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USSR Report

ENGINEERING AND EQUIPMENT

(FOUO 4/82)

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AERONAUTICAL AND SPACE

UDC 629.7.064.2

ROCKET SYSTEM GAS GENERATORS

Moscow GAZOGENERATORY RAKETNYKH SISTEM in Russian 1981 (signed to press 14 Apr 81) pp 2-4, 152

[Annotation, preface and table of contents from book "Gas Generators for Rocket Systems", by Al'bert Alekseyevich Shishkov and Boris Vasil'yevich Rumyantsev, Izdatel'stvo "Mashinostroyeniye", 1183 copies, 152 pages]

[Text] The book gives a systematic presentation of principal designs, characteristics and peculiarities of working processes in gas generators using chemical fuel (liquid, solid and mixed) that are sources of power and gas jets on flightcraft and in ground-based rocket systems. Methods of experimental development of gas generators are briefly covered.

The book is intended for engineers and designers in the field of rocketry.

Preface

Gas generators are extensively used in rocketry. Their major components are in many ways similar to those of the main rocket engines; however, the working processes in gas generators have important distinguishing features that must be taken into consideration in design and development.

In connection with the development of rocketry and expansion of areas of application of gas generators in recent years, a number of patents and journal articles have been published with results of research on gas-generator devices [Ref. 12]. Condensed information on gas generators is given in handbooks on the principles of rocket engine design [Ref. 2, 3]. However, as a whole, the published materials on gas generators are disconnected, fragmentary and procedurally nonuniform.

In this book the authors attempt to systematize the designs and particulara of working processes of gas generators using different fuels based on major principles of the theory of rocket engines.

The book has five chapters. Chapter 1 presents the basic characteristics of gas-generator devices and the fuel compositions used in them, giving individual attention to methods of model, laboratory and stand tests.

Chapter 2 deals with peculiarities of gas flows in gas generator, gas line and exhaust nozzles, and methods of calculating gasdynamic characteristics of gas generators, and also takes up problems of filtration of gas generation products and gas-thermodynamic processes in the devices that are used.

A survey of materials on the peculiarities of devices and principles of calculation of one-component and two-component liquid gas-generators, as well as gas generators based on fluidized powder fuel is contained in chapter 3.

Chapter 4 is devoted to designs, methods of intraballistic calculation and different possible methods of regulating solid-fuel gas generators (particularly with end-burned charges), including gas generators that can be repeatedly energized. A solution is given for the problem of transient processes with variation of certain parameters.

The last chapter deals with problems of development of a variety of combined gas generators using solid (with separate components), quasi-hybrid and hybrid fuels, vapor-gas generators, and gas generators for straight-flow rocket engines and turborocket engines; engineering methods are given for calculating the major characteristics of some gas generators.

Chapters 1, 4 and 5 were written by both authors jointly, chapter 2 -- by A. A. Shishkov and chapter 3 -- by B. V. Rumyantsev.

The authors are sincerely grateful to the reviewer, Doctor of Technical Sciences A. P. Tishin, for constructive recommendations that have improved all parts of the manuscript, and to Candidate of Technical Sciences M. Ye. Yevgen'yev for useful advice in solving the problems in various sections.
We would be grateful to readers for comments and suggestions, which should be sent to the publishers: Mashinostroyeniye, 107076, Moscow, Stromynskiy per. 4.

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CSO: 1861/192

NUCLEAR ENERGY

UDC 621.039.616

FAST HYDROGEN ATOM INJECTORS

Moscow INZHEKTORY BYSTRYKH ATOMOV VODORODA in Russian 1981 pp 2-11, 18, 152

[Excerpts from the book "Fast Hydrogen Atom Injectors", by N. N. Semashko, A. N. Vladimirov, V. V. Kuznetsov, V. M. Kulygin and A. A. Panasenkov, Energoizdat, 168 pages]

[Excerpts]

Annotation

Equipment for producing fast hydrogen (deuterium) atoms is described for heating plasma to thermonuclear temperatures in closed and open magnetic systems. Powerful injectors for thermonuclear research, demonstration and reactor installations are considered. Methods of producing flows of low-energy atoms based on charge exchange of positive ions and flows of medium-energy atoms based on conversion of negative ions are described. The book is intended for scientific workers and engineers working in the field of controlled thremonuclear fusion, accelerator technology and space technology. It may be useful to students and graduate students of physical engineering specialties.

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CHAPTER 1

INJECTION OF FAST ATOMS, METHOD OF CREATING THERMONUCLEAR PLASMA

Injection of fast hydrogen (deuterium) atoms into magnetic systems is now regarded as one of the main methods of creating a thermonuclear plasma. This method was born during the early 1960s when the Phoenix [1], Ogra-2 [2] and 2X [3]--open traps designed to store a plasma formed upon ionization of fast atoms entering an installation--appeared. The initial captures made by the Lorentz ionization and an exponential increase of density then occurred due to ionization upon collisions of the beam of atoms with particles of the formed plasma.

The injectors of that time were comparatively low-power; their rapid development was largely stimulated by interest in additional heating of plasma in tokamaks. Installations of this type, developed at IAE [Institute of Atomic Energy] imeni I. V. Kurchatov and that demonstrated good capabilities in producing a dense hot plasma, spread rapidly throughout the laboratories of the world. The corresponding intensive research led to conclusions on the inadequacy of Joule heating, ordinary for tokamaks, and the need for additional input of energy, for example, by high-frequency heating or injection of powerful beams of fast hydrogen atoms. The results of efforts to develop injectors meeting the needs advanced by tokamaks are presented in Figure 1.1, which illustrates the sharp increase of flows of atoms injected into installations throughout the decade from 1969 through 1979.

These results were reflected in plasma parameters achieved in tokamaks with injectors--a record value of β (ratio of plasma pressure to magnetic field pressure) in the T-11 [4] and an absolute ion temperature record in the PLT [5]. The main thing is that the classical mechanism of energy transfer from a beam trapped in a tokamak to a plasma is confirmed and it is shown that its temperature increases in proportion to the output of the injected beam; no additional losses related to the presence of a group of fast ions occur.

The appearance of powerful injection systems again aroused flagging interest in open traps with injection since the hopeful results obtained in the 2XIIB [6] permit one to talk about a new quality of an old combination.

Thus, experiments of the last few years confirmed the practical importance of the injection method to solve the problem of controlled thermonuclear fusion; injectors of fast atoms will apparently be one of the main parts of the demonstration thermonuclear reactor of the near future.

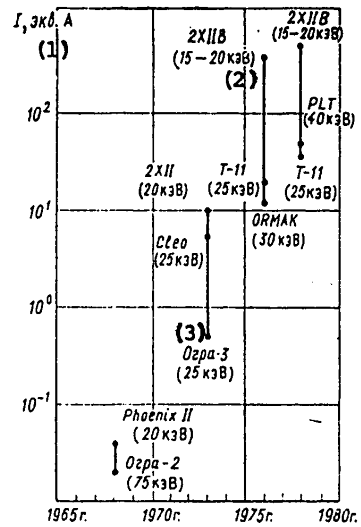

Figure 1.1. Growth of Flows of Atoms Injected Into Installations

Key:
1. Equivalent amperes
2. keV
3. Ogra

1.1. Types of Thermonuclear Installations With Injection

As already noted above, fast atoms are injected into both open and closed magnetic systems. It is assumed that both methods may lead to the goal--development of a thermonuclear reactor, initially a demonstration reactor and then a commercial reactor that is economically advantageous and capable of competing with other energy sources. Tokamaks are presently the most advanced representatives of closed systems toward this goal.

According to modern concepts, at least one of three varieties of a thermonuclear reactor based on a tokamak with injection can be achieved: 1) a reactor with ignition, 2) a reactor with constant heating and 3) a two-component reactor.

The first version is the most preferable. In this case the injectors should be switched on only during the initial operating cycle. By using them, the plasma temperature is brought up to the required temperature for "ignition" of an intensive thermonuclear reaction, after which the injectors are switched off, while "combustion" is self-sustained due to the energy of α-particles formed as a result of the reaction itself. It is clear that the α-particles must not leave the system without having released their energy to the plasma and that the rate of energy loss from the plasma to the reactor wall not exceed some maximum value to realize this attractive possibility.

If heat transfer is so high that the reaction is not self-sustaining, the reactor will be able to operate only provided there is continuous makeup of the departing energy due to continuous injection of high-energy particles into the plasma, i.e., we have a system with continuous heating in this case.

It should be noted that practical realization of a thermonuclear reactor can be an intermediate version of these two varieties: a powerful heating injection system at the beginning of the operating cycle brings the reactor "almost to ignition," after which the required energy balance is maintained by a comparatively weak control injector. This scheme is convenient from the viewpoint of monitoring the operating mode, since it permits control of reactor "criticality" by changing the output of the injected flow.

Finally, the third possible variety--a two-component tokamak reactor [7]--is a system in which energy is produced in reactions occurring upon collisions of high-energy injected particles with particles of a comparatively cold plasma that serves as the target bombarded by a beam impinging on it rather than in a thermonuclear reaction, i.e., not in the combination of nuclear reactions in a plasma heated to a high temperature. A two-component tokamak, being a powerful neutron generator, can be used specifically not only in the form of a purely plasma-beam system but also as a hybrid reactor in whose jacket additional energy is produced as a result of neutron irradiation of fissionable materials with simultaneous breeding of a nuclear fuel--plutonium.

If we can talk about a plasma heating method alternative to injection in tokamaks--introduction of a flow of high-frequency electromagnetic radiation, then the practically interesting idea of using an open trap without injectors apparently does not exist at the given moment. Moreover, as already noted, it is the modern level of developments in the field of injection systems that made possible the appearance of such very promising proposals as systems with reversal of the magnetic field and ambipolar traps.

The idea of reversing the magnetic field inside an open trap has existed since the time of the astron [8] in which it was suggested that this reversal be accomplished by creating a conducting layer of fast electrons injected into the trap. Realization of this old idea on a new base--due to the dimagnetism of a plasma created in a trap with a powerful injector--now appears quite realistic. The results of experiments on the 2XIIB [6] indicate that total repulsion of the magnetic field from the center of a plasma has now been fully achieved in this installation. If a magnetic field reversed to the external field can be produced in the center of the plasma, then a qualitatively new field configuration is achieved--the magnetic system is closed due to the dimagnetic current in the plasma itself (Figure 1.2).

An ambipolar trap was suggested comparatively recently at Novosibirsk by G. I. Dimov et al [9] and independently by Fowler and Logan and is being actively analyzed in both our country and abroad. It consists of three parts--two small traps into which are injected high-energy particles (1 MeV) are joined to an ordinary open trap, while medium-energy particles (10 keV) are injected into the main trap. By corresponding selection of the injection currents, one can achieve a plasma density in the outer traps n_k higher than in the central

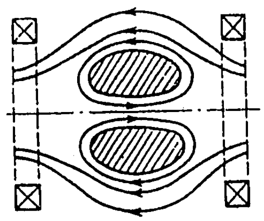

Figure 1.2. Reversal of Magnetic Field in a Reflected Trap and Formation of a Closed Configuration

trap n_0 (Figure 1.3) and in this case the ambipolar electric field forms a potential barrier in the axial direction to ions located in the central trap.

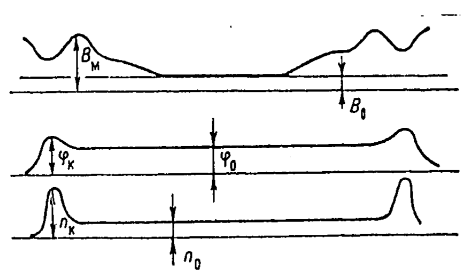

Figure 1.3. Distribution of Magnetic Field, Potential and Density of Plasma Along Axis of Ambipolar Trap

1.2. Requirements on Injection Systems

The requirements placed on an injection system during design are determined by the type, designation and scales of the installation. Let us enumerate the main characteristics primarily stipulated by these requirements: 1) the output of the flow of atoms injected into the installation, 2) the total injected flow of atoms, 3) the energy of the beam particles, 4) the injection pulse length, 5) the geometric characteristics of the beam of atoms, 6) the accompanying flow of cold gas, 7) the energy efficiency of the injection system and 8) the operating reliability and service life of the system (including its efficiency under conditions of neutron and gamma-radiation).

It seems at first glance that one could limit oneself only to two of the first three beam characteristics. However, depending on the type of installation for which the injector is being designed, pair combinations different for them emerge to the forefront. Thus, the specific parameter in a heated tokamak (with ignition or with constant heating) is the output of the beam injected into the installation. The particle energy must then be selected so as to provide the particles with sufficiently deep penetration into the plasma and capture more or less uniform through the pinch cross-section. It is natural that in this case the total injected flow of atoms is a deviation. If the injector is designed for a two-component tokamak, then one must first select an energy so as to achieve the maximum yield of nuclear reactions with established energy distribution of trapped fast particles. The design intensity of the reaction with selected injection energy permits one to determine the total injected current. Thus, beam intensity is now a derivative.

The remaining parameters can remain unchanged with formulation of the initial requirements on the injector.

Table 1.1. Parameters of Injection Systems of Some Existing Installations and Those Under Construction

Установка (1)	Мощность инжекции P_0, МВт (2)	Энергия атомов E_0, кэВ (3)	Число инжекторов (4)	Мощность инжектора, МВт (5)	Число источников в инжекторе (6)	Ионный ток с источника I^+, А (7)	Длительность импульса, с (8)	Длина тракта инжектора, м (9)	Размеры входной апертуры, см (10)
PLT	4	40	4—(6)	0,6—0,9	1	60	0,3	3,6	20
D-III	10	50	16	0,6—0,9	1	60	1,0	4,0	15×30
PDX	4—10	40	4—6	0,6—0,9	1	60	0,3	3,6	20
TFTR	20	120	4—(6)	5	3	70	0,5	8,5	40×70
T-10M	4,5	40/80	2—(4)	2,5	3	35	1,5	6	20×80

Key:
1. Installation
2. Injection power P_0, MW
3. Energy of atoms E_0, keV
4. Number of injectors
5. Injector power, MW
6. Number of sources in injector
7. Ion current from source I^+, A
8. Pulse length, s
9. Length of injector channel, meters
10. Dimensions of inlet aperture, cm

Injection systems capable of injecting flows of hydrogen or deuterium atoms with power of 10-20 MW with particle energy of 40-120 keV and injection pulse length up to several seconds are required for research installations of the next few years. Some parameters of these systems in installations already operating and those being designed are presented in Table 1.1 [10]. The devices of the next generation, which will actually be demonstration reactors, will be designed for outputs of injected fluxes on the order of hundreds of megawatts. The efficiency of the injection systems will be of determining significance for these installations.

1.3. Design Principles of Injection Systems

The physical prinicple of producing energetic beams of atoms, used in modern injectors, includes acceleration of ions with subsequent conversion of them to atoms. This principle can be realized by two schemes--through positive or negative ions. Specific selection of one or another scheme is determined by the desire to produce the maximum energy efficiency of the injector. The fact is that it is easier to generate positive hydrogen ions than negative ions, but the charge exchange cross section of positive ions on the target, which determines the coefficient of converting them to atoms, decreases strongly with an increase of energy (Figure 1.4). At the same time, the "stripping" cross-section of negative ions that converts them to atoms is weakly dependent on energy up to very large values (Figure 1.5). Therefore, it is much more advantageous with respect to energy to use positive ions to produce comparatively low-energy fluxes of atoms and negative ions if high-energy atomic fluxes are required. The dependence of the energy efficiency of injectors designed according to the two schemes discussed above are presented in Figure 1.6.

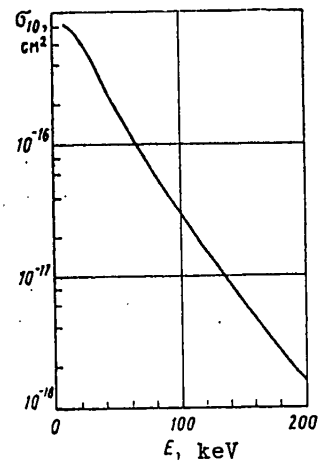

Figure 1.4. Dependence of Resonance Charge Exchange Cross-Section of Fast Hydrogen Ions to Atoms on Energy

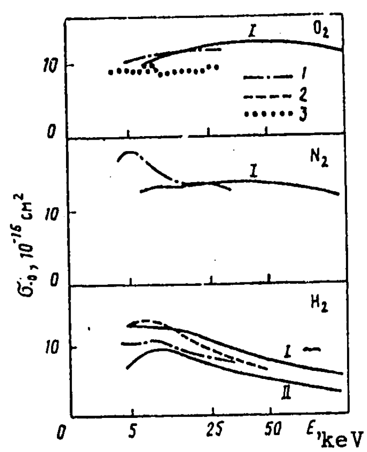

Figure 1.5. Dependence of "Stripping" Cross-Section of Fast-Negative Hydrogen Ions on Hydrogen, Nitrogen and Oxygen Molecules: 1--quasi-classical approximation; II--Born approximation; 1-3--experimental data from [11]

Technical embodiment of one or another physical scheme is always related both to fundamental limits imposed by the laws of nature and to real technological capabilities. For example, the restrictions of the typical dimensions of the ion-optical system (IOS) of the ion source--the most important component of the injector--are directly related to the maximum achievable precision of manufacturing its parts on metal-cutting lathes, to the mechanical strength of the material and to the capabilities of heat dissipation. Thus, the size of the accelerating gap is determined by its high-voltage strength, which is very strongly dependent on the quality of machining the electrode surfaces. These restrictions in turn limit the ion current density from the source at which the beam can be formed by the ion-optical system in a satisfactory manner. An ion-emission surface must be developed, the number and dimensions of the input apertures in the installations must be increased and so on to produce large fluxes of fast atoms.

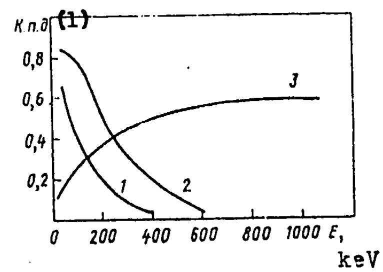

Figure 1.6. Dependence of Injector Efficiency on Energy E_0 for Deuterium: 1--injector based on direct charge exchange of positive ions; 2--the same injector with recovery (85 percent) of ion energy; 3--injector using negative ions with double charge exchange on a sodium target

Key:
1. Efficiency
2. keV

Analysis of similar relationships and also the requirements of operating reliability and operating convenience lead to formulation of the following structural principle of designing injection systems for thermonuclear installations: the injection system should consist of several injection zones, each of which combines several injectors having self-contained vacuum systems; several ion sources with self-contained electric power supply circuits and a working gas supply are installed in each injector. This situation is actually the modular principle that permits independent optimization of the parameters of separate components and that increases the operating reliability of the entire system as a whole, since a failed or worn-out component can be repaired or replaced without interrupting the functioning of all the remaining components.

The sodium charge exchange target (Figure 6.7) consists of a steam generator 1 with supersonic nozzle.

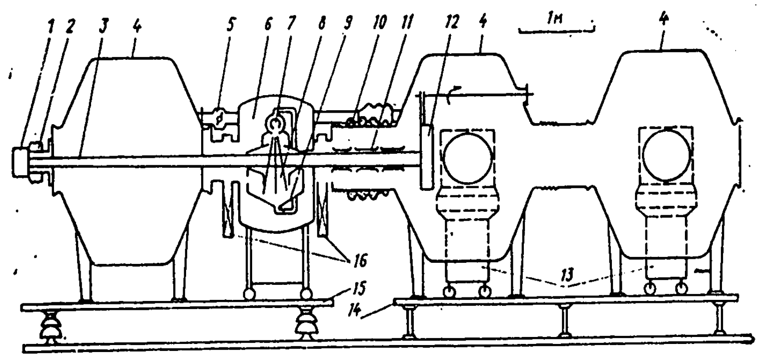

Figure 2.8. Experimental MIN Injector: 1--positive ion source; 2--insulator; 3--ion beam; 4--vacuum chamber; 5--evacuation pipe with slide valve; 6--vacuum chamber of sodium target; 7--steam generator; 8--supersonic jet of sodium vapors; 9--condenser; 10--insulator; 11--preacceleration system; 12--diagnostic detector; 13--vacuum pumps; 14--platform under ground potential; 15--platform under potential of 100 kV; 16--vacuum slide valves

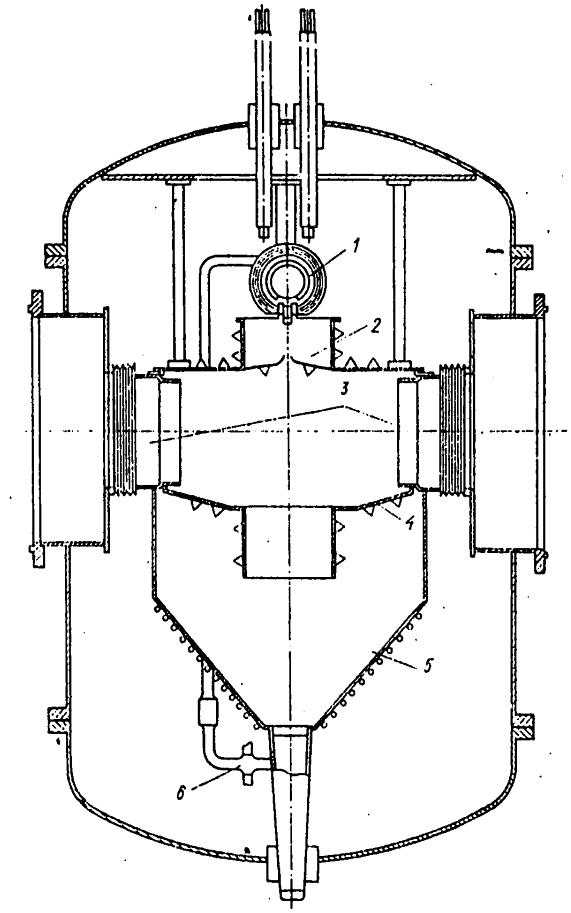

Figure 6.7. Design of Continuous Charge Exchange Target: 1--steam generator; 2--separator; 3--windows for beam travel; 4--intermediate condenser; 5--condenser; 6--electromagnetic pump

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CSO: 1861/188

UDC 621.039.516.22

USE OF BURNING ABSORBERS IN RBMK

Moscow ATOMNAYA ENERGIYA in Russian Vol 51, No 5, Nov 81
(manuscript received 13 Feb 81) pp 304-307

[Article by V. K. Vikulov, S. A. Dorshakov, V. I. Ilyukhin and Yu. I. Mityayev]

[Text] The intensive development of nuclear power is making even small improvements in power reactor characteristics significant. One such improvement in the RBMK [1] fuel cycle is enhancing uranium enrichment. This enables a significant increase in its burnup, which in turn noticeably lowers the need for natural uranium and reduces the fuel component of electric power production cost. However, it is not always possible to enhance uranium enrichment. In the continuous fuel recharging mode, and when bringing the reactor to this mode, the fuel channels of the core have a different power output (burnup), from zero (fresh) to that of the spent ones. Therefore, the content of fissile isotopes in these channels is unequal, causing an additional power variation factor due to overload [2]. Enhancing the enrichment raises the overload factor, which can lead to unevenness of power distribution that is unacceptable in terms of the cooling conditions. Such a constraint is especially significant for reactors with high specific fuel power, such as the RBMK-1500 and the RBMKP. One possible method of overcoming such a limitation is to use a self-shielded burning absorber inserted as an absorbing rod in the fresh fuel assembly during loading into the reactor. Implementing this method is aided by the fact that the design of individual RBMK fuel assemblies provides a central hole 6.5 mm [3] in diameter.

Usually, burning absorbers [4-7] are used to lower the reactivity, compensated by mobile control elements, where its burnup reserve is large. In such a case, the reactor operates in the continuous recharging mode, and the compensating capacity of the regulation elements is known to be sufficient. The capacity for local balancing of the power distribution is limited, however, since each of them affects a rather large region of the neutron field that includes both the freshly loaded and burned up fuel channels, and the effect of the control elements on fuel channel power is insufficiently differentiated. Burning absorbers are thus used in this case only for local balancing of the power distribution.

The authors were faced with two tasks: to determine the density of the absorbing material and diameter of the absorbing rods inserted in the center of the fuel assembly which will provide maximum reduction in the power of the loaded fuel channels with the least total neutron absorption in the absorber during the time of operation; and, given the existing power engineering constraints on the maximum fuel channel power, to find those conditions of channel utilization allowing enhancement of enrichment and burnup of uranium, resulting in improved economic indices of the fuel cycle.

The theoretical approach to solving the first problem, that of choosing the optimal burning absorber with the best coordination of its burnup with that of the fuel, is the same as that given in works [6,7]; however, the method of solution is different. It is based on using the VOR [2] computer program, enabling calculation of the spatial distribution of the thermal neutrons of the fuel channel cell in a P_3 approximation with part of the calculation grid in the region of the absorbing rods (such that the distance between the calculation nodes is much less than the neutron path length). It thus becomes possible to directly calculate the "burning" [7] of the rods using a small time step, with simultaneous burnup of the uranium, and correspondingly allow for the change in the neutron-physical characteristics of the fuel channel cells with the run*.

The solution of the second problem, requiring a heterogenous design of the reactor, used a two-dimensional diffusion grid HODIHER program [8], whose two-group cell macrosections are calculated in the first part.

The experiments showed that when absorbing rods 5.6 mm in diameter are inserted in the central fuel assembly hole, the power of the fresh fuel channels declines by 10-15%. In both experiments and calculations the absorber was gadolinium oxide GD_2O_3, having two strongly absorbing isotopes ^{155}Gd and ^{157}Gd. If an identical uranium and absorber burnup rate is assumed, then the reactivity and power of the fresh fuel channels and the absorbing rods will remain consistent until the absorber burns up. The power distribution variation factor in the reactor will thus drop by the same 10-15%. In practice, it is impossible to maintain an identical uranium and absorber burnup rate during the entire burning time of the absorbing rods. However, it follows from the results of work [7] and this research, that the discrepancy in the burnup rates can be reduced to a minimum given a certain gadolinium concentration. This is confirmed by graphs of Fig. 1, illustrating the varying effect of the self-shielded burning absorbers on the RBMK fuel channel reactivity in the process of simultaneous burning of uranium and gadolium. Uranium enrichment in the fuel channels is 2.6%. Given a low

* The authors first used such a calculation method in 1966, in designing the first loading of the reactor of the second stage of the BAES power plant. Experiments during the physical start-up confirmed the basic calculation results.

absorber concentration, the burnup of the rods is faster than that of the uranium, and a fast growth in reactivity, and thus power (curve 2), is observed. On the other hand, given a high concentration of gadolinium, the rods burn too slowly (curve 4), and although the maximum fuel channel power N_{max} declines (it is defined by the initial point), the reactivity loss is too great, which unjustifiably reduces the uranium burnup. There is obviously an absorber concentration at which the reactivity loss will be minimal (curve 3) given the same action on N_{max}. Calculations show that for RBMK fuel channels the optimum absorber concentration corresponds to a Gd_2O_3 density of 0.4-0.5 g/cm^3, and is almost independent of the rod diameter if it changes in the range of 3-12 mm. The initial influence of the rods on the fuel channel reactivity and power is basically determined by their diameter, and increases with it.

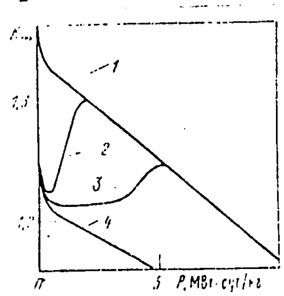
Under conditions of continuous recharging of the fuel channels and burning absorbers, the diameter of the rods with an optimal absorber concentration unambiguously determines such mutually connected characteristics as N_{max}, rod burnup time t_0 and uranium burnup time P. The greater the diameter, the lower N_{max} and P and the greater t_0. Spatial calculations of the reactor have been performed to establish the relation between such characteristics. Fig. 2 gives curves for determining the possibility of lowering N_{max} and the relative underburning of uranium when using rods of different diameter in fuel channels with 2.6% enrichment. The calculations were performed according to an iteration scheme worked out by the authors. Assuming a continuous recharging mode, it allows for underburning of the spent uranium given the use of absorbing rods. The power of the fuel channels containing the rods is assumed equal to N_{max}, and does not change until their complete burnup. This enables determination of the fuel channel run, t_0, and thus the proportion of the fuel channels containing the absorbing rods; i.e., formulation of the reactor composite lattice whose calculation is used to find N_{max}.

The calculations done with allowance for the different diameter of the rods and the enrichment of the recharged uranium showed the possibility of a maximum increase in its enrichment, given a predetermined constraint on the maximum fuel channel power depending on the diameter. It is more convenient to use the power distribution variation factor K_E or overload factor K_{ov} than N_{max} as the constraint. It can be seen from Fig. 3 that despite a certain loss of uranium burnup due to the additional neutron absorption resulting from the use of rods, its enrichment and burnup increase noticeably. Naturally, less uranium burns up in this case than in the absence of the constraint (curves 1, 2).

These results describe only the technical aspect of enhancing uranium enrichment and burnup, and are still insufficient for a clear conclusion on the economic advisability of using absorbing rods. Data are needed on the economic and natural indices of the fuel cycle. To provide them, the calculations of the relative change in fuel components of electric power production cost and adjusted costs for producing electric power, and the uranium fuel rate required for annual make-up of the reactor, have been done as a function of the rod diameter. The calculation results show (Fig. 4) that in the RBMK-1500 and RBMKP it is advisable to use absorbing rods up to 10 mm in diameter. The greatest technical and economic effect is achieved using rods 4-6 mm in diameter, which is very important from the practical standpoint, since it enables their installation into currently manufactured fuel assemblies without any design modifications. The effect is greater the more rigid the constraint on power distribution variation. For example, at $K_{ov} = 1.27$, the fuel components of power production cost and adjusted costs for electric power production can be reduced by 7-8 and around 2%, respectively; 2-2.5% less natural uranium is consumed.

Finally, it is interesting to point out certain results of a calculated study of the effect of absorbing rods on the space-energy distribution of thermal neutrons in the fuel channel cell. This effect, studied using the NEKTAR [9] program, turns out to be negligible (cf. table), which makes it possible to simplify the rod burnup calculations and preparation of the two group constants. In particular, it turns out to be possible to use the single-rate VOR program, to whose sections are added the necessary thermalization corrections. Deformation of the neutron distribution due to the use of rods can cause a relative increase in the power of the fuel elements of the outer row of the fuel assembly; however at a 5-6 mm diameter it does not exceed 2%.

The results given are also valid for the use of other self-shielded thermal neutron absorbers, for example those containing ^{113}Cd or ^{149}Sm.



P, MWt. day/kg

Fig. 1. Dependence of the neutron breeding factor on an infinite lattice in an RBMK fuel channel K_∞ on uranium burnup P at a Gd_2O_3 density of 0 (1), 0.2 (2), 0.45 (3) and 1 g/cm^3 (4).

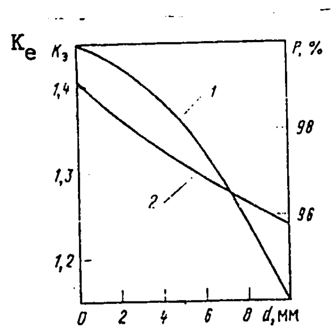


Fig. 2. Effect of the absorbing rod diameter on uranium burnup (1) and power distribution variation factor (2).

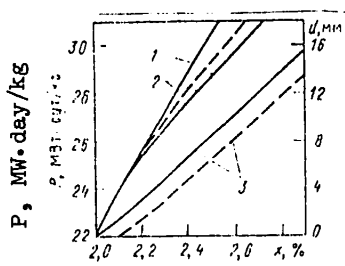


Fig. 3. Dependence of spent uranium burnup on its enrichment x in the absence (1) and presence (2) of a constraint on N_{max}, ensured by the corresponding diameter of the absorbing rods (3). K_{ov} = 1.27 and 1.3, respectively

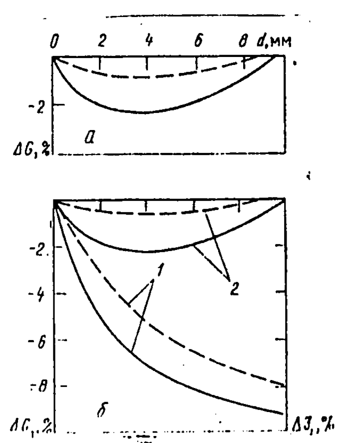


Fig. 4. Effect of the absorbing rod diameter on the relative change in fuel cycle indices: a - natural uranium consumption G; b - fuel component of electric power production cost C_T (1) and adjusted costs Z_T (2). K_{ov} = 1.27 and 1.3, respectively

TABLE

Distribution of thermal neutron fluxes F* (relative units) and neutron gas temperature $T_{n \cdot g}$ (K) in a fuel channel cell as a function of absorbing rod diameter (density of Cd_2O_3 0.45 g/cm^3).

d, мм	a. Твэлы **		b. Вода		c. Труба канала		d. Графит	
	Ф	$T_{н.г}$	Ф	$T_{н.г}$	Ф	$T_{н.г}$	Ф	$T_{н.г}$
0	0,598/0,511	858/848	0,668	762	0,805	811	1,039	847
4	0,590/0,484	861/856	0,652	767	0,800	812	1,041	848
6	0,586/0,471	862/860	0,644	768	0,799	813	1,042	849
10	0,580/0,449	863/862	0,633	770	0,796	814	1,043	849

a. **Fuel elements b. Water c. Channel pipe d. Graphite

* The average values by volume of the indicated elements are given for fluxes normalized to the average value by cell volume.

** The data in the numerator are for the 12 outer fuel elements; those in the denominator, for the 6 inner fuel elements.

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CHOICE OF EXTRACTORS AND FILTERS FOR PLANTS REPROCESSING SPENT FUEL ELEMENTS OF ATOMIC POWER STATIONS

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[Article by A. M. Nudel', A. P. Roshchin and V. V. Dolgov]

[Text] The chemical conditions of processing spent nuclear power plant effluents have specific equipment requirements, such as nuclear safety, remote control servicing and maintenance, possibility of decontamination, air-tightness, a minimum of radioactive wastes, etc. This complicates the design of the equipment and increases its cost and operational expenditures. Consequently, equipment design significantly influences the effectiveness of the radio-chemical plants being examined here. In principle, the optimal construction or type of equipment for given conditions may be selected on the basis of technical and economic calculations. However, the necessity to take into consideration complex and contradictory requirements, and the absence of reliable mathematical models of the process makes it necessary to introduce varying and often subjective hypotheses that lower the reliability of such calculations. Even with adequate information on the process as a whole, the reliability of the comparative evaluation of equipment may be improved through application of empirical methodology. For example, a system of criteria for selecting extractors was suggested [1, 2]: value intervals for each criterion selected in advance are evaluated by a specific number of points and the sum of points received by each compared piece of equipment determines its advantages.

The basic inadequacy of present methodologies is that their authors attempt to derive an absolute evaluation of equipment that does not depend on what it is being compared with; consequently, after a single comparison, the criteria tables themselves become unnecessary since the evaluation remains unchanged. In addition, it is impossible to suggest a system that would permit an intuitive determination of the selected interval values of the criteria and the number of points they were assigned. One more substantial inadequacy is that it is impossible to determine the expediency of utilizing different types of equipment in a single technological chain. For example, the ability of an extractor to handle unclarified solutions is evaluated by the highest mark 5, while inability to do so receives no mark at all [2]. However an arrangement including a filter and high-efficiency extractor may turn out to be more advantageous than a less effective extractor without solution clarification. It's also impossible to compare versions in which different types of equipment are used in the extraction and reextraction stage.

The following principles serve as the basis of the suggested methodology: equipment for a specific technological process is compared; the comparison of equipment assures that required technological parameters will be met and their operational characteristics learned and verified. Also taken into consideration is its connection to related equipment in the technological chain. The final evaluation of the equipment is not absolute but its superiority or inferiority is determined only in relation to other equipment being compared. The methodology has been developed to compare extractors and filters--the basic tools utilized in modern radio-chemical reprocessing of atomic power plant fuel elements. Other equipment may be compared similarly.

Basic criteria influencing effectiveness (Tables 1, 2) were selected in advance. Selection was based on known and generally accepted evaluations recommended for technical/economic calculations. Each compared piece of equipment is characterized by an absolute value on the basis of any criterion but its link with related equipment in the technological chain is characterized by criteria that take into consideration the value of auxilliary equipment for transmitting the effluents and operating expenditures. Criteria are ranged by their value and rated by coefficient r. Evaluation of the equipment's water phase B according to any criterion is determined by taking into consideration absolute values of the latter with regard to the instruments being compared

$$B_{pi} = \frac{N_p}{N^i_{max}} r_i,$$

where N_p - is the value of the criterion for the instrument being examined; N^i_p - is the highest criterion among the instruments being compared; p - the number of instrument being examined; i - is the ordinal criterion number.

Table 1

Filter Evaluation Criteria

	Критерий Criterion	r_i	$R_i \cdot 10^2$
1	Средняя скорость фильтрования *	2,46	1,82
2	Число фильтров, необходимое для обеспечения непрерывной работы отделения фильтрования с заданной производительностью	4,87	3,61
3	Объем сбросных растворов и пульпы промытого осадка, выдаваемых на переработку отходов **	5,35	3,97
4	Поверхность фильтрующей перегородки, требующей замены, в расчете на 8 тыс. ч работы фильтра	5,37	3,98
5	Объем вспомогательных растворов ***	5,49	4,1
6	Объем сбросных растворов и пульпы осадка, выдаваемых на переработку перед локализацией **	6,49	4,8
7	Годовые затраты на эксплуатацию основного и вспомогательного оборудования	7,34	5,44
8	Стоимость фильтра (суммарная стоимость фильтров отделения)	7,46	5,53
9	Высота производственных помещений	9,87	7,32
10	Площадь занимаемых производственных помещений	9,96	7,38
11	Стоимость вспомогательного оборудования (насосы для перекачки растворов и суспензий внутри отделения)	11,0	8,15
12	Затраты на подготовку растворов к фильтрованию (флоккулянты, вспомогательные фильтрующие вещества) ***	11,33	8,4
13	Количество воздуха, сбрасываемого из отделения на очистку	11,74	8,7
14	Стоимость системы КИПиА	11,79	8,74
15	Стоимость емкостной аппаратуры, необходимой для нормальной работы фильтров	11,81	8,76
16	Стоимость контейнеров и приспособлений для дистанционной замены дефектных узлов и деталей	12,55	9,31

* При расчете относительной оценки критерия используется величина, обратная его абсолютному значению; ** на 1 м³ фильтрата; *** на 1 м исходной суспензии.

Key follows on next page

Key to Table 1

1. Average filtration rate*
2. Number of filters necessary to assure filtration department's continuous handling of assigned production
3. Volume of effluent solutions and washed sediment pulp provided to waste processing**
4. Filtering partition surface requiring replacement calculated for eight thousand hours of operation per filter
5. Volume of auxilliary solutions***
6. Volume of effluent solutions and pulp sediment provided to reprocessing prior to localization**
7. Annual expenditures for operational and auxilliary equipment
8. Filter cost (total cost of filters for department)
9. Height of production premises
10. Total area used by production premises
11. Cost of auxilliary equipment (pumps to transfer solutions and suspensions inside the department)
12. Expenditures for preparation of filtration solutions (flocculants, auxilliary filtering substances)***
13. Amount of air discharged for cleaning by department
14. Cost of the control and measurement instrumentation and automation system.
15. Cost of storage equipment necessary for normal filter operation
16. Cost of containers and equipment for remote substitution of defective units and parts

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* When calculating the relative evaluation of the criterion, a value inverse to its absolute value is utilized

** Per 1 m^3 of filtrate

***For 1 m of initial suspension

Table 2

Extractor Evaluation Criteria

	Критерий Criterion	r_i	$R_i \cdot 10^2$
1	Ресурс работы аппарата до капитального ремонта *	5,76	2,49
2	Время пребывания фаз в аппарате	6,08	2,63
3	Механический унос органической фазы при номинальной производительности	6,12	2,65
4	Число аппаратов, необходимых для заданной производительности	7,79	3,37
5	Продолжительность работы аппарата до зачистки при переработке растворов, содержащих твердую фазу *	8,21	3,55
6	Минимально допустимая разность плотности фаз	8,43	3,65
7	Время выхода аппарата на стационарный режим после остановки на 12 ч и более	9,18	3,97
8	Число текущих и планово-предупредительных ремонтов аппарата (аппаратов) в расчете на 8 тыс. ч работы	9,45	4,1
9	Длительность ремонта аппарата в случае аварийного выхода из строя его основных узлов	9,75	4,22
10	Количество органической фазы в аппарате (аппаратах)	10,26	4,44
11	Количество сжатого воздуха, поступающего на очистку при работе аппарата (аппаратов)	11,37	4,92
12	Минимально допустимое отношение потоков органической и водной фаз	11,38	4,93
13	Минимально допустимое отношение потоков водной и органической фаз	11,98	5,19
14	Затраты (годовые) на эксплуатацию основных и вспомогательных аппаратов, системы автоматического управления и регулирования	12,23	5,29
15	Минимально необходимая производственная площадь для размещения аппарата (аппаратов)	12,85	5,56
16	Минимальная высота производственных помещений для размещения аппарата (аппаратов)	13,46	5,83
17	Стоимость аппарата (аппаратов)	13,85	6,0
18	Стоимость вспомогательного оборудования, контактирующего с радиоактивными растворами (пульсаторы, насосы и т. п.)	14,33	6,2
19	Стоимость системы автоматического регулирования и управления работой аппарата (аппаратов)	15,23	6,59
20	Стоимость контейнеров и других приспособлений для дистанционного монтажа и демонтажа аппарата, его узлов и вспомогательного оборудования	16,63	7,2
21	Расход электроэнергии при работе аппарата (аппаратов)	16,66	7,21

Key follows on next page

* * При расчете относительной оценки критерия используется величина, обратная его абсолютному значению.

Key to Table 2

1. Operative life of equipment prior to overhaul
2. Time that phases remain in equipment
3. Mechanical entrainment of organic phase at nominal capacity
4. Number of devices necessary for assigned production scale
5. Length of equipment operation prior to stripping when reprocessing solutions with solid phase content.
6. Minimally permissible difference in phase density*
7. Time for equipment to reach steady operation after shutdown of 12 hours or more.
8. Number of routine and preventive maintenance service calls of equipment (devices), calculated for eight thousand hours of operation.
9. Duration of equipment repair in the instance of accidental failure of its basic units.
10. Organic phase quantity in the equipment (devices).
11. Amount of condensed air released for cleaning when equipment is in operation.
12. Minimum permissible ratio of organic and water phase flows.
13. Minimum permissible ratio of water and organic phase flows.
14. Annual expenditures for operating basic and auxilliary equipment, and automatic operation and control systems.
15. Minimum production area necessary to set up equipment (devices).
16. Minimum necessary height of production premises to set up equipment (devices)
17. Cost of equipment (devices)

*When calculating the relative criterion evaluation, a value inverse to its absolute value is utilized.

Key to Table 1 (continued)

18. Cost of auxilliary equipment coming in contact with radioactive solutions (pulsators, pumps, etc).

19. Cost of automatic operation and control system of the equipment (devices).

20. Cost of containers and other devices for remote assembly and disassembly of the equipment, its units and auxilliary equipment.

21. Use of electric energy when the equipment is in operation.

Values of B_{pi} are totaled utilizing all criteria. Since criteria are formulated in a manner where the least value corresponds to the best piece of equipment, and the more important the criterion, the lower the value of the coefficient, it follows that equipment receiving the least sum of evaluations is given preference over all the others. For example, two pieces of equipment are compared where the absolute criterion value with significance coefficient 0.2 is equal to 10 and 8. Then $B_{1i} = 8 : 10 \cdot 0.2 = 0.16$, and $B_{2i} = 10 : 10 \cdot 0.2 = 0.2$ (to facilitate calculation, each evaluation based on all of the criteria may be multiplied by a constant greater than unity, which will not affect the results).

The suggested methodology is more flexible than those presently known, as it permits the relative advantages of devices being compared to be evaluated. In addition, criteria intervals and their evaluations are not included here since they would be impossible to substantiate.

Obviously, comparative evaluation depends first of all on correct determination of the significance coefficient r_i, which is impossible to determine by strict calculation. For this reason, a professional poll of experts working in this area was conducted. Experience, intuition, and understanding of the essence of the problem permitted them to evaluate the significance of individual criteria, and they were able to decrease uncertainty and subjectivity by assessing the evaluations with known methods. The experts were given questionnaires containing the criteria shown in Tables 1 and 2. The arrangement of criteria in the questionnaire was arbitrary. Each expert was required to assign a number to each criterion, beginning with the most significant (in his view). It was also permitted to assign the same number to any number of criteria. Numbers did not have to be consecutive, however they were not to exceed the total number of criteria. It was also permitted to delete those criteria which the expert considered unnecessary and to add those which in his opinion had been omitted from consideration. In addition, he had to evaluate his own competence according to a scale: am completely familiar with planning issues of the divisions and with development and operation of the equipment ($\nu = 3$, the "weight" of the expert); am well acquainted with planning issues of divisions and development of equipment ($\nu = 2$); am acquainted with the operation of the equipment ($\nu = 2$); am acquainted in general with planning and operation issues ($\nu = 1$). In response to the questionnaire, 35 were received concerning filters and 26 concerning extractors. Professional positions held by the experts ranged from engineers to directors, with more than 10 years of on-the-job experience. They were employed by design organizations, scientific research institutes, and plants. The majority of experts rated their own competence at $\nu = 2$.

The questionnaires were handled in the following manner [3, 4]: criteria were set up in order of significance, in relation to the number of ranges determined by the expert (with the most significant listed first). When some of the criteria received the same evaluation, they were then marked with identical standardized ranges determined by the equation

$$r = \frac{k + (k+1) + \ldots + (k+b)}{b+1},$$

where k, (k + 1),...,(k + b) are positions occuped by criteria with an identical evaluation. Standardized ranges were utilized in further calculations.

After ranging each expert's criteria, the r_i mean total range was determined for criteria provided by all of the experts:

$$r_i = \sum_{j=1}^{m} r_{ji}\nu_j / \sum_{j=1}^{m} \nu_j,$$

where m - the total number of experts; j = 1.2,..., m - the expert's number; i = 1.2,..., n - the criterion number. The unanimity of the experts was evaluated by utilizing the concordance coefficient ω, determined by Kendall's equation

$$\omega = \frac{S}{\frac{1}{12}\left(\sum_{j=1}^{m} \nu_j\right)^2 (n^3-n) - \sum_{j=1}^{m} T_j},$$

where

$$S = \sum_{i=1}^{n} (r_i - \bar{r}) \left(\sum_{j=1}^{m} \nu_j\right)^2;$$

$$\bar{r} = \sum_{i=1}^{n} r_i / n \sum_{j=1}^{m} \nu_j;$$

$$\sum_{j=1}^{m} T_j = \frac{1}{12} \sum_{j=1}^{m} \nu_j (t_j^3 - t_j);$$

t_j - the number of identical ranges, given by the j-th expert.

The confidence level (significance) of the coefficient ω was evaluated with the aid of chi-square distribution, with n - 1 the number of degrees of freedom.

$$\chi^2 = \frac{S}{1/2mn(n+1) - \frac{1}{1-n}\sum_{j=1}^{m} T_j}.$$

To derive the relative importance of r_i criteria values were normalized:

$$R_i = \frac{r_i}{\sum_{i=1}^{n} r_i}.$$

Calculation results are shown in Tables 1 and 2 (criteria are arranged in the order determined by the experts).

For filters, the concordance coefficient was ω = 0.438, with a confidence level p > 0.99; for extractors ω = 0.287, confidence level p > 0.99. The unanimity of the experts was not very high, however it cannot be considered insignificant. Overall, the arrangement of criteria by their significance should be acknowledge as successful and logical. Thus, for filters, the criterion taking into account the equipment's intensity of operation was rated first; criteria characterizing a single filter's maximal potential performance, and one of the more important indicators, the amount of waste, followed next. For extractors, the criterion determining its operational life stood first. This reflected the tendency presently in existence to utilize the most labor-intensive extractors in the radio-chemical industry, those equipped with a mechanical energy feed (centrally powered, mixer-settlers with mechanical mixing, and so forth) for which duration of operational life is precisely one of the basic indices. The criterion characterizing the duration of the equipment's time phase, i.e. high performance, came next. The arrangement of other criteria is also non-contradictory in both cases.

Of major interest is the comparison of expert' opinions from other countries. Experts from the Atomic Energy Commission of France agreed to respond to the questionnaire. The responses were processed in the same manner. The unanimity of French experts was higher than that of our experts (ω = 0.793 for filters, ω = 0.59 for extractors). Characteristically, the differences in extractor evaluations was higher here as well. Apparently, this is due to the greater complexity of the extraction process. The distribution of the criteria provided by the French experts approximates that shown in Tables 1 and 2, even though there are differences determined by the differing approaches to problems in other countries. The degree of correspondence in the responses of the experts from both countries can be conveniently evaluated by Spearman's rank correlation ρ:

where x, y are the number of ranks assigned to each criterion by Soviet and French specialists, respectively.

For filters, $\rho = 0.64$; for extractors, $\rho = 0.82$. This shows a rather high unity of views.

Naturally, the given criteria systems and their evaluations reflect current views and approaches to the problem. As science and technology develops, i.e. as new information appears, conceptions may change; this will decrease the level of uncertainty in evaluations based on new polls of the experts.

Thus, the suggested methodology does not require designation of an unsubstantiated number of points for evaluating equipment under comparison; it also permits the advisability of utilizing equipment of various types and design to be evaluated with minimum uncertainty, as well as the advisability of their application within the framework of plants reprocessing atomic power station fuel elements.

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RADIATION DAMAGE TO STEEL IN WATER-MODERATED, WATER-COOLED REACTOR VESSELS

Moscow RADIATSIONNYOYE POVREZHDENIYE STALI KORPUSOV VODO-VODYANYKH REAKTOROV in Russian 1981 (signed to press 27 May 81) pp 2-7

[Annotation, table of contents, and introduction from book "Radiation Damage to Steel in Water-Moderated, Water-Cooled Reactor Vessels", by Nikolay Nikolayevich Alekseyenko, Amir Dzhabrailovich Amayev, Igor' Vasil'yevich Gorynin and Vladimir Aleksandrovich Nikolayev, edited by Igor' Vasil'yevich Gorynin, corresponding member of USSR Academy of Sciences, Energoizdat, 1250 copies, 192 pages]

[Text] The results of research of a number of scientific and engineering problems connected with serviceability of the vessels of water-moderated, water-cooled power reactors at nuclear power plants are generalized and analyzed. A study is made of the requirements imposed on materials used in water-moderated, water-cooled power reactor vessels, material selection principles and also the properties of used and proposed materials. The greater part of the paper is devoted to the behavior of vessel materials under neutron radiation conditions, in particular, in contact with the reactor coolant. The basic laws of variation of strength, ductility and toughness of steel and the metal of welded joints subjected to neutron radiation are described as a function of heat treatment, structure, alloying, nature and amount of impurities and a number of other factors. Information is also given on the specific behavior of hydrogen in irradiated materials.

The book is designed for scientific workers and engineers dealing with the problems of building nuclear power plants and also for instructors, postgraduates and students in the corresponding specialties at the institutions of higher learning.

There are 27 tables, 94 illustrations and 321 references.

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Introduction

Twenty-five years have gone by since starting up the First Nuclear Power Plant in the USSR. Whereas at the beginning of the 1950's nuclear power engineering had taken the first steps in the direction of serving peaceful purposes, in the past two and a half decades it has been able to win strong positions in the fuel and energy system of the country and it has become an independent branch of the national economy.

The 26th CPSU Congress stated the goal of generating 1550 to 1600 billion kilowatt-hours of electric power in 1985, including 220 to 225 billion kilowatt-hours at the nuclear power plants and 230 to 235 billion kilowatt-hours at the hydroelectric power plants. The increase in electric power production in the European part of the USSR will take place primarily at the nuclear and hydroelectric power plants. New capacity in the amount of 24 to 25 million kilowatts will be put into production at the nuclear power plants [1-3].

A significant part of the program for development of nuclear power engineering in the Soviet Union and the majority of other industrially developed countries is based on the use of two-circuit water-moderated, water-cooled power reactors (VVER). Reactors of this type are among the best assimilated. They are distinguished by great compactness, high use coefficient of the power, comparatively low cost, reliability and simplicity in operation and maintenance. With respect to specific power intensity of the core and economy, the VVER are at the present time some of the best [4, 5].

With respect to chronological attributes and technical indices the VVER can be provisionally divided into three generations [6-8]. The first generation reactors (VVER-210, VVER-70, VVER-365) confirmed in practice the correctness of the scientific, technical and design development solutions used as the basis for them. They promoted the accumulation of experience in the industrial operation of VVER nuclear power plants and offered the possibility of training highly qualified service personnel. The second generation is made up of the VVER-440 reactors used as the basis for the large series of nuclear power plants already in operation or being built in the Soviet Union, German Democratic Republic, Finland and in a number of other countries. Economic calculations and operating experience indicate that the technical and economic indices of power plants equipped with these reactors are in no way inferior (and in a number of cases are even superior) to the corresponding indices of classical thermal electric power plants operating on imported organic fuel. The next qualitative step in the development of VVER (third generation) is manifested in the VVER-500 and the VVER-1000 reactors with a unit electric capacity of 500 and 1000 megawatts respectively.

One of the most responsible elements of the structural design of VVER to a significant degree determining the unit power and operating safety is the reactor vessel. Direct monitoring of the change in the mechanical properties of the vessel material of an operating reactor and necessary repairs impose great technical difficulties. High manufacturing quality, which can be insured only under the conditions of a specialized plant, can serve as a definite guarantee of the serviceability of a reactor vessel. In the Soviet Union, the possibility of plant manufacture of a vessel subsequently transported by rail, has formed the basis for all structural designs since the very beginning of the development of VVER. This has led to the necessity for solving a large number of problems in the areas of material science, metallurgy and welding technology.

First of all, it was necessary to develop andindustrially assimilate high quality, low-alloy steel which has high metallurgical adaptability to manufacture, weldability in large thicknesses and mechanical properties that insure operating reliability of the vessel for no less than 40 years. The high volume of welding involved in manufacturing a reactor vessel has led to the necessity for creating new welding equipment, the development of welding materials, welding conditions and technology guaranteeing the mechanical properties of the welds and high efficiency of the welding operations. High corrosion resistance of the vessel material has been insured by anticorrosive surfacing of the inside surface. In addition, the possibility of operating VVER vessels without anticorrosive protection has been proven. The problems of quality control of the base metal and welds by highly effective x-ray and gamma techniques, ultrasonic, color and luminescent flaw detection have also been solved.

In contrast to ordinary high-pressure vessels, the reactor vessels are subjected to powerful neutron and gamma radiation causing significant changes in properties of the metal. The most unfavorable of these changes is the loss of ductility and increase in inclination of the steel to brittle failures. Accordingly, the problem of radiation strength of reactor vessel material from the point of view of prolonged serviceability has acquired primary significance. The first person to point to the necessity for studying radiation strength of reactor materials in the Soviet Union was I. V. Kurchatov. By his initiative a number of scientific research complexes were set up which are equipped with specialized materials science reactors and "hot" chambers offering the possibility of studying the properties of materials in intense ionizing radiation fields. In the final analysis, the broad development of such research has led to the creation of a new field of science -- solid-state radiation physics, a component part of which is radiation materials technology.

The general problems of the influence of reactor irradiation on the properties of metals are fully covered in the book by S.T. Konobeyevskiy [9]. Various theoretical and engineering aspects of the influence of neutron irradiation on the behavior of materials designed for making VVER vessels have been discussed in the reports by a number of international and all-union conferences and numerous periodicals. However, the increasing scale of the construction of nuclear power plants equipped with VVER has put the question of the necessity for systematization and generalization of scattered information and the investigation of the problems of radiation strength of reactor vessels as a whole on the agenda. The effort to solve this problem has been undertaken in the book now brought to the attention of the specialists.

In conclusion, the authors consider it their duty to express their deep appreciation to P. A. Platonov, G. P. Karzov, Yu. I. Zvezdin and V. V. Rybin for useful discussion of individual sections of the book, and also to L. M. Lebedev, V. I. Badanin, E. P. Usatov, A. A. Kuznetsov and A. M. Morozov who participated directly in the experiments.

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RADIATION SAFETY AND PROTECTION OF NUCLEAR POWER PLANTS

Moscow RADIATSIONNAYA BEZOPASNOST' I ZASHCHITA AES in Russian 1981 pp 2, 64-78

[Annotation and article "Tritium at Nuclear Electric Power Plants" by V. V. Badyayev, Yu. A. Yegorov, V. P. Sklayrov and G. F. Stegachev from book "Radiation Safety and Protection of Nuclear Power Plants", edited by Professor Yu. A. Yegorov, Atomizdat, 256 pages]

[Text] The fifth edition includes part of the papers discussed at the Second All-Union Scientific Conference on Protection of Nuclear Engineering Facilities from Ionizing Radiation (Moscow, MIFI [Moscow Engineering Physics Institute], December 1978) and the Branch Conference on Exchange of Experience in Radiometric and Dosimetric Research at Nuclear Power Plants (Moscow, NIKIET [expansion unknown], May 1979). Reports by American specialists heard at the Soviet-American Symposium on the Problem of Fast Neutron Reactor Shielding (Obninsk, FEI [expansion unknown], November 1978) were also included. The remaining materials will be published in the sixth edition.

The results of studying the radiation conditions at nuclear power plants and the problems of predicting activity are investigated. Beginning with this edition, a new section has been initiated where the environmental protection problems of operating nuclear power plants are discussed. Materials on passage of radiation through reactor shielding and methods of calculating shielding are presented.

The book is designed for specialists in radiation safety and protection of nuclear power plants.

There are 59 tables, 70 illustrations and 317 references.

Tritium at Nuclear Power Plants, by V. V. Badyayev, Yu. A. Yegorov, V. P. Sklyarov, G. F. Stegachev

Tritium, which is a superheavy radioactive hydrogen nuclide, plays an important role in the problems of radiation safety of nuclear power plants. Tritium has a halflife of 12.26 years, it decays with the emission of β-particles with maximum energy of 18.6 kiloelectron volts (average energy 5.8 kiloelectron volts). Tritium belongs in the category of so-called global radionuclides for it has a quite long halflife, it can enter into the composition of organic and inorganic compounds and when it gets into the environment it can spread to significant distances.

Tritium exists primarily either in gaseous form or in the form of tritium water, but it can also enter into the composition of various organic compounds. The toxicity of tritium as a radionuclide depends to a great extent on the form in which it is found in the environment and in which it enters into the organism of man: in gaseous form (HT, T_2) or in the form of tritium water (HTO, T_2O). Gaseous tritium modifies the skin and mucous membrane of the respiratory tracts of man by its β-radiation (external irradiation) and causes damage to the internal organs of man, being found in the chemically bound state in these organs. Upon entering the environment in the form of tritium water, tritium can enter the organism of man both through the respiratory tracts (tritium water vapor) and through the skin and gastrointestinal tract (with food and water). It is noted that HTO penetrates the organism through the entire skin surface of a human just as efficiently as through the lungs: up to 10 microcuries/minute with a tritium concentration in the air of 1 microcurie/liter [1]. When tritium water enters the gastrointestinal tract with food, it is quickly assimilated by the organism, and a few minutes after ingestion it can be detected in the venous blood. It is also noted [2] that 24 hours after tritium enters the organism, the greatest concentration of it is detected in the blood, liver and small intestine, and after 5 days the tritium concentration in all of the internal organs is almost identical. According to the most recent findings, the process of elimination of tritium from the human organism is described by a curve with three exponential segments. At first, the tritium is eliminated quite intensely (effective halflife ~10 days). This is connected with the water exchange of the organism. Later on, the process of tritium elimination decelerates. It has been established that the elimination rate of organically bound tritium depends on the type of organ and biological tissue: the effective halflife for the majority of organs varies from 10 to 100 days, but there are organs for which the effective halflife exceeds 5 years.

According to the recommendations of the International Commission on Radiological Protection, it is considered that on irradiation of the organism of man with β-particles with an energy to 0.1 megaelectron volts, their energy is absorbed by the skin, that is, only the skin is irradiated. Considering that the skin of the entire body of man belongs to group III of critical organs (tissues), the maximum permissible dosage of irradiation for which is 30 rem per year, the NRB-76 established the following mean annual permissible concentrations (MAPC): $2\cdot10^{-6}$ curies/liter in the air of the work place (for personnel) and $6.6\cdot10^{-8}$ curies/liter in atmospheric air (for individuals of the population). For tritium in the form of T_2O or HTO, the MAPC are $4.8\cdot10^{-9}$ curies/liter in the air of the work place, $1.6\cdot10^{-10}$ curies/liter in atmospheric air and $3.2\cdot10^{-6}$ curies/liter in water. Such a large difference in values of the MAPC for different forms of existence of tritium arises from the difference in radiotoxicity of these forms.

Usually tritium of natural and artificial origin is distinguished. Natural tritium is formed in the upper layers of the atmosphere as a result of the interaction of cosmic radiation with ^{14}N and ^{16}O nuclei. Natural tritium exists primarily in the form of HT or HTO. The natural tritium content in atmospheric air is 1 atom per 10^{14} hydrogen atoms, and in water 1 tritium atom occurs for 10^{18} hydrogen atoms, which corresponds to a specific concentration of $3.2 \cdot 10^{-12}$ curies/liter. The total natural tritium content on the earth is 20 to 80 million curies.

The primary source of artificial tritium until recently was thermonuclear weapons testing (to 10 million curies per megaton of thermonuclear blast), as a result of which the tritium concentration in entities of the environment increased significantly: the total tritium concentration on the earth in 1973 was 2900 ± 700 million curies, which is appreciably higher than the natural level [3]. Thus, in the 1960's the tritium concentration in rainwater in the northern hemisphere was on the average $1.6 \cdot 10^{-9}$ curies/liter [4]. Detailed studies of the tritium pollution of the water of the Baltic Sea in 1972 [5] demonstrated that on the average the tritium concentration in the sea was $(1.98 \pm 0.73) \cdot 10^{-10}$ curies/liter. Curtailment of thermonuclear weapons testing is naturally accompanied by a decrease in the tritium concentration in the environment. However, in connection with the development of nuclear power engineering, nuclear power plants and plants for processing irradiated nuclear fuel are becoming contributors of tritium to the environment. Processing of uranium fuel to generate 1 megawatt of power is accompanied by the release of 19 curies/gram of tritium; the same figure for plutonium fuel is 36 curies/gram [3].

An analysis performed in reference [6] shows that in 1980 the tritium production at nuclear power plants and plants for processing nuclear fuel will be 6.7 megacuries, and 34 megacuries of it will be accumulated for all years of operation of nuclear power plants. The same values for the year 2000 will increase to 140 and 720 megacuries, respectively. Inasmuch as the greater part of this tritium is discharged into the environment, it can become a radionuclide significantly influencing the radiation situation, especially if we consider the possibility of interference of the discharge of nuclear power plants which in the future will be located comparatively close to each other. All of this indicates the necessity for careful study of sources and means of entry of tritium into the environment, accumulation and migration of it, the forms of its existence in the environment, possibly the development of means of localizing tritium in order to remove it from the nuclear power plant waste and other radioactive waste [7].

In nuclear power plant reactors tritium is formed in the following processes:

Directly on fission of the fuel nuclei as a product of ternary fission;

As a result of capture of neutrons by deuterium nuclei in the coolant--water in the form D_2O;

On capture of neutrons by boron or lithium nuclei in the coolant--water (boron control, adjustment of the water regime);

As a result of boron nucleus reactions in the control rods;

During capture of fast neutrons by construction materials.

The contribution of each of these processes to the total amount of tritium formed in a nuclear power plant is determined by the type of reactor, its parameters, operating conditions and materials used in the reactor.

One of the primary sources of tritium is formation of it in the fuel elements during ternary fission; the triton yield per fission of ^{235}U [8] is on the average $8.7 \cdot 10^{-3}$%, that is, $1.23 \cdot 10^{-2}$ curies of tritium is formed per megawatt of thermal power of the reactor in the fuel elements per day. The fission process as a source of tritium can be characterized by the following example: by 1980 an accumulation of about $3 \cdot 10^6$ curies of tritium is expected in the fuel elements of the nuclear power plants of Great Britain, and by the year 2000, about tenfold more [9].

As a result of diffusion through the fuel jacket and also through cracks and microcracks in the jackets, tritium can get from the fuel to the coolant. It is natural that under other equal conditions the tritium leakage out of the jackets depends on the amount accumulated in the fuel element. Estimates made on the basis of experimental data [10] indicate that the tritium leakage rate from fuel jackets with a microcrack surface area of 1% of the entire jacket surface for fuel in the form of sintered uranium dioxide is characterized by a coefficient of 0.86 per day. According to other data [11], the tritium leakage from fuel jackets made of corrosion resistant steel is 1%, and from zirconium alloyed jackets, 0.1% of the total amount of tritium under the jacket. Considering these data, the amount of tritium getting from the fuel elements into the coolant can be defined by the ratio [12] $Q_T = 1.32 \cdot 10^{-16} W$, curies/sec, where W is the thermal power of the reactor, watts (zirconium fuel jackets).

Table 1. Thermal Neutron Reactions with Tritium Formation

Reaction	Content of target nuclide in a natural mixture, %	Thermal neutron reaction cross section, σ
$^{2}_{1}H(n, \gamma)\,^{3}_{1}H$	0,015	$5{,}7 \cdot 10^{-4}$
$^{3}_{2}He(n, p)\,^{3}_{1}H$	$1{,}3 \cdot 10^{-4}$	$5{,}4 \cdot 10^{3}$
$^{6}Li(n, \alpha)\,^{3}_{1}H$	7,52	$9{,}36 \cdot 10^{2}$
$^{7}Li(n, n', ^{4}_{1}He)\,^{3}_{1}H$	92,48	0,02*
$^{10}B(n, 2\alpha)\,^{3}_{1}H$	19,8	0,02*

*Cross sections are averaged with respect to fission neutron spectrum.

The probability of tritium formation as a result of other reactions is characterized by the data presented in Tables 1 and 2. It is obvious that the basic reactions of tritium formation in nuclear reactors are the deuteron and ^{10}B nucleus (n, γ) reactions. The D(n, γ)T reaction leads to the formation of tritium directly in the coolant, and the amount of tritium formed by this reaction in an ordinary water-cooled reactor can be defined by the formula

$$Q_T = 27.6 n_H V \phi \cdot 10^{-19} \text{ curies/sec},$$

where n_H is the nuclear density of hydrogen in the coolant, cm^{-3}; V is the coolant volume in the reactor core, cm^3; ϕ is the average thermal neutron flux density with respect to the coolant, sec^{-1}, cm^{-2}.

The ^{10}B nucleus reactions, as has been noted, are possible during boron control--in the coolant and in the rods of the control and safety rod system of the reactor.

Table 2. Fast Neutron Reactions with Tritium Formation

Reaction	Neutron energy, Mev	Target nuclide content in natural mixture, %	Average value of the reaction cross section, σ
$^{14}N(n, t)^{12}C$	5—10	99,63	$3 \cdot 10^{-2}$
$^{6}Li(n, \alpha)^{3}H$	0,01—10	7,52	1,0
$^{7}Li(n, n', t)^{3}He$	6—14	92,48	$4 \cdot 10^{-1}$
$^{10}B(n, \alpha)T$	1—9	19,8	$7,5 \cdot 10^{-2}$
$^{32}\delta(n, t)^{30}R$	14,7	95,018	$2 \cdot 10^{-5}$
$^{40}Ca(n, t)^{38}K$	14,7	96,97	$0,1 \cdot 10^{-3}$
$^{54}Fe(n, t)^{52}Mn$	14,6	5,84	$7,5 \cdot 10^{-5}$
$^{54}Fe(n, t)^{52m}Mn$	14,8	5,84	$6 \cdot 10^{-4}$

The contribution of the various tritium formation reactions to the total amount of tritium in the coolant was estimated in reference [12] in the example of the Rheinsberg nuclear power plant for its operating period during boron control testing in 1970:

Yield from fuel elements	0.5
Yield from the rods of the control and safety rod system	0.15
Formation in coolant as a result of the following reactions:	
$D(n, \gamma)T$	0.18
$^{10}B(n, 2\alpha)T$	5.23
$^{10}B(n, n', \alpha)T$	10^{-4}

It is obvious that more than 85% of tritium is formed as a result of the fast neutron $^{10}B(n, 2\alpha)T$ reaction. The authors of reference [12] note that this does not agree with the conclusions, for example, of reference [13], in which it was demonstrated that the primary reaction of tritium formation under such conditions must be the interaction of thermal neutrons with ^{10}B.

The results of the approximate calculation [4] of the specific activity of tritium in the coolant of the Kol'skaya Nuclear Power Plant also demonstrated that in VVER-440 water-moderated, water-cooled power reactors with boron control (boric acid concentration 3.4 g/kg) the primary reaction of tritium formation is the

$^{10}B(n, 2\alpha)T$ reaction: during the operating time of the reactor of 7000 effective hours, the specific tritium activity in the coolant of the primary circuit as a result of this reaction is approximately 10^{-3} curies/liter, at the same time as the specific activity will be about $5 \cdot 10^{-6}$ curies/liter as a result of deuterium activation and double activation of protium.

For nuclear power plants with VVER-50 reactors [12], the estimates show that in one run, the coolant picks up as much as 260 curies of tritium: 5.23 curies as a result of ternary fission, 3.22 curies as a result of protium activation and 252 curies as a result of the ^{10}B reaction (during boron control).

The presented calculated values agree qualitatively with the data on the actual tritium content in the coolant of a number of operating nuclear power plants with pressurized water reactors gathered, for example, in reference [14] (curies/year): Indian Point-1 (W=163 megawatts), 500; Yankee (W=183 megawatts), 1300; Haddon Rock (W=462 megawatts), 1733. All of these nuclear power plants have boron control and fuel elements in corrosion-resistant steel jackets. In reference [15] it is also pointed out that the specific tritium activity in the water of the primary circuit of nuclear power plants with PWR falls within the limits of $(0.1 \text{ to } 5) \cdot 10^{-3}$ curies/liter as a result of introduction of boron into the coolant. The tritium activity in the water of the primary circuit of the Novovoronezh Nuclear Power Plant [16]* is $2.2 \cdot 10^{-6}$ curies/liter for block I, $21 \cdot 10^{-6}$ for block II, $72 \cdot 10^{-6}$ for block III and $95 \cdot 10^{-6}$ curies/liter for block IV. The maximum recorded tritium concentrations in these blocks in the period from 1973 to 1975 was 3, 10, 520 and $340 \cdot 10^{-6}$ curies/liter, respectively [17]. At the Kol'skaya Nuclear Power Plant, 3 months after beginning operation, the tritium concentration in the coolant was $31 \cdot 10^{-6}$ curies/liter [17].

It is natural that for nuclear power plants with reactors of other types, the amount of tritium formed as a result of one reaction or another and, consequently, its concentration in the coolant will be different. Thus, for nuclear power plants with heavy water reactors the primary tritium formation reaction is deuterium activation. Estimates show that in heavy water reactors up to 30 curies/year of tritium are formed per megawatt of electric power. At nuclear power plants with boiling reactors, the tritium formation as a result of ternary fission and as a result of reactions on the boron nuclei in the control rods is significant, but the primary source of tritium is deuterium activation. Thus, calculation of the amount of tritium formed by the $D(n, \gamma)T$ reaction at the Dresden-1 Nuclear Power Plant demonstrated [1] that the rate of tritium formation is $6 \cdot 10^{-8}$ curies/sec, and experimental information on the accumulation of tritium in the coolant gives a rate equal to approximately $5 \cdot 10^{-8}$ curies/second, that is, it is obvious that the tritium is formed primarily as a result of deuterium activation. Estimates made for nuclear power plants with RBMK-1000 reactor also demonstrated that in the multiple forced circulation circuit tritium is formed primarily as a result of the $D(n, \gamma)T$ reaction, its specific

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*See also the survey: N. G. Gusev, "Provision for Radiation Safety at Nuclear Power Plants," ATOMNAYA ENERGIYA [Nuclear Power], Vol 41, No 4, 1976, p 254.

activity in the multiple forced circulation circuit is low, and it amounts to about $4 \cdot 10^{-7}$ curies/liter.

The actually measured tritium concentrations in the coolant of nuclear power plants with boiling reactors are characterized by the following values:

Blocks I and II of Beloyarskaya Nuclear Power Plant $20 \cdot 10^{-6}$ and $28 \cdot 10^{-6}$ curies/liter [16], maximum recorded concentration for block I $29 \cdot 10^{-6}$ curies/liter [17];

For block I of Leningrad Nuclear Power Plant $0.48 \cdot 10^{-6}$ curies/liter [17];

At the nuclear power plant with RBMK-1000 reactor (on the average) during the initial operating period at stable power level, in the water of the multiple forced circulation circuit and in the saturated steam condensate (1 to 2)$\cdot 10^{-7}$ curies/kg;

At a nuclear power plant with VK-50 reactor the maximum recorded concentration is $4 \cdot 10^{-6}$ curies/liter [17].

The amount of tritium produced by nuclear power plants with boiling reactors in a year is characterized by the following data [14]: Dresden-1 (W=200 megawatts), 2.9 curies; Humboldt Bay (52 megawatts) and Big Rock Point (W=50 megawatts), 20 curies each.

At nuclear power plants with channel boiling reactors, in addition to the multiple forced circulation circuit, tritium is also formed in the control and safety rod system circuit and in the gas circuit, that is, in the gas blown through the graphite stacking of the reactor. In the control and safety rod system circuit at a nuclear power plant with RBMK-1000 reactor, about 2.5 curies of tritium are formed per year as a result of reactions on the boron nuclei (under the assumption that the tritium leakage from the jackets of the rods of the control and safety rod system is 0.1%) and about 1 curie as a result of deuterium activation. The tritium concentration in the control and safety rod system loop in the absence of leaks is $\sim 5 \cdot 10^{-6}$ curies/liter. Tritium is formed in the gas circuit as a result of reactions on helium nuclei. At the nuclear power plants with RBMK-1000, up to 800 curies of tritium are formed in the circuit, and its average concentration in the gas will be $\simeq 6 \cdot 10^{-7}$ curies/liter. The results of measurements of the tritium concentrations give the following values: in the control and safety rod system circuit (2 to 6)$\cdot 10^{-7}$ curies/liter; in the gas circuit, $\sim 10^{-9}$ curies/liter of tritium in the form of tritium water vapor.

In general, it is impossible to note good convergence of the calculated and measured tritium concentrations in the various circuits of nuclear power plants. At the same time it is necessary to note defined scattering of the experimental data on concentrations. All of this is connected, on the one hand, with the fact that it is difficult to consider all the factors influencing the tritium concentration in the calculations, for example, the state of the core, the actual magnitude of the leaks: the water chemical regime of the coolant obviously influencing the chemical forms of existence of tritium, the amount of impurities in the scavenging gas, and so on.

On the other hand, the majority of experimental data are the results of measuring the tritium concentration in oxidized form, that is, in the form of tritium water. The discovery of the influence of various operating conditions of nuclear power plants and the state of its components on tritium concentration in the process circuits is one of the problems which must be solved for reliable prediction of the influx of tritium into the environment and the development of devices for localization of tritium if the necessity for such arises. It is now possible to state that at nuclear power plants with boiling channel RBMK-1000 reactors the tritium concentration is below the MAPC of tritium for open bodies of water, and at the nuclear power plants with water-moderated, water-cooled power reactors it is higher than the MAPC by 10 to 100 times and no more.

The tritium formation in the coolant circuit gives rise to its presence in the gas and liquid effluents of nuclear power plants, and also, primarily as a result of coolant leaks, in the air of the work places of the nuclear power plants. Thus, the activity of the liquid discharge of many nuclear power plants with PWR is 85 to 99.9% determined by tritium, and in 1973 it was from 16 to 4000 curies/liter [18]. The maximum tritium activity in the liquid discharge of the Novovoronezh Nuclear Power Plant (in 1973 to 1975) and the Kol'skaya Nuclear Power Plant were below the MAPC of tritium in open bodies of water and equal to 10^{-7} to 10^{-6} and up to $2.5 \cdot 10^{-6}$ curies/liter, respectively [17]. The tritium discharge with liquid waste from nuclear power plants with boiling reactors is with respect to activity also a large portion--from 9 to 90%-- and at the Vermont Yankee Nuclear Power Plant, 99.9%, but with respect to absolute magnitude, the activity of the discharged tritium is appreciably less: from 4 to 50 curies less (for 1973), for example, (at the Vermont Yankee Nuclear Power Plant, 0.2 curies/year) [18]. It is natural that the tritium concentration in the liquid waste of nuclear power plants with boiling reactors is also lower: maximum tritium concentrations in the waste water of the Beloyarsk and Leningrad Nuclear Power Plants do not exceed 0.26 to 0.27 curie/liter [17], in the industrial sewage of Leningrad Nuclear Power Plant $(6 \text{ to } 7) \cdot 10^{-9}$ curies/liter, that is, significantly below the MAPC for water. This conclusion agrees with the conclusion drawn in reference [19] that the tritium concentration at the liquid waste discharge point into the Kaporskaya Bay on the Gulf of Finland is more than an order below the permissible concentrations in potable water.

The tritium discharge with liquid waste from nuclear power plants in the majority of countries has not been standardized. In the United States before 1971 it was permissible to discharge liquid waste with tritium concentration of $3 \cdot 10^{-6}$ curies/liter, but from the materials of the Third Geneva Conference on Peaceful Uses of Atomic Power it is known that in recent years this permissible discharge has been decreased: the tritium concentration in the waste before reservoir dilution must not exceed $5 \cdot 10^{-9}$ curies/liter, that is, approximately no more than 25 times the average tritium concentration in the water of the Baltic Sea.

In the gas effluents of nuclear power plants the tritium content is much less than in the liquid effluents, which is naturally explained by the inclination of tritium toward oxidation. Thus, the activity of tritium discharged in 1973 by nuclear power plants with boiling reactors falls within the limits from fractions of a curie to 75 curies [18], and in percentage ratio this is 10^{-3} to 10^{-4}% of the total discharged activity. In reference [20] a theoretical estimate was made of the power of the tritium discharge for nuclear power plants with VVER-440 reactor. Ternary fission of ^{235}U, activation of deuterium and basic reactions on ^{10}B nuclei were considered as the means of formation of the tritium. It was established that in the absence of unorganized primary circuit coolant leaks, the primary source of tritium at nuclear power plants is the water purification ponds, in the water of which the tritium concentration can reach $1.8 \cdot 10^{-4}$ curies/liter. Tritium gets into the discharge from the ponds with water vapor, and the power of the discharge can be about 130 curies/year. In the case of unorganized leaks of 0.2 tons/hour as a result of partial evaporation the discharge power increases to 160 curies/year.

The presence of tritium in the air of the work places is prompting some specialists to consider constant monitoring of the tritium content in the organizing of workers at nuclear power plants necessary, just as is done, for example, for iodine radionuclides. Actually, at the nuclear center in Karlsruhe, 7.2% of the personnel have from 0.1 to 1 times the permissible amount of tritium in the organisms, and 60% of the workers are tritium carriers; at radioisotope production facilities, the internal irradiation by tritium is also on the maximum permissible level [21].

At the Kozloduy Nuclear Power Plant (People's Republic of Bulgaria), during the initial period of operation, the tritium concentration in the air of various rooms was $(1.3 \text{ to } 3.4) \cdot 10^{-13}$ curie/liter [22]; according to the estimates of reference [20], under the above-described conditions in the facilities of nuclear power plants with VVER-440 tritium will be detected, but its concentration will not exceed the MAPC for work places established by the NRB-76. The measurements made at the Novovoronezh Nuclear Power Plant demonstrated that the maximum recorded tritium concentrations in the work places of the nuclear power plant fall within the limits of $(64 \text{ to } 180) \cdot 10^{-9}$ curies/liter. At nuclear power plants with boiling channel reactors, these values are lower: at the Beloyarsk Nuclear Power Plant, at the background level, and at the Leningrad Nuclear Power Plant, to 10^{-9} curies/liter [17].

As is obvious, the data presented here on the actual liquid and gas discharge of tritium by the nuclear power plants do not permit establishment of any laws. On the contrary, the significant scattering of the information about the effluents even for nuclear power plants with like reactors attracts attention. This is caused not only by defined differences in the process flow charts of the nuclear power plants and not only by insufficiently complete study of the sources of tritium at the nuclear power plants, as was noted above, but also by insufficient study of the ways that tritium gets into the waste facilities, the laws of its accumulation in the coolant. It is natural that the absence of such information complicates even more the prediction of the accumulation of tritium in the biome as a result of operation of the nuclear power plants and obtaining detailed information about the interrelation of the process flow chart of a nuclear power plant, its operating conditions with the formation and delivery of tritium to the waste units--the primary element of the tritium problem at nuclear power plants.

The experimental study of the behavior of tritium in the process circuits of nuclear power plants and in the environment presupposes the taking of samples from the media in the circuits and the objects of the experimental environment and measurement of the tritium concentration in them. It is possible to note the following peculiarities of this experimental work:

1) The investigated media can be different (water, steam, gas, air), and tritium can be found in the same medium in different chemical forms (for example, tritium water and gaseous tritium in the water of the multiple forced circulation circuit, in atmospheric air);

2) The activity of the samples, as is obvious from what has been stated above, is low, but the measurement results must be obtained with good accuracy inasmuch as they are needed for forecasting purposes;

3) The energy of the tritium β-particles is quite low; therefore for reliable recording of them, detectors and equipment with low intrinsic noise level and low natural background are needed.

However, at the present time reliable methods of taking samples of various media, concentration of them (if necessary) and measurement of the activity, which will permit us to obtain qualitative information about tritium at the nuclear power plants, have been developed and are in use in practice [10, 23-25].

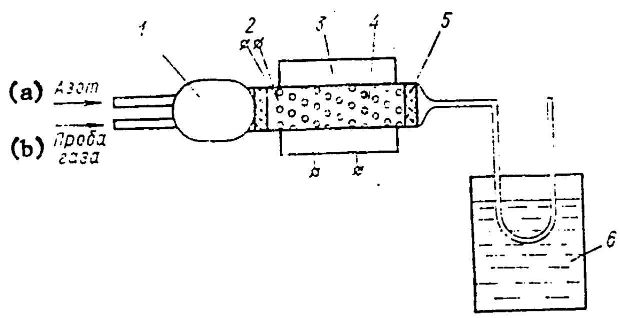

Figure 1. Diagram of a device for taking tritium gas samples from air: 1 -- mixing chamber; 2 -- thermocouple; 3 -- furnace; 4 -- tube with pelletized copper oxide; 5 -- quartz wool; 6 -- Dewar flask with liquid nitrogen

Key:

a. Nitrogen
b. Gas sample

If it is necessary to determine the tritium water content in water, the sample is taken directly, that is, a vessel is filled to the appropriate volume, mechanical impurities and other radioactive nuclides are removed, for example, using an ion-exchange resin, and the sample is stored in a sealed container until analysis in order to exclude evaporation, inasmuch as evaporation can distort the results as a result of isotopic effects. When determining the tritium gas concentration in water, the sample is taken sealed. Gaseous tritium is extracted from the sample by bubbling. A sample of atmospheric moisture is collected either by freezing out using cooling units or by pumping air through water absorbers, for example, through concentrated sulfuric acid or silica gel [24]. On 2 kg of silica gel it is possible to collect up to 250 ml of water. The water is isolated from the silica gel by a closed current of dry air.

The gas samples for determining the concentration of gaseous forms of tritium are taken with the help of a compressor in bottles or elastic shells. However, if a sample must be stored for a prolonged period of time, this method of sampling can lead to errors as a result of uncontrolled tritium loss by diffusion through the walls of the sampling vessel. It is more reliable to take the sample by final oxidation of volatile forms of tritium in a special device through which gas or air is pumped. Final oxidation of tritium is carried out in a nitrogen current on pelletized copper oxide (Figure 1) at a temperature of 600°C. The vapor in the formed water is frozen out in a U-tube. By selecting the oxidation conditions (catalyst, temperature, and so on) properly, it is possible to obtain information about the tritium content in various chemical forms.

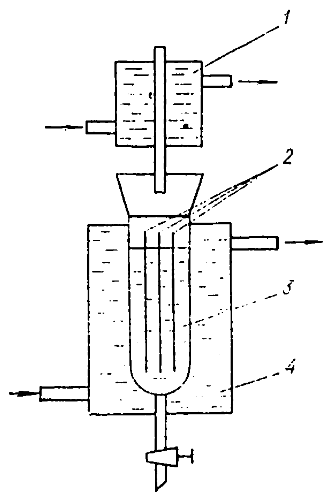

Figure 2. Diagram of a hydrolyzer for concentrating tritium samples: 1 -- cooler; 2 -- electrodes; 3 --- water sample subject to concentration; 4 -- vessel for cooling the electrolysis cell.

If it is known in advance that tritium activity in the sample is small, the sample is concentrated. The difference in hydrolytic decomposition rate of the molecules of ordinary, heavy and tritium waters is used for concentration. Concentration is carried out in special devices called hydrolyzers (see Figure 2), and if necessary the process is repeated several times. Usually eightfold to tenfold enrichment of the sample with tritium water molecules is achieved in one cycle.

For determining the tritium activity in the sample, the scintillation or ionization method is used. For implementation of the ionization method, either proportional internal filling meters (sometimes flow meters) or ionization chambers (for measuring high tritium activities in the sample) are used.

In the scintillation method usually liquid scintillators based on dioxane are used, for it is possible to add a sample to dioxane in a volume of up to 20% of the volume of the dioxane without having significant negative effects on its conversion efficiency. Various scintillating additives and spectrum displacers are used; ready-made ZhS-7 and ZhS-8 (dioxane+α-methylnaphthaline+BPO) liquid scintillators are suitable for recording tritium radiation. The conversion effectiveness of liquid scintillators becomes noticeably worse on "poisoning" of them with oxygen and various impurities, especially impurities containing sulfur and chlorine. Therefore before introduction into the scintillator, the sample is purified of its impurities on ion exchange resin or by distillation, and after introduction of it into the scintillator, the mixture ready for measurements is purged with argon to remove the oxygen.

The activity of the sample is measured on units with 1, 2 or 3 photomultipliers. The advantage of units with one photomultiplier is simplicity. However, in such units it is necessary to use a photomultiplier with high sensitivity of the photocathode and low energy equivalent of the intrinsic noise. Sometimes the photocathode is cooled for this purpose. On the USS-1 unit without cooling of the photocathode of the photomultiplier, a sensitivity of $\sim 10^{-7}$ curies/liter of tritium is achieved by applying a scintillator based on dioxane with naphthaline (100 g/liter), PPO (8 grams/liter) and POPOP (0.2 grams/liter). On units with several photomultipliers noise pulses can be avoided to a significant degree, and it is possible to determine activity on the level of 10^{-10} curies/liter. The known standard foreign-made devices (TRICARB, SL-20, and so on) have a sensitivity of $\sim 10^{-9}$ curies/liter.

Devices for preparing a sample with measuring by the ionization method and proportional internal filling counters with active shielding made of gas discharge counters are described in reference [23]; they fully satisfy the requirements of the problems solved when investigating the formation and transport of tritium at nuclear power plants. Such devices T-1 and T-2, which are distinguished by small structural changes from those investigated in [23], are described below. These devices are used in operations performed at the nuclear power plants*, and they are distinguished by their sensitivity: the T-1 is a device with relatively low sensitivity, to 10^{-8} curie/liter.

In the T-1 device, an internal filling counter with a volume of 600 cm^3 (Figure 3) with cathode in the form of a corrosion-resistant steel cylinder and with anode made of gold-plated tungsten wire 20 microns in diameter, is used. In contrast to the counter described in reference [23], the flange seal of its ends has been replaced by a seal using a nut, which has made it possible to reduce the overall dimensions of the counter and to use mat for its protection and lead shielding from the UMF device. In the massive organic glass end covers, devices are mounted for putting tension on the filament--anode--an intake connection for the investigated gas, a socket for connecting the electronic unit and thin-walled window of lavsan film (50 microns thick) for admitting ^{55}Fe x-radiation into the counter

*The authors express their appreciation to L. I. Gedeonov, V. I. Blinov, V. P. Tishkov and their colleagues for consultation and assistance in adjusting the units.

(E_γ=5.9 kev) used to select the operating conditions of the counter. The mat is assembled from the MS-6 type gas discharge counters, and it is included anti-coincident with the main counter. The lead shielding is 50 mm thick. The system for filling the counter with the working gas mixture is analogous to that described in reference [23]. The working gas mixture is prepared from propane (90%) and the investigated sample of hydrogen with tritium (10%); during the measurements a pressure of 1 to 1.2 atmospheres is maintained in the counter volume.

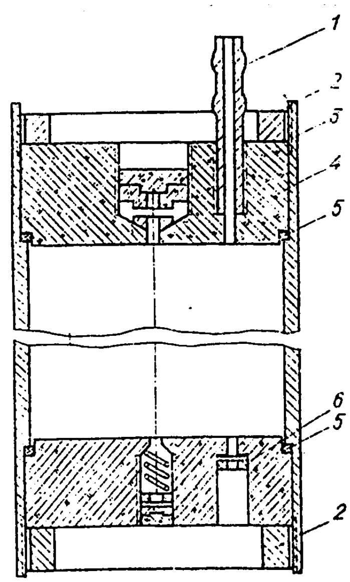

Figure 3. Structural design of a gas discharge internal filling counter: 1 -- counting gas intake connection; 2 -- pressure nuts; 3 -- counter housing; 4 -- end type organic glass insert; 5 -- rubber gaskets; 6 -- lavsan film

The T-2 device consists of an internal filling counter with a volume of 4500 cm^3 (diameter 120 mm, length 650 mm) of analogous structural design placed in the mat of gas discharge counters type SI-6G. The counting module is placed in the cavity (300×300×900 mm), in lead shielding 150 mm thick with sliding door. Just as in the T-1 device, the known [23] system for preparing the sample and filling the working volume of the counter with it is used (see Figure 4).

For measurements on the T-1 device the samples are prepared in advance by decomposition of water by calcium oxide and zinc; on the T-2 device the sample is prepared either in the counter filling system by decomposition of water by magnesium [22] in a special tube furnace or in advance.

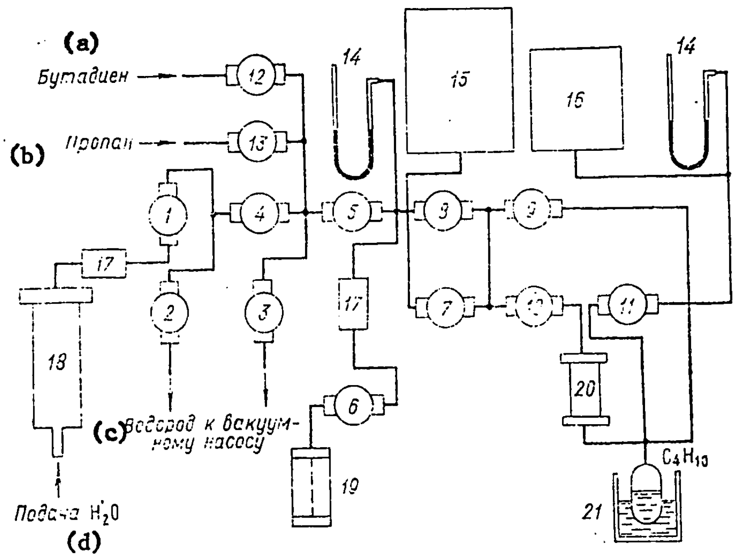

Figure 4. Diagram of a stand for preparing low-active tritium samples for measurements:
1-13 -- vacuum valves; 14 -- vacuum meter; 15 -- vessel for mixing butadiene and hydrogen; 16 -- vessel for storing excess gas; 17 -- H_2 absorption; 18 -- H_2 generator; 19 -- counter; 20 -- catalyst; 21 -- liquid nitrogen

Key:
- a. Butadiene
- b. Propane
- c. Hydrogen for a vacuum pump
- d. H_2O feed

The electronic modules of both devices are the same, they have a total channel gain of ~1000 with stability of no worse than 1%, deviation from linearity of the amplitude characteristics in the tritium β-particle energy range does not exceed $\pm 1\%$.

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NON-NUCLEAR ENERGY

UDC 532.538.4

INDUSTRIAL MAGNETOHYDRODYNAMIC EQUIPMENT AND PROCESSES

Kiev TEKHNOLOGICHESKIYE MGD USTANOVKI I PROTSESSY in Russian 1980 (signed to press 21 Oct 80) pp 2, 189-190

[Annotation and table of contents from book "Industrial Magnetohydrodynamic Equipment and Processes", by Anatoliy Fedorovich Kolesnichenko, Ukrainian SSR Academy of Sciences, Electrodynamics Institute, Izdatel'stvo "Naukova dumka", 1000 copies, 191 pages]

[Text] In this monograph a study is made of qualitative and spatial conversions in limited domains filled with electrically conducting drop liquid or electrically conducting gas and placed in physical fields--electric, magnetic and gravitational. Among such conversions are transformation and transport of energy--conversion of electromagnetic energy to heat and mechanical work, transport of the heat and material mass of a liquid conductor under the effect of various combinations of electromagnetic, capillary and thermoconvective effects. The results of theoretical and experimental studies of a new class of magnetohydrodynamic flows--capillary flows that occur during arc and induction working of metals--are discussed. New industrial procedures for working alloys based on the application of MHD phenomena are described.

The book is designed for scientific and engineering-technical personnel interested in the development and application of magnetohydrodynamic devices.

There are 82 illustrations, 1 table and 119 references on pages 182-188.

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INDUSTRIAL TECHNOLOGY

UDC 539.3+519.2

STRENGTH AND RELIABILITY OF TECHNICAL DEVICES

Kiev PROCHNOST' I NADEZHNOST' TEKHNICHESKIKH USTROYSTV in Russian 1981 (signed to press 8 Jan 81) pp 2, 184-191

[Annotation and abstracts of articles from collection "Strength and Reliability of Technical Devices", editor-in-chief V. S. Gudramovich, Institute of Technical Mechanics, UkSSR Academy of Sciences, Izdatel'stvo "Naukova dumka", 1300 copies, 192 pages]

[Text] Current problems of strength and reliability of technical devices are discussed. Data are presented on theoretical and experimental studies of the load-bearing capacity and strength of thin-walled systems under static and dynamic loads with consideration of plastic deformations and creep strains with complex loading histories. Contact problems of the theory of shells are discussed as well as questions of optimum design of structural members. Some problmes of the theory of accelerated tests are outlined. Some questions of checking the working condition of technical devices are examined.

The collection is intended for engineers and researchers specializing in the field of strength and reliability, and also for undergraduate and graduate students with the corresponding majors in colleges and universities.

UDC 539.374

OSCILLATIONS OF SYSTEM WITH PHYSICALLY NONLINEAR CHARACTERISTICS UNDER PULSE LOADING

[Abstract of article by Velichkin, V. A., Gudramovich, V. S., Konovalenko, V. Ya., Makeyev, Ye. M., Pilipenko, V. V., Popov, A. I., Semenenko, V. P. and Fediy, S. P.]

[Text] An analysis is made of oscillations of a complex mechanical system that consists of two masses with plates on springs fastened to an imponderable hinged beam of finite stiffness under pulsed action with consideration of physically nonlinear properties of the spring material.

Accelerations are studied as a function of the magnitude and time of application of pulses, ratio of stiffnesses and plastic properties of the springs, and clearances between the plate and the beam.

Calculated and experimental values of accelerations are compared.

Figures 5, references 4 Russian.

UDC 519.2+624.011:539.4

PROBABILISTIC PROPERTIES OF FREQUENCIES AND REDUCED MASSES OF TRANSVERSE OSCILLATIONS OF THIN-WALLED FLUID-FILLED CYLINDRICAL SHELL

[Abstract of article by Velichkin, V. A. and Fediy, S. P.]

[Text] The paper formulates the problem of finding the probabilistic characteristics of a dynamic system by the method of statistical modeling.

An analysis is made of the probabilistic properties of frequencies and reduced masses of transverse oscillations of a dynamic system based on the example of a reinforced cylindrical shell filled with liquid.

The conclusions are illustrated by results of numerical calculation.

Figure 1, table 1, references 3 Russian.

UDC 539.384.6:624.074.4

STRESS-STRAIN STATE OF CYLINDRICAL SHELL UNDER LOADING VIA CIRCULAR ELASTIC SADDLE SUPPORTS

[Abstract of article by Gayduchenko, A. P., Katan , L. I. and Makeyev, Ye. M.]

[Text] The authors give the results of an experimental study of the deformed state of a cylindrical shell lying on elastic circular saddle supports of finite dimensions. Calculated data found by a previously developed method are compared with experimental results, and an estimate is made of the influence that accounting for the interaction of support and shell has on the stressed state of the shell in the vicinity of the support based on the example of a tested structure.

Figures 3, table 1, references 6: 4 Russian, 1 Polish, 1 Western.

UDC 539.374

EXPERIMENTAL STUDY OF CYLINDRICAL SHELL LOAD-BEARING CAPACITY UNDER LOCAL LOAD

[Abstract of article by Gerasimov, V. P., Gudramovich, V. S.]

[Text] The paper gives the results of experimental studies of the load-bearing capacity of cylindrical shells deformed beyond the elastic limit under the action of local loads of various classes applied to the surface of shells by rigid saddle-support punches.

Figures 5, tables 3, reference 1 Russian.

UDC 539.374

EXPERIMENTAL STUDIES OF BAUSCHINGER EFFECT FOR AMG-6M ALUMINUM ALLOY

[Abstract of article by Konovalenkov, V. S]

[Text] The author gives the results of experimental studies of the Bauschinger effect for AMG-6M. The tests were done on specimens of circular cross section. Results of processing experimental data are given in the form of tables and a graph.

Figures 2, table 1, references 3 Russian.

UDC 539.384.6:624.072.4

EXPERIMENTAL STUDY OF DEFORMATION IN BULKHEADS REINFORCING CYLINDRICAL SHELL UNDER TRANSVERSE LOCAL LOADING

[Abstract of article by Makeyev, Ye. M. and Semenenko, V. P.]

[Text] The paper gives the results of an experimental study of the deformed state of circular bulkheads that reinforce a cylindrical shell under local loads of various kinds with different laws of distribution in different combinations with consideration of their deformation together with the shell. Calculated data are compared with experimental results, and an analysis is made of the effect of some simplifying assumptions that are generally made in approximate solutions.

Figures 4, tables 4, references 4 Russian.

UDC 624.074.4

MODEL OF MATCHING REQUIREMENTS IN MULTICRITERIAL STRUCTURAL DESIGN PROBLEMS

[Abstract of article by Pochtman, Yu. M. and Skalozub, V. V.]

[Text] The authors discuss the feasibility of using an n-person game model with a considerable set of criteria in multicriterial problems of optimizing structural members. It is noted that the model permits simultaneous analysis of stages of "external" and "internal" design; some conditions are presented that are sufficient for constructing a minimax solution with required properties. A technique is given for reducing certain problems of optimizing beams, plates and shells to game problems. Numerical examples are given.

Figure 1, table 1, references 10 Russian.

UDC 539.376

STABILITY OF STRUCTURALLY ORTHOTROPIC SHELLS UNDER CREEP CONDITIONS

[Abstract of article by Poshivalov, V. P.]

[Text] The author derives equations of stability of structurally orthotropic shells under conditions of creep. It is assumed that during creep the stresses and strains in the shell differ little from the stresses of the basic zero-moment state. Consideration is taken of the biaxial stressed state of the shell and the uniaxial stressed state of the reinforcing elements.

References 4 Russian.

UDC 539.3

STRESSES IN REGION OF JOINING BETWEEN PIPE AND CYLINDRICAL SHELL

[Abstract of article by Sel'skiy, Yu. S.]

[Text] An examination is made of the stressed state of a joint for small ratios of pipe and shell radii.

Coefficients of concentration are obtained for cases of internal pressure, uniform temperature differential, and also the longitudinal force and torques applied to the pipe.

It is proposed that a ray method be used for problems with arbitrary boundary shape (as applied to intersections at an arbitrary angle with small ratios of radii).

Figures 4, tables 2, references 10: 8 Russian, 2 Western.

UDC 539.3

PRINCIPAL EQUATIONS IN THEORY OF SHELLS WITH DIFFERENT MODULI IN TENSION AND COMPRESSION

[Abstract of article by Tamurov, N. G. and Turovtsev, G. V.]

[Text] The paper proposes a model for a material that has different elastic characteristics in compression and tension. Without discussing the details of deriving the model, the authors find the principal equations of a theory of thin shells in the geometrically nonlinear approximation that are converted to the corresponding classical expressions in the case of a single-modulus material. A method is outlined for solving problems with the proposed physical relations.

References 7: 5 Russian, 2 Western.

UDC 539.384.6:624.072.4

DESIGNING REINFORCEMENT RINGS WITH MAXIMUM BENDING STIFFNESS

[Abstract of article by Tkacheva, T. V.]

[Text] The Prager optimality criterion is used to solve the problem of optimizing the cross sectional shapes of rings that reinforce a cylindrical shell under loading by a concentrated transverse force. The optimum area of the supporting layers of a sandwich ring is determined. An example of calculation is given.

Figures 3, references 6: 4 Russian, 2 Western.

UDC 624.072.4

PROPERTIES OF OPTIMUM RING FRAME DESIGNS UNDER COMBINED FORCE AND TEMPERATURE LOADING

[Abstract of article by Binkevich, Ye. V. and Dzyuba, A. P.]

[Text] The problem of designing nonuniformly heated ring frames loaded by concentrated forces is numerically solved on the basis of a maximum principle with consideration of limitations on deviation of the shape of the axis from circular, and also strength and structural deviations. The properties of optimum designs are studied. An analysis is made of the particulars of using necessary conditions of optimality in design. Results of digital computer calculations are given. Figures 4, table 1, references 3 Russian.

UDC 539.374

DYNAMIC BEHAVIOR OF PLASTIC SPHERICAL SHELL UNDER REPEATED PULSED LOADING

[Abstract of article by Gudramovich, V. S. and Shatsillo, S. I.]

[Text] A formulation is given for the problem of dynamic analysis of the motion of a plastic spherical shell under repeated pulsed loading. The factor that accounts for the nonuniformity of properties that is acquired during preceding loading is the fields of residual stresses. The finite element method is used for the snalysis; the finite elements are selected in the form of frusta of cones. An examination is made of the problem of motion of an initially homogeneous shell with pulsed loading on the basis of isotropic flow theory. References 14: 12 Russian, 2 Western.

UDC 539.3

CYLINDRICAL SHELL BEHAVIOR WITH DYNAMICALLY APPLIED NONUNIFORM EXTERNAL PRESSURE

[Abstract of article by Makarenko, A. D. and Makarenko, N. B.]

[Text] The authors consider the problem of stability of a cylindrical shell with dynamic application of nonaxisymmetric external pressure. Results of calculations are given that were obtained with the use of a specially compiled algorithm enabling investigation of the process of deformation and loss of stability of shells with respect to space and time coordinates. Figures 3, references 3 Russian.

UDC 519.248:62-192

DETERMINING INVARIANCE IN ACCELERATED TEST THEORY

[Abstract of article by Avramenko, V. I.]

[Text] A definition of invariance of a factory that produces goods by batches is proposed on the basis of a statistical probability approach that is different from the existing definition. A definition of invariance of the properties of items relative to a load is introduced. An illustration is given of the feasibility of using these concepts to solve problems of the theory of accelerated tests. References 4 Russian.

UDC 519.2.48:62-192

MATHEMATICAL MODEL OF SYSTEM DEGRADATION

[Abstract of article by Belosvetov, S. A. and Dyachenko, V. Ya.]

[Text] Based on an approach to systems analysis as comprising a paired random process and level, the authors propose mathematical models of processes of structural and phase degradation of a system as a basis for developing algorithms and for machine simulation of real degradation processes to evaluate the reliability of a system. Reference 1 Russian.

UDC 629.4:519.2-192

DEFINITION OF RATIONAL MAINTENANCE SYSTEM FOR COMPONENTS OF TRACTION UNITS

[Abstract of article by Bosov, A. A. and Khandriga, A. G.]

[Text] A method is proposed for determining the periods of renovating components of traction units. An example is given of designing and setting up a system for maintenance of the DT-9N traction engine. Figures 5, tables 4, references 2 Russian.

UDC 519.24

SOME PARTICULARS OF USING STANDARDIZATION IN COMPUTERIZING REGRESSION ANALYSIS ALGORITHMS

[Abstract of article by Dolgiy, V. I., Perlik, V. I. and Sokolov, A. S.]

[Text] An examination is made of the effect of the sequence of linearizing and standardizing variables in regression analysis algorithms. It is shown that when constructing incomplete second-order models, algorithms based on standardizing variables before linearizing are more correct in all cases. Table 1, references 4 Russian.

UDC 539.2

EVALUATING DURABILITY OF MATERIALS UNDER AGING CONDITIONS

[Abstract of article by Pereverzev, Ye. S.]

[Text] Approximate analytical expressions are given for evaluating the durability of materials under aging conditions. Approximate methods are considered for taking account of damage accumulation due to mechanical loading and natural aging. References 4 Russian.

UDC 621.3

RELATION BETWEEN THERMODYNAMIC AND STATISTICAL EQUIVALENCE PRINCIPLES

[Abstract of article by Pereverzev, Ye. S.]

[Text] Based on a thermodynamic approach, conditions are found for which the thermodynamic principle of equivalence is identical to the statistical method of equal probabilities. It is shown that in this case the failure rate is proportional to the rate of increase in entropy. References 4 Russian.

UDC 519.281

STATISTICAL ESTIMATION OF TECHNICAL SYSTEM RELIABILITY INDICES BY EFFICIENCY FUNCTION METHOD

[Abstract of article by Perlik, V. I.]

[Text] The author considers the statistical aspects of the method of efficiency fucntions; the main principles of the method have been formulated in previous papers by this author. First an investigation is made of the theoretical sequence of statistical estimation of the index of reliability of a system, which in principle enables derivation of the exact solution. Practical implementation of this sequence is difficult because of insurmountable mathematical difficulties. The process of approximate statistical estimation recommended in the article is based on a moment approach. A detailed examination is made of problems involved in getting the initial information. References 3 Russian.

UDC 519.24

STATISTICAL METHOD OF EVALUATING SCALE FACTOR FOR ROCK STRENGTH PROPERTIES

[Abstract of article by Rubets, G. T.]

[Text] Various generalizations of the Weibull theory of strength are examined on the basis of the theory of distributions of ordered statistics.

It is shown how these generalizations can be applied to estimation of the scale factor of rock strength characteristics, as well as those of structural components in underground structures. References 10: 5 Russian, 5 Western.

UDC 519.248:62-192

MODEL TO ACCOUNT FOR A PRIORI INFORMATION IN ACCELERATED TEST PROBLEM

[Abstract of article by Rybalka, K. P.]

[Text] The author examines a method of determining how the mathematical expectation of time to failure ($M\xi$) depends on the load vector ($\vec{\varepsilon}$) from results of accelerated tests using a priori information on the distribution law for time to failure and the function of change in $M\xi$ as dependent on $\vec{\varepsilon}$. References 3 Russian.

UDC 621:004.17:620.199

BAYES MODEL FOR SEQUENTIAL EVALUATION OF TECHNICAL SYSTEM RELIABILITY INDEX

[Abstract of article by Savchuk, V. P.]

[Text] A mathematical model is proposed for evaluating the lower confidence limit of the reliability index for predetermined confidence coefficient. The model uses a Bayes approach. Inversion of the model enables planning of the number of demonstration tests when confirming reliability. It is shown that carrying out tests of long duration considerably reduces the number of test specimens as compared with the conventional method. Table 1, references 3 Russian.

UDC 621.3.019.3:519

DETERMINING RELIABILITY REQUIREMENTS FOR ITEMS FROM PREDETERMINED LEVEL OF SYSTEM RELIABILITY

[Abstract of article by Skripnik, V. M. and Spirkov, S. N.]

[Text] An expression is derived for determining guaranteed up times of items in a complex system with predetermined system reliability requirement when the distribution of service lives of the items conforms to Weibull law. Figure 1, references 3 Russian.

UDC 620.171.311.2-192

POSSIBLE METHODS OF RECALCULATING RELIABILITY CHARACTERISTICS UNDER DIFFERENT LOADING CONDITIONS

[Abstract of article by Stepanov, V. V.]

[Text] The author considers some possible methods of recalculating reliability characteristics of technical devices under different loading conditions. A relationship is established between the proposed formulas and known principles of reliability theory. Figures 2, references 5 Russian.

UDC 62:519.2

READINESS FACTOR AND OVERHEAD COST OF TECHNICAL DEVICE FOR COMBINED MONITORING AND VARIOUS RATES OF RESTORING OPERABILITY

[Abstract of article by Chumakov, L. D.]

[Text] Analytical expressions are found for the readiness factor, operating costs and average number of restorations of operability in the case of exponential distribution of the time of fail-free operation and restoration. References 3 Russian.

UDC 62-192:539.3

DURABILITY EVALUATION METHOD USING SEMI-MARKOV MODEL

[Abstract of article by Shiyan, O. V.]

[Text] An examination is made of a method of accelerated tests -- the stepwise loading method. A method is proposed for estimating the parameters of the failure distribution function under operating conditions using the Korolyuk theorem of mean sojourn time of a semi-Markov process in a fixed set of states and the principle of maximum likelihood. References 3 Russian.

UDC 62:519.2

EVALUATING DURABILITY OF TECHNICAL DEVICE WITH PERIODIC WORKABILITY CHECK

[Abstract of article by Chumakov, L. D.]

[Text] An estimate is obtained for the durability of a technical device with periodic monitoring of its working state when the mathematical expectation of the time of fail-free operation is known. References 3 Russian.

6610
CSO: 1861/189

TURBINE AND ENGINE DESIGN

UDC 621.165000.5

OPTIMAL LAST STAGE DESIGN OF HIGH-POWER STEAM TURBINES

Kiev OPTIMAL'NOYE PROYEKTIROVANIYE POSLEDNEY STUPENI MOSHCHNYKH PAROVYKH TURBIN in Russian 1980 (signed to press 7 Jul 80) pp 2, 226-227

[Annotation and table of contents from book "Optimal Last Stage Design of High-Power Steam Turbines", by Leonid Aleksandrovich Shubenko-Shubin, Anatoliy Alekseyevich Tarelin and Yuriy Petrovich Antiptsev, Ukrainian SSR Academy of Sciences, Problems of Machine Building Institute, Izdatel'stvo "Naukova dumka", 1000 copies, 228 pages]

[Text] This monograph discusses the problems of optimal last-stage design of high-power steam turbines. A study is made of the physicomathematical and engineering principles of optimal turbine stage design, optimization of the thermal gas dynamic process and blade design. The following are presented: a mathematical model of the thermal gas dynamic process and its investigation using the existing mathematical methods of finding optimal solutions and modern computer engineering means. The methods of constructing and optimizing the blade cascades are presented which permit economical profiles to be obtained in a wide range of variation of the initial specifications and geometric characteristics. A formalized blade design process is proposed which offers the possibility of using simple methods of optimizing various types of designs which are convenient in practical application.

The book is designed for scientific and engineering-technical personnel specializing in turbine building.

There are 54 illustrations, 3 tables, and 59 references on pages 222-225.

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NAVIGATION AND GUIDANCE SYSTEMS

UDC 531.55:521.1

CORRECTING INERTIAL GUIDANCE SYSTEMS BY USING COMBINED SUBSIDIARY POSITIONAL AND VELOCITY INFORMATION

Moscow IZVESTIYA AKADEMII NAUK SSSR: MEKHANIKA TVERDOGO TELA in Russian No 5, Sep-Oct 81 (manuscript received 9 Oct 79) pp 12-19

[Article by V. I. Kalenova, V. M. Morozov, N. A. Parusnikov and A. G. Shakot'ko, Moscow]

[Text] An investigation is made of the problem of correcting an inertial guidance system with combined utilization of redundant positional and velocity information. In connection with the solution of this problem the authors discuss one of the possible correcting algorithms that realize decomposition of the problem with respect to components of the correction vector.

In present-day navigational systems that are based on the inertial guidance method, the inertial information is supplemented by information of noninertial nature to improve accuracy properties. The problem of correcting inertial guidance systems by using redundant information can be formulated as a problem of estimating the vector of state of a linear system from given measurements [Ref. 1, 2].

To evaluate the possibilities of correcting such systems with respect to subsidiary information of a given type, the linear theory of observability should be used [Ref. 3, 4]. Analysis of the observability of the system enables discrimination of the variables that can be evaluated, determination of the limiting accuracy of evaluation of such variables, and also construction of a class of suitable algorithms of evaluation that are subject to further detailed study. In addition, analysis of observability enables estimation of errors due to various simplifications in constructing the correction algorithm.

In Ref. 2, 5 an analysis was made of observability, and some correction algorithms were constructed for inertial guidance systems using either positional or velocity subsidiary

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information. We discuss below some modifications of combined use of this information, and propose corresponding estimation algorithms. In doing so, we use one of the methods of decomposing the observed space with respect to the components of the correction vector.

1. Consider the linear system

$$d\xi/dt=G\xi+q, \quad \sigma=H\xi+r \tag{1.1}$$

Here ξ is an n-dimensional vector of state, σ is an s-dimensional vector of measurement, G, H are fixed matrices of the corresponding dimensionality, q, r are random vector processes of the white noise type with given intensities. It is assumed that the pair (G, H) is observable. It is required to get an estimate ξ° of quantity ξ by measurement σ on segment $[t_0, t]$.

In constructing practical estimation algorithms, it is often advisable to decompose (split) the estimation problem with respect to the components of the measurement vector so that this problem is reduced to a series of subproblems with scalar measurements. Generally speaking, the decomposition can be carried out in different ways [Ref. 6]. Let us consider the problem of decomposition for a two-component measurement vector $\sigma=(\sigma_1, \sigma_2)^T$, having in mind a situation where the time interval on which information σ_1 and σ_2 is used in conjunction is preceded by a long time interval with arrival of only information σ_1.

Let $\sigma_1=h_1^T\xi$, $\sigma_2=h_2^T\xi$ and let {X}, {Y} be spaces observable by using measurements σ_1 and σ_2 respectively.

We use the notation $\dim\{\xi\}=n$, $\dim\{X\}=m$, $\dim\{Y\}=l$. Obviously $m\leqslant n$, $l\leqslant n$, $m+l\geqslant n$. Let us also introduce the notation

$$\begin{gathered} a_1=h_1,\ a_2=G^Ta_1,\dots,\ a_{j+1}=G^Ta_j,\dots \\ b_1=h_2,\ b_2=G^Tb_1,\dots,\ b_{j+1}=G^Tb_j,\dots \end{gathered} \tag{1.2}$$

The vector $x=(x_1x_2\dots x_m)^T$ made up of components observable by means of quantity σ_1 takes the form

$$x=L_x\xi,\quad L_x=C_1(a_1a_2\dots a_m)^T \tag{1.3}$$

where C_1 is some nondegenerate square matrix whose selection is not directly related to the decomposition problem. In virtue of the linear dependence of vectors $a_1, a_2,\dots, a_m, a_{m+1}$ we have the representation

$$a_{m+1}=\vartheta_1a_1+\vartheta_2a_2+\dots+\vartheta_ma_m \tag{1.4}$$

Here the θ_i are constant coefficients. From (1.1)-(1.4) we get

$$dx/dt=G_xx+q_x,\quad \sigma_1=g_x^Tx+r_x \tag{1.5}$$

$$G_x=C_1\theta C_1^{-1} \qquad q_x=L_xq \quad \theta=\begin{Vmatrix} 0 & 1 & 0 & \dots & 0 \\ 0 & 0 & 1 & \dots & 0 \\ \cdot & \cdot & \cdot & \cdot & \cdot \\ 0 & 0 & 0 & \dots & 1 \\ \vartheta_1 & \vartheta_2 & \vartheta_3 & \dots & \vartheta_m \end{Vmatrix} \qquad g_x^T=(10\dots0)\,C_1^{-1}$$

The standard algorithm that yields estimate x° of vector x is described by the equation

$$dx^\circ / dt = G_x x^\circ + K_x(\sigma_1 - g_x^T x^\circ) \qquad (1.6)$$

Vector K_x is unambiguously determined either by the Kalman optimum filtration method, or from the condition of assignment of roots of characteristic equation $|pE - G_x + K_x g_x^T| = 0$. At sufficiently low intensity of the measurement error r_x, the initial condition $x^\circ(t_0) = x_0^\circ$ should be given as $x_0^\circ = C_1(\sigma_{10} 0 0 \ldots 0)^T$.

Let us introduce vector $z = (z_1 z_2 \ldots z_{n-m})^T$, $n - m \leqslant l$ so that

$$z = L_z \xi, \quad L_z = C_2(b_1 b_2 \ldots b_{n-m})^T \qquad (1.7)$$

where C_2 is a square nondegenerate matrix.

Since the set of vectors $a_1, a_2, \ldots, a_m, b_1, b_2, \ldots, b_{n-m}$ can serve as a basis of the space $\{\xi\}$, vectors ξ and $\left\|\begin{matrix} x \\ z \end{matrix}\right\|$ are related by the mutually equivalent transformation

$$\left\|\begin{matrix} x \\ z \end{matrix}\right\| = L\xi, \quad L = \left\|\begin{matrix} L_x \\ L_z \end{matrix}\right\|, \quad \det L \neq 0 \qquad (1.8)$$

By analogy with (1.5), we get

$$dz / dt = G_{z1} x + G_{z2} z + q_z, \quad \sigma_2 = g_z^T z + r_z \qquad (1.9)$$

$$G_{z1} = C_2 \Phi_x C_1^{-1}, \; q_z = L_z q, \; G_{z2} = C_2 \Phi_z C_2^{-1}, \; g_z^T = (10 \ldots 0)^T C_2^{-1} \qquad (1.10)$$

$$\Phi_x = \left\|\begin{matrix} 0 & 0 \ldots 0 \\ 0 & 0 \ldots 0 \\ \cdot & \cdot\;\cdot\;\cdot\;\cdot \\ 0 & 0 \ldots 0 \\ \varphi_1 & \varphi_2 \ldots \varphi_m \end{matrix}\right\|, \quad \Phi_z = \left\|\begin{matrix} 0 & 1 & 0 & \ldots 0 \\ 0 & 0 & 1 & \ldots 0 \\ \cdot & \cdot & \cdot & \cdot\;\cdot\;\cdot \\ 0 & 0 & 0 & \ldots 1 \\ \varphi_{m+1} & \varphi_{m+2} & \varphi_{m+3} & \ldots \varphi_n \end{matrix}\right\|$$

and the elements of matrices Φ_x, Φ_z are determined from the relation

$$b_{n-m+1} = \varphi_1 a_1 + \ldots + \varphi_m a_m + \varphi_{m+1} b_1 + \ldots + \varphi_n b_{n-m}$$

We seek the estimate z° of vector z in the form

$$dz^\circ / dt = G_{z1} x^\circ + G_{z2} z^\circ + K_z(\sigma_2 - g_z^T z^\circ) \qquad (1.11)$$

estimate x° being given by algorithm (1.6). Vector K_z is uniquely defined if roots are assigned for characteristic equation $|pE - G_{z2} + K_z g_z^T| = 0$. Estimate ξ° of vector ξ is found by using relation (1.8)

$$\xi^0 = L^{-1} \left\|\begin{matrix} x^0 \\ z^0 \end{matrix}\right\| \qquad (1.12)$$

The algorithm described by relations (1.6), (1.11), (1.12) has a two-stage structure. Let us make the following remarks about algorithms of this kind.

If information σ_1 arrives on interval $[t_0, t_1]$, then for sufficiently small errors q_x, r_x the error of the estimate $\Delta x = x - x^\circ$ can also be made sufficiently small by time t_1 by suitable selection of K_x. This reduces the influence of this error on the estimate of quantity z found by algorithm (1.11) with combined use of information σ_1 and σ_2 on interval $[t_1, t]$.

If vector z does not have excessively high dimensionality, then the time interval necessary for getting an acceptable estimate z° may not be too long. The latter circumstance is often decisive for practical use of an estimation algorithm.

Comment. A more general case of realization of the idea of "two-stage structure" is this: vector x is defined as before by relation (1.3); vector z is defined by transformation $z = L_z^\circ \xi$ such that the rows of matrix L_z° and the rows of matrix L_z are linearly independent, and therefore matrix $L^\circ = \left\| \begin{matrix} L_x \\ L_z^\circ \end{matrix} \right\|$ is a square degenerate matrix (otherwise matrix L_z° is arbitrary). In this case, by analogy with (1.9) we get

$$dz/dt = G_{z1}^\circ x + G_{z2}^\circ z + q_z, \quad \sigma_2 = g_{z1}^T x + g_{z2}^T z + r_z$$

where G_{z1}°, G_{z2}°, g_{z1}^T, g_{z2}^T are determined from (1.10) by transformation L_z°. The corresponding algorithm of estimation of quantity z takes the form

$$dz^\circ / dt = G_{z1}^\circ x^\circ + G_{z2}^\circ z^\circ + K_z^\circ (\sigma_2 - g_{z1}^T x^\circ - g_{z2}^T z^\circ) \qquad (1.13)$$

It is essential that the estimate x° given by algorithm (1.6) must not depend on measurement σ_2. As a consequence of this circumstance, algorithm (1.13) is generally speaking less preferable than (1.11) since the error of the estimate $\Delta x = x - x^\circ$ influences the error of the estimate $\Delta z = z - z^\circ$ with weighting factor K_z° -- the gain selected from the condition of a sufficiently high degree of asymptotic stability of the equations of errors of the estimate yielded by algorithm (1.13).

Let us give a useful modification of algorithm (1.6), (1.11), (1.12) for a special case. Let the set of vectors $a_1, a_2, \ldots, a_i, b_1, b_2, \ldots, b_j$ for some $i < m$ form the basis of space $\{\xi\}$. Let us introduce the vectors $u = L_u \xi$, $L_u = C_{11}(a_1 a_2 \ldots a_i)^T$; $v = L_v \xi$, $L_v = C_{12}(a_{i+1} a_{i+2} \ldots a_m)^T$; $w = L_w \xi$, $L_w = C_{22}(b_1 b_2 \ldots b_j)^T$, where C_{11}, C_{12}, C_{22} are nondegenerate square matrices. The behavior of vectors u, v, w conforms to equations

$$\begin{aligned} du/dt &= G_{u1} u + G_{u2} v + q_u \\ dv/dt &= G_{v1} u + G_{v2} v + q_v \\ dw/dt &= G_{w1} u + G_{w2} w + q_w \end{aligned} \qquad (1.14)$$

$$G_{u1} = C_{11} F C_{11}^{-1}, \quad G_{v1} = C_{12} \Theta_u C_{11}^{-1}, \quad G_{w1} = C_{22} D_u C_{11}^{-1}$$

$$G_{u2} = C_{11} F^\circ C_{12}^{-1}, \quad G_{v2} = C_{12} \Theta_v C_{12}^{-1}, \quad G_{w2} = C_{22} D_w C_{22}^{-1}$$

$$q_u = L_u q, \quad q_v = L_v q, \quad q_w = L_w q$$

$$F = \left\| \begin{matrix} 0 & 1 & 0 & \ldots & 0 \\ 0 & 0 & 1 & \ldots & 0 \\ . & . & . & . & . \\ 0 & 0 & 0 & \ldots & 1 \\ 0 & 0 & 0 & \ldots & 0 \end{matrix} \right\|, \quad F^\circ = \left\| \begin{matrix} 0 & 0 & \ldots & 0 \\ 0 & 0 & \ldots & 0 \\ . & . & . & . \\ 0 & 0 & \ldots & 0 \\ 1 & 0 & \ldots & 0 \end{matrix} \right\|$$

$$\Theta_u = \begin{Vmatrix} 0 & 0 & \dots & 0 \\ \dots & \dots & \dots & \dots \\ 0 & 0 & \dots & 0 \\ \vartheta_1 & \vartheta_2 & \dots & \vartheta_i \end{Vmatrix}, \quad \Theta_v = \begin{Vmatrix} 0 & 1 & \dots & 0 \\ \dots & \dots & \dots & \dots \\ 0 & 0 & \dots & 1 \\ \vartheta_{i+1} & \vartheta_{i+2} & \dots & \vartheta_m \end{Vmatrix}.$$

$$D_u = \begin{Vmatrix} 0 & 0 & \dots & 0 \\ \dots & \dots & \dots & \dots \\ 0 & 0 & \dots & 0 \\ \delta_1 & \delta_2 & \dots & \delta_i \end{Vmatrix}, \quad D_w = \begin{Vmatrix} 0 & 1 & \dots & 0 \\ \dots & \dots & \dots & \dots \\ 0 & 0 & \dots & 1 \\ \delta_{i+1} & \delta_{i+2} & \dots & \delta_n \end{Vmatrix}$$

and the elements of matrices D_u, D_w are determined from the relation

$$b_{j+1} = \delta_1 a_1 + \dots + \delta_i a_i + \delta_{i+1} b_1 + \dots + \delta_n b_j$$

Let us note that the dimensionality of system (1.14), equal to $m+j$, is greater than that of initial system (1.1) by an amount $m-i$, and vectors ξ and $\begin{Vmatrix} u \\ w \end{Vmatrix}$ are related by the reciprocally equivalent transformation

$$\begin{Vmatrix} u \\ w \end{Vmatrix} = L_* \xi, \quad L_* = \begin{Vmatrix} L_u \\ L_w \end{Vmatrix}$$

Components σ_1, σ_2 of measurement vector σ are: $\sigma_1 = g_u^T u + r_1$, $g_u^T = (1\,0 \dots 0) C_{11}^{-1}$, $\sigma_2 = g_w^T w + r_2$, $g_w^T = (1\,0 \dots 0) C_{22}^{-1}$.

The corresponding algorithm for estimation of vector ξ takes the form

$$du^\circ / dt = G_{u1} u^\circ + G_{u2} v^\circ + K_u (\sigma_1 - g_u^T u^\circ)$$
$$dv^\circ / dt = G_{v1} u^\circ + G_{v2} v^\circ + K_v (\sigma_1 - g_u^T u^\circ)$$
$$dw^\circ / dt = G_{w1} u^\circ + G_{w2} w^\circ + K_w (\sigma_2 - g_w^T w^\circ)$$
$$\xi^\circ = L_*^{-1} \begin{Vmatrix} u^\circ \\ w^\circ \end{Vmatrix}$$

Let us make a comment regarding possible selection of the quantity $i = \dim u$. Heuristic considerations reinforced by practical experience show that when estimating components of the vector of state that are derivatives of an observable quantity, the time required for estimating a given component with predetermined accuracy increases with the derivative order of this component. Therefore it is desirable if possible to form the basis of space $\{\xi\}$ by a set of vectors $\{a_1, a_2, \dots, a_i, b_1, b_2, \dots, b_j\}$, such that $\max(i, j)$ is minimum.

2. Consider a two-component inertial guidance system with levelable platform in which altitude information is provided by altimeters or given a priori. The following trihedral representations are used in describing inertial systems [Ref. 1]: ideal trihedron $Mx_1x_2x_3$(Mx), point M being identified with the moving object, Mx_3 is the direction of the local vertical, orientation of trihedron Mx in the azimuth is given; instrumental trihedron $Mz_1z_2z_3$(Mz), which is the material realization of trihedron Mx in the navigation system; model trihedron $My_1y_2y_3$(My), which is the numerical pattern of the ideal

trihedron (the orientation of the trihedron is determined by numerical information contained in the computer of the inertial system).

The mutual orientation of trihedra Mx, Mz is determined by the vector of small rotation $\alpha_x = (\alpha_1 \alpha_2 \alpha_3)^T$; for trihedra My, Mz -- by vector $\beta_x = (\beta_1 \beta_2 \beta_3)^T$; for trihedra Mx, My -- by vector $\gamma_x = (\gamma_1 \gamma_2 \gamma_3)^T$. The quantities α_i, β_i, γ_i $(i = 1, 2, 3)$ are projections of the corresponding vectors on the axes of trihedron Mx. Let us denote the vector of absolute angular velocity of trihedron Mx in projections on its own axes by $\omega_x = (\omega_{x1} \omega_{x2} \omega_{x3})^T$. To be specific, we will assume that trihedron Mx is azimuthally free $(\omega_{x3} = 0)$. Having in mind mainly aircraft guidance, we will assume that the following conditions are met.

1. The velocity of motion of the vehicle is small compared with orbital velocity: $\omega_{x1}^2 + \omega_{x2}^2 \ll \omega_0^2$ (ω_0 is the Schuler frequency).

2. Evolutions of the vehicle with respect to altitude h are limited: $h \leqslant e^2 a$, $h^{\cdot} < 0.1 a \max (\omega_{x1}^2 + \omega_{x2}^2)^{1/2}$ (a is the semimajor axis of the terrestrial ellipsoid, e^2 is the square of eccentricity of this ellipsoid).

The equations of errors of the inertial guidance system in this case can be written as [Ref. 2]:

$$\begin{aligned} &\alpha_1^{\cdot} = \delta p_1 + \nu_{x1}, \quad \alpha_2^{\cdot} = \delta p_2 + \nu_{x2} \\ &\delta p_1^{\cdot} = -\omega_0^2 (\alpha_1 + \varepsilon_2) - \omega_{x2} \nu_{x3} \\ &\delta p_2^{\cdot} = -\omega_0^2 (\alpha_2 - \varepsilon_1) + \omega_{x1} \nu_{x3} \end{aligned} \tag{2.1}$$

$$\begin{aligned} &\beta_1^{\cdot} = -\omega_{x2} \beta_3 + \nu_{x1}, \quad \beta_2^{\cdot} = \omega_{x1} \beta_3 + \nu_{x2} \\ &\beta_3^{\cdot} = \omega_{x2} \beta_1 - \omega_{x2} \beta_2 + \nu_{x3} \end{aligned} \tag{2.2}$$

$$\gamma_1 = \alpha_1 - \beta_1, \quad \gamma_2 = \alpha_2 - \beta_2 \tag{2.3}$$

Here δp_1, δp_2 are variables that have the sense of velocity impulse, ν_{x1}, ν_{x2}, ν_{x3} are errors of inertial information about the projections of angular velocity of the instrumental trihedron, ε_1, ε_2 are the normalized errors of inertial information about the projections of the external force applied to point M on axes Mz_1, Mz_2.

We are considering the problem of correction of inertial guidance systems by using redundant positional and velocity information. The vector of velocity correction $\sigma_x^c = (\sigma_{x1}^c \sigma_{x2}^c)^T$ takes the form [Ref. 2]:

$$\begin{aligned} &\sigma_{x1}^c = \delta p_1 + u_{x3} \alpha_2 + u_{x2} \beta_3 - u_{x3} \beta_2 + w_{x1}^c \\ &\sigma_{x2}^c = \delta p_2 - u_{x3} \alpha_1 + u_{x3} \beta_1 - u_{x1} \beta_3 + w_{x2}^c \end{aligned} \tag{2.4}$$

The vector of positional correction takes the form [Ref. 2]:

$$\sigma_1^M = \gamma_1 + w_1^M, \quad \sigma_2^M = \gamma_2 + w_2^M \tag{2.5}$$

where u_{x1}, u_{x2}, u_{x3} are projections of the angular velocity of the earth on the axes of trihedron Mx, w_{x1}^c, w_{x2}^c, w_1^M, w_2^M are the normalized instrumental errors of the corresponding redundant information.

We will solve the correction problem with the following assumptions.

1. Motion of the vehicle is along "steady" trajectories $(\omega_{x1}^{*2}+\omega_{x2}^{*2})^{1/2}<\max(\omega_{x1}^2+\omega_{x2}^2)$ -- this condition is met in particular when the vehicle moves along a great circle at constant velocity.

2. The correction time t_k is small compared with the period associated with proper motion of the vehicle: $t_k \ll 2\pi/(\omega_{x1}^2+\omega_{x2}^2)^{1/2}$.

3. The model assumed for instrumental errors is: quantities ν_{xj}, ε_i are practically constant on the correction interval; $w_i^M=w_{i0}^M+r_i^M$, $w_{xi}^c=w_{xi0}^c+r_{xi}^c$, w_{i0}^M, w_{xi0}^c are nearly constant quantities on the correction interval; r_i^M, r_{xi}^c are random processes of the white noise type with given intensities ($i=1, 2$, $j=1, 2, 3$). We introduce dimensionless time $\tau=\omega_0 t$, and dimensionless quantities

$$\pi_j=\delta p_j/\omega_0,\ \sigma_j^c=\sigma_{xj}^c/\omega_0,\ \omega_i=\omega_{xi}/\mu\omega_0,\ u_i=u_{xi}/\mu\omega_0$$

$$\nu_i=\nu_{xi}/\omega_0\quad (i=1,2,3;\ j=1,2);\quad u=\max\sqrt{\omega_{x1}^2+\omega_{x2}^2}/\omega_0$$

Under the above conditions, $\mu\leqslant 0.1$. The characteristic values of variables α_j, π_j, ε_j, ν_i usually are of the same order ρ; the characteristic values of the variables β_i, γ_i by the instant of start of correction may be of order ρ/μ; the characteristic values of quantities ω_i, u_i are of the order of unity; the characteristic value of the quantity $\tau_k=\omega_0 t_k$ lies in a range of 1-3.

Let us introduce two groups of variables x_i ($i=1, 2,\dots, 6$) and y_j ($j=1, 2,\dots, 8$). Variables x_i are associated with analysis of observability of system (2.1), (2.2) by subsidiary velocity information (2.4). Variables y_j are associated with analysis of observability of system (2.1), (2.2) by subsidiary positional information (2.5):

$$\begin{aligned}
&x_1=\alpha_1+\varepsilon_2,\ x_2=\pi_1+\nu_1\\
&x_3=\mu(u_2\beta_3-u_3\beta_2)+w_{10}^c-\nu_1+\mu u_3\varepsilon_1\\
&x_4=\alpha_2-\varepsilon_1,\ x_5=\pi_2+\nu_2\\
&x_6=\mu(u_3\beta_1-u_1\beta_3)+w_{20}^c-\nu_2+\mu u_3\varepsilon_2
\end{aligned}\tag{2.6}$$

$$\begin{aligned}
&y_1=\gamma_1+w_{10}^M,\ y_2=\pi_1+\mu\omega_2\beta_3\\
&y_3=\alpha_1+\varepsilon_2,\ y_4=\pi_1+\nu_1\\
&y_5=\gamma_2+w_{20}^M,\ y_6=\pi_2-\mu\omega_1\beta_3\\
&y_7=\alpha_2-\varepsilon_1,\ y_8=\pi_2+\nu_2
\end{aligned}\tag{2.7}$$

Leaving the former notation for differentiation with respect to dimensionless time τ, and disregarding terms of order $\mu^2\rho$, we get the following relations:

$$\begin{aligned}
&x_1^{\cdot}=x_2,\ x_2^{\cdot}=-x_1-\mu\omega_2\nu_3\\
&x_3^{\cdot}=-\mu^2\omega_2(u_1\beta_2-u_2\beta_1)+\mu(u_2\nu_3-u_3\nu_2)\\
&x_4^{\cdot}=x_5,\ x_5^{\cdot}=-x_4+\mu\omega_1\nu_3\\
&x_6^{\cdot}=\mu^2\omega_1(u_1\beta_2-u_2\beta_1)+\mu(u_1\nu_3-u_3\nu_1)\\
&\sigma_1^c=x_2+x_3+\mu u_3 x_4+r_1^c,\ \sigma_2^c=x_5+x_6-\mu u_3 x_1+r_2^c
\end{aligned}\tag{2.8}$$

$$
\begin{gathered}
y_1^{\cdot}=y_2,\ y_2^{\cdot}=-y_3+\mu^2(\omega_1 y_5-\omega_2 y_1)\omega_2+\mu\omega_2^{\cdot}\beta_3 \\
y_3^{\cdot}=y_4,\ y_4^{\cdot}=-y_3-\mu\omega_2\nu_3,\ y_5^{\cdot}=y_6 \\
y_6^{\cdot}=-y_7+\mu^2(\omega_2 y_1-\omega_1 y_5)\omega_1-\mu\omega_1^{\cdot}\beta_3 \\
y_7^{\cdot}=y_8,\ y_8^{\cdot}=-y_7+\mu\omega_1\nu_3 \\
\sigma_1^M=y_1+r_1^M,\ \sigma_2^M=y_5+r_2^M
\end{gathered} \tag{2.9}
$$

We note that equations (2.8), (2.9) are inhomogeneous relative to variables x_i, y_j, but the terms that introduce inhomogeneity are of order $\mu\rho$. The quantity μ is treated here as a small parameter.

Analysis of observability of relations (2.8), (2.9) shows that variables x_i, y_j are readily observable variables when the appropriate redundant information is used, i. e. they permit an estimate with acceptable accuracy on the correction interval. In this regard, it is feasible for practical purposes to use the algorithms described in section 1 as estimation algorithms.

3. A comparison of the two groups of observable variables (2.6) and (2.7) shows that the space that is observable by using positional information is to a considerable extent overlapped by the space that is observable using velocity information. Besides, there are identical variables among those introduced above:

$$
\begin{gathered}
x_1=y_3=\alpha_1+\varepsilon_2,\ x_2=y_4=\pi_1+\nu_1 \\
x_4=y_7=\alpha_2-\varepsilon_1,\ x_5=y_8=\pi_2+\nu_2
\end{gathered} \tag{3.1}
$$

These variables are the dynamic errors of the inertial systems.

Combined utilization of positional and velocity information can be approached from two standpoints. According to one point of view, when positional information is available there is almost no need for velocity information. On the other hand, as a rule positional information is available for only a short period of time (about 30 minutes), whereas velocity information arrives over a much longer time interval, including the period preceding the interval of positional correction. Therefore it is advisable first to use velocity information to estimate the variables observable in this case, and then as the positional information begins to arrive, to use it for estimating the remaining observable variables. Such an approach is in some sense natural. First those variables are estimated that change fairly rapidly (with the Schuler frequency) and then the slowly changing variables are estimated.

In accordance with the results of sections 1 and 2, the following algorithm is proposed for combined processing of positional and velocity data:

$$
\begin{gathered}
x_1^{\circ\cdot}=x_2^\circ+K_1^c(\sigma_1^c-x_2^\circ-x_3^\circ-\mu u_3 x_4^\circ) \\
x_2^{\circ\cdot}=-x_1^\circ+K_2^c(\sigma_1^c-x_2^\circ-x_3^\circ-\mu u_3 x_4^\circ) \\
x_3^{\circ\cdot}=K_3^c(\sigma_1^c-x_2^\circ-x_3^\circ-\mu u_3 x_4^\circ) \\
x_4^{\circ\cdot}=x_5^\circ+K_4^c(\sigma_2^c-x_5^\circ-x_6^\circ+\mu u_3 x_1^\circ) \\
x_5^{\circ\cdot}=-x_4^\circ+K_5^c(\sigma_2^c-x_5^\circ-x_6^\circ+\mu u_3 x_1^\circ) \\
x_6^{\circ\cdot}=K_6^c(\sigma_2^c-x_5^\circ-x_6^\circ+\mu u_3 x_1^\circ)
\end{gathered} \tag{3.2}
$$

$$
\begin{aligned}
y_1^{\circ\cdot} &= y_2 + K_1^k(\sigma_1^M - y_1^\circ)\\
y_2^{\circ\cdot} &= \mu^2\omega_2(\omega_1 y_5^\circ - \omega_2 y_1^\circ) + K_2^k(\sigma_1^M - y_1^\circ) - x_1^\circ\\
y_5^{\circ\cdot} &= y_6^\circ + K_5^k(\sigma_2^M - y_5^\circ)\\
y_6^{\circ\cdot} &= \mu^2\omega_1(\omega_2 y_1^\circ - \omega_1 y_5^\circ) + K_6^k(\sigma_2^M - y_5^\circ) - x_4^\circ
\end{aligned}
\qquad (3.3)
$$

In the simplest feasible version, coefficients K_i^c, K_j^k are chosen so that the characteristic polynomials of subsystems of equations that describe the behavior of errors of the estimate Δx_1, Δx_2, Δx_3 and Δx_4, Δx_5, Δx_6 take the form $(p+\lambda_c)^3$ (disregarding relations between subsystems), while the characteristic polynomials of subsystems of equations relative to the quantities Δy_1, Δy_2 and Δy_5, Δy_6 take the form $(p+\lambda_k)^2$, where $\lambda_c > 0$, $\lambda_k > 0$ are the degrees of attenuation. On interval $[t_0, t_1]$, when only velocity information is coming in, $-K_1^k = K_2^k = K_5^k = K_6^k = 0$. The algorithm described by expressions (3.2), (3.3) was simulated on a digital computer at different noise intensities r_i^c and r_i^M and different values of t_0, t_1, λ_c, λ_k. The results of simulation showed that the algorithm is completely acceptable for practical implementation.

Let us note that relations (3.2), (3.3) can easily give a correction algorithm that accounts for all possible situations that might be encountered in the given problem, such as: only positional information; only velocity information but preceded by a run of positional information; neither positional nor velocity information after a combination correction run, etc.

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FLUID MECHANICS

UDC 533.6.013.14:629.7.025.3

SUPERSONIC FLOW PERTURBATIONS WITH INJECTION OF MASS AND HEAT

Kiev VOZMUSHCHENIYA SVERKHZVUKOVOGO POTOKA PRI MASSOTEPLOPODVODE in Russian 1980 (signed to press 9 Jan 80) pp 2-4

[Annotation and table of contents from book "Supersonic Flow Perturbations With Injection of Mass and Heat", by Nikolay Dmitriyevich Kovalenko, UkSSR Academy of Sciences, Izdatel'stvo "Naukova dumka", 1150 copies, 224 pages]

[Text] The monograph examines interaction of a supersonic stream with local sources of mass and heat. Models are described for various cases of interaction. The author considers the way that the physical properties and parameters of the injected and oncoming streams influence the flow pattern. Particular attention is given to investigation and development of engineering methods of calculating perturbations of pressure and aerodynamic forces on a surface in a flow as various gases and fluids are injected, and also to the study of the way that efficiency in setting up aerodynamic forces is influenced by the parameters of the injected streams and the injecting devices.

Intended for engineers and scientists working in the field of aerodynamics, and may be of use to instructors, undergraduate and graduate studnets in institutions of higher education in corresponding areas. Figures 99, references 140.

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TESTING AND MATERIALS

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INVESTIGATING EFFICIENCY OF SLIDING BEARINGS IN HELIUM ENVIRONMENT

Moscow ENERGOMASINOSTROYENIYE in Russian No 3, Mar 82 pp 36-39

[Article by Candidates of Technical Sciences R. G. Bogoyavlenskiy, Yu. D. Nikoforov, Yu. V. Makarov and A. T. Yermakov and engineer A. B. Anapol'skiy]

[Text] Important significance is now given to development of the energy production direction in nuclear power engineering on the basis of high-temperature gas reactors with helium coolant [1]. Investigations are being conducted in our country to develop an experimental chemical power plant with the VGR-50 reactor for simultaneous production of electric power and radiation products. A number of complex design, production and metallurgical problems must be solved for manufacture and further emergency-free operation of the plant. One of the problems is to ensure the efficiency of the bearings of the production circuit mechanisms.

The conditions of their operation are characterized by a highly pure helium medium, elevated temperature (approximately 300°C), the presence of radioactive radiation and the requirement of failure-free operation over a prolonged time (up to 400 days). This combination of operating conditions of bearings eliminates the use of liquid or plastic lubricants both because of their insufficiently high temperature stability and radiation resistance and because of the possibility of contaminating the coolant.

In the given case solid lubricant materials, among the wide nomenclature of which are temperature- and radiation-resistant materials, including neutral gas environments, are promising. Of special interest are the dichalcogenides of the transition metals of groups 5 and 6 of the periodic table and primarily the most thoroughly studied, widespread and economical of them--molybdenum disulfide (MoS_2).

Its basic advantages are the high temperature stability in a vacuum and in inert media (up to 800-1,000°C) [2], radiation stability [3] and the capability of ensuring the operation of a dry friction assembly over a wide range of speeds and loads. However, analysis of data presented in the literature [4] shows that the development of bearing assemblies with long service life (in time and method of friction), even when using MoS2 solid lubricant, is a problem.

Table 1

(1) Обозначение покрытия	(2) Химический состав покрытия	Температура стабильная (3) в He, °C
М-801	MoS_2	800
М-802-1	$MoS_2 + 20 \div 25$ объемных % Pb (4)	800
М-802-2	$MoS_2 - ZnS$ (твердый раствор) (5)	400—450

Key:
1. Notation of coating
2. Chemical composition of coating
3. Stable temperature in helium, °C
4. MoS_2 + 20-25 percent lead by volume
5. Solid solution

This is explained by the comparatively short life of molybdenum disulfide solid lubricant coatings (TSP) and the low wear resistance of self-lubricating structural materials (KSM) based on MoS_2. The significant effect of the counterbody material on the nature and intensity of wear of molybdenum disulfide TSP is shown in [5] and it is concluded that the optimum selection of the counterbody material with respect to specific operating conditions is an effecting means of increasing the life of solid lubricant coatings.

According to this conclusion, the effect of counterbody material on the friction characteristics of TSP and KSM based on MoS_2 in a helium environment at approximately 300°C was investigated to solve the postulated problem. Molybdenum disulfide solid lubricant coatings of the diffusion type produced by sulfiding molybdenum parts in sulphur-containing media, were selected as the main object of investigations among the group of TSP [4]. These coatings have comparatively high thickness (60-100 microns) and durability, high temperature stability in a vacuum and inert media (up to 800°C) during prolonged operation and are capable of tolerating large specific loads [6].

Several versions of the technology for producing coatings based on MoS_2 of the diffusion type, distinguished by composition and properties (Table 1), have been developed at VNIIOFI [expansion unknown].

M-810 and M-812 composites, also developed at VNIIOFI, were selected as the object of investigations from the group of isotropic KSM. The solid-lubricant base of M-810 is comprised of MoS_2-PbS and that of M-812 is comprised of MoS_2-PbS-$(FeMo)S_2$; the binders in both cases are metallic lead [7, 8].

Preliminary investigations of friction characteristics during friction with different counterbodies in a helium environment and at elevated temperatures (approximately 300°C) were conducted on the VVT-1M laboratory installation [9]. The installation consists of a system of evacuation pumps, friction assembly, motor with electromagnetic coupling, loading system, heater, cooled chamber, inlet system and system of instruments that record the force of friction, load, temperature in the contact zone, number of revolutions and gas vacuum and pressure.

The pumping system ensures preliminary evacuation in the test chamber of the installation in the range of 10^{-5} to 10^{-3} Pa. The drive system permits transfer of the torque from the DC motor by the electromagnetic coupling to the shaft being tested in the speed range of 0.05-0.7 m/s. The friction assembly consists of a housing, shaft to be tested (10 mm in diameter and 40 mm long), two working bushings attached to self-adjusting holders and a center bushing connected to the loading system and normal load measuring system.

The tests were conducted in two steps. The efficiency of M-801, M-802-1, M-802-2 coatings and M-810 and M-812 KSM were compared during the first step in different combinations at sliding rate of 0.5 m/s and at specific load of 016 MPa. The best pairs were tested at sliding rates of 0.05, 0.5 and 0.7 m/s and at specific load of 0.3 MPa during the second step to evaluate the effect of sliding rate and load on friction characteristics. Kh18N10T steel and S-820 pseudoalloy consisting a porous N18K9M3T body of martensite-aging class impregnated with molten lead containing 25 percent copper by volume, were used along with the enumerated TSP and KSM as the counterbody materials. Thus, the suggestion of the possible lubrication of the surfaces by the molten lead was checked (test time was 100 hours or until seizing occurred.

Highly pure helium (TU 51-689-75) was used in the investigations. The evaluation criteria were the friction coefficient and wear intensity. The test results for the first step are presented in Table 2 and those for the second step are presented in Table 3.

Table 2

(1) Сочетание материалов		(2) Фрикционные характеристики			
(3) Вал	(4) Втулка	(5) Коэффициент трения	(6) Износ, мкм/км (3) Вал	(4) Втулка	(7) Время работы, ч
(8) С-820	М-810	0,18—0,57	—	—	2,0 (схватывание) (9)
	М-812	0,18—0,57	—	—	1,5
	М-802-1	0,2—0,6	—	—	1,5
	М-802-2	0,26—0,78	—	—	1,5
	М-801	0,2—0,4	—	—	4,0
(10) Х18Н10Т	М-801	0,06—0,07	—	0,03	80 (схватывание)
	М-810	0,05—0,06	—	0,43	100
М-801	М-810	0,12—0,18	0,65	0,72	100
	М-812	0,1—0,15	0,51	0,68	100
	М-802-1	0,09—0,11	0,13	0,02	100
	М-802-2	0,1—0,12	0,12	0,015	100

[Key on following page]

Key [Continued from preceding page]:

1. Combination of materials
2. Friction characteristics
3. Shaft
4. Bushing
5. Friction coefficient
6. Wear, microns/km
7. Operating time, hr
8. S-820
9. Seizure
10. Kh18N10T

The investigations conducted during the first step made it possible to establish that the presence of lead in the composition of the contact materials is a sharply negative factor that significantly limits durability. The combination of solid lubricant coatings with different properties on a molybdenum base showed the most encouraging results.

A combination of M-802-1 and M-801 coatings was selected on the basis of the given laboratory investigations for bench tests under conditions that approximate full-scale conditions to the maximum.

Service life bench tests of sliding bearings were conducted at VNIIAM [expansion unknown] on a specially developed UTG-1 installation. The installation consists of a working section, loading devices, electric drive, electromagnetic coupling, electric heater, cooled chambers, evacuation and filling system and monitoring and measuring instruments.

Table 3

(1) Сочетание материалов		(2) Условия трения		(3) Фрикционные характеристики		
(4) Вал	(5) Втулка	(6) Нагрузка, МПа	(7) Скорость, м/с	(8) Коэффициент трения	(9) Износ, мкм/км (4) Вал	(9) Износ, мкм/км (5) Втулка
M-801	M-802-1	0,3	0,05	0,11—0,13	0,07	0,012
		0,3	0,5	0,1—0,13	0,065	0,02
		0,3	0,7	0,1—0,13	0,072	0,018
		0,6	0,5	0,09—0,11	0,13	0,02
	M-802-2	0,3	0,05	0,12—0,16	0,065	0,012
		0,3	0,5	0,12—0,15	0,08	0,017
		0,3	0,7	0,11—0,13	0,076	0,011
		0,6	0,5	0,1—0,12	0,12	0,015

Key:

1. Combination of materials
2. Friction conditions
3. Friction characteristics
4. Shaft
5. Bushing
6. Load, MPa
7. Rate, m/s
8. Friction coefficient
9. Wear, microns/km

The working section (Figure 1) is a sealed cylindrical chamber 1 manufactured of stainless steel, inside which a shaft 2 is installed on rocker bearings 3 and the assembly of the bearing to be tested 4 and 5. The arrangement of the shaft is vertical. The chamber is enclosed by two flanges sealed by vacuum rubber. An electromagnetic coupling 9, through which hermetic entry of rotation to the chamber is accomplished, is attached to the bottom flange. The loading devices are lever systems with set of weights. Radial loading of the bearing is accomplished through connecting pipes welded to the middle part of the chamber. The axial load is transmitted through rod 8 and ball support 7 to sleeve 6, to which the outer bushing of the bearing to be tested 5 is attached. A bellows serves as the flexible coupling element. The inner bushing of the bearing 4 is installed on shaft 2 resting on two roller bearings 3.

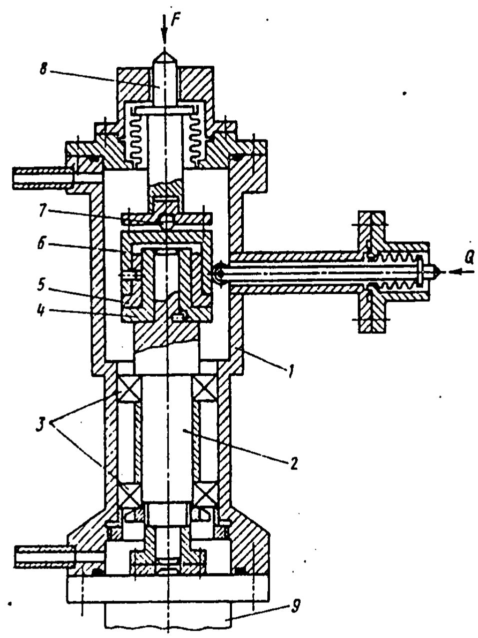

Figure 1. Working section of UTG-1 Installation

Thus, the axial and radial loads applied to the sliding bearing to be tested are transmitted through shaft 2 to roller bearings 3 and are enclosed in a housing 1.

The self-lubricating roller bearings designed by VNIPP [All-Union Scientific Research, Design and Technological Institute of the Bearing Industry] with bronze separator in which a composite consisting of 50 percent fluoroplastic and 50 percent molybdenum disulfide is pressed [10], are used in the UTG-1 installation. Two water chambers are available to cool the flanges of the working section and also the roller bearings. The working part of the section (the zone where the bearing assembly to be tested is located) is heated by a three-sectional electric heater. The temperature of the housing of the working section is measured in three cross sections by Chromel-Kopel thermocouples which are caulked in the wall of the housing. A potentiometer of type KSP is used as the recording instrument. The chamber is filled with helium from a tank through a gas reducer and cut-off valve.

The order of conducting the tests on the UTG-1 installation was as follows: the bushings of the bearing to be tested, after being wiped with acetone and alcohol, were installed in the working section, the chamber was sealed and evacuated to residual pressure of 10^{-1} to 10^{-2} Pa and was filled with helium to a pressure of 0.1 MPa. The chamber was again evacuated but upon reaching a pressure of 10^{-2} Pa the chamber housing was heated to a temperature of 150-200°C, it was held at this temperature for 1.5-2 hours, the chamber was filled with helium to a pressure of 0.1-0.2 MPa and was purged with helium three times.

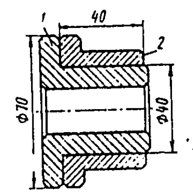

Figure 2. Sliding Bearing.

This operation was repeated no fewer than two times. The required helium pressure and temperature in the chamber were established after the last filling and the installation was ready to conduct the tests. The temperature and pressure in the chamber and the operating time of the bearing were monitored during the testing. The temperature was regulated and was held at a given temperature by voltage transformers and millivoltmeter. The helium pressure in the chamber was monitored by a pressure gauge and was kept constant by means of a gas reducer connected to the helium tank. The operating time of the bearing was recorded by an hour-meter, connected to the electric drive circuit of the installation.

Experimental radial-thrust sliding bearings consisting of two bushings (Figure 2) were manufactured to conduct the service life tests. The inner bushing 1 was made of molybdenum with M-801-1 coating and the outer bushing 2 was made of molybdenum with M-801 coating. This design of the friction assembly ensures high stability of the radial clearance of the bearing elements over a wide temperature range. The service life tests provided evaluation of the effect of the thickness of solid lubricant layers on the durability of the bearing. A bearing with thickness of coatings approximately 30 microns (M-802-1) and approximately 20 microns (M-801) were initially tested.

The efficiency criterion of the bearing was the time until total wear of the coatings. After 5,000 hours of operation (which corresponds to a friction path of 7,500 km), the thickness of the sulfide layers of the end surfaces of the bearing bushings was measured on a profilograph-profilometer, for which scratches were made with a needle to the entire depth of the coating (Table 4). It turned out that the thickness of the remaining layer of the coating of the

Test Conditions

Surrounding medium	Highly pure helium (TU-51-689-75)
Medium:	
pressure, MPa	0.14
temperature, °C	300
Load on bearing, N:	
axial F	400
radial Q	100
Rotational speed, rpm	200

Table 4

Combination of Materials	Number of Bearing	Thickness of Coating, microns — Friction path, km: 0	7,500	10,500
Outer bushing M-801	1	20	10	0
Inner bushing M-802-1	1	30	5	0
Outer bushing M-801	2	65	40	25
Inner bushing M-802-1	2	40	35	25

outer bushing comprises approximately 10 microns (with initial thickness of approximately 20 microns), while that of the inner bushing comprises approximately 5 microns (with initial thickness of approximately 30 microns). The tests were then continued and the total time to complete wear of the coatings comprised 7,080 hours (10,620 km). Since the life of the tested bearing was less than the required life (12,000 hours--18,000 km), the thickness of solid lubricant layers was increased to 65 microns (M-801) and 40 microns (M-802-1). Running-in over a period of 1,000 hours was carried out under the previously adopted conditions and the rotational speed was then increased to 540 rpm. The data of measuring the thickness of solid lubricant layers after 2,500 hours (7,500 km) and 3,250 hours (10,500 km) of operation of the bearing are presented in Table 4.

Based on the results, one can suggest that a bearing with total thickness of sulfide layers (M-801 and M-802-1) greater than 100 microns provides the required service life.

Thus, the investigations indicate the prospects of MoS_2 coatings of the diffusion type for the bearings of mechanisms of nuclear power plants with helium coolant.

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ADAPTIVE MEASURING INSTRUMENTS

Kiev SAMONASTRAIVAYUSHCHIYESYA IZMERITEL'NYYE PRIBORY in Russian 1981 (signed to press 8 May 81) pp 2, 202-203

[Annotation and table of contents from book "Adaptive Measuring Instruments", by Sergey Glebovich Taranov, Institute of Electrodynamics, UkSSR Academy of Sciences, Izdatel'stvo "Naukova dumka", 2350 copies, 204 pages]

[Text] The book presents the theory and principles of constructing wide-band and selective adaptive instrument amplifiers, and also stabilized sources of alternating current. A classification is given for adaptive measuring instruments. A method is given for analyzing the stability of adaptive measuring instruments that are nonlinear, nonautonomous, unsteady automatic control systems described by nonlinear differential equations with variable coefficients. A method of time quantization is proposed for analyzing transient processes. The informational characteristics of instruments are defined.

For scientists, engineers and technicians specializing in information-measurement technology, as well as specialists in development of electronic measuring devices. Figures 40, references 195.

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