0 is the fraction of ^{37}Ar extracted, t is the extraction time, and $t_0 = 1.4$ h. According to the relation obtained, 95% of the ^{37}Ar is extracted in roughly 4.2 h. The efficiency with which the detector records neutrons from the Pu–Be source is ~1%. The present-day technique of low-background measurements allows ^{37}Ar activity as low as ~1 decay/day to be recorded [8]. The detector presented here is capable of recording fluxes of fast neutrons with an energy $\gtrsim$ 2 MeV down to $\sim 10^{-2}$ neutrons/cm^2·day. It should be noted that the detector is not sensitive to γ rays with an energy of up to 10 MeV and is not intended for measurements of the flux of fast neutrons from the earth.

In conclusion, the authors express their profound gratitude to Professor R. Davis for his useful discussions.

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COMPARISON OF CALCULATIONS ON A STANDARD FAST REACTOR

A. I. Voropaev, A. A. Van'kov, and A. M. Tsibulya

UDC 621.039.51

Results have been published [1] from calculations on a standard fast reactor (Baker model) performed in various countries; new data have now become available. The Argonne National Laboratory in the U.S.A. has published results from use of a library of nuclear data ENDF/BIV for three reactor models in order to determine the criticality of variations in the enrichment [2]. The Power Physics Institute at Obninsk has performed analogous calculations with the BNAB-78 system of constants, which are currently used in calculations on fast reactors. A difference from the BNAB-microsystem [1] is that the BNAB-78 system is consistent with basic integral experiments on fast critical assemblies.

Table 1 gives calculated values for the critical concentration, critical mass, and breeding characteristics; the definitions of these quantities are found in [1]. The CARNAVAL-IV data (France), the KFK-INR data (Federal German Republic), and the FD-5 results (Britain) are produced for convenience in comparison. The following conclusions are drawn: The maximum difference in the equivalent critical loads M_{eq} is about 40 kg in all cases, while the maximum difference in the physical breeding factor B is about 0.03 in all cases, whereas the difference is twice this in the excess breeding factor G.

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TABLE 1. A Comparison of Theoretical Characteristics for a Standard Fast Reactor

Constant system	Nuclear concns. 10^{21} cm^{-3}			Critical load,* kg			Excess breeding coeff.†			Physical breeding coeff.†		
	^{239}Pu	^{240}Pu	^{238}U	M_9	M_0	M	G_{co}	G_b	G	B_{co}	B_b	B
						Model A						
BNAB-78	0,953	—	6,246	946	—	946	−0,228	0,568	0,340	0,713	0,575	1,288
ENDF/B IV	0,983	—	6,217	975	—	975	−0,257	0,565	0,308	0,713	0,565	1,278
CARNAVAL-IV	0,957	—	6,243	946	—	946	−0,238	0,613	0,375	0,702	0,610	1,312
KFK-INR	0,952	—	6,248	945	—	945	−0,195	0,584	0,390	0,730	0,588	1,319
FD-5	0,936	—	6,264	929	—	929	−0,229	0,598	0,369	0,707	0,613	1,320
	Mean					948	−0,229	0,586	0,356	0,713	0,590	1,303
						Model B						
BNAB-78	1,030	—	5,450	1022	—	1022	−0,362	0,550	0,188	0,592	0,553	1,145
ENDF/B	1,042	—	5,338	1034	—	1034	−0,385	0,567	0,181	0,595	0,570	1,165
CARNAVAL-IV	1,026	—	5,454	1018	—	1018	−0,368	0,594	0,226	0,587	0,586	1,173
KFK-INR	1,024	—	5,456	1016	—	1016	−0,329	0,566	0,237	0,610	0,566	1,176
FD-5	1,004	—	5,476	997	—	997	−0,358	0,576	0,218	0,593	0,582	1,175
	Mean					1017	−0,361	0,571	0,210	0,595	0,571	1,167
						Model C						
BNAB-78	0,950	0,475	5,055	943	473	1013	−0,187	0,544	0,357	0,704	0,598	1,302
ENDF/B IV	0,964	0,482	5,034	957	480	1031	—	0,564	—	0,701	0,616	1,317
CARNAVAL-IV	0,956	0,476	5,043	948	476	1013	−0,199	0,587	0,389	0,686	0,629	1,316
KFK-INR	0,940	0,470	5,070	933	468	1018	−0,196	0,563	0,366	0,701	0,610	1,311
FD-5	0,938	0,469	5,073	931	467	990	−0,176	0,567	0,391	0,705	0,624	1,329
	Mean					1013	−0,189	0,565	0,376	0,699	0,615	1,315

*The subscripts 9 and 0 refer to ^{239}Pu and ^{240}Pu.
†co, core; b, blanket.

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EFFECTS OF COOLANT INPUT PARAMETERS ON THE THERMOHYDRAULIC CHARACTERISTICS OF A FIELD STEAM-GENERATING TUBE

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UDC 621.039.534.25

A steam generator heated by liquid metal currently employs Field tubes with natural circulation in the steam–water loop. Design work on steam generators required definition of the stability of the thermal and hydraulic characteristics of the evaporators in order to define the theoretical or emergency values for the input parameters of the heat carriers.

Well-tried methods [1, 2] incorporate the heat leakage from the annular channel to the tubing, and these have been used in an analysis of the thermal and hydraulic characteristics of Field tubes. The calculations have been performed for direct-flow and countercurrent modes of motion in the annular channel. The parameters of the carriers cover the ranges occurring in actual or projected systems. The pressure in the steam–water loop was varied from 2.3 to 7.2 MPa. The ratio of the cross sections of the annular channel and the pipe was taken as constant:

$$S_{an}/S_{op} = 3{,}4; \quad l = 7\,\mathrm{m}.$$

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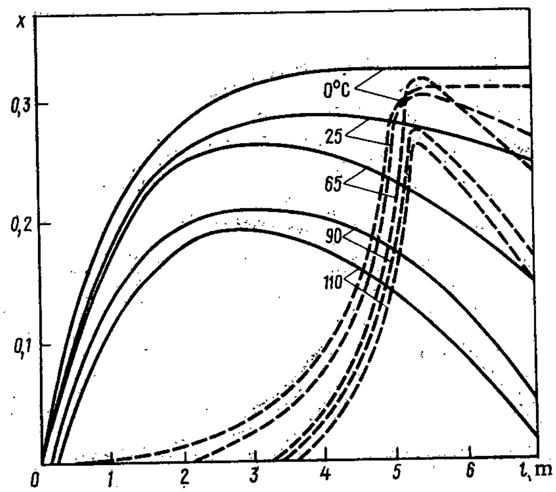

Fig. 1

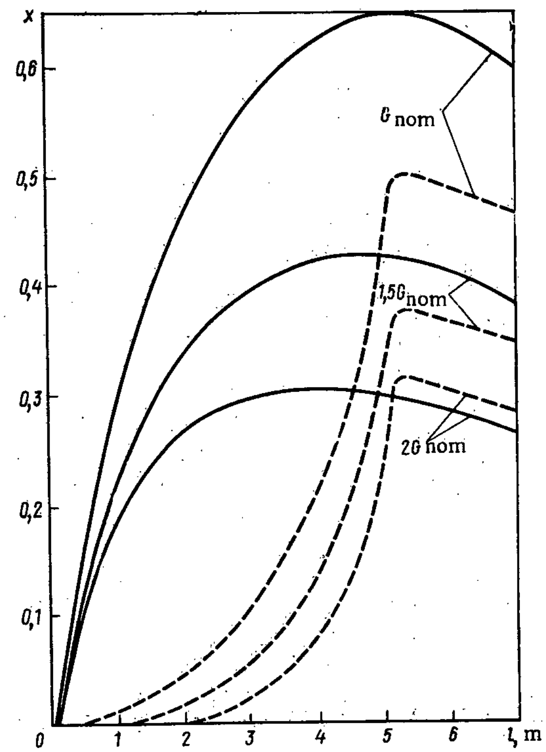

Fig. 2

Fig. 1. Distribution of the steam content along the length of the channel in relation to water temperature at the inlet to the tube in direct-flow mode (solid line) and in countercurrent mode (broken lines) for various values of the temperature deficiency in the water at the inlet (figures on the curves).

Fig. 2. Distribution of the steam content along the length of a tube in relation to heating carrier flow rate (numbers on curves) in direct-flow mode (solid lines) and countercurrent mode (broken lines).

The temperature of the water at the inlet to the evaporator system may differ from the theoretical value under actual conditions on account of inadequate mixing of the boiler and feed components in the evaporator volume, and therefore that temperature may lie between the saturation temperature t_s and the feedwater temperature. Calculations show that a fall in water temperature at the inlet alters the heat-transfer conditions in the evaporator channel within the working pressure range, and this gives importance to a feature of the heat transfer in Field tubes, namely the considerable heat leakage. The countercurrent scheme is more stable against fluctuations in the inlet temperature. Figure 1 shows that the current and exit values for the steam content in the countercurrent case are largely unaffected when the inlet water temperature fluctuates between t_s and 100°C below this (x_{out} of 0.15-0.27). The corresponding range for the direct-flow scheme is 0.05-0.35. The effects of the inlet temperature on the thermohydraulic characteristics have also been examined, and it is found that a rise in temperature at the inlet to 550°C reduces the number of recirculations required in the steam–water loop to 1.2-1.5 with this geometry for the channel in the working pressure range. It has been found [3] that a boiling crisis of the second kind can occur at the exit from the annular channel in this range of characteristics, which is due to detachment of the liquid film from the wall. Therefore, stable natural circulation in the loop involving a Field tube (absence of crises) required the inlet temperature of the heating carrier to be in the range 380-450°C. On the other hand, a Field tube works more efficiently at 450-500°C. The steam content attains its maximum at the end of the evaporation section. The leakage effects have only a slight effect on the overall heat transfer, although they are particularly noticeable in the direct-flow scheme. Any reduction in the inlet temperature below 400°C raises the natural-circulation repetition factor to 10-15 even for pressure in the steam–water loop of 2.3 MPa, and the input heat may go only to heat the water in the inlet tube.

The hydrodynamic features of the design have to be incorporated in designing and operating steam generators with Field tubes. There is nonuniformity in the flow of the carrier along the radius, while the spaces affect the hydraulic resistance and the turbulence, and emergency shutdown may affect part of the carrying

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cross section and influence the flow distribution. This means that the actual flow of the heating carrier in the individual channels may differ from the theoretical one. The nominal flow rate is taken as the flow rate through an evaporating component that provides a speed for the carrier close to the speed in the evaporators used in existing or projected systems. Figure 2 shows theoretical results for the effects of the flow rate of the hot carrier on the distribution of steam content along a Field tube. If the flow rate is raised to 1.5 times the nominal value, there are no substantial changes in the thermohydraulic characteristics of the element, but increase in the flow rate by more than a factor 1.8 may cause some danger of unstable natural circulation on account of elevated steam contents in the Field tube. Similarly, reduction in the hot-carrier flow rate by more than a factor 1.25, which resembles a reduction in the inlet temperature below 400°C, can increase the condensation of the steam-water mixture in the outlet tube, and as a consequence can cause the steam content at the outlet to fall to $x=0$. The Field tube works more stably in the countercurrent scheme and the output steam content varies from 0.28 to 0.47, as against from 0.25 to 0.57 for the direct-flow scheme.

This analysis of steam generators with Field tubes shows that any deviations in the parameters of the heating carrier and the working body from the nominal ones may cause instability in the natural circulation or accentuate the heat leakage from the annular channel to the outlet tube, and this leak can become comparable with the entire heat input to the evaporating component. A countercurrent scheme is more stable than a direct-flow scheme under fluctuations in the input parameters of the carriers.

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EIGHTIETH BIRTHDAY ANNIVERSARY OF ACADEMICIAN NIKOLAI ANTONOVICH DOLLEZHAL'

The eminent Soviet scientist and designer, Academician Nikolai Antontvich Dollezhal', was 80 years old on Oct. 27, 1979.

N. A. Dollezhal' is widely known in both the Soviet Union and abroad as one of the closest companions of I. V. Kurchatov, as the main designer of many types of nuclear reactors constructed in the Soviet Union, and as one of the founders of the new branch of science and technology, associated with the mastery and practical utilization of atomic energy.

N. A. Dollezhal' is a shining example of scientist and organizer of the Soviet structure. The distinguishing features of Nikolai Antonovich Dollezhal' are the innovation, search and boldness of the true scientist; originality and persuasiveness of the talented designer; energy, purposefulness, grandeur and will of the re-

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markable organizer; the generosity, sensitiveness and fastidiousness of the talented teacher and tutor; the depth, captivating strength and clarity of the advocate of scientific ideas and scientific–technical skills.

The working, creative, and vital path of N. A. Dollezhal' is closely connected with the biography of the Soviet Union, with the birth, shaping, and development of Soviet science, with the solution of the most important scientific and engineering problems, having paramount importance also for increasing the prosperity of the Soviet society, and for consolidating the effort and might of the Motherland.

In 1923, after completion at the Moscow Higher Technical College, N. A. Dollezhal' worked in the Podol' locomotive factory, and participated in the restoration and modernization of the means of communication of the young Soviet State.

In the following years he worked as a designer in the organization "Heat and Power," created for the achievement of the GOELRO Lenin Plan, and the restoration of the thermotechnical and power generation economy of establishments. By his active and direct participation, the most important scientific–technical problems of the young power generation of the Soviet Union were solved – the combustion of fuel in the powdered state, the combined generation of heat and electric power, central heating, increasing the efficiency of the power generation cycle by the use of high pressure, etc.

After a scientific mission in 1929 to Germany, Czechoslovakia, and Austria, N. A. Dollezhal' carried out a number of important projects on the construction of special plant for power generation, the chemical industry and other branches of the national economy. He was soon appointed a member of the Technical Council for the National Commissar of Heavy Industry. In these years N. A. Dollezhal' designed the first Soviet high-pressure compressors for the chemical industry and carried out the basic work on the theory of automatic valves. In 1935 Nikolai Antonovich was appointed to the office of the chief engineer of a large-scale chemical engineering factory, and in 1942 he was entrusted with the organization of the Scientific Research Institute of Chemical Engineering. By this time N. A. Dollezhal' had already developed as an eminent scientist in the field of thermophysics, mechanics, and material behavior, and as a talented designer of specific equipment and assemblies operating under complex conditions.

When the problem of creating a nuclear technology was set before Soviet science, N. A. Dollezhal' immediately advanced into the first ranks of its creators. In this profession, the talent of Nikolai Antonovich Dollezhal' was manifested with particular brilliance as an academic-fundamental inaugurator, creator of designs, having no equals in world practice, organizer and director of large-scale projects requiring a complex approach to their achievement and the coordinated efforts of a large staff of scientific and engineering-technical workers. To N. A. Dollezhal' belongs the fundamental research on the interrelation of thermophysical and nuclear-physical phenomena. By his active participation, the staff under his direction carried out a large group of scientific-research and experimental-design projects, the results of which formed the basis of reactor construction in the Soviet Union.

The channel type of nuclear reactor, regularly designated abroad as the Soviet type, is widely used in Soviet power generation. Its design was suggested by A. N. Dollezhal'. This design principle was used subsequently by him as the main designer for the reactor of the world's first nuclear power station at Beloyarsk, and the Bilibin' nuclear-thermal power station, etc.

The important advantage of this type of reactor is the possibility for achieving large unit capacities based on existing engineering factories. The channel type of reactor has allowed Soviet nuclear power generation to emerge in the foremost position in the assimilation of nuclear power station power units with a unit capacity of 1 million kW. The first of these power units was brought on stream in 1973 at the V. I. Lenin nuclear power station, Leningrad. It was the head of a series of functioning and under-construction power units with rated power concentrations of 4-6 million kW in the Leningrad, Kursk, Chernobyl', and Smolensk nuclear power stations, etc. Thanks to this type of reactor, the Soviet Union leads in work on the construction of nuclear power units for the Ignalin nuclear power station with a capacity of 1.5 million kW, which is still unknown in foreign nuclear power generation.

The first research nuclear reactors and many other important advances in the creation, establishment and development of reactor technology are inseparably linked with the name of Academician N. A. Dollezhal'.

The Motherland has valued highly the services of N. A. Dollezhal' to science, the Soviet nation, and the state. The title of Hero of Socialist Labor has been conferred on him; Nikolai Antonovich – Lenin Laureate and five State Prizes – he has been rewarded with four Orders of Lenin, Orders of the October Revolution, Labor Red Standard, and Red Star. In 1953 N. A. Dollezhal' was chosen as a Corresponding Member of the Academy of Sciences of the SSSR and in 1962 he was chosen as Academician.

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N. A. Dollezhal' paid serious attention to teaching work. He organized and headed the Faculty of Power Plant of the N. É. Bauman Higher Technical School, Moscow. N. A. Dollezhal' was a member of the Examiner Council of High Academic Courses of the Soviet Union; he conducted a great deal of work on the propagation of technical knowledge, he was a member of the editorial staff of a number of journals and was a deputy chairman of the board of the All-Union Society "Znanie."

For almost 20 years N. A. Dollezhal' has been a member of the editorial staff of this journal and by his active work contributed to the strengthening of its scientific authority.

Today, Nikolai Antonovich is looking into the future. He is excited by the new problems being faced by nuclear power generation: The increase of the unit capacity of the nuclear power station units, the different ecological-economical aspects of the disposition of the increasing volume of nuclear electric power generating capacities and plants for the reprocessing of nuclear fuel, the use of nuclear reactors in centralized heat supply and high-temperature industrial technology. And, as always, Academician N. A. Dollezhal' is striving not only to understand the progressive trends in nuclear science and technology, but also to reflect them in design developments many of which have qualitatively new technical solutions and which open up possibilities new in principle for the future efficient development of reactor technology.

On this anniversary day, we wish Nikolai Antonovich good health and further successes in his manifold scientific, teaching, and public activities.

VIKTOR ALEKSEEVICH SIDORENKO

On Oct. 17, 1979, the eminent Soviet scientist—engineer, Doctor of Technical Sciences, Professor Victor Alekseevich Sidorenko celebrated his 50th birthday.

The scientific activity of V. A. Sidorenko is associated with the I. V. Kurchatov Institute of Atomic Energy, in which he started to work in 1952.

In the 1950s, during the formation of nuclear power generation, V. A. Sidorenko participated directly in the development of water-cooled—water-moderated power (VVÉR) reactors and directed such important courses as the monitoring and control of reactors, safety assurance of nuclear power stations, the physics and thermophysics of the cores, etc. V. A. Sidorenko was the scientific director of the power-startup of the first unit of the Novovoronezh nuclear power station and the Rheinsberg (GDR) nuclear power station.

For the development and construction of the VVÉR type of reactor for the first unit of the Novovoronezh nuclear power station, Viktor Alekseevich Sidorenko, together with others, was awarded the State Prize of the Soviet Union.

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The great scientific and organizational activity of V. A. Sidorenko has been associated with the development and assimilation of the VVÉR-440 units, providing the first stage of the extensive use of economically competitive nuclear power stations. The investigations carried out by V. A. Sidorenko and colleagues showed the possibility of increasing the capacity of the VVÉR-440 series-production unit by a factor of more than two by comparison with the first unit of the Novovoronezh nuclear power station for the same dimensions of the reactor vessel.

Viktor Alekseevich Sidorenko is an acknowledged authority in the field of research on the cardinal problem of nuclear power generation – the safety assistance of nuclear power stations. Under his guidance, the scientific principles of the problem of the operating safety of VVÉR reactors have been developed, a new approach to the problems of nuclear safety during the mass utilization of nuclear power stations has been developed and introduced into practice. The many years of research on this problem by V. A. Sidorenko are recorded in the monograph "Problems on the Safe Operation of VVÉR Reactors," published in 1977.

An important project of recent years, carried out by V. A. Sidorenko under the immediate direction of Academician A. P. Aleksandrov, is a new trend for the use of nuclear power generation – nuclear heat supply stations based on water-cooled–water-moderated reactors.

The well-known specialist, Director of the Division of Nuclear Reactors in the I. V. Kurchatov Institute of Atomic Energy, V. A. Sidorenko pays great attention to the training of scientific staff, having been Professor of the Faculty of Nuclear Power Stations of the Moscow Power Institute. V. A. Sidorenko is a member of the Examiner Council of High Academic Courses of the Soviet Union, of the editorial board of the journal Atomnaya Énergiya, and the editorial council of Atomizdat.

Viktor Alekseevich is a spirited and cheerful person, is distinguished by his broad learning, direct approach to the solution of scientific problems, moderation of the director-communist to unite the staff, and to direct his efforts to carrying out the problems posed.

The services of V. A. Sidorenko are highly valued by the government. He has been rewarded with the Order of Lenin and the Order of Labor of the Red Standard.

Viktor Alekseevich is full of creative force and energy. The editor, his staff, and students congratulate Viktor Alekseevich with affection, and wish him good health and future creative successes.

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SECOND EUROPEAN NUCLEAR CONFERENCE

I. D. Rakitin

The Conference under the motto "Nuclear Power – the Choice for Mankind," organized by the European Nuclear Society (ENS) together with the American Nuclear Society (ANS) took place in May 1979 in Hamburg (Federal Republic of Germany). Simultaneously a meeting of representatives of the nuclear industry of Western Europe and the USA, FORATOM 7, took place. About 2400 specialists in all participated in the meetings, from more than 30 countries and international organizations.

At the plenary sessions and the section sessions, more than 350 reports were presented on the following main subjects: General problems of design and construction of nuclear power stations; control of nuclear power stations; fuel technology; reactor physics, numerical methods and mathematical models; fuel reprocessing and the use of plutonium; safety, shielding from radiation and estimation of the degree of risk (including loss of coolant and core melting); protection of the environment; quality assurance and reliability of the structural components of nuclear power stations; treatment, transportation, and removal of wastes; the use of nuclear power in ship installations and for the provision to users of low- and high-potential heat.

The Soviet delegation presented reports on controlled thermonuclear fusion, development of the VVÉR-1000 and VTGR (high-temperature gas-cooled reactor), experience in the operation of the BN-350 and maritime nuclear power plants, and also certain aspects of the optimum control of the RBMK-1000.

As would be expected, great attention was paid at the Conference to events connected with the recent accident on the second unit of the American nuclear power station at Three-Mile Island.* Therefore, the previously prepared program was altered, and a special plenary session was devoted to a discussion of the circumstances and analysis of this accident.

The incident in Pennsylvania showed the great effect on the development of nuclear power generation in the USA and certain other countries. Thus, in the USA, a revision of all Babcock and Wilcox nuclear power stations has been halted, a decision on the introduction of new nuclear power stations has been withheld pending a refinement of requirements, in Sweden a new referendum on nuclear power generation has been announced, and in the Federal Republic of Germany the "freezing" of a decision on the installation of a large-scale center for the reprocessing and storage of fuel in Gorleben, etc. At the same time, the policies in principal of the Federal Republic of Germany on nuclear power generation are unchanged and 2 billion marks are being assigned additionally to work for ensuring the safety of nuclear power stations. Likewise, it was clearly and definitely stated that the program for nuclear power generation in France and in certain other countries will be maintained without changes.

More than 70 reports at the Conference were devoted to safety, shielding from radiation, and to different approaches to the assessment of risk. The different approach was noted to the role of the operator in ensuring the overall safety structure. Thus, if in the USA in this sense the operator has great potentialities, then in the West German concept carried out by the firm of "Kraftwerk Union" on reactors of the "Biblis" nuclear power station type, the emergency arrangement and automation of the scram systems is considerably higher. According to estimates, the development of incidents similar to what occurred at Harrisberg, could not happen in these reactors. In the Federal Republic of Germany, and Great Britain, it is assumed that in emergency situations the provision of safety should be totally entrusted to automation (analog and digital electronic circuits with emergency arrangements).

Many papers were devoted to a discussion of problems associated with thermohydraulics during the loss of coolant and with core melting. Attention was drawn to the corresponding "nonnuclear" experiments on the loss of coolant in a special American research reactor. Measures were described for ensuring the safety of the commercial "Superphoenix" fast sodium reactor (France) under construction, and the prevention

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of loss of sodium in the core. Three independent groups of scram rods ensure the reactor shutdown even in the case of deformation of the core. The system for the removal of the residual heat release of this reactor consists of four primary loops of sodium—water—steam (fourfold emergency arrangement) and two auxiliary emergency cooling systems.

In some reports, devoted to the development of general safety criteria of nuclear power stations, their considerable dependence is noted not only on the type of reactor, but also on the conditions of siting. Thus, the special features of the design and planning of the Swedish reactor SECURE, intended for heat supply, increases its safety because of its location in the vicinity of heavily populated points. In more than 50 reports, different aspects of light-water reactors were described. The tendency to standardization and unification of reactors and plant was noted. Thus, the firm of "Kraftwerk Union," based on unified plant have built a four-loop facility of 1300 MW, a three-loop facility of 1000 MW and a two-loop facility of 700 MW. The firm of "Combustion Engineering" (USA) with improvement of the reactor steam-generating plant (System-80) reinforce the definite conservatism of the decisions taken. Intrareactor-vessel devices have been converted for the installation of a larger number of control rods, the intracassette nonuniformity of the heat release has been reduced, the guiding assemblies have been remodelled, the flow of water in the steam generators has been improved, and the scram system has been modernized. The potentialities of operation of nuclear power stations under variable load conditions have been extended. The latter is characteristic also for development in the Federal Republic of Germany and especially in France, where in the mid-1980s up to 50% of the installed capacity will come from nuclear power stations.

As in the USA, so also in France and the Federal Republic of Germany, serious attention has been paid to the development of intracore monitoring systems for the neutron flux and the coolant flow rate, and also to special systems for the extracore monitoring of the neutron flux. Thus, for the control and safety system installed in the four units of 900 MW (electrical) of the "Tricastin" (France) nuclear power station, multisection extracore detectors are used, the readings of which are processed by special algorithms in microprocessore. In France a digital scheme of emergency shielding was developed and has operated during 5 years on "Phoenix." It should be mentioned that there is an ever-increasing tendency toward the use in unit computers of quite complex programs (simulators) of three-dimensional neutron-physics calculation using modern rebalance and the so-called nodal methods in a coarse network. The effective diffusion small-group macroconstants are obtained by means of folding, using first collision probability methods or Monte Carlo. In this case such conventional methods of calculation of meshes as the method of spherical harmonics, S_n, DS_n are almost unmentioned. Obviously, this is because of the tendency to the most adequate geometrical description of the complex polymeshes (cassettes) of modern commercial reactors.

For reactors of the "Biblis" nuclear power station type, the firm of "Kraftwerk Union" use program-simulators not only for interpolation of the power distribution fields together with readings of the detectors, but also for optimization of the different recharging conditions.

A considerable number of reports was devoted to fuel for light-water reactors, mainly in the form of uranium dioxide. Thus, the French specialists presented interesting data on the characteristics of the "Caramel" fuel, representing pellets (thin parallelepipeds) of uranium dioxide with a zirconium cladding, which are packed in the form of plates. The deep burnup of up to 30 MW·day/kg U or more, is achieved with a high power-loading (up to 100 kW/kg). The fuel is characterized by increased safety and enrichment up to 8%. The firm of Combustion Engineering has developed a so-called completely zircalloy fuel, in which even the spacing grids and the directing tubes are made of Zircalloy-4, which has led to a reduction of enrichment by 0.12%. This fuel is already being used in seven reactors, and it has shown a high reliability with good operating characteristics. The firm of "Kraftwert Union," using a similar approach to this problem, expect to achieve a burnup of up to 45 MW·day/kg U.

It was mentioned at the Conference that the growth of power requirements with the limited potentialities of light-water fuel cycles inevitably will lead to the intensive construction of fast neutron reactors. It is expected that the startup of the "Superphoenix" reactor, in the main repeating the successful solutions found for "Phoenix," will take place in 1981-1982. The further development of commercial reactor-breeders in France will be determined mainly by the experience in operating this facility. For fast reactors an oxide fuel is mainly being considered. The possibility of using gel-spheres is being studied, not only in vibrofilled fuel elements but also in the reprocessing of spent fuel (in the form of intermediate products). The use of uranium monocarbide is still being withheld in consequence of the increased cost of production.

Great attention was paid to the future developments and experience in the operation of prototypes of liquid-metal fast, and high-temperature gas reactors. When considering an international program of investi-

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gations of high-temperature gas-cooled reactors (USA, France, Federal Republic of Germany, Switzerland), intensified interest should be assigned to the use of these reactors, in the first place in nuclear power stations. It is expected that reactors with a steam cycle in a vessel of prestressed concrete will enter industrial operation in 1990. The high level and large volume of material-study and technological work should be mentioned, leading to the development of a quite reliable technology of spherical fuel elements for high-temperature gas-cooled reactors in the Federal Republic of Germany.

The conference demonstrated the high level of development in the design of modern nuclear power generating facilities, the increased responsibility for considering the problems associated with safety and the nonproliferation of nuclear weapons, and undoubtedly has been an important stage on the path for strengthening international cooperation in solving the complex power-generation problems facing the world.

SOVIET – FRENCH SEMINAR ON SODIUM – WATER STEAM GENERATORS

P. L. Kirillov

The seminar, held in July 1979 in FÉI,* was devoted to problems related with emergency shielding systems of sodium–water steam generators for nuclear power stations with fast reactors. The use of sodium in the second circuit of these facilities in conjunction with water under high pressure in the third circuit has posed new problems. The special feature of the present day structures consists in that the structural materials used in steam generators are relatively rapidly destroyed at the site of a defect, which leads to the self-development of a small leakage of water into the sodium. A large leak creates a danger of damage to the adjacent pipeline and subsequent failures in the steam generator. Any leakages of water into the sodium are assumed to be unacceptable, and therefore the steam generator must be provided with a sensitive and fast-response monitoring system which would allow it to be switched off and thereby prevent the development of a hazard and failure of the structure. Present-day technology has coped successfully with this problem, which is confirmed by the operation of the BOR-60 and BN-350 reactors in the Soviet Union and "Phoenix" in France. Nevertheless, certain problems require additional investigation and development and on which specialists of the Soviet Union, France, and other countries are working. The seminar which was held, not only allowed the investigations of recent years to be summed up, but also allowed experience to be exchanged in the operation of steam generators for the facilities under development.

As noted in the reports, depressurization of the BN-350 evaporator tubes in the initial period of operation was caused by the failure of components. Depressurization started, as a rule, with the appearance of a small leak. As it increased, a rapid increase of the pressure in the gas cavity of the evaporator and an increase of the concentration of hydrogen in the gas was observed. The increase of hydrogen above the rated concentration serves as the basis for shutting down the steam generator.

In "Phoenix," no water leakage into the sodium was recorded. Erosion-corrosion damage to the pipelines was noted at the inlet to the steam generator module. At the present time, detectors based on monitoring the hydrogen in the liquid sodium and in the protective gas are in the most widespread use. In the hydrogen detection system in "Phoenix," a nickel membrane is provided, with a thickness of 0.3 mm, through which the hydrogen diffuses and is detected with a quadrupolar mass-spectrometer. Sampling of the sodium is conducted after each module of the steam generator through a special selector. The hydrogen concentration in the sodium is known to the operator. With the appearance of an alarm signal, the selector is switched on and the operator has available the information about any leak after 5 min. If the leakage rate is small, the operator can localize the defective module. When the leakage rate is ≥ 0.18 g/sec, the steam generator is shut down and the water and sodium drained off. Great attention has been paid to the processes of hydrogen diffusion through ferrite steel walls with a thickness of 3.6 and 4 mm. The addition of hydrazine to the water sharply increased the hydrogen diffusion; however, its rate decreased with time and with a hydrazine concentration of 5 μg/liter it was established at the level of ~ 120 $\mu g/m^2 \cdot h$.

In the reports of the Soviet specialists, the results were given of experiments on the self-development of a small water leak into the sodium with different structural materials. These data have been used for

*Physico-Power Institute.

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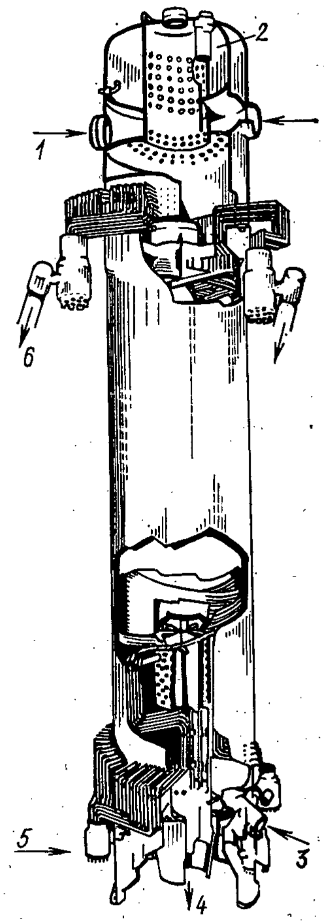

Fig. 1. "Superphoenix-1" steam generator: 1) sodium inlet at 525°C; 2) argon; 3) bursting membrane; 4) sodium outlet at 345°C; 5) sodium inlet at 235°C; (4 nozzles); 6) steam inlet at 184 bar and 490°C (4 nozzles).

meeting the requirements for hydrogen detectors in the gas and sodium, as applicable to the steam generator of the BN-350 and the BN-600 reactors. Hydrogen in the gas and sodium in the BN-350 is monitored by a thermal conductivity analyzer. Monitoring systems have been adapted in which magnetic discharge pumps are used as sensors. Mention was made in the reports of the results of tests of these systems and their operating characteristics, including under leakage conditions.

At present, in some countries, inertialess acoustic methods of leak detection are being developed. However, there is no unified opinion concerning the optimum frequency range and the most suitable algorithm for processing the signals. This situation is due to the technical difficulties for carrying out the experiments and with the difference in the facilities. The possibility of detecting leaks by an acoustic method is being investigated on a model of the BN-600 steam generator, which is being tested in the BOR-60 reactor. The leaks are simulated with argon.

A report on the design and manufacturing technology of the cased steam generators for "Superphoenix-1" (see Fig. 1) created great interest. The four steam generators, with a capacity of 750 MW (thermal) each, are being produced in the factory at Chalons-sur-Sol. The manufacture of each one occupies about 20 months. The safety systems of the steam generator have been calculated on the entry of 500 kg water into the sodium (~400 tons) in the case of leakage in a single vessel. The thermal fluxes do not exceed 450 kW/m^2.

STEAM GENERATOR PARAMETERS

Capacity, MW	750
Casing, m:	
diameter	2.898
height	22.44
Pipelines:	
diameter, mm	25 × 2.6
number	357
material	Nikalloy-800
mass	140 tons

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Declassified and Approved For Release 2013/02/15 : CIA-RDP10-02196R000800020004-1

The long-term development program of steam generators for nuclear power stations with fast reactors in France includes the planning and investigation of various alternatives for "Superphoenix-2." Technical solutions are being considered, similar to those accepted for the "Superphoenix-1" steam generators, but with some other steel and with tubes of some other size [25 × 2.7 mm of alloy-800; 30 × 3.7 mm of ferrite steel 1Kh2M or Kh9MV (EM12)]. Modular straight-tube steam generators are also being considered, with tubes of size 15 × 2.5 mm of ferrite steel 1Kh2M. The French specialists assume that the optimum capacity of the module will be within the limits of 100-200 MW (thermal). As other possible alternatives of steam generators, they are considering a steam generator with double tubes and a layer of helium between them (the commercial production of these tubes has started on the order of the firm "Westinghouse," and their heat-transfer characteristics are being determined) and a steam generator with a built-in pump, which will allow the secondary circuit to be simplified and will allow it to be made more cheaply.

At the present time in the Soviet Union and in France, the greatest experience has been built-up in the operation of nuclear power stations with fast reactors. By general acknowledgement, experience has shown that the reliability of nuclear power stations with these reactors is comparable with the reliability of other nuclear power stations, and their readiness factor is higher.

The seminar was useful to the specialists of both countries. The French specialists visited the FÉI laboratory, the BN-350, and the factory manufacturing the steam generators for the BN-600.

INTERNATIONAL SYMPOSIUM ON THE PHYSICS AND CHEMISTRY OF FISSION

G. B. Yan'kov

The Symposium on the Physics and Chemistry of Fission is traditional and is organized by the IAEA approximately once every 5 years. This Symposium was the fourth and took place from May 14-18 at Julich (Federal Republic of Germany); it was held in a jubilee year — 40 years since O. Hahn and F. Strassman discovered the phenomenon of the fission of heavy nuclei.

About 240 representatives of different countries and international organizations participated in the Symposium; 55 reports were heard, of which 13 were review reports on the basic problems of the physics of fission. All the reports presented were published by the organizing committee before the start of the symposium in a collection of comprehensive summaries of the reports; the IAEA are planning to issue proceedings, containing the complete texts at the end of 1979.

The following problems were considered: Fission barriers and the probability of fission, spontaneous fission and fission barriers, shell effects due to potential energy and density of levels, fission under the action of heavy ions, different characteristics of fission fragments and the emission of particles, and dynamic theories of fission.

When disucssion fission barriers and the probability of fission, much attention was drawn to the investigation of potential energy surfaces associated with fission. For nuclei of the radiothorium group, the second peak in the barrier was produced in the form of two components, corresponding to symmetrical and asymmetrical fission. It was shown that two parallel barriers exist: The normal mass-asymmetrical barrier and a somewhat higher mass-symmetrical barrier. This confirms the experimental data on subbarrier resonances during fission by neutrons of ^{231}Th and ^{234}Th. The values of the heights of the barriers were also considered on the assumption of triaxial asymmetry for nuclei from radium to californium, which allowed much better agreement to be obtained with the experimental values; based on this, secondary subbarrier resonances are predicted for radium. The structure of the fission cross section of even—even and odd nuclei, resonances and anomalies in the prompt and delayed fission cross sections and other problems were studied in detail in the discussion on photofission.

In considering spontaneous fission and fission barriers, attention was directed at the study of the spectroscopic characteristics of fission isomers and the theoretical verification of a new, experimentally obtained classification of spontaneous fission half-lives, confirming the increase of the half-lives with approach to the shell with N = 184. For spontaneously fissile isomers, the half-lives and spectra of the conversion

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electrons were measured on the basis of which the decay scheme was constructed for the rotational band in the second well, and the quadrupolar moments were determined. According to these data, the quadrupolar moments in the isomer state exceed by a factor of 3-4 the quadrupolar moments in the ground state.

The discussion of the shell effects associated with the potential energy and density of the levels was related with a detailed analysis of the various methods of theoretical calculations of the deformation of heavy nuclei. The considerable discrepancies between the theoretical predictions and the experimental data for the light actinides were noted. The reason for these discrepancies may be the nonconformity of the liquid and shell drop models. Part of the theoretical reports of this section was devoted to the study of the dependence of the properties of the excitation spectrum of nuclei on the deformation and also estimates of the fission barriers of nuclei with very high angular moments. According to the calculations, the fission barriers for large angular moments can be not only conserved by may even be increased: e.g., for ^{232}U the height of the second barrier is conserved at the level of 5 MeV right up to $l = 50\hbar$, and then with increase of l it increases by ≈ 1 MeV and only when $l > 100\hbar$ does it fall sharply.

On the problem of fission by the action of heavy ions, considerable and unique data has been accumulated in recent years, but its explanation has proved to be not so simple as appeared to be at first sight. Clearly, the small number (5 in all) of reports presented at the symposium can be explained by this. Nevertheless, a detailed analysis of the experimental data on the determination of the fission cross section of the light and heavy nuclei formed in different combinations of target – ion with different angular moments, deformations and excitation energies, is given in the papers. It was shown that the fission of such light systems as ^{16}O + ^{40}Ca is possible with a high probability. An important conclusion concerning the independence of the total kinetic energy of the fission fragments on the angular moment of the fissile compound nucleus was demonstrated in two papers. It was interesting to note that in the U + U reaction, in which the compound nucleus clearly is not formed, the mechanism of formation and decay of the heavy nuclei are subject to the rules previously obtained for the fission of compound nuclei.

In the greatest number of papers presented at the symposium, the different characteristics of the fission fragments and the emission of particles were considered. Considerable attention was paid to the problem of nuclear viscosity in fission – the relation between the collective and single-particle degrees of freedom. Much new specific data have been accumulated on the relations concerning the yield of the different masses of the fission fragments, on the charge distributions, and on the kinetic energy of the fission products. The field of study of the characteristics of the fission fragments is now being directed to fermium and mendelevium. In particular, for ^{259}Mv formed in the reaction ^{248}Cm (^{18}O, α, 3n), a symmetrical mass distribution was observed.

Several reports at the symposium were devoted to the different problems of the dynamics of the theory of fission and fission induced by muons. The explanation of the fission mechanism from the saddle point to the instant of break-up of the nucleus into the fragments is of the greatest importance in the investigation of the dynamics of the fission process.

The reports of the Soviet scientists touched upon urgent problems in fission (a review report on photofission, the nonconservation of party during fission of nuclei, the investigation of the spontaneous fission of certain isotopes of the heavy elements, the fissility of the subactinide nuclei, etc) and created great interest and lively discussion.

SECOND INTERNATIONAL CONFERENCE ON THE USE OF NUCLEAR METHODS OF ANALYSIS IN ANALYTICAL CHEMISTRY

V. P. Varvaritsa and Yu. F. Rodionov

Specialists from the member-countries of COMECON and Yugoslavia, and also Austria, the Netherlands, Norway, France, and the Federal Republic of Germany participated in the work of the conference, held in March 1979 in Dresden (German Democratic Republic). Some 109 reports were presented, of which 12 were review reports. The five sections considered the application of nuclear methods of analysis in power generation, industry, geology and space chemistry, medicine, biology and for monitoring the environment, and also individual procedural and equipment problems.

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The majority of the reports was devoted to the use and development of activation, x-ray, and radiometric analyses.

The report of H. Engelman (France) was devoted to the complex consideration of activation analysis as applicable to monitoring the purity of different materials. In it, he reported on the possibilities and use of cyclotrons, betatrons, microtrons, and linear accelerators, and presented the results of the determination of more than 30 impurity elements (Al, Si, Ca, Ag, Th, Dy, Ta, etc.) in thin layers by the p, γ-reaction, microimpurities of elements by the γ,γ', γ,n and γ,p, γ,f reactions in the energy range of 1-15, 15-40, and 5-10 MeV, with a sensitivity of 0.1-0.001 $\mu g \cdot g^{-1}$.

A large number of reports devoted to activation analysis for monitoring the quality of manufacturing production for the electronics industry and the determination of trace quantities of elements in semiconductor materials. In particular, the French specialists reported on the use of activation analysis, using low-energy charged particles for monitoring semiconductors of gallium arsenide. In particular, the determination of oxygen ^{16}O (^{3}H, n)^{18}F and carbon ^{12}C ($^{2}_{1}H$, n)^{13}N in gallium arsenide was discussed. For activation, nuclei of tritium and deuterium with an energy of 3 MeV were used, and the error of the analysis amounted to $\pm 10\%$ for a concentration of these elements of $10^{-5}\%$.

The use of activation analysis in geochemistry and metallurgy of metals was described in more than 20 reports. A quite thorough analysis of the present-day use of activation analysis and also certain other nuclear methods in geochemistry was given by R. Hubner (GDR). In the report, attention was drawn to the possibility of using an activation method based on the photoexcitation of nuclei (γ, γ') for determining uranium at the level of tenths of g/ton. Part of the reports concerned the use of individual methods for the solution of specific problems and was interesting from the point of view of estimating the applicability of the methods: The determination of boron in geological materials (G. Lutzidr, GDR); the determination of the distribution of chlorine in silicon oxides (L. Rowinska, Polish Peoples' Republic); elimination of the matrix effect when determining oxygen in metals, by activation with 14-MeV neutrons (Z. Czopa et al., Polish Peoples' Republic); the application of neutron-activation analysis to the investigation of the geochemistry of regional metamorphosis (G. Luch et al., GDR); the determination of osmium in geological samples (H. Schelhorn et al., GDR), and the investigation of zirconium alloys (K. Krogner et al., GDR), etc.

About 20 reports in the biology, medicine, and preservation of the environment sections were devoted to the development of methods of analysis of trace concentrations of elements in biological samples: Activation analysis in oncology (K. Baldauf et al., GDR) and metals in human hair (E. Lantzel et al., Austria); the use of certain facilities for the neutron-activation analysis of dry biological materials (N. Daz, The Netherlands) study of atmospheric fallout of heavy metals by means of neutron-activation and atomic-absorption analysis of peat samples (E. Steinse, Norway) etc. A report by L. Mozulishvili (SSSR) created interest, in which a method was described for the preparation of biological samples. The use of this method is very effective when using high-intensity irradiation fluxes when the normally used polyethylene containers start to melt.

A considerable part of the reports at the conference was devoted to nuclear-physics methods of analysis, based on the excitation of x-ray characteristic radiation. The reports presented reflected the progress in the development of this method, inc. ding the increase of sensitivity of the analysis due to the use as the primary radiation source of streams of heavy charged particles. In consequence of this, the analytical potentialities of the method have increased significantly, which has determined the prospects for its application in those fields which were not conventional, first and foremost in medicine, biology, and for environmental monitoring. The reports showed that work on the development of x-ray fluorescence methods of analysis, using heavy charged particles, is being extensively carried out, and that the greatest successes have been achieved in the GDR and Poland. The investigatigations in Poland are being conducted with both the use of radioisotopic sources of α-radiation and with proton generators and cyclotrons. In one of the reports, the possibility was considered of using a silicon—lithium spectrometer and a radiation source of ^{238}Pu for the analysis of concentrations of aqueous solutions (including blood) in Fe, Cr, Mn, Ni, Cu, and Zn of up to $10^{-7}\%$. The measurements were carried out on "thin" and "intermediate" samples, obtained by the method of chemical deposition of the elements from solutions, by means of sodium diethyldithiocarbamate.

The concluding meeting differed from the first in the considerable number of papers devoted to the application of analytical monitoring methods in nuclear power generation. Reports concerned with the monitoring of spent fuel and also monitoring at different stages of technology in the Purex-process, occupied an important place in the program. Great interest was created by the report of F. Baumgartner (Federal Republic of Germany) concerning analytical problems in this technology and by the report of R. Seluk (Czecho-

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slovakia) concerning the high-speed analysis of highly active solutions, obtained during nuclear fuel regeneration by extraction technology. The method is based on the separate extraction from the solutions of relatively pure ^{90}Sr fraction. First of all, ^{137}Cs is removed from the solution by means of a mixture of nitrobenzene with carbon tetrachloride. Then, by means of polyethylene glycol, the strontium is separated. The advantage of the method is that there is no necessity for analysis on a β-spectrometer. The measurements are carried out by an integrated count over a very short time.

In the report by Zh. Krtil' et al., (Czechoslovakia), information was given concerning a modified radiometric method for determining the content of plutonium in spent nuclear fuel. The essence of the method consists in the measurement of the total α-activity of plutonium and the ratio of the α-activity of ^{238}Pu/(^{239}Pu + ^{240}Pu). A calibration curve of the α-activity of ^{239}Pu + ^{240}Pu on burnup is constructed on the basis of mass-spectrometer measurements. In the report by A. Hermann et al., (GDR), correlations are given of the degree of burnup of 2% enrichment of ^{235}U with the ratio of the concentration of fission products ^{134}Cs/^{137}Cs and ^{154}Eu/^{137}Cs for VVÉR-70, and also the isotopic correlations between these ratios of fission products and ratios of Pu/U. The results obtained by γ-spectrometry show that by using the correlation functions found, the content of uranium and plutonium in spent fuel can be determined with an accuracy of ±2.5-4.7%. The discrepancy between the experimentally determined uranium content and that calculated by the COHN and GRUPA program does not exceed 1-2%, whereas for plutonium it reaches ±8%. V. Gruner (GDR) proposed a high-speed method for the electrolytic preparation of uranium and plutonium samples from organic solutions for α-spectrometry. This method allows thin and uniform layers of the sample to be obtained on a large surface area. F. Sus (Czechoslovakia) reported on the results obtained within the framework of the SROK-1 international laboratory experiment. With the participation of laboratories of the Soviet Union, Czechoslovakia, GDR, Poland, Hungary, and Romania, work has been carried out on the determination of the content and isotopic composition of uranium in a solution of VVÉR spent fuel. The errors in determining the uranium concentration amounted to 0.34 and 1 rel. %, respectively, for mass-spectrometric and chemical analyses. Some reports on this subject were devoted to chromatographic and polarographic, and electromigration methods, and also to the method of isotopic dilution.

The report of G. N. Flerov (Soviet Union) concerning the prospects for the development of nuclear physics methods of analysis created great interest. In the report the main attention was focused on new trends for the development of these methods, using laser sources and synchrotron radiation, on the prospects for the use of microtrons in activation analysis, etc.

The reports presented at the Conference reflect the recent achievements in the development of nuclear-physics methods of analysis and are of great scientific and practical interest for specialists.

TWENTY-EIGHTH SESSION OF THE SCIENTIFIC COMMITTEE OF THE UNITED NATIONS ON THE EFFECT OF NUCLEAR RADIATION

A. A. Moiseev

The meeting was held in June 1979 in Vienna (Austria). Fifty-nine experts from 20 countries and certain international organizations participated in its work (IAEA, UNEP, WHO, ICRP, and IARP). The purpose of the Meeting was to start the preparation of the next Report of the Committee to the General Assembly of the United Nations at the beginning of 1981.

The physics subgroup considered eight documents – projects headings of the coming report: models for estimating radiation dosage; technological changes of dosage due to the nature of the radiation background; radon and its decay products; radioactive contamination of the environment due to nuclear explosion tests; radioactive contamination of the environment caused by the production of nuclear power, and occupational irradiation dosage and medical irradiation.

The biological subgroup discussed three documents: the dose-effect relation for radiation-induced cancer; late nontumorous consequences of whole body irradiation, and nonstochastic effects of local irradiation.

The genetics group considered one document – the genetic consequences of irradiation.

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In discussing the documents, the committee paid great attention to modernization of the mathematical models discribing the migration mechanisms of natural and artificial radionuclides in the environment, and the formulation of the external and internal irradiation dosage of the population of the earth and of its individual regions, and the refinement of the transfer parameters of radionuclides through the food chain.

In recent years the Scientific Committee on the Effects of Nuclear Radiation (SCENR) has focused special attention on the estimation of the irradiation dosage to the population due to the technologically increased or changed radiation background of the earth (i.e., due to the use of materials with an elevated content of natural radionuclides in the construction of dwelling houses and industrial buildings; the more intensive use in agriculture of fertilizers containing natural radionuclides; the uses for heating buildings – of natural gas containing radon and its daughter products; the use of water with a high radon content from boreholes for drinking water supply; passenger aircraft flights at higher altitudes, etc.), and also the comparative assessment of nuclear power stations and thermal power stations operating on natural fossil fuel sources (bituminous coal, petroleum, natural gas) as sources of radioactive contamination of the environment. In the documents considered, quite considerable attention was paid to these problems. It was shown that the level of radioactive contamination of the environment due to effluents from thermal power stations (predominantly due to the discharge of uranium, thorium, radium-226, polonium-210 and lead-210) having increased markedly in recent years, despite the improvement of the scrubbing system for ash and fossil fuel combustion products.

The committee took the decision to give data in the special section of the forthcoming report about the combined effects of ionizing radiation and other factors of a nonradiation nature (general principles of interaction, different forms of interaction of a group of factors of a radiation and nonradiation nature: synergism, antagonism, additivity, models and criteria of the most important characteristics of synergism, etc.). It was noted that data about the prolonged effect of different factors at the lower levels which are characteristic for the typical occupational activity of mankind are of the greatest interest (ionizing radiation, shf, and uhf radon, its decay daughter products and dust, etc.).

The main task on the preparation of the report will be carried out at the next meeting of the Scientific Committee, which will be held in September 1980 in Vienna.

INTERNATIONAL CONFERENCE "NEUTRINO-79"

A. A. Pomanskii

The conference, held in June 1979 in Norway, was very well represented in the authority and number of its participants, and in the topics of the problems considered it may be even more timely than the conferences of recent years. Of the 250 participants, 65 specialists represented the USA, ≈ 25 scientists from the Federal Republic of Germany, France, and Italy, and 23 participants represented CERN.

The theme of the conference was the discussion of the state of the unified theory of electromagnetism – weak and strong interactions and the consequences resulting from it. For the first time, at "Neutrino-79" the positive aspects of the unified theory and also certain of its contradictory aspects were discussed in so much detail and so thoroughly. The general conclusion from the data of the conference consisted in that the unified theory gives a fine and plausible explanation to the many different and at first sight, unrelated phenomena. Thus, CP-violation, appearing in kaon decay, according to the unified theory rests in the interaction of heavy bosons X, due to which the union of three forces takes place. The unified theory explains also such a fundamental quantity as the ratio of the number of baryons to the number of γ-quanta in the universe. If we assume CP-violation and the nonconservation of the baryon number during the interaction of massive X-bosons, then the required n_b/n_γ is obtained as equal to 10^{-8}-10^{-10}. Before the appearance of the unified theory, it was assumed that this value is a condition for the existence of the universe. This theory predicts the mass of quarks, the value of the parameter $\sin^2\theta_W$, agreeing with experiment (the average result according to the Conference data is equal to 0.23-0.01).

The unified theory groups fermions into families. Recently, after the discovery of the τ-lepton (its mass according to the data of two groups is 1782^{+2}_{-7} and 1787^{+10}_{-18} MeV), it obviously became possible to speak about the appearance of a third family in addition to the two previously established, although the existence of the t-quark has still to be verified experimentally. It must also be substantiated that ν_τ is an independent fermion.

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From the results given at the conference, it follows that the nonidentity $\nu_\tau \neq \nu_\mu$, $\nu_\tau \neq \bar{\nu}_\mu$, and $\nu_\tau \neq \bar{\nu}_e$ is proven; however, the question of the nonidentity of ν_τ and ν_e still remains open. It is true that the unified theory cannot explain why the families are families and how many of them there may be. The limit here may be obtained from cosmology, as according to the model of the hot universe, not more than four massless fermions can exist. Up to the present time, there are no adequate arguments in favor of the zero mass of the neutrino. If, however, it is nonzero, then the situation is changed sharply. Thus, up to 13 neutrino states with mass ~10 MeV and up to 10^6 with mass ~100 MeV can exist. Therefore, the importance of experiments to determine the mass of the neutrino has now increased considerably. The Canadian specialists presented a paper at the conference in which, using a Si(Li)-detector, they studied the β-decay of tritium introduced into this detector. The result of $m_\nu = 0$ was obtained with one standard deviation of 45 eV and two standard deviations of 70 eV.

The results were given at the conference of the verification of the predictions of the unified theory of effects related with weak interactions in atoms. First and foremost here, reference may be made to the experiments conducted in SLAK (USA) to investigate the nonconservation of parity in deep inelastic scattering of longitudinally polarized electrons by protons and deuterons. The results coincide with the calculations for $\sin^2\theta_W = 0.2$ in both sign and in value. As before, the verification of the rotation of the plane of polarization in bismuth vapor, predicted by the theory, remains unresolved. The results of different groups are contradictory. As concerns the study of the $6^2P_{1/2}-7^2P_{1/2}$ transition in thallium for $\lambda = 293$ nm, quite good agreement is obtained between experiment and theory for $\sin^2\theta_W = 0.25$; however, it will be necessary to improve the accuracy of the data obtained.

One of the principal predictions of the unified theory is baryon instability. The lifetime of the proton depends on the mass of the boson unifying the interaction. Predictions, made on the basis of SU(5) and SU(10) models, give $M_X \simeq 10^{15}$ GeV and $\tau_p \simeq 10^{33\pm1}$ years. At the conference the reports of two groups of USA scientists were presented, concerning plans of underground facilities, of which the basic purpose is to detect proton decays or, at least, to establish a new limit for their lifetime. According to one of the plans, the detector located at a depth of 1500 m of water equivalent, will contain 10,000 tons of water filled into a cubical container with an edge of 21 m. Some 1350 photomultipliers are arranged uniformly over all six surfaces. Recording of decays by the main channel $p - e^+ + \pi^0$ allows the cosmic radiation background to be defined, if the coinciding signals from the photomultipliers are used, recording the Cherenkov emission of the shower from e^+ and the two γ-quanta from the decay of π^0. By using this detector, it will also be possible to investigate the high-energy muons and neutrinos formed in the earth's atmosphere, and in particular to verify the hypothesis about neutrino oscillations. The limit on the difference in mass of the oscillating particles may be reduced to 10^{-2}, which is better by two orders of magnitude than the existing value. The authors assume that in the main experiment they will be able to detect the decay of protons if their lifetime is less than $3 \cdot 10^{33}$ years.

A separate plenary session was devoted to new accelerators. The greatest interest was shown toward LEP (Federal Republic of Germany), a large electron–positron storage ring with particle energies of up to 70 GeV, built in CERN. This considerably exceeds the energy achieved at the present time on PETRA in the Federal Republic of Germany (2 · 19 GeV) and therefore new possibilities are opened up in the investigations of the structure of quarks and leptons. For example, this will make it possible to detect Z^0-besons, the mass of which is less than $M_{Z^0}^{WS}$ 37.4 GeV/$\sin\theta\cos\theta \simeq 90$ GeV and, possibly, also $W^\pm$-besons and Higgs particles. Major physics will be revealed also ahead by the projects for EP-colliding electron (20 GeV) and proton (270 GeV) beams and pp-colliding proton–antiproton beams, which are intended to be brought into operation in 1981. By bringing about the new accelerators, an immense jump can take place in the knowledge of the structure of protons and quarks.

On the problem of solar neutrinos, it was reported that the average value of the experimental solar neutrino flux during measurements since 1970 amounts to 2.2 ± 0.4 SNU (solar neutrino units) as against 1.75 ± 0.4 SNU at the "Neutrino-78" Conference. The standard solar model makes it possible to vary the flux within the limits from 3 to 10 SNU. New experimental data on the velocity of the reaction $^3He + ^4He \rightarrow Be + \gamma$ shows that extrapolation to the energy in the central region of the sun, as assumed previously, has given a rate of formation of ^7Be which is overestimated by a factor of 1.5 and thus an overestimated flux of "boron" neutrinos, leading to the formation of ^{37}Ar from ^{37}Cl. In this connection, the startup of a gallium detector, sensitive to the neutrinos from the principal reactions of the hydrogen cycle, $p + p \rightarrow d + e^+ + \nu_e$, is opportune. A report was made to the conference on the staging plan for conducting a Ga–Ge experiment with 50 tons of gallium. In conducting this experiment, planned to start in 1983, three USA research centers are participating, and also one West German and one Israeli center. It may be said with confidence that the carrying out of this experiment will make possible a correct interpretation of the Cl–Ar experiment and, possibly, it will resolve the problem of the solar neutrinos.

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Part of the reports at the conference was devoted to the experimental data on neutral and charged streams. In many papers, new results were not given but thanks to increasing statistics the accuracy of the experimental data has been increased markedly. For example, for the inclusive processes on isoscalar targets the following results have been obtained: $R = \sigma(\nu \to \nu)/\ \sigma/(\nu \to \mu^-) = 0{,}307 \pm 0{,}008$; $\bar{R} = \sigma(\bar{\nu} \to \bar{\nu})/\sigma(\bar{\nu} \to \mu^+) = 0{,}337 \pm 0{,}025$. The distribution with respect to y at the present time coincides with the standard distribution for $\sin^2\theta_W = 0.25$.

New data were given about the interaction of reactor antineutrinos with deuterons: $\bar{\nu}_e + d \to n + p + \bar{\nu}_e$ (neutral stream) and $\bar{\nu}_e + d \to n + n + e^+$ (charged stream). The value obtained for $\sigma_{n.str.} = (3.8 \pm 0.9) \cdot 10^{-45}$ $c^2/\bar{\nu}_e$ is found to be in agreement with "solution A" according to Hwang and Sakurai terminology but exceeds the prediction of "solution B" by a factor of three. This is a confirmation of the result concerning the singleness of the solution, which was obtained earlier in elastic ν_e $(\bar{\nu}_\mu)$p-scattering and testifies in favor of the Salam–Weinberg model. The cross section with the charges streams $\sigma_{ch.str.} = (1.5 \pm 0.4) \cdot 10^{-45}$ $cm^2/\bar{\nu}_e$ also coincides with the standard model within the limits of error.

In the work carried out on the LAMPF meson factory (Federal Republic of Germany), the interaction of ν_e with deuterons has been studied: $\nu_e + d \to p + e^- + p$. A value has been obtained of $\sigma = (0.56 + 0.16) \cdot 10^{-40}$ cm^2. This is the first study of inverse β-decay with a neutrino (and not with an antineutrino). The reaction $\bar{\nu}_e + p \to e^+ + n$ has also been studied here. The upper limit on the flux $\bar{\nu}_e$ is simultaneously the limit on the possible oscillations of $\bar{\nu}_\mu \to \bar{\nu}_e$. For a typical neutrino momentum of 40 MeV/c and with a distance from the source to the counter of 8 m, we obtain for the nonzero mixing parameter, the limit on the mass of the neutrino < 1 eV in the oscillations

The reports of different groups concerning the "beam dump" experiments created great interest; the preliminary results mainly concern non- and single-muon events. Targets of different density were used and extrapolation was made to density ∞. It may be said with definiteness that the signals recorded were from $(\nu_\mu + \bar{\nu}_\mu)$ and $(\nu_e + \bar{\nu}_e)$. The results coincide with the fact that the flux $\nu_\mu + \bar{\nu}_\mu$ is equal to the flux $\nu_e + \bar{\nu}_e$. No indications of the presence of ν_τ were obtained and their expected flux is small. The results of the experiments of different groups coincide.

Summing up, it may be said that the conference proceeded at a high level and contributed to the further development of predictions about the microworld.

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S. N. Kraitor

DOSIMETRY IN THE CASE OF RADIATION ACCIDENTS*

Reviewed by G. V. Shishkin

The measures for the protection of staff from radiation taken during the designing, construction, and operation of nuclear facilities ensure the radiation safety of staff in such a way that the mean annual radiation dosage as a rule amounts to less than the 5 rem, permitted by radiation safety standards. At the same time, international experience shows that accidents in nuclear facilities, although rare, do occur. In this case, the irradiation of staff may considerably exceed 5 rem. In these cases a rapid and accurate estimate of the dosage received by the victims is required from the dosimetric services, in order to choose the most appropriate methods of medical treatment. At the present time, methods have been developed for determining the emergency radiation dosage. However, until now in the Soviet literature, there has been no sufficiently complete monograph which generalizes and systemmatizes the research experience in this field. The book being reviewed has been issued for precisely this purpose.

The book consists of six chapters. In the first chapter, examples are given of typical radiation accidents of spontaneous chain reactions (SCR) in critical assemblies, research reactors and in systems with solutions of fissile materials. These accidents are accompanied by intense short fluxes of neutron and γ-radiations from the cores and lead to an absorbed radiation dose by the victims of up to several hundreds of rad (sometimes thousands of rad). In this case the radiation is of a nonuniform nature and the dosage differential with respect to height and depth of the body is very important.

The second chapter is devoted to the characteristics of the n-γ-radiation fields at the locations of critical assemblies and reactors, and to the dose distribution in the human body. Great attention is paid in it to the spectral-angular and spatial distribution of the neutron fluence at the locations, as the dependence of the absorbed dose on these factors in the case of neutron irradiation is much stronger than in the case of photon irradiation. The spectra of typical reactors and critical assemblies are given and the nature of their variation with distance from the core is traced. The dosimetric characteristics of the radiation fields at the locations of the facilities are considered in detail, also the effect of scattered radiation, neutron spectra originating at the surface of the human body, and the dose-depth distribution. The contribution is shown of neutrons of different energy groups to the total dose as a function of the type of reactor, the ratio of the neutrino and γ-components to the total dose for different levels of moderator in a liquid assembly, or the thickness of the reflector in a metal assembly. These data are extremely useful both for researchers in the field of radiation safety and shielding physics, as well as for workers in the dosimetric services.

In the third chapter, methods of γ-radiation individual dosimetry are considered (from the conventional photofilm type to the exotic type using paramagnetic resonance). Data are given concerning the international experimental comparisons of accident dosimeters of different types, which have shown that radiothermoluminescent dosimeters, in particular dosimeters based on alumophosphate glasses of the type IKS-A, are the best at present in accuracy, reliability, and convenience of operation.

The fourth chapter is the most "authorative." It describes the currently existing types of individual neutron dosimeters, to the development of which the author made a significant contribution. The track neutron dosimeter DINA is incorporated in the assembly of the GNEIS γ-neutron individual spectrometric accident dosimeter, developed by I. A. Bochvar, I. B. Keirim-Markus, S. N. Kraitor, and L. N. Uspenskii, and which is now the principal dosimeter for the accidental irradiation of personnel, and which has been accepted in the Soviet Union. In addition to measurements of the absorbed dose of thermal, intermediate, and fast neutrons, the GNEIS assembly makes it possible to measure the γ-radiation dose at the body and the $\beta-\gamma$ radiation at the exposed parts of the skin. For this, the GNEIS incorporates a thin plate of IS-7 glass, sealed with a cover layer of thickness 7 mg/cm^2, which simulates the horny epithelial layer. As the determination of the dosage of intermediate and fast neutrons by the DINA dosimeter is quite prolonged, for a rapid first estimate the

*Atomizdat, Moscow (1979), 280 pp., 3 rubles 10 kopecks.

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GNEIS assembly contains an isomeric rhodium detector, for which the measurement of the readings is carried out on a slit rhodium spectrometer.

In the fifth chapter, neutron spectrometry methods are considered for radiation accidents, and which have independent value, as a knowledge of the spectra allows the neutron distribution in the body of the victim to be calculated and the dose to be established most accurately. The detailed discussion of neutron-activation spectrometry methods and the description of spectrometric sets of neutron detectors with fissile nuclides allows these methods to be used in the practical work of the dosimetric services. The greatest attention is paid to the DISNEI and DISNEI-1 neutron spectrometers, developed by the author and coworkers on the basis of the fissile isotopes ^{235}U, ^{238}U, ^{237}Np and threshold detectors based on ^{32}S and ^{27}Al. These spectrometers allow neutron spectra to be measured in the energy range 0.025 eV to 10 MeV in continuous and pulsed irradiation conditions for the investigation of radiation facilities at locations with reactors and critical assemblies, for simulation of accidents, and for the calibration of neutron dosimeters.

In the sixth chapter recommendations are contained for the organization of accident dosimetric monitoring and the examination of victims resulting from radiation accidents based on a system of IKS-A–GNEIS individual accident dosimetric monitoring and with the inclusion of additional methods.

The deficiencies of the book include the extremely brief first chapter and the absence of illustrationmaterial on the mutual disposition of the radiation sources and the victims for radiation accidents. Separate data of this nature have been published in periodicals but collected together they would be very relevant in the mon graph being reviewed. There are individual slight inaccuracies, about which the reviewer has already reported to the publisher. The deficiencies mentioned in no way detract from the great positive importance of the book published by Atomizdat, and which is a valuable and essential textbook for specialists who, to some or other degree, encounter the necessity for recording accident dosage and for simulating radiation accidents.

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