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LOAD CAPACITY AND GEOMETRICAL PARAMETERS OF GEARING FOR SPEED REDUCERS ON SHIPS

Leningrad SUDOSTROYENIYE in Russian No 7, Jul 82 pp 20-24

NACHINKIN, V. P. and KRUGLOV, A. V.

[Abstract] A comparative study is made of 30 different main turbodrive speed reducers installed and operating in various tankers ("Sofia", "Krym", "Tokio Maru", "Idemitsu Maru", "Iova Maru", "Druzhba", "Mir"), with low-pressure turbine and with high-pressure turbine. Three arrangements are considered for kinematic interaction of gear and pinions in the first stage and in the second stage (gearing on tanker "Krym" has a third stage with idler), the first stage either with single meshing or with bifurcation of power flow. A statistical analysis of design and performance has yielded a correlation between available nominal load capacity, on the basis of stress concentration factors, and ratio of tooth width to pitch circle diameter, also simple formulas for the required pitch circle diameter as a function of the torque to be transmitted. The results indicate that the kinematic arrangement with axial correction of pinion teeth is best for heavy duty application. Figures 5, tables 2, references 4 Russian. [296-2415]

CEMA SYMPOSIUM ON NUCLEAR FUEL PROCESSING AND WASTE DECONTAMINATION

Prague ISSLEDOVANIYA V OBLASTI PERERABOTKI OBLUCHENNOGO TOPLIVA I OBEZVREZHIVANIYA RADIOAKTIVNYKH OTKHODOV in Russian Vol 1, 1981 pp 4-7, 11-12, 13

[Welcoming address by CEMA Secretariat representative L. Molnar, councillor of the Department for Peaceful Use of Atomic Energy, opening speech by I. L. Rybalchenko, representative of the IAEA (International Atomic Energy Agency) and table of contents from collection "Research in the Area of Reprocessing Irradiated Fuel and Decontaminating Radioactive Waste, Vol. 1. Materials of the Fifth CEMA Symposium, Marianske Lazne, Czechoslovakia, 7-10 April 1981", Czechoslovak Atomic Energy Commission at the Nuclear Information Center, 400 copies, 341 pages]

[Text] Welcoming Address by CEMA Secretariat Representative L. Molnar, Councillor of the Department for Peaceful Use of Atomic Energy

Esteemed Comrade Representative, esteemed comrades, honored guests.

Speaking on behalf of the Secretariat of the Council of Economic Mutual Assistance, let me welcome the participants to the Fifth Symposium of CEMA Member Nations on Research in the Area of Reprocessing Irradiated Fuel and Decontaminating Radioactive Wastes, thank the Czechoslovak delegation in the Commission and the Organizational Committee for convening this symposium in Marinaske Lazne, and wish its participants success in their work.

Also at this symposium of CEMA member nations I would like to welcome the delegation of specialists from the Socialist Federated Republic of Yugoslavia who, in accordance with a resolution of the CEMA Executive Committee, have extended their participation to the activity of the CEMA Subcommittee on Cooperation in the Area of Peaceful Use of Atomic Energy. Let me also welcome Doctor Rybalchenko, representative of the International Atomic Energy Agency.

Participants in the symposium will be discussing the results of research done in CEMA member nations on various aspects of the fuel cycle of the nuclear power industry over the last five years. The results of this discussion will enable specialists to map out ways for future resolution of problems involved in making reliable equipment, for developing ways and means of monitoring and controlling technological processes, and for working out normative-technical documentation for all issues that are discussed.

Esteemed comrades, in accordance with the Long-Range Target Program of Cooperation in the Area of Energy, Fuel and Raw Materials approved at the Twenty-Third CEMA Session (June 1978), a great contribution to resolution of the energy problem in European CEMA member nations and the Rebpulic of Cuba will be made by the construction of nuclear electric facilities on their territory with total power of about 37,000 MW before 1990 with the technical assistance of the Soviet Union. The total power of nuclear electric facilities in CEMA member nations including the Soviet Union will reach 110,000-130,000 MW, which corresponds to a savings of 240 million metric tons of ideal fuel per year. Nuclear power in CEMA member nations will become one of the most important sources for increased production of electric energy.

The strategy of CEMA member nations is to develop nuclear power with the use of thermal reactors with subsequent gradual displacement of some of them by fast reactors. This will appreciably expand the fuel base of nuclear power.

However, realization of the advantages of this strategy hinges on successful resolution of problems of the external fuel cycle, including questions of transporting spent nuclear fuel, its radiochemical reprocessing and decontamination of radioactive wastes, which is the topic of discussion at this symposium.

Esteemed comrades, the urgent necessity of accelerated development of scientific and technical progress raises the problem of expeditious acquisition of results of the present stage of the scientific and technical revolution in a number of very important areas of socialist economic integration. In the declaration of the Twenty-Third Session of CEMA on the Thirtieth Anniversary of the Council of Economic Mutual Assistance it is persuasively stressed that in the forthcoming decade the member nations of CEMA deem it necessary in a number of main areas to direct their cooperation toward all possible means of accelerating scientific and technical progress. The communiqués of the Twenty-Fourth CEMA Session additionally call attention to the need for emphasizing cooperation on specialization and integration as well, on intensifying comprehensive coverage by coordinating research, technical advances, development and production of equipment.

In recent years the character of steps being taken by CEMA member nations in the area of scientific-technical and production cooperation has been distinguished by an increase in the time scope of their effects, greater stability, increasing significance of the principle of the comprehensive approach to planning and organization. This has been fostered to a decisive extent by the increasing role of agreements and contracts both in the area of international and scientific-technical cooperation, and in the area of international specialization and integration of production.

The much-increased volume of cooperation means in essence a transition to a qualitatively new level, which naturally raises some problems. Among these is the problem of ensuring coordination and mutual conditionality of scientific-technical and industrial cooperation.

We must direct our efforts toward solution of a problem that is common to all CEMA member nations: attainment of unity in the cycle "science - engineering - production - exchange - utilization."

Esteemed comrades, in conclusion, let me once again wish you success in your work. It is my hope that this symposium will make a further contribution to resolution of the complex scientific and technical problems of nuclear power, and will further reinforce cooperation among CEMA member nations and between CEMA and the IAEA.

Thank you for your attention.

Opening Speech by I. L. Rybalchenko, Representative of the IAEA (International Atomic Energy Agency)

The International Atomic Energy Agency welcomes the participants to the Fifth Symposium of CEMA Member Nations on Research in the Area of Reprocessing Irradiated Fuel and Decontaminating Radioactive Waste, and wishes successful work for all those taking part.

Since 1975, the IAEA has had an agreement with CEMA on mutual cooperation, and it is the mission of representatives of these organizations to strengthen and expand this cooperation.

Worldwide development of nuclear power faces some difficulties conditioned by technical as well as organizational and political aspects.

The main difficulties to development of nuclear power on an international scale are due to the following:

--enriched uranium must be had for fabricating fuel for water-cooled water-moderated and water-cooled graphite-moderated reactors;

--a closed fuel cycle (regeneration) leads to formation and accumulation of plutonium, as well as formation of high-level wastes;

--uranium enrichment and plutonium generation are so-called "sensitive" points of the fuel cycle, bringing about conditions for possible uncontrolled proliferation of technology and of the very materials that can be used for making nuclear weapons.

These points have been considered in working out some international programs for development of nuclear power and its fuel cycle that have been carried out recently with the participation of the IAEA. Most significant have been the Program for Elaborating the Concept of Regional Centers of the Nuclear Fuel Cycle (RCNFC, 1976-77), the program of International Evaluation of the Nuclear Fuel Cycle (IENFC, 1977-80), the program of International Storage of Spent Fuel (ISST, begun in 1979) and the program of International Storage of Plutonium (ISP, started in 1979).

The IAEA is directing its efforts toward solving existing technical and organizational problems to promote successful development of nuclear power for peaceful purposes with strict observance of international obligations associated with non-proliferation of nuclear weapons in the world.

Treatment of spent fuel and decontamination of radioactive waste is a problem whose solution in large measure will determine the successful development of nuclear power in the world.

This symposium should contribute to solution of these problems.

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PROBLEMS OF REPROCESSING IRRADIATED FUEL FROM THERMAL REACTORS

Prague ISSLEDOVANIYA V OBLASTI PERERABOTKI OBLUCHENNOGO TOPLIVA I OBEZVREZHIVANIYA RADIOAKTIVNYKH OTKHODOV in Russian Vol 1, 1981 pp 16-25

[Article by V. Ya. Galkin, V. M. Dubrovskiy, V. I. Zemlyanukhin, L. N. Lazarev, R. I. Lyubtsev, V. N. Romanovskiy and N. S. Tikhonov]

[Text] This paper gives a survey of research and development done in CEMA member nations in 1976-1979 on problems having to do with reprocessing irradiated fuel elements from nuclear electric plants with thermal reactors. It is pointed out that the main topics of research in this area have been opening and preparation of irradiated fuel elements for processing, improvement of technology and development of equipment for regenerating spent VVER fuel elements, development of methods and means of monitoring and controlling technological processes of regeneration. The paper gives the results of work on developing technology and making equipment for removing fuel cladding, continuous dissolving of fuel elements, clarifying initial process solutions. An examination is made of research on redox reactions of actinides, kinetics of phase transitions in extracting actinide elements and fission products, chemical and radiation properties of new extractants and diluents, extraction mixture of TBP diluent, theory of extraction processes, development of methods and technology for isolating transuranium elements and various fission fragments, development of individual forms of equipment, methods and instruments for laboratory and remote monitoring of the content of valuable elements in the solution, remote monitoring and control of technological processes of recycling irradiated fuel elements. Research that has been done is evaluated and directions for future development are outlined.

One of the most important scientific and engineering problems that to a great extent determines the development of nuclear power in the USSR and nations building socialism is development of optimum technology for regenerating spent nuclear fuel of electric power plants and atomic heat supply stations. The Permanent Committee of CEMA on Peaceful Use of Atomic Energy feels it necessary

place this problem before the research organizations of these nations for the following reasons:

--some nations have specialized engineering facilities equipped for doing research with highly radioactive materials, as well as highly skilled specialists;

--nuclear power is developing at a steady pace in the USSR and nations building socialism;

--we have accepted the concept of developing nuclear power with a closed fuel cycle, providing for regeneration of spent nuclear fuel with reuse to cut down requirements for natural uranium.

Successful resolution of such a complex and specific problem on a multilateral basis has required first of all a rapid and fairly complete assessment of the mutual capabilities of the nations, determination of areas of research that are of the greatest interest for the most nations, selection of the specific research jobs that should be done on a cooperative basis. In this respect, the period between the fourth and fifth symposiums of CEMA member nations takes a prominent place in development and perfection of multilateral cooperation on the problem of regenerating spent fuel. It was during this period that the Coordination Council passed a resolution providing for inclusion of specific developments in the working plans of projects only after the council had examined the corresponding technical assignments, noted some jobs that were recommended for specialists of nations to do on the basis of cooperation, and defined the major problems to which specialists of the nations should direct their efforts in 1977-1980.

Just as in the preceding half-decade, research done by CEMA member nations in accordance with working plans on topics of problem 1-6 "Research on Reprocessing Fuel Elements of Nuclear Electric Plants" for 1976-1980 has covered practically all aspects of technology of regenerating spent fuel elements, beginning with opening and preparation of the fuel and ending with extractive processing. The survey of methods of opening spent fuel elements given at the fourth symposium included:

--the mechanical method;
--chemical opening;
--electrochemical dissolving;
--thermal removal of cladding;
--use of lasers;
--use of ultrasound.

The intervening years have not altered the predominance of the mechanical method, which is used for opening fuel elements on practically all existing and planned radiochemical plants and facilities for processing irradiated fuel elements. However, the disadvantages of the mechanical method--complexity of equipment, requirements for reliability of their working components and for methods of remote servicing and repair--have made it necessary to look for new methods of opening fuel elements. New methods of getting at irradiated

fuel are also attractive from the standpoint of their use in developing technology for regenerating fast reactor fuel. One of these methods is thermal opening, and ways to improve this technique are being studied in the Soviet Union. In the course of this research, stand tests have been completed on high-temperature radiators for induction melting devices on equipment for thermal opening of fast reactor fuel assemblies. The capacity for work and causes of wear of graphite and molybdenum radiators with silicide coating have been determined in different gases. Experiments have been done on thermal opening of irradiated RB-5 and BOR-60 fuel elements. The direct yield of fuel upon opening was 99.9%, steel content in the fuel was 1-5%, total losses of fuel with waste were less than 0.1%.

An analysis has been made of the process of separating molten steel and oxide fuel. The conditions of removing molten steel from fuel have been studied. The patterns that have been revealed have enabled us to select optimum conditions for thermal opening of fuel elements, and to foresee the equipment for the process.

At present, both periodic and continuous processes of dissolving fuel are being considered for use in equipment for reprocessing irradiated power reactor fuel elements.

The disadvantages of periodic-action dissolvers are large overall dimensions, difficulties of ensuring nuclear safety, the necessity of using high-capacity gas cleaning systems due to the nonuniformity of liberation of gases and vapors in periodic dissolution. These disadvantages are eliminated in continuous-action dissolvers. However, most models of continuous-action dissolvers still need additional work and improvement from the standpoint of remote control, automation and simplification of design. It is in this direction that Polish specialists have concentrated their attention in efforts to perfect their drum continuous-action dissolver. Based on work done previously, an experimental stainless steel model of this drum dissolver has been made at the Institute of Nuclear Research of Poland. An electronic system has been developed for controlling the motor and fuel batcher. The facility has been successfully tested on fuel simulators. Apparently this work should be continued in the same vein in future.

An important question in preparing solutions for extractive reprocessing is absence of macrosuspensions and colloidal particles, which are strongly detrimental to the indices of extractive processes. A report at the fourth symposium mentioned an experiment on using cartridge filters with cermet baffles. In the intervening period, this work has been continued in Poland, where an experimental model of a filtration unit has been made with a filtering diaphragm of acid-resistant steel powder alloy 3 mm thick with pore size from 5 to 20 μm. This equipment has been successfully tested on model sediments with granulometric composition from 1 to 30 μm. These tests showed that vibration and back pressure increase the working time of the filtration baffle by a factor of nearly 50. Analysis of the research shows that just as before it is urgent to study extension of the service life of filters, to perfect methods of regeneration, and to develop methods of utilizing the sediments obtained in the filtration process.

One of the ways to improve filtration efficiency and at the same time remove certain impurities and fission fragments (carbon, zirconium, molybdenum, silicic acid) that complicate the extractive process is flocculation. In 1977-1978 the materials tested as flocculants in the USSR were phosphoderivatives of polyvinyl alcohol, diacetyl, crotonic and cinnamic aldehydes and some other compounds. Polyvinyl alcohol derivatives that showed the greatest flocculating action with respect to zirconium were used in experimental reprocessing of VVER fuel elements, resulting in conversion of 70-80% of the zirconium to filtered residue.

Currently the extractive method is considered the most completely developed of all techniques at the disposal of applied radiochemistry. Practically all existing and planned facilities (plants) for reprocessing nuclear fuel of electric power stations use technological procedures based on extractive processes. At the same time, it would be untrue to say that this method has reached its pinnacle and that its capabilities have been completely exhausted --extractive technology continues to be developed in two main areas. The first involves improvement of the process of isolating uranium, plutonium and neptunium, and has the aim of reducing the number of extractive operations while simultaneously improving the major indices: degree of extraction and purification.

The second area involves isolation of all isotopes of practical value, including transplutonium elements, rare earth elements, strontium, cesium and so forth, leading to expansion of technological facilities.

If the first area is to be further developed the principal topics of research and development will be:

--on creating high-efficiency extractive equipment;
--on working out so-called salt-free methods and processes as applied to extraction;
--on selecting new dissolvers that are more resistant to radiation;
--fundamental research involving in-depth investigation of new extractive systems, problems of kinetics and mass transfer;
--development of systems for monitoring and controlling technological processes and creating improved devices for these purposes.

These trends in development and perfection of aqueous technology for regenerating fuel have been reflected in research done in CEMA member nations (both on the basis of multilateral and bilateral cooperation) in the period of 1977-1980.

For example, work on developing and testing centrifugal extractors, among the most promising forms of equipment, has been done in the Soviet Union, East Germany and Poland. Experimental models have been developed that cover a range of productivity from a few liters to 5 m^3/hr. It is important to note that a prominent place in the research work goes to development of facilities for remote servicing and replacement, as well as to auxiliary equipment.

Individual models of such extractors are being studied for kinetics and mass transfer in two-phase systems, and the most important hydrodynamic characteristics are being investigated.

As before, specialists have their eye on another universally acknowledged type of extraction equipment--columns. Pulsation columns and columns with vibrational intermixing of phases designed for use in technology of regenerating fuel of nuclear electric plants have been described in papers of Soviet and Czechoslovak specialists at the fourth symposium. The combination of high productivity of equipment of column type with simplicity of design and reliability puts these devices into the category of the most promising equipment for reprocessing fast reactor fuel as well.

At the same time, it should be noted that still lacking for either type of equipment--centrifugal extractors and columns--is development of monitoring and measurement instrumentation, control facilities and other auxiliary equipment with adequate consideration of the specific conditions of extractive technology (flowmeters, batcher pumps, instruments for determining phase retention time and so forth).

As to the fundamental theoretical and applied work in the field of extractive technology, it can be stated that this has become traditional for nearly all CEMA member nations.

During the period between the fourth and fifth symposiums, this research has developed in the following areas:

--in-depth detailed investigation of physicochemical properties of traditional extractive systems based on TBP;

--search for new extractants, diluents, sorbents and investigation of their properties;

--investigation of redox reactions of actinides for use in separative equipment;

--development of new methods and technological arrangements for isolating transplutonium elements and individual elements from fission products.

To illustrate this area of research we might mention investigations done in Poland to study the radiation stability of the TBP-diluent system, where n-dodecane, n-paraffin (C_{10}-C_{14}), aromatic diluents, carbon tetrachloride and silanes were used as the diluent. The researchers determined the influence of nitric acid on the radiation resistance of such systems, as well as the influence of zirconium and palladium on retention of uranium and plutonium in the extraction mixture. Recommendations have been made on extractant regeneration.

In the USSR, methods have been developed for synthesizing polyhalide-substituted organophosphorus compounds that are resistant to the action of radiation and can be used as TBP diluents, e. g. chlorine derivatives of trifluoromethylbenzene.

An investigation has been made of the extractive capacity of sulfur-containing phosphates of the type $R,S(CH_2)_n(PO)_2PO$, where $n = 0, 1, 2$. The role of the substituent R,S has been demonstrated, as well as its influence on the properties of the extractant.

Research has continued on other extractants--alkyl dioxides that contain PO_4-SO groups, diphosphine dioxides of various structure and so on.

Systematic studies of different extractants are conducive to the development of methods of isolating and separating elements from fission products, including rare earth and transuranium elements.

Experimental work has been supplemented by theoretical research and calculations aimed at elucidating the causes for anomalous aryl fixing of complexes of transplutonium and rare earth elements with extractants of this class (aryl diphosphine dioxides). Specific recommendations have been prepared on directional modification of the structure and composition of extractants.

Among other fundamental studies, we should single out investigations of the kinetics of phase transitions in extraction of actinide elements and fission products done in Poland and the USSR, taking note both of their high efficacy and procedural achievements (in particular the use of a photographic method for determining the interphase surface).

Further improvement of extractive technology is rightfully associated with development of methods and means of stabilizing valence forms of individual elements, and in particular plutnoium and neptunium. In the period between the last two symposiums, serious attention has been given to investigation of redox reactions of actinides with the aim of developing effective means of separating these elements. Comprehensive studies have been done on electrochemical reduction of plutonium both in electrolytic cells and in extractors of the mixer-settler type. An investigation has been made of the effectiveness of reducing Pu(IV) at constant potential on the working electrode, constant voltage across electrodes, and constant current density. The method with constant current density has been found to be most convenient.

Traditional research has also continued on investigation of the equilibrium and kinetics of the direct and inverse reaction of Np(V) and Pu(III) with V(V), and also interaction in systems Pu(VI)-NH_2OH, Pu(IV)-Pu(V)-HNO_2-HNO_3, Np(IV)-H_2O_2 and so on. In all cases the activation parameters have been determined, the mechanism of interaction has been examined, and practical recommendations have been made on carrying out processes. The importance and practical significance of research in this area give a sufficient reason for including such studies in the plan of scientific research work for 1981-1985.

It is impossible to imagine fuel regeneration technology without further improvement of methods and means of monitoring technological processes. It is gratifying to note that this area has been developing successfully within the scope of the problem, and is broadly represented on the agenda for 1981-1985. And although a special report is to be given at the symposium on the state of the art and ways to improve monitoring in the technology of fuel regeneration at nuclear electric plants, we would like to take up some general questions right now.

The intensive development of instrumental analytical methods, advances in the field of electronics, automation facilities and equipment, mathematics

and programming have given us a new approach to development of technological processing monitoring: automated systems monitoring based on a combination of remote "on-line" monitoring and automated express lab analysis. Such an automated monitoring system is based on extensive use of microprocessors with various levels of hierarchy, and when appropriate algorithms are incorporated it should be possible to automate control of the technological process.

It is quite obvious that development of optimum monitoring systems and the corresponding methods and means requires close cooperation between experts of various nations specializing in a number of fields.

It should be pointed out in this connection that there is quite a representative base for developing such cooperation and testing newly proposed methods and means on actual highly radioactive technological products.

Here we have reference to the radiochemical department of the Radium Institute imeni V. G. Khlopin (USSR) where rather broad capabilities have been made available for research in the area of fuel regeneration technology and facilities for monitoring and controlling processes. The department building provides facilities for delivery of depleted fuel specimens from nuclear electric plants, a process equipment complex including an automated extraction facility, computing center and experimental facilities for waste treatment, including by vitrification. Prolonged experience in using this research complex has shown that the existing experimental facilities and stands can be sucessfully used for research on a multilateral and bilateral basis involving development of monitoring systems and equipment, development and testing of individual devices and complex systems for control, and also investigation of technological operations and processes.

As was pointed out above, one of the ways to improve the fuel cycle and increase its economic effectiveness is a comprehensive approach to reprocessing irradiated nuclear fuel. Such an approach involves the development of technological arrangements that not only provide for isolation of the principal target components of uranium, plutonium and neptunium, but also ensure extraction of transplutonium elements and a host of fission products: strontium, cesium, palladium, technetium, certain lanthanides, etc.

These components to a greater or lesser degree have found or are beginning to find application in different areas of the national economy. Since their main source of supply is high-level liquid wastes--refinates after extractive treatment of irradiated fuel solutions--the methods of isolating individual elements inevitably become interrelated with the general problem of handling wastes. In particular, the extraction of actinides from highly radioactive refinates is usually considered from the standpoint of complete isolation of uranium, plutonium and neptunium, as well as from the standpoint of localizing transplutonium elements as the most long-lived radionuclides when preparing wastes for prolonged safe storage.

This aspect complicates still further the already complex problem of developing a comprehensive technological arrangement for regenerating spent fuel, which makes a number of demands on methods and reagents that are used. Nevertheless,

during the period under analysis (which can be taken as exploratory from the standpoint of developing a comprehensive arrangement) there have been a number of promising proposals on isolating actinides and certain fission products of practical significance.

Among these, we should take note of systematic studies done in the USSR on the distribution of transplutonium and rare earth elements in a system comprising nitric acid, an agent for salting out, a complexon and TBP over a wide range of concentrations of system components: nitric acid from 0.1 to 0.5 mole/dm^3, salting-out agent (sodium nitrate) from 3 to 6 moles/dm^3, TBP from 20 to 80%. Based on the resultant data, empirical relations are derived that give the coefficients of distribution of americium and europium in terms of the initial concentrations of acid, extractant and salting-out agent. The results of these studies have been taken as a basis for developing a technological arrangement for additional extraction of uranium and plutonium, and isolating fractions of transplutonium and rare earth elements from highly radioactive refinates after processing fuel from nuclear electric plants. This process includes the following basic stages:

--extracting uranium and plutonium with 40% TBP from the refinate followed by re-extraction and return to the head of the recycling process;

--combined isolation of transplutonium and rare earth elements from the solution that has been corrected for concentrations of nitric acid and salting-out agent, and rinsing of the extractant;

--re-extraction of transplutonium elements with a complexon solution (DTPA) that contains an agent for salting out;

--re-extraction of rare earth elements with a nitric acid solution.

This process was checked out on an experimental facility in hot cells of the radiochemical department at the Radium Institute imeni V. G. Khlopin, using irradiated VVER fuel as the initial material for the solution. In verification of the process it was found that americium extraction reaches 99%, the coefficients of separation of the lanthanide(III)/americium(III) pair being: 77 for cerium, 18 for neodymium, 12.5 for promethium, 4.2 for europium. The coefficient of removal of zirconium and ruthenium from americium was 200 and 50 respectively. The test results confirmed the previously obtained data. The given arrangement has a number of advantages over conventional methods of reprocessing refinates, among which we should mention the use of an extractant that is common with that of the main process, and fairly high degrees of extraction and separation of transplutonium and rare earth elements.

In recent years at the Institute of Nuclear Research (Czechoslovak SSR) and at the Radium Institute (USSR) research has been continuing on the possibilities of using polyhedral carboranes for extractive isolation of cesium and strontium from high-level liquid wastes. Equilibrium data have been determined for operations of extraction, re-extraction and regeneration of the extractant in the process of isolating cesium and strontium with chlorinated carborane solutions. The theoretical flows of extractant, rinsing and re-extraction

solutions have been calculated on the basis of the equilibrium data. Tests have been done and an estimate has been made of the explosive and fire hazard of organic systems that contain a chlorinated carborane, polyethylene glycol, nitrobenzene and carbon tetrachloride in contact with nitric acid.

On the experimental facility in hot cells of the Radium Institute, tests were done on a technological arrangement for isolating cesium and strontium by using a polyhedral carborane with refinates after processing irradiated VVER fuel as the initial solution. In checking out the arrangement, high indices of extraction of cesium and strontium were attained (more than 99%), as well as high purification from principal concomitant radionuclides (10^3-10^4). The main advantage of the proposed method of isolating cesium and strontium, besides the data obtained on extraction and purification, is the feasibility of using highly acidic and saline initial solutions without any preparation. The test results revealed advantages of the scheme for isolating cesium and strontium from the standpoint of compatibility with the main technology of fuel reprocessing.

A method of isolating cesium from acidic highly radioactive tail-water solution by sorption on an inorganic solvent--ammonium 12-molybdophosphate--developed at the Central Institute of Isotopic and Radiation Research (East Germany) has been brought to the stage of construction of an experimental facility and testing on model solutions. The method includes the following operations:

--sorption of cesium on the ammonium 12-molybdophosphate;
--rinsing with a 0.2 M solution of nitric acid;
--dissolving the sorbent in a 0.5 M ammonia solution;
--filtering the solution thorugh a type KPS Vofatit cation exchange resin.

The proposed method is also of interest from the standpoint of compatibility with the main technology since it does not require neutralization of the initial solution, and apparently does not introduce reagents into the process that complicate fuel processing. Sorption of cesium on ammonium 12-molybdophosphate is one of the methods that have been proposed by East German specialists for setting up comprehensive technology for reprocessing highly radioactive refinates. Not all of these methods can be considered equivalent for combining with the main technology of regenerating irradiated fuel since they call for the use of a host of dissimilar operations and reagents. In particular, the use of trioctylamine with subsequent re-extraction has been suggested for isolating technetium and palladium. Processes utilizing anion exchange resins, chelate resins and metal ferrocyanides have also been studied for selective isolation of palladium. A method of liquid chromatography under pressure has been used for fine separation of a mixture of rare earth elements, americium and curium.

Quaternary ammonium bases have also been studied for separating americium and curium. Laboratory facilities that realize reflux and semi-reflux processes have been made for checking these methods. Various amines have also been studied in the form of solid extractants. A coefficient of removal of americium from curium of 2, and removal of curium from americium of up to 4.5 have been attained when a solid extractant based on a tetraalkylammonium nitrate is used with magnesium nitrate as the agent for salting out.

The action of high levels of radiation has been studied in application to the process of chromatographic separation of rare earth and transplutonium elements. It has been shown that when using KU-2 cation exchange resin and DPTA as the elutriating agent, doses of up to 10^7 Grad cause destruction of the sorbent, increased pH of the elutriating agent and other effects that must be taken into consideration in developing technology.

To evaluate the prospects of different extractants for isolating palladium from nitrate media, research has been done with the use of carboxylic acids, dioctyl phosphoric acid, TBP, higher alcohols, phosphonates, phosphine oxides and quaternary ammonium bases. The greatest attention has been given to tri-1-methylheptylphosphine oxide, for which an investigation has been made of the influence that concentration of the salting-out agent, temperature and other factors have on the coefficient of distribution.

In studying methods of isolating technetium, binuclear complexes have been obtained with metal-metal bond and degree of oxidation of 2.5 in which the cations are organic compounds. The conditions and compositions of these compounds have been determined. In studying the mechanism of thermal destruction of cluster compounds [and] temperature of the onset of processes of dissociation, it has been established that the behavior of clusters is determined by the sum of processes of hydrolysis, disproportionation and oxidation.

Summing up the cooperation of nations in the area of regenerating depleted nuclear fuel of electric power plants, we can state that up until now the extractive scheme has been selected and mainly carried out on consolidated pilot facilities for isolating uranium, plutonium and neptunium from solutions of irradiated fuel elements and deep removal of fission fragments, major types of equipment have been singled out that are promising for use in technology, ways have been pinned down for extracting transuranium and individual fission fragments from the technological process that are promising for use in the national economy, and the basic principles of the systems approach to the monitoring problem have been formulated.

In the period up to 1985, the efforts of specialists of nations should be concentrated on solving the following major problems on the basis of cooperation:

--research in the area of hydrodynamics of mass exchange and kinetics of extractive processes aimed at optimizing processes and perfection of structural components of equipment;

--finding and testing new methods of separating actinide elements that do not complicate the basic extraction technology;

--developing and testing new methods of selective isolation of individual elements from highly radioactive tail-water solutions;

--a complex of research and development aimed at coming up with a system for monitoring the process of recycling spent nuclear fuel.

Experience accumulated up to the present shows that research and development on a bilateral basis has been most effective. In this connection, one of the aims of specialists of nations represented in the Council on Reprocessing and Decontaminating Radioactive Products should be an assessment of research, nations interested in carrying it out, and presentation of the appropriate recommendations for inclusion in working plans of bilateral cooperation to the responsible organizations.

Discussion

V. Vesely: Are provisions being made for checking out the method of cesium extraction with ammonium 12-molybdophosphate developed in East Germany under realistic conditions?

V. I. Zemlyanukhin: I believe that there is a need for verifying this technique on actual solutions. There has been an agreement for such a test between the Central Institute of Isotopic and Radiation Research (East Germany) and the Radium Institute (USSR).

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CSO: 8144/1829

STATE OF RESEARCH ON REPROCESSING FUEL ELEMENTS FROM NUCLEAR ELECTRIC PLANTS WITH FAST REACTORS

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[Article by V. I. Zemlyanukhin, A. S. Nikiforov, G. P. Novoselov, V. N. Prusakov, O. V. Skiba, A. F. Tsarenko and V. S. Shmidt]

[Text] This paper gives a survey of new data obtained in the area of aqueous and non-aqueous methods of reprocessing fast reactor fuel elements. With respect to aqueous methods, data are given on the following questions:

--methods of calculating the distribution of valuable components in reflux extraction (with consideration of new data on hexavalent element distribution);
--the behavior of individual elements--fission products--during extraction, including new data on extraction of lead;
--the behavior of fluorine ions during extraction under conditions of reprocessing fast reactor fuel elements;
--conditions of isolating a third phase in extraction;
--chemistry of processes of dissolving fast reactor fuel elements;
--problems of radiation stability of extractants and radiative transformations of the aqueous phase under processing conditions.

With respect to non-aqueous methods, the following questions are considered:

--distribution of fission products in fuel elements;
--new data on thermal opening of fuel elements with mixed fuel.

At the conclusion of the paper an examination is made of the principal areas of future research on technology of reprocessing fast reactor fuel elements.

1. Introduction

An extensive program for building nuclear electric plants with thermal reactors has been developed in the USSR and CEMA member nations. The volume of work on nuclear electric plants with fast reactors is also expanding. In 1980 the Soviet Union started operation of the world's largest reactor of this type, the BN-600, at Beloyarsk Nuclear Electric Plant imeni I. V. Kurchatov. In the offing is further construction of nuclear electric plants with fast reactors [Ref. 1].

In this connection there has been an increase in urgency of research on regeneration of fast reactor fuel. As before, work is being done in two areas: on development of aqueous (extractive) and non-aqueous (mainly gas-fluoride) methods of reprocessing, which will be examined in sequence below. These two areas are not isolated, since aqueous methods inevitably use non-aqueous facilities and operations such as removal of the cladding, oxidation of the fuel and so on.

2. Aqueous Methods of Reprocessing

Aqueous technology for reprocessing fast reactor fuel elements coincides in large measure with the technology of reprocessing thermal reactor fuel; therefore the use of aqueous methods is facilitated by the availability of considerable experience on industrial reprocessing of thermal reactor fuel elements. Aqueous methods are considered in the USSR as the most ready for industrial introduction at the present time [Ref. 2-5].

The basic arrangement for reprocessing fast reactor fuel elements that has been developed in the USSR includes the following operations. The fuel assemblies are treated with steam or carbon dioxide to remove traces of coolant (sodium), and are sent to mechanical chopping when reprocessing fuel elements that have been held for a year or more, and to melting of the cladding for elements that have been held for a shorter time. The opened fuel is dissolved in nitric acid, the solution goes to clarification, and the undissolved residue goes to control dissolution in nitric acid with the addition of hydrofluoric acid, the cladding hulls are removed from the solution and sent to burial. Before burial, these hulls may be converted to compact ingots by an additional melting operation, which considerably reduces the volume of required disposal depots [Ref. 6]. The clarified solution goes to the operation of iodine elimination, and in case of necessity, to stabilization of the valence of plutonium and neptunium and correction of acidity. The prepared solution is sent to extraction with 30% TBP in paraffin diluent.

Extraction in the uranium line includes three cycles. On the first and second cycles, plutonium is extracted along with the uranium. On the second cycle the plutonium may be completely removed from the uranium by reduction re-extraction and sent to the plutonium refinement cycle (in the version of separate output of uranium and plutonium goal products), or converted to the aqueous phase together with a considerable part of the uranium (in a ratio of U:Pu$>$ 3:1 if a mixed uranium-plutonium product is required) by a displacement re-extraction method; in this case the mixed solution is also purified on a

refinement cycle and the product goes to precipitation. The remainder of the uranium, completely free of plutonium, goes to the third uranium cycle--final purification--after which it is sent to oxide production. Nearly all impurities remain in the first-cycle refinate, which goes to evaporative concentration, and then to possible final extraction of valuable elements--fission products.

The technology has been checked out on pilot facilities, confirming the possibility of realizing the process for recycling irradiated fast reactor fuel elements.

To optimize the technology, research has been done, and further research is being done involving known peculiarities [Ref. 2-7] specifically inherent in fuel elements of this type: a higher concentration than in thermal reactor fuel elements of plutonium and some fission products (palladium, molybdenum, zirconium, iodine, technetium and others), a greater amount of insoluble residue after dissolution which must be separated during clarification of the solution, higher level of radioactivity of solutions and probable increased action of radiation on the extractant and diluent, a greater concentration of valuable components (in addition to uranium and plutonium) in the fuel, and also impurities introduced on various stages of storage, transportation and processing.

2.1. Peculiarities Associated with Elevated Plutonium Content

The basis of computational selection of optimum conditions for carrying out extraction processes at the present time is provided by mathematical models of reflux processes [Ref. 7-10] that are also a necessary component of automated systems for control of these processes. The models of reflux processes are based on mathematical description of the way that the coefficients of distribution of uranium and plutonium depend on the composition of the aqueous and organic phases [Ref. 7]. Previous research has given fairly detailed and reliable data on the distribution of plutonium at relatively low concentrations (less than 20 g/ℓ) in the presence of U(VI) and nitric acid (as applied to extractive reprocessing of thermal reactor fuel elements with the use of 30% TBP), and appropriate mathematical models of distribution have been developed [Ref. 7, 8-17]. However, there has not been enough research on the distribution of Pu(IV) at the higher concentrations that will occur when reprocessing fast reactor fuel elements (especially the concentrations at which the extractant is close to saturation in the presence of U(VI)) or on the distribution of Pu(VI) in extraction with 30% TBP; there have been no reliable experimental data, and hence no mathematical model, for equilibria. This gap has basically been closed by Ref. 4, 5, 18, 19. A variant of a mathematical model has been developed that gives a fairly accurate description of available experimental data on the reflux distribution of plutonium under the conditions of the first extraction cycle as well [Ref. 4]. An important job that still remains is development of a model of the process of reductive re-extraction of plutonium that is acceptable over a wide range of conditions.

Simultaneously with data on the distribution of plutonium at high concentrations, data have been obtained on its maximum permissible content in the

extract at which homogeneity of the organic phase is assured [Ref. 20-22]. The concentrations of Pu(VI) in first-cycle extracts under the accepted conditions are 20-25 g/ℓ; the capacity of a 30% solution of TBP in dodecane is 40-45 g of Pu per liter, which means that the process can be done without isolating the third phase. A slight increase in the temperature of the system also helps to retain homogeneity of extracts.

It is obvious that when fast reactor fuel elements are processed, in connection with the high concentration of plutonium in the solutions, plutonium reduction with saline reducing agents leads to an elevated salt content in the radioactive wastes. Therefore an important job is to develop salt-free methods of reductive re-extraction of plutonium [Ref. 23, 24]; research in this area has been intensively continuing in recent years. Considerable attention has also been devoted to analysis of the feasibility of separating plutonium and uranium without using reducing agents: by displacing Pu(IV) with an excess of U(VI) from the organic phase [Ref. 5]. It has been established that total mutual separation of these elements by such a method at a sufficiently high level of extraction in the process cannot as yet be accomplished with adequate reliability, but this technique can be used to separate the main mass of uranium, which reduces the required amount of reducing agent. This method is particularly effective in the case where it is necessary to increase output of mixed uranium-plutonium product, i. e. where it is not required to completely remove the uranium from the re-extracted plutonium, but it is only necessary to remove plutonium from the uranium remaining in the extract.

Research is continuing on the influence that alpha radiation in aqueous nitric acid solutions has on the valence state of plutonium [Ref. 25]. Results obtained so far have enabled evaluation of the need for adjusting plutonium valence on various stages of the process.

In recent years detailed investigations have been made of the influence that high plutonium concentrations have on alpha radiolysis of tributyl phosphate; the mechanism of processes that occur has been studied [Ref. 26]. It has been shown that dissociation of the extractant under the action of alpha radiolysis between cycles of soda regeneration, even for the relatively long time of contact between phases typical of extraction in mixer-settlers, is within permissible limits and does not cause any effects that are qualitatively different from those observed in reprocessing thermal reactor fuel elements. Data on the way that the degree of dissociation of TBP depends on the dose of alpha radiation and conditions may serve as a basis for future development of a mathematical model of the accumulation of DBP and MBP in the extraction cycle for conditions of reprocessing fast reactor fuel elements.

Since it may be necessary to introduce fluorine ions on the stage of opening for complete extraction of valuable components from insoluble residues, it has been of interest to study the behavior of hydrofluoric acid in the process of nitric acid extraction, and in particular its influence on plutonium extraction. It has recently been demonstrated that HF is extracted with TBP solutions in the form of the compounds HF·TBP and HF·2TBP [Ref. 27, 28]; the extraction of HF may be suppressed by adding aluminum nitrate [Ref. 29], preventing occurrence of fluorine ions in re-extractates when processing solutions that contain such ions.

2.2. Peculiarities Associated With Elevated Content of Fission Products and Some Impurities

These peculiarities of fast reactor fuel first of all make it necessary to study stability of the extractant and diluent with respect to the action of beta and gamma radiation of radionuclides--fission products--in the presence of weighable amounts of the latter. The processes that occur in the TBP-hydrocarbon diluent system under the action of these factors and their influence on the indices of extractive technology have been studied in Ref. 21, 30. Analysis of conditions of possible formation of compounds of acid products of dissociation of TBP with some fission-fragment elements was done in Ref. 31. The resultant data, like the data of alpha radiolysis, may serve in future as a basis for a mathematical model of the process of formation of acid products of dissociation of TBP, their interaction with fission products, and removal from the extractant with soda rinsing in the cycle. The radiation stability of hydrocarbon diluents of various compositions and the influence of fission products on their stability and operational characteristics are examined in Ref. 21 with consideration of theoretical concepts [Ref. 33] about the influence that the nature of diluents has on extraction equilibria. Particular emphasis in radiation-chemical reserach of recent years has been given to investigation of the influence of macroconcentrations of fission-fragment elements [Ref. 34, 35].

Research coverage shows that whereas reprocessing of thermal reactor fuel requires practically no deep regeneration of the extractant with removal from the cycle, and its efficacy is ensured by intracycle soda or alkali rinses, it is possible that deep regeneration with periodic removal of part of the extractant from the cycle will be necessary under conditions of fast reactor fuel reprocessing. It has been shown that it may be effective to use sorption [Ref. 37, 28] and electrochemical methods for deep regeneration along with the conventional techniques based on rinsing the organic phase with solutions of various reagents [Ref. 36] or distillation.

Since the service life of the extractant in the cycle with prolonged use depends on the dose of irradiation, the problem of reducing the time of contact with the highly radioactive aqueous phase on one cycle of extraction remains current. In this connection, research and development is continuing on extraction columns and centrifugal extractors, as well as in studies of extraction kinetics [Ref. 39] and re-extraction kinetics [Ref. 40] as a basis for design of such facilities. Since the macroconcentrations of fission products may have a catalytic effect on redox processes, researchers are also giving attention to this problem.

It is of interest to study the effect that some inactive impurities present in the solutions have on the indices of extraction processes. Among these are components of stainless steel that enter the solution as a consequence of partial dissolution of cladding that has been corroded at points of contact with the fuel, residues of coolant (sodium), and also in isolated cases the residues of lead used as a sealant for leaky fuel elements. It has been established that all these impurities remain practically unextracted, and also that they have no effect on indices of the process with the exception of Fe(III) which may influence redox reactions [Ref. 41].

Of considerable interest is the extraction of certain elements of value from the refinates of extractive reprocessing of fast reactor fuel elements. In addition to nuclides that are sources of radioactive emission (strontium, cesium, rare earth and transplutonium elements) for which methods of extraction from refinates have been widely described [Ref. 24, 42-47], such components include valuable elements like palladium, rhodium, technetium, which will quite probably find applications in the national economy in future. Some research done in recent years in the Soviet Union and CEMA member nations has dealt with finding methods of extracting these elements from nitric acid solutions. It has been demonstrated that it is possible to extract technetium with tributyl phosphate and trioctylamine [Ref. 48-50], palladium with solutions of dialkyl and diaryl sulfides [Ref. 51], tributyl phosphate and tributyl iodophosphate [Ref. 52], and also that the sum of these elements and rhodium can be extracted by anion-exchange techniques [Ref. 53], and that an ion-exchange method can be used to extract technetium [Ref. 54, 55] and palladium [Ref. 52, 56].

2.3. Non-Aqueous Processes Involved in the Aqueous Reprocessing Arrangement

Among such processes is removal of the cladding by melting, and also oxidation of fuel to remove gaseous and volatile fission products as well as thermal operations on the stage of obtaining the final products. As noted above, such processes may precede the aqueous scheme of reprocessing when handling fuel that has been held for less than a year.

Some theoretical principles of the method of melting and operation of the equipment used are described in Ref. 57-59. Experimental verification of the melting method and of fuel oxidation has been done mainly in connection with subsequent gas fluoride processing, and therefore is described in more detail below in examination of non-aqueous methods. The results are completely suitable as well for use of these processes in combination with subsequent dissolving of the opened fuel in aqueous nitric acid solutions.

In the area of thermal processes for getting the goal products of the scheme--uranium and plutonium oxides--research has been done on various compounds of these elements that can be isolated from aqueous solutions, and their subsequent thermal dissociation. In addition to hydroxides or ammonium salts, studies have been done on basic salts that frequently produce larger crystals and are more readily filtered out, and salts of lower carboxylic acids that in some cases are more easily subjected to thermal dissociation than the most extensively studied oxalate compounds of actinides. The literature describes the production and properties of hydroxonitrates and hydroxoacetates of Pu(IV) [Ref. 60, 61], series of thorium hydroxo salts (as an analog of Pu(IV)) with monocarboxylic acids [Ref. 62], acetates and formates of U(VI) [Ref. 63]. An interesting observation is made in Ref. 64, in which it is shown that attachement of the carbamide molecule to the molecule of dioxouranium(VI) formate further reduces the temperature of thermal dissociation of this compound, which should be conducive to production of more chemically active oxides.

The results of the research coverage described above may be used for further improvement of aqueous extraction technology in fast reactor fuel processing.

3. Non-Aqueous Methods of Reprocessing Fast Reactor Fuel Elements

In accordance with the program of research and development on regeneration of depleted nuclear fuel [Ref. 3] on the "Fregat" facility, a series of experiments has been done on thermal opening of irradiated fuel elements and oxidative recrystallization and fluorination of fuel.

3.1. Thermal Opening of Irradiated Fast Reactor Fuel Elements

Experiments on opening of fuel elements have been done on the standard equipment of the "Fregat" facility [Ref. 65]. Maximum burnup of uranium-plutonium fuel assemblies irradiated on the BOR-60 reactor and opened by melting reached 135,000 MW-days/kg of $UO_2 + PuO_2$ with cooling for 213 days after a run. Weight ratio of masses of steel and fuel 1:3. Measurements showed that the volume of gas phase accumulated under the fuel element cladding was about 2.3 liters per kilogram of fuel.

In melting, the steel ran down satisfactorily through the filter into the crucible. The denuded fuel was in the form of sintered rods with axial channel aperture. The rods were broken by slight mechanical action. Upon discharge from the crucible they were converted to grit.

The direct yield of product attained in the opening operation was 99.9±0.1% with steel content in the fuel of 3.2% or less, i. e. melting the cladding gives fuel extraction near 100% with fairly effective separation from steel. A certain increase of steel content in uranium-plutonium fuel as compared with uranium fuel can apparently be attributed to the high wettability of the fuel due to the presence of plutonium. The plutonium content in the steel ingot was 0.005%, and in the ceramic liner--0.025%.

The results of analysis of the distribution of fission products and yield of volatile components from uranium-plutonium fuel showed that they differ but little from previously published indices [Ref. 3] for uranium fuel.

3,2, Oxidative Recrystallization of Fast Reactor Fuel

This operation following thermal opening of fuel elements solves several problems: dispersion of fuel to the dimensions of particles suitable for subsequent fluorination; separation of fuel from residues of construction materials; isolation of gaseous and volatile fission products.

On unexposed specimens of nuclear fuel an investigation was made of the rate of oxidation of briquettes of solid solution of uranium dioxide and plutonium dioxide (15 mass% PuO_2), briquettes of uranium dioxide, and the same briquettes after thermal opening of fuel elements, and also investigation of the conditions of producing U_3O_8 powder with a certain particle size [Ref. 65]. The briquettes were oxidized at a temperature from 350 to 550°C.

The studies showed that with increasing temperature there is a reduction in the duration of the oxidation process. Oxidation of briquettes after thermal opening of fuel elements also takes place more rapidly, since these have a

larger specific surface due to micro- and macrocracks. With a reduction of oxidation temperature there is an increase in the fraction of fine fractions of powder. As oxidation temperature increased for briquettes of solid solution $(U, Pu)O_2$, there was an increase in the rate of the process, but it was less than for UO_2 briquettes. The products of oxidation of solid solution mainly consisted of U_3O_8 with plutonium dissolved in its rhombic lattice. No independent PuO_2 phase was observed.

Research done on oxidative recrystallization of irradiated uranium oxide fuel under static conditions has shown that the degree of dispersion of uranium dioxide resulting from oxidation is somewhat lower than expected, although uranium dioxide is fairly easily oxidized at parameters near those used for unirradiated material. Apparently this is the result of the presence of up to 10-12% by mass of fission products in the fuel, some of which are in the oxidized state.

An increase in duration of the process and rise in temperature have little effect on increasing the percentage of fine fractions of powder, although there is a noticeable increase in the completeness of fuel oxidation. Apparently the presence of solid solution of plutonium dioxide in the structure in the aggregate with a considerable amount of fission products affects the mechanism of the process of oxidation of the solid solution, impeding rearrangement of the structure of the fluorite lattice.

The latest experimental results from the USSR and elsewhere [Ref. 65] evidence a high yield of gaseous and volatile fission products in the process of oxidative recrystallization of uranium-plutonium fuel at elevated temperatures. The extraction from fuel has reached up to 99% for tritium, from 40 to 98% for krypton and xenon, from 70 to 98% for iodine, up to 90% for ruthenium, etc.

From the research results it is concluded that intensification of volumetric oxidation of $(U, Pu)O_2$ solid solution will necessitate the use of mechanical action, and specifically vibration, on the fuel during oxidation.

3.3. Fluorination and Separation of Fluorides

On the "Fregat" facility, accumulation of statistical data has continued on basic technological indices of the process of fluorination of irradiated uranium fuel and removal of volatile fluorides of fission products from the uranium hexafluoride [Ref. 67-69]. A series of experiments has been done on six fuel assemblies of the BOR-60 reactor in which several suspensions of uranium dioxide with overall weight of more than 12 kg were fluorinated; fuel burnup--10-12%, cooling--from 4 to 48 months. Before starting the experiments, the specific radioactivity of the irradiated uranium dioxide was quite high, and amounted to $8.5 \cdot 10^{12}$-$1.6 \cdot 10^{14}$ s^{-1} per kg.

Most of the experiments on the "Fregat" facility, as in earlier experiments, included operations of: fuel fluorination--sorption of uranium hexafluoride on sodium fluoride, desorption and resorption of UF_6 on NaF, desorption--condensation of uranium hexafluoride and finally, recondensation in the shipping cask. In some experiments the uranium hexafluoride was collected in the

condenser immediately following the fluorinator. This was necessary for establishing the feasibility of removing fission product fluorides from uranium hexafluoride by a condensation method.

As in the first experiments, a high degree of uranium extraction and removal of fission products was achieved in the regeneration process. Losses of uranium in the solid residues of the fluorination reactor and in the sorbent of the purification sorption column (unrecoverable losses) did not exceed 0.4%. Technological peculiarities that could be established in this series of experiments are as follows:

1) there are signs that there could be a considerable increase in the efficiency of condensation of fluorides of ruthenium and niobium, and consequently removal from uranium hexafluoride by combining the precondenser for precipitating these compounds with the fluorination reactor;

2) in the condensation-evaporative version of technology the advisability of using sorption cleaning of uranium has been confirmed: after vaporization of uranium from the condenser, the sorbent held an appreciable amount of ruthenium-rhodium fluorides;

3) using circulation of fluorine-nitrogen mixture on operations of desorption of uranium hexafluoride enabled a reduction in fluorine expenditures in the process, and simultaneously reduced the consumption of exhaust gas absorbers. The specific consumption of fluorine when using circulation was 0.97 kg/kg of processed fuel; the specific expenditures of chemical absorber (when about 50% was used) amounted to 1.6 kg/kg of regenerated uranium;

4) sorption columns of sodium fluoride in repeated use withstand at least six sorption-desorption cycles. The expenditures of sorbent in this case decrease to 1.75 kg/kg of reprocessed uranium (previously this coefficient had been 5);

5) it has been confirmed that reliable filtration of gas flows is necessary to prevent penetration of radioactive aerosols into the final product.

After completion of the "uranium" part of the experimental program on technological investigation of the process of fluoride regeneration of irradiated fuel on the "Fregat" facility, studies were begun on the behavior of plutonium. To do this, the "Fregat" facility was modified, and fuel fluorination equipment was changed. At this time the plutonium program is realized with the use of artificial mixtures of uranium and plutonium oxides and real uranium-plutonium composites that have undergone prolonged irradiation in the BOR-60 reactor. At the same time, considerable research is being done on reducing the volumes of radioactive waste disposal, methods of converting waste to a form suitable for prolonged storage and burial.

3.4. Salt Process of Recycling Fluorination Residues and Sorbents that Contain Plutonium

To extract the small fraction of valuable components contained in fluorination residues, these materials can be dissolved in molten fluoride salts with subsequent isolation of solid oxides of valuable elements from the melt by using

fluorine-oxygen exchange based on the effect that added oxides of aluminum or calcium have on fluorides of uranium and plutonium [Ref. 70-72]. In the case of an excess of added oxides, fairly complete isolation of valuable components from the melt is achieved with considerable purification from rare earth elements and some other fission products, as well as from alkali and alkali earth elements.

The use of fluorine-oxygen exchange in fluoride melts may in future be one of the possible solutions for the problem of complete extraction of plutonium from nonvolatile products of gas-fluoride technology. In this connection it can be expected that there sill also be a solution for the problem of utilizing valuable fuel components (in addition to uranium and plutonium). For example, after isolating plutonium oxides from the melt, the further addition of oxide reagents enables successive precipitation of americium and rare earth elements from the melt [Ref. 73, 74]. The feasibility of effective realization of the salt processes also, as for other processes of non-aqueous technology, depends on future development of appropriate equipment suitable for remote servicing.

3.5. Prospects for Further Research on Gas-Phase Technology

The next stage of development of research to develop fluoride technology for regenerating highly radioactive fast reactor uranium-plutonium fuel is tied up with the program of development of the "Fregat-2" facility. This remote-controlled automated facility is being developed conjointly by the USSR State Committee on Atomic Energy and the Czechoslovak Atomic Energy Commission [Ref. 75] to be accommodated in specially made chambers at the Scientific Research Institute of Nuclear Reactors imeni V. I. Lenin. The facility is intended for comprehensive investigation of the fluoride scheme in all its technological aspects based on the example of regenerating BOR-60 reactor fuel elements. Its productivity is about 3 kg of irradiated fuel per hour. The facility realizes the following technological operations: melting the cladding from the full-scale fuel assembly of the BOR-60 reactor, high-temperature fluorination of uranium-plutonium fuel to produce hexafluorides of uranium and plutonium, thermal isolation of plutonium tetrafluoride with final purification and conversion to oxides, rectification of uranium hexafluoride and other operations. The project provides for cryogenic removal of oxygen from fluorine for more complete (near 100%) use of the latter in the technological process.

It is to be hoped that in the next few years a final answer will be found to the question of the possibility of practical realization of the technological advantages of the gas fluoride method of regenerating fast reactor fuel elements.

Discussion

I. Peka: What amount of fuel remains when residues are dissolved in nitric acid with the addition of fluorine ion?

V. I. Zemlyanukhin: When residues of dissolution of fast reactor fuel are dissolved in nitric acid for more complete extraction of plutonium,

hydrofluoric acid is added. As a result, the unextracted plutonium will be a fraction of a percent.

I. Peka: What amounts of fast reactor fuel have been used in doing experiments on extractive processing? How long was the fuel cooled before reprocessing?

V. I.Zemlyanukhin: Experiments have been done with kilogram amounts of BOR-60 reactor fuel with holding for 1-2 years.

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STORAGE OF SPENT FUEL AT NUCLEAR ELECTRIC PLANT WITH VVER-440 REACTORS

Prague ISSLEDOVANIYA V OBLASTI PERERABOTKI OBLUCHENNOGO TOPLIVA I OBEZVREZHIVANIYA RADIOAKTIVNYKH OTKHODOV in Russian Vol 1, 1981 pp 62-76

[Article by V. M. Sedov, A. N. Kondrat'yev, I. Ye. Kiyko, S. G. Danilov, N. M. Dusayev and B. I. Snaginskiy (paper delivered by V. V. Morozov)]

[Text] This paper outlines basic decisions on storage of spent fuel and problems for further research on prolonged storage as applied to VVER-440 reactors.

1. Introduction

The plans of nuclear electric facilities with VVER reactors to be constructed in Bulgaria, Hungary, East Germany, Czechoslovakia and other CEMA member nations provide for storage facilities for spent fuel at the reactor sites that are designed for holding fuel for three years. Shipping casks are being designed for transporting spent fuel with such holding to regeneration plants. However, the indicated fuel holding time is advisable only during the period when mass construction will start on fast reactors in which plutonium will be used that is of major value in VVER reactors. Considering the delay of series construction of fast reactors in the Soviet Union and other CEMA member nations, the decision has been made to build additional spent fuel storage facilities apart from the rezctor that are designed for approximately ten years of nuclear electric plant operation, and that will be used for the entire operating period of the plant (~30 years).

The first such facility for the Kozloduy Nuclear Electric Plant in Bulgaria is being developed jointly by Soviet and Bulgarian specialists. The engineering plan has now been approved, and the working drawings for the storage facility are being developed. The decisions on this project are being recommended as standard for Hungary, East Germany, Czechoslovakia, the USSR and in case of necessity for other CEMA member nations as well.

This paper outlines the major decisions on storage of spent fuel, and problems for further study on questions of prolonged storage as applied to VVER-440 reactors.

2. Transport-Technological Part

Additional storage of spent fuel for nuclear electric plants with VVER-440 reactor units has been developed on the basis of experience in designing and

operating similar facilities in the USSR, as well as the experience of other nations. This storage facility is intended for complete reception of depleted fuel over a period of about ten years from four reactors, which amounts to about 5000 fuel assemblies, or 600 metric tons of fuel.

2.1. Layout and Equipment

Decisions on configuration, technological equipment and process have been made with consideration of maximum use of standard, typical and nonstandardized equipment used in facilities of this type for spent fuel assemblies of reactor units in the Soviet Union.

The additional storage facility for spent fuel is a separate building that consists of three main departments:

--reception, transfer and distribution of shipping casks;
--fuel storage;
--technological systems and services for support of fuel storage conditions.

The department for reception and transfer of shipping casks consists of a transport corridor and a transfer room. The transport corridor runs beneath the transfer room and is connected to the room by a covered opening for passage of containers.

Situated in the transfer room is a cask transfer compartment, a compartment for washing and transferring fuel assemblies, pits for washing and decontaminating casks, shafts for storage of attachments and tools, and also a section for minor repairs of technological equipment. The transfer and washing compartments are connected by a transfer corridor to the compartments for storage of spent fuel assemblies.

The transfer room is equipped with:

--a special overhead crane with lifting capacity of 125/20 metric tons, that has auxiliary reduced speeds snd other devices for work with special attachments;

--a special platform with facilities for removing covers from casks;

--a transverse beam for transferring containers, a rod for covers of containers and coffins, and other tools necessary for transfer work.

The department for storage of spent fuel consists of a pool filled with water and a transport room. The pool has four compartments in which coffins with spent fuel assemblies are placed for storage, and a transfer corridor that interconnects all pool compartments, and also compartments for transfer and washing of fuel assemblies.

The pool compartments are connected to the transfer corridor by hydraulic locks for sealing off each compartment according to technological conditions or for carrying out repairs.

The walls and bottom of the pool are lined with a double casing of carbon steel and stainless steel. Possible leaks through the inner casing enter the gap between liners, are absorbed and returned to the pool compartments, enabling a check on the state of the casings of each compartment of the pool, and practically eliminating outside leaks from the pool.

The pool compartments have a slotted cover, providing normal conditions for personnel operations. The slots in the cover are transport passages for conveyance of fuel coffins, and provide the necessary disposition of coffins in pool compartments. The transport room of the storage facility is situated above the pool.

The transport room of the facility is equipped with:

--a special overhead crane with lifting capacity of Q = 15 metric tons with auxiliary reduced speeds;

--rods and grippers for transferring coffins;

--lighting and portable television equipment for underwater inspection of coffins and fuel assemblies.

The department of technological services and systems for supporting spent fuel storage conditions is an annex to the storage facility, accommodating systems for:

--cooling and purifying pool water;
--technological and sanitation ventilation;
--electric power supply for the storage facility;
--washing and decontaminating casks, special equipment and enclosures;
--panels and operator rooms for monitoring and control;
--sanitary passage, administrative and other areas.

3. Major Transport-Technological Operations

3.1. Reception of Spent Fuel in the Storage Facility

Fuel is transported from the reactor units of nuclear electric plants to the storage facility in the TK-6 shipping cask in which a coffin is installed for 30 spent fuel assemblies. After holding for three years in the pool at the reactor, the spent fuel is loaded into the cask, which is transferred by a specially equipped trailer to the transport corridor of the storage facility. The overhead crane in the transfer room lifts the container into the room and puts it into a water-filled transfer compartment. The overhead crane in the transport room takes the coffin with spent fuel assemblies out of the container cask, and transfers it to the pool compartment, where it is placed for storage. The container cask is moved from the transfer compartment to washing and decontamination, after which an empty coffin is installed. The ready cask is sent to the reactor facility for a new load. About a month is required for transferring a year's supply of spent fuel assemblies from one reactor, placing them in storage and removing the same number of assemblies.

3.2. Storage of Spent Fuel

The storage of spent fuel is handled without the constant presence of service personnel. A panel at one of the modules of the nuclear power plant displays secondary monitoring signals on the state of principal technological parameters that determine normal conditions of storage of fuel assemblies: water temperature of pool compartments, water level in compartments, information on operation of cooling and purification systems, condition of air atmosphere of the pool.

3.3. Removal of Depleted Fuel From Storage Facility

Depleted fuel is removed from the storage facility in the TK-6 shipping cask. The cask with unloaded coffin is driven into the transport corridor and taken to the transfer room, where the empty coffin is removed. The cask is placed in the transfer room, where it is loaded with a coffin containing fuel assemblies. The loaded cask is washed, decontaminated and placed on a technological monitoring stand where it is brought to a steady technological state with respect to temperature, and checked for gas-tightness. The ready container is placed on a trailer and hauled away from the storage facility. Four casks can be prepared for dispatching simultaneously in the storage facility.

3.4. Method of Storing Spent Fuel

Spent fuel is stored under a shielding layer of water (~3.0 m over the active part of the fuel assembly). Such a technique ensures reliable storage of spent fuel, does not need complicated equipment, and at the same time meets some important conditions on providing:

--biological shielding in performing all technological operations with direct visual control;

--constant reliable removal of heat from the fuel assemblies;

--carrying out technological processes of transfer and storage of fuel with a small amount of simple equipment.

Possible accidents with spent fuel as it is being transferred and transported as well as accidents with the technological systems do not lead to a critical emergency in the storage facility and can be handled with a minimum of reserves, with restoration of normal conditions in a normal environment.

The only disadvantage of such a method of storage is that there is as yet no technological design solution for pool casings that would guarantee total water-tightness without leaks throughout the service life of the storage facilities of ~30 years.

The use of a two-layer casing in the pool provides rather good prevention of water leakage into the ambient environment, but does not guarantee total exclusion. Therefore research and development is needed on new, more technological and economic solutions of waterproofing that guarantee exclusion of leaks throughout the service life of the storage facility.

Spent fuel assemblies can be stored in pools by two basic methods:

--individually on racks;
--in multiplace coffins installed on the bottom of the pool.

The storage facility that we are considering here uses the method of storing spent fuel assemblies in coffins in which they are placed in the pool and transported in the TK-6 container. The use of coffins of this type gives a number of important technological and economic advantages over the rack method:

--it provides a high degree of safety against damage of fuel assemblies since the coffin shields the assemblies from mechanical factors and eliminates the additional overloads that are inevitable in the rack method of storage;

--there is no need for an overhead coordinate reloader since the processes of transfer and transport of fuel assemblies are handled by the overhead crane alone. The same crane is also used for repairs and preventive maintenance in the pool;

--the coffin method enables rapid and efficaceous preparation of the pool compartments for repair by moving coffins to a spare compartment.

In the rack method, a compartment would be emptied by moving individual fuel assemblies. This takes thirty times longer, and besides, after the transfer it is necessary to remove the racks from the pool bottom, which involves a lot of work. It should be noted here that it takes about the same amount of stainless steel to make racks or coffins.

4. Chemical-Technological Part

The purpose of chemical-technological services of the storage facility consists in the following:

--maintaining standard water indices;

--keeping the water transparent for carrying out transport operations deep in the pool;

--maintaining corrosion resistance of structural materials of fuel assemblies and the lining of pools;

--keeping a normal radiation environment by removing radioactive contaminants from the water and decontaminating equipment and enclosures;

--rinsing shipping casks with fuel assemblies to the levels provided for by shipping rules.

Based on experience in operating pool storage facilities in the USSR, the quality of water is normalized in accordance with the following indicators:

pH = 6.0-7.5
$Cl^- + P^-$ ≤500 μg/kg
electrical conductivity ≤3 μS/kg
(1-2 mg/kg of dissolved salts)
corrosion products <0.5 mg/kg

These water quality indices ensure:

--transparency with visibility of objects measuring 100 x 100 mm at depth of 7.5 m;

--high corrosion resistance of fuel assembly materials (zirconium) and liner materials (stainless steel).

The corrosion rate of materials is:

--for stainless steel--1 mg/m^2hr, with uniform corrosion. This corresponds to a reduction in pool liner thickness of less than 0.01 mm/yr;

--for zirconium--0.5 mg/m^2hr.

These quality standards are attained by purifying the water. As a rule, the purification is done by a two-stage scheme: on the first stage the suspended corrosion products are removed from the water; on the second stage, dissolved salts are removed.

Radioactive contaminants are simultaneously removed on both stages.

The first stage uses equipment and filtering materials of two types:

--wash filters of cartridge type with the use of auxiliary filtering material, --perlite;

--loose filters with the use of cation exchange resin as the filtering material.

The second stage is made either in the form of two filters with different charges of cation and anion exchange resins, or in the form of a single mixed-action filter.

The resins are regenerated with 5% solutions of nitric acid and caustic soda.

Wastes from the facility in the form of regenerates and pulp are sent to evaporative condensation and solidification.

Productivity of the purification facility is determined by calculation.

The productivity of the first stage is calculated by the formula:

$$Q = \frac{\Sigma r_i F_i}{C - C_1} \times 1 \cdot 10^{-3} \text{ m}^3/\text{hr},$$

where r_i is the rate of yield of corrosion products of the i-th material in mg/m hr; F_i is the surface of the i-th material in m^2; C is the permissible

concentration of corrosion products in the water in mg/ℓ; C_1 is the concentration of corrosion products in the purified water in mg/ℓ.

The productivity of the second purification stage is calculated from the material balance between impurities entering the water and removed from it.

The main sources of entry of impurities are makeup water added to compensate for evaporation from the surface of the pool, suction of cooling water into the heat exchangers, and also other sources such as contaminants introduced when doing transport operations.

The total productivity of the purification facility is determined with respect to the largest calculated value.

It is difficult to calculate the productivity with respect to water purification from radioactive contaminants since it is impossible to determine the rate of output of radioactive isotopes into the water of the storage facility under conditions of prolonged storage. However, experience in using storage pools has shown that for a selected productivity (40 m /hr) for maintaining normalized water quality the content of radioactive impurities in the water is simultaneously maintained on a level no higher than $3.7 \cdot 10^4$ Bk/ℓ. This creates a normal radiation environment for pool operation.

The following decisions were made in the project of the Kozloduy Nuclear Electric Plant.

Productivity of 40 m^3/hr, first stage of purification--loose filter 2.6 m in diameter with KU-2 cation exchange resin, second stage--separate ion exchange filters 1 m in diameter charged with KU-2-8 cation exchange resin and AV-17-8 anion exchange resin.

Washing of shipping casks to the levels specifed in "Rules of Transportation" at Kozloduy Nuclear Electric Plant is planned as follows.

The cask is lowered into a special shaft equipped with electromechanical hydromonitors. Composites of various compositions can be used as washing solutions. Provisions have been made for heating, circulating and repeat use of solutions.

After rinsing with solutions, the cask is washed with water, air-dried and checked for contamination. In case of necessity, individual parts of the cask may be cleaned with steam ejectors, and individual sections of the covering can be removed and reapplied.

5. Ensuring Work Safety in the Storage Facility

The plan for the storage facility provides for the following steps to assure radiation safety and environmental protection:

--location of the facility in a sanitary-protection zone;
--zonal configuration of rooms with organization of sanitary passages and locks;
--radiation monitoring inside and outside of the building;

--biological shielding that reduces the level of radioactive emission to permissible levels;
--storage of coffins with fuel, and also work on loading and unloading casks under a shielding layer of water;
--organization of the special ventilation system;
--system for purifying and cooling pool water;
--collection and removal of the liquid and solid radioactive wastes;
--system for decontaminating transport-technological equipment and transport facilities.

In normal operation of transport-technological systems, no appreciable contamination of the environment is expected.

The project has considered the following possible emergency situations inside the building of the storage facility:

--hanging and dropping of loaded cask;
--hanging and dropping of loaded coffin;
--operational failure of pool cooling system;
--rupture of pipeline of pool cooling system;
--leakage of water from the pool;
--failure of electric supply.

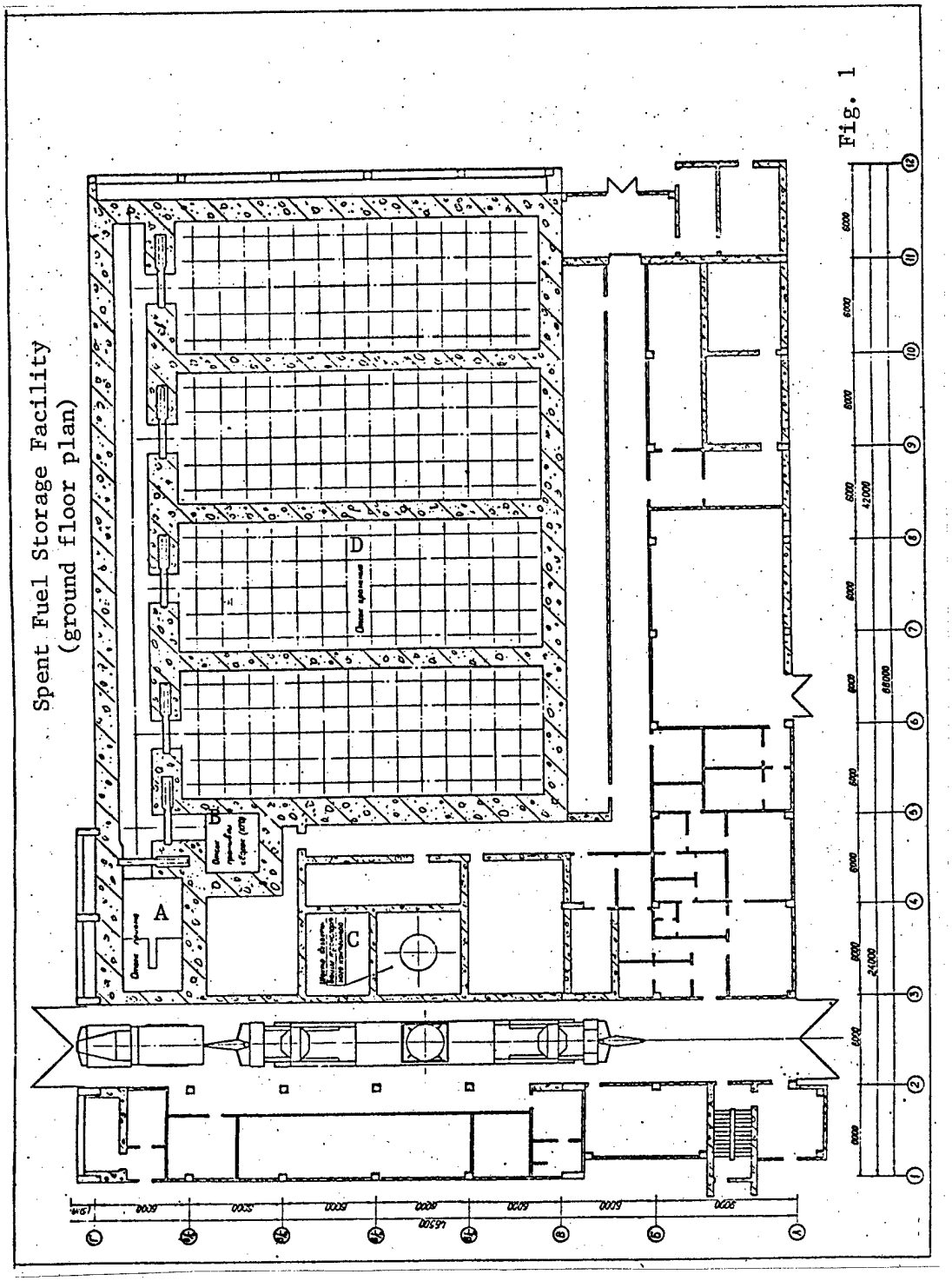
An analysis done on the consequences of possible accidents shows that the technology assumed in the project for receiving, storing and distributing spent fuel assemblies corresponds to sanitary-technical requirements, and the configuration of the rooms and equipment ensures safe working conditions for personnel both inside the building and in the environs of the storage facility.

All compartments in which work is done with depleted fuel are planned in such a way that under conditions of normal operation and in emergency situations the formation of critical systems is precluded. This is achieved by safe distribution of fuel assemblies in the coffin of the TK-6 [cask] for which $K_{ef} = 0.9$ (permissible $K_{ef} \leqslant 0.95$). The storage facility provides for all necessary steps to prevent emergency situations that might lead to a self-sustaining chain reaction, i. e. situations involving the escape of fuel assemblies from coffins or breakage of fuel assemblies and fuel elements.

5.1. Residual Heat Release

Residual heat release in spent fuel assemblies is due to decay of radionuclides. To remove the heat for the purpose of maintaining the temperature state of the pool water after placing fuel assemblies in the pool, the project provides for a forced cooling system.

The problem of removing residual heat emissions has been solved differentially for each compartment of the storage pool depending on the concentration and the absolute level of radionuclides, possible consequences of disruption in the removal of residual heat emissions and the permissible water temperature (taken as 50°C in the project).



A--Reception compartment; B--Compartment for washing fuel assemblies; C--Pit for decontaminating shipping casks; D--Storage compartment

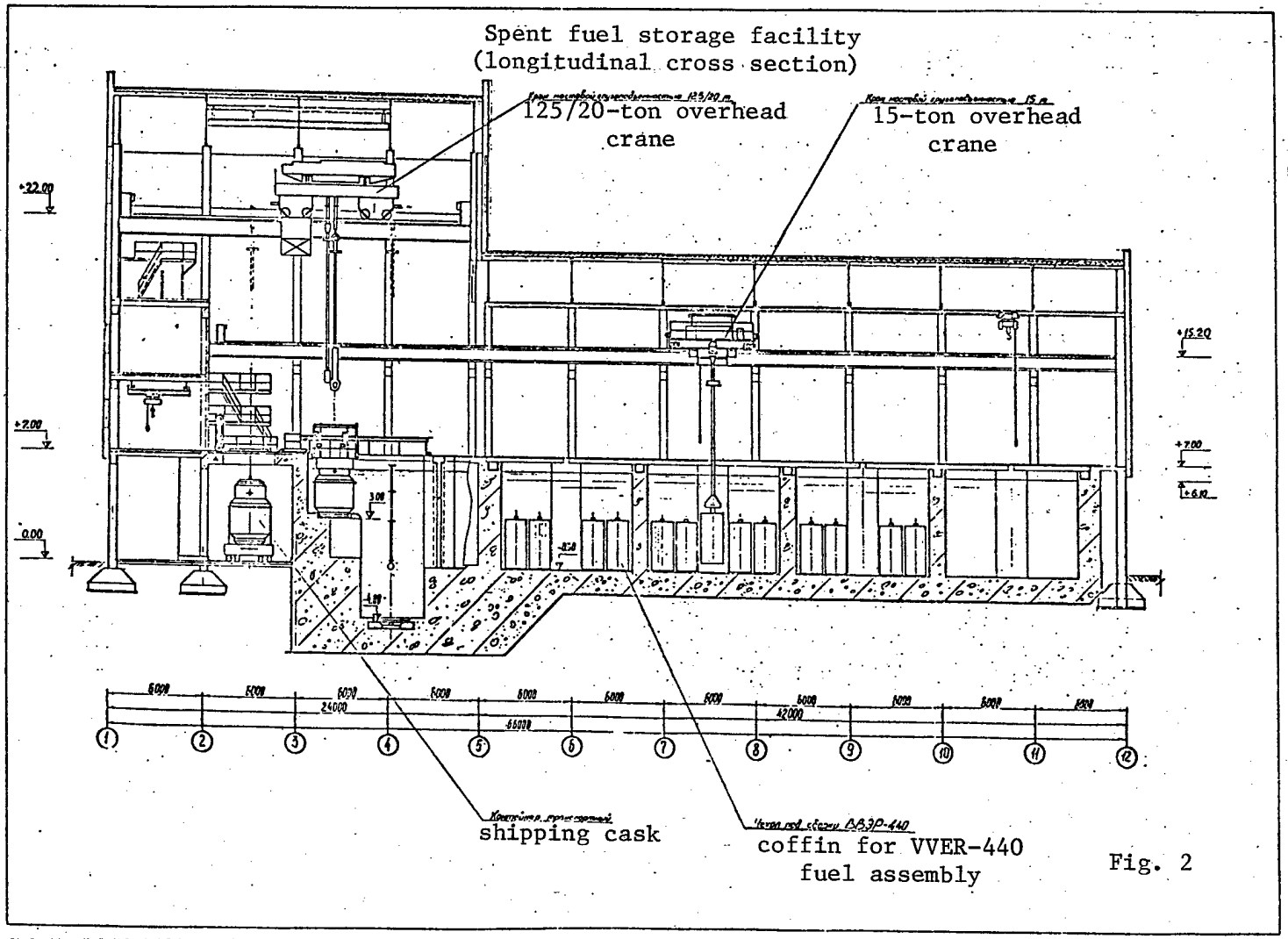

Spent fuel storage facility
(longitudinal cross section)
125/20-ton overhead crane
15-ton overhead crane
shipping cask
coffin for VVER-440 fuel assembly

Fig. 2

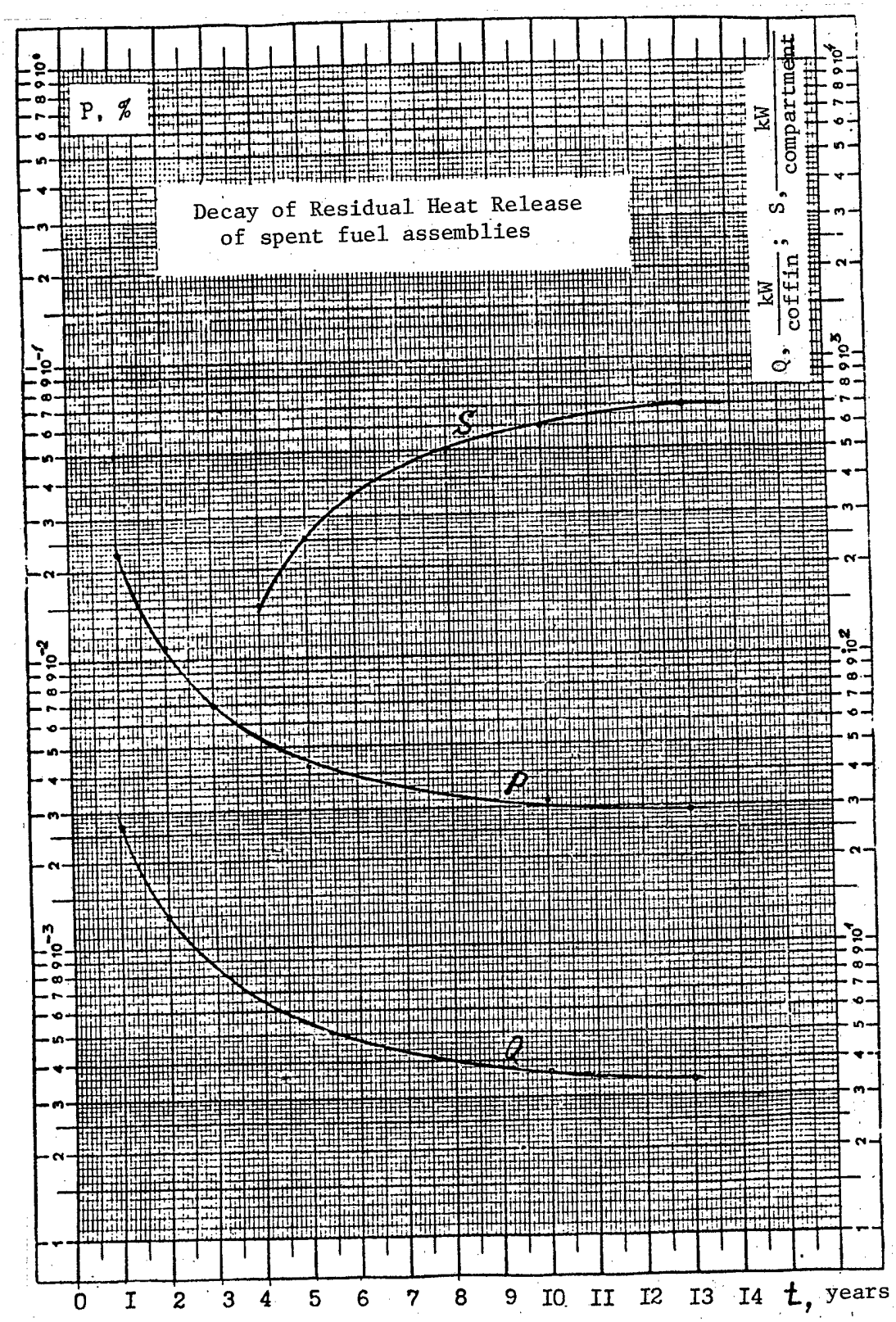

Decay of Residual Heat Release
of spent fuel assemblies
P, %
Q, kW/coffin; S, kW/compartment
S
P
Q
0 I 2 3 4 5 6 7 8 9 I0 II I2 I3 I4 t, years

Fig. 3

Note should be taken of the considerable inertness of the system, which means that failures in the cooling systems lead only to a slow rise in temperature, and as noted above, sufficient time always remains for taking the proper preventive steps.

The residual heat release of depleted fuel assemblies of the VVER-440 reactor is shown (in % and kW/coffin) as a function of holding time on Fig. 3.

Under conditions of uniform arrival of depleted fuel at the building of the storage facility at a rate of 16 coffins per year, the heat release in one compartment at the instant when it is filled up (56 coffins) is about 380 kW. Fig. 3 shows the curve for the increase in $S = f(t)$ of residual heat release over the entire facility for storage of depleted fuel after the first year of loading (absolute holding time $\leqslant 4$ years).

5.2. Liquid Radioactive Wastes

In accordance with the technology assumed in the project, various kinds of intermediate and low-level liquid wastes are formed in the process of operating the storage facility, and special systems are provided for collecting and removing them.

The main fraction of liquid radioactive wastes comes from the water of the storage compartments, decontamination waters and solutions, concentrates of filters in the water purification system.

The radioactivity of holding pool water is caused by radioactive products of activation and fission that are released during storage of fuel assemblies. For pre-cooled fuel, the most typical isotopes from the standpoint of handling wastes are cesium-137, cobalt-60, strontium-90 and praseodymium-144.

Planned or emergency dumping of a compartment is accomplished by a system for transferring water into the special water purification plant at the nuclear electric facility. In addition, this plant handles the decontamination rinsing water ($3.7 \cdot 10^3$ Bk/ℓ) and trap water (10^2-10^3 Bk/ℓ). It is expected that the volume of liquid radioactive wastes from the station will be 650 m^3 per year.

5.3. Special Ventilation and Emissions Into the Atmosphere

The following provisions have been made to prevent radioactive contamination from spreading through the rooms of the storage facility:

--zonal distribution of rooms with organization of directed airflow from rooms with a lower level of contamination toward those with a possibly higher level;

--separate special ventilation systems;

--a system of scavenging blowers on various installations and vessels with cleaning by aerosol filters;

--hermetically sealed containers (shipping cask, filters);

--emission of ventilation air into the atmosphere at an altitude of at least 35 meters.

Ejection of radioactive aerosols into the atmosphere from the storage facility under normal operating conditions is expected to be about $6.3 \cdot 10^5$ Bk/day (including $1.2 \cdot 10^5$ Bk/day of strontium-89 and strontium-90), which is less than 2% of the rated emissions of long-lived aerosols from the Kozloduy plant when two generating facilities are operating simultaneously in the normal mode. The concentration of radionuclides in the ground layer of the atmosphere at a distance of about 100 m from the axis of the stack is expected to be $9.2 \cdot 10^{-8}$ Bk/ℓ, the permissible concentration for category B being $4.8 \cdot 10^{-6}$ Bk/ℓ.

5.4. Emission of Tritium and Hydrogen

The only possible source of tritium occurrence in the storage facility comes from the depleted fuel elements. If an average of about $2.2 \cdot 10^{12}$ Bk of tritium accumulates in each fuel assembly of a VVER-440 reactor during a run, then a leak in the fuel element cladding in the core of the reactor releases the main mass of the tritium not bound to the fuel into the coolant of the primary circuit, and only a small amount may be released in the water of the holding pool during the first two months after removing the fuel assembly from the reactor.

In removing the fuel assembly, part of the primary circuit coolant is mixed with the water of the holding pool, and it may be contaminated with tritium. Tritium from the reactor building of the nuclear electric facility cannot get into the water of the pool in the long-term storage facility since the coffins with the fuel assemblies are filled with clean water before transfer to the building of the storage facility rather than with water from the holding pool.

Tritium release in the pools can occur in appreciable amounts only in the case where fuel elements are damaged in the storage facility itself as a result of melchanical factors in carrying out transport-technological operations. Experience in operating storage facilities in the Soviet Union and elsewhere has shown that this kind of damage to fuel elements is highly unlikely.

Hydrogen may be formed in storage pools as a result of radiolysis of water. Calculations show that the maximum possible release of hydrogen into the cavity above the water surface in one completely filled compartment is 0.26 nm^3/hr. This means that even in an emergency situation involving failure of the ventilation system, the volume of explosive mixture of hydrogen with air will be 1/10 of the permissible 5% of the volume of the given space.

6. Problems That Need Further Study

--systematic investigation of the stability of coverings of fuel elements and fuel with prolonged storage of fuel assemblies in water, paying particular attention to the behavior of leaky fuel elements. It is important to know whether prolonged storage will lead to further loss of integrity;

--further improvement and elaboration of scientifically substantiated norms of quality of water for storage facilities under various working conditions and with the use of different structural materials;

--investigation of radiative fusion of nitrogen and oxygen dissolved in water, and the resultant nitrite-nitrate corrosion;

--investigation of formation of scum on the surfaces of equipment and the bottom of the pool, the corrosion that takes place under scum, and ways to deal with it;

--development of highly productive and economically effective means of purifying water in pools (in particular when two filtration stages are combined in a single facility).

7. Conclusions

1. The delay in series construction of fast reactors has made it a sound decision to build additional storage facilities for spent fuel on nuclear electric facilities with VVER reactors.

2. All major engineering features incorporated into the design of the storage facility for nuclear electric plants with VVER reactors will enable safe storage of fuel assemblies and safe handling of transfer operations.

3. Further research needs to be done in the area of improving water chemistry, investigation of corrosion of structural materials and development of steps to reduce corrosion.

Discussion

F. Takach: According to plan, the main unit of storage is the coffin of the shipping cask. Has any consideration been given to including an absorber in the coffin so as to increase the number of fuel elements that can be stored and transported?

V. V. Morozov: The construction of the coffin and the TK-6 shipping cask is such that there is no room for additional fuel assemblies. This is not a question of nuclear safety, but merely geometry. The coffin is almost completely filled up, and the clearance between fuel assemblies is already at a minimum. It would be difficult to make the structure more compact.

A. Shtolmar: 1. In what form has the decision been made to construct additional storage facilities, bearing in mind that building such facilities increases capital investments on nuclear electric facilities of CEMA member nations?

2. What is the basis taken for calculating the total reception of spent fuel over the course of about ten years from four reactors?

3. In selecting the current method of prolonged storage of nuclear fuel, why was not consideration taken of:

a) experience of some western nations in storage with the use of absorbers (with packed grids)?

b) IAEA recommendations on setting up regional storage facilities?

V. V. Morozov: 1. I agree that construction of additional storage facilities will to some extent increase capital expenditures on building nuclear electric plants. But according to calculations, such a storage facility will not increase capital investments by much (no more than a few percent). As you know, storage difficulties have come up not only here, but in the West as well. Various plans have been considered, and fuel storage has been accepted as an interim solution. There have been difficulties in recycling fuel because of the unsolved problems of waste handling.

2. There are 120 fuel assemblies unloaded from the reactor in a year, four coffins from one generating unit. Everything is based on this calculation. The ten-year period is taken as a first step. In future, radiochemical plants will develop at a faster pace both in the United States and in our nations. We feel that continuous operation of nuclear electric plants will be achived in this period. Other decisions will be made at a later date.

3. The version of storage in shipping coffins was accepted as the basis, and when all calculations had been done it was considered to be the most convenient and technologically feasible. From the standpoint of fuel shipment, no additional technological operations are necessary. The fuel from a generating unit can be transported in coffins to the storage facility in one month. It was decided to make coffins without absorbers for VVER-440 reactors. As to VVER-1000 reactors, the coffins will definitely be made with absorbers. This question is automatically decided.

With regard to regional storage centers: The question of regional storage facilities is only now under discussion, and specific decisions have not been made either here or in other nations.

V. Velart: Is there to be a safety report, and what kinds of accidents will be analyzed?

V. V. Morozov: All possible accidents have been considered in the technical project of the storage facility, and it has been concluded that all structures and systems of the facility are reliable and will provide safe operation for 30 years.

E. Gladki: 1. Does the anticipated amount of liquid waste refer to what arrives at the special water purification system, or the final amount of liquid waste for storage in tanks?

2. The report gives data on aerosol emissions for normal conditions. Has an analysis been made to estimate emission in case of accident?

V. V. Morozov: 1. The amount of waste is quoted for the volume going to reprocessing. There will not be much salt, and the final volume of wastes will be small--650 m^3/year.

2. It is true that the individual problems considered here have assumed normal conditions. But no serious additional emissions are expected under emergency conditions because the fuel has already been held for a considerable time. Leakage of fuel in the storage facility is highly unlikely. And even if this does occur, judging from materials already available, there will be no tritium migration from fuel elements. The tritium will be confined to the fuel elements. And there will be no emissions of such isotopes as cesium, cerium and promethium.

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PROBLEMS OF TRANSPORTING SPENT NUCLEAR FUEL

Prague ISSLEDOVANIYA V OBLASTI PERERABOTKI OBLUCHENNOGO TOPLIVA I OBEZVREZHIVANIYA RADIOAKTIVNYKH OTKHODOV in Russian Vol 1, 1981 pp 78-87

[Article by V. Fisher, K. Rudol'f, V. N. Yershov, B. V. Kalinin, Yu. V. Kozlov, A. N. Kondrat'yev and Ye. I. Yulikov, GDR, USSR]

[Text] This report analyzes experience in transporting spent fuel assemblies from the Novovornezhsk, Rheinberg and Bruno Leuschner nuclear electric plants. Based on experience of shipments from these facilities, problems are formulated that must be solved to ensure more effective transportation of spent fuel. Data are given from tests of a new transport packaging facility and carrier designed for shipments of spent fuel from nuclear electric plants in CEMA member nations with VVER-440 reactors. The report examines basic directions and results of stages of the work that have been completed on transportation of spent nuclear fuel in the Soviet Union and East Germany. In particular, results are given on research now in progress: investigation of conditions of transporting nuclear fuel with damaged fuel elements, on studying the temperature conditions of casks in "wet" and "dry" shipments, including investigation of the stability of fuel assemblies in gas and gas-vapor atmospheres at high temperatures, investigation of the strength of shipping casks, analysis of questions of nuclear safety of packaging under normal and emergency shipping conditions, investigation of the explosion hazard of the gas mixture in water-filled containers, investigation and analysis of waterway transport of spent fuel.

Improvement of safety in transportation of spent nuclear fuel is becoming more and more important worldwide. This is due on the one hand to the unabated increase in the number of nuclear electric plants, and the consequent increase in intensity of spent nuclear fuel shipments. The increased traffic, as well as the necessity of shipments in the near future involving fuel with higher residual heat release, high specific activity and intense neutron flux, including fuel from fast reactors, requires the solution of more and more complicated engineering and organizational problems.

On the other hand, the current trend in some nations toward a pessimistic outlook on use of nuclear power in general is reflected in tightening of conditions for safety of nuclear facilities and associated operations, including transportation of spent nuclear fuel, although it should be noted that according to data of A. M. Platt [Ref. 1] for example, the United States has not recorded a single case of environmental radioactivity from type B packaging in accidents during shipment of radioactive materials. In some nations the requirements of IAEA Rules on Safe Shipment of Radioactive Materials are treated as only the necessary minimum for ensuring safe conditions of transportation. There is an ever sharper trend toward ensuring safety in transportation of spent fuel by improved reliability of packaging facilities. In this connection it has been proposed that packaging be calculated for a drop from a height of more than 9 m, flame action at the center of a conflagration for 1.5 hours rather than 30 minutes and so on. A re-examination of the IAEA Rules is now being prepared. It is proposed that there will be refinements in the Rules on conditions of heat tests, gas-tightness criteria, provisions for additional requirements on shipments by water and by air. The same goal of improving reliability is served by programs being introduced in some nations for guaranteed quality in making and using packaging, i. e. programs for bringing the quality of the actual packaging structures into line with their prototypes approved by responsible agencies.

Worldwide, shipments of spent fuel are currently handled by three modes of transport: truck, railroad and water transportation. The decision to use truck or railroad transportation depends on the specific transport conditions of the nation, the availability of railroad approach tracks, and the appropriate equipment at the nuclear electric plant, the reprocessing plant or intermediate storage facility, size and type of container, and economic aspects. Railroad transportation is becomingly increasingly more important for shipping heavy containers massing 80-100 metric tons. Since 1975, the volume of railroad shipments in France has increased from 68 to 80%, and in West Germany--from 10 to 50%.

In France, special trains are used for railroad transportation of spent fuel, and truck shipments are handled only on secondary roads.

In Great Britain, transportation is handled by ordinary trains at normal speed [Ref. 2].

In all other Western European nations, trucks are being increasingly used to ship spent nuclear fuel [Ref. 3].

In the United States, spent fuel from nuclear electric plants without approach tracks is handled by trucks, using light casks massing no more than 40 metric tons. These shipments are handled mainly by carriers that have at their disposal the necessary transport facilities for casks type NAC-1, NFS-4, NLI 1/2, TN-8 and TN-9 [Ref. 4].

Since 1979, shipment schedules in the United States have been changed in the respect that all shipments of spent fuel are now reported to the Nuclear Regulatory Commission (NRC). The shipment is routed through regions of low

population. Agreements have been made with police that the shipment will not be stopped enroute. In addition, the truck must be in a convoy and be provided with radio communication, and also be secure against theft. In railroad shipments, trains with spent fuel can be handled like ordinary freight traffic if the container conforms to national safety requirements. NLI-10/24 and IF-300 containers meet these requirements.

Marine shipment is governed mainly by the geographic location of the recycling plant and the nuclear electric facility, e. g. shipment between Western Europe and Japan. As a rule, shipment of a small amount of spent fuel is treated as separate cargo on ordinary freighters, railroad ferries or vessels of type Po-Po.

The first specialized ships were used for shipping spent magnox fuel from Italy, and also for the first shipments of spent fuel from Japanese light-water reactors. New ships with displacement of approximately 3000 metric tons that can handle 60 metric tons of spent fuel per passage have been ordered and already built to transport the large amount of spent nuclear fuel called for by contract from Japanese nuclear electric plants to Windscale (Great Britain) and Hague (France) [Ref. 5]. High safety requirements are almost completely met on these ships to avert disasters. Auxiliary equipment is used primarily for maintaining normal shipping conditions (temperature, power, radiation dose), and also for immobilizing containers and reducing the risk of shipping accidents (collisions, going aground, capsizing, fires and so on). Since December 1977, the specialized ship "Hindoura-Maru" [Ref. 6] has been handling shipments between Japanese nuclear electric plants and the recycling plant in Tokaj.

Shipping casks of different designs are used for spent fuel transport. Ref. 9 provides an extensive survey of spent fuel containers that are used at present and that are in various stages of manufacture or development. Here we should point out some features and differences of new containers of type HZ-75T (Japan), NAC-3 (United States) and "Castor" (West Germany). All three types are heavy casks massing 75-100 metric tons, and will be transported by railroad or marine shipping.

The HZ-75T cask [Ref. 7] is a horizontal cylindrical vessel with lead shielding massing 80 metric tons, and designed for transporting seven fuel assemblies of PWR reactors, or 17 fuel assemblies of BWR reactors. Transportation may be "dry" or "wet".

Annular steel ribs can handle thermal power of 84 kW. During transportation, the container is cooled by an external system (two air blowers) with cooling pipes that also pass through the neutron shielding consisting of a 45-55% water-ethylene glycol solution between the outer jacket and the lead shield.

The fuel assemblies are accommodated in a stainless steel coffin with boral partitions. The cask meets safety schedules of the United States, Japan, the IAEA, and is classified as a type B(M) container.

The United States is developing the new NAC-3 cask with mass of 100 metric tons for handling 12 fuel assemblies of PWR reactors or 32 fuel assemblies of

BWR reactors, and also a smaller version, the NAC-2 with mass of 80 metric tons for shipping seven fuel assemblies of PWR reactors or 21 fuel assemblies of BWR reactors [Ref. 10]. These casks are made of low-alloy steel with wall thickness of 360 mm with inner and outer shells of stainless steel. The neutron shield is made of WEP (water-extended polymer) combined with uranium or lead. The polyester in the form of individual blocks is fastened to the housing of the cask between ribs during transportation of spent fuel with large burnup. The cask meets safety requirements of the United States and West Germany.

West Germany is developing a series of containers with the name "Castor" [Ref. 9-11]. An experimental model of ductile cast iron massing 80 metric tons was subjected in 1979 to tests scheduled for type B packaging. Before testing for a fall from a height of 9 m the vessel was cooled to -40°C [Ref. 11]. Until now, the design of Castor containers has been based on one-year cooling of spent fuel. For such a holding time, the heat release of fuel assemblies is 25-50 kW per container. In "dry" shipment, the maximum temperature in the container does not exceed 350°C. As the neutron shield, a light hydrogen-containing material that does not lose its neutron-absorbing properties even in case of fire should be placed in openings in the wall of the cask. Casks massing from 60 to 105 metric tons are being developed for different types of fuel assemblies. The capacity of these containers is 4-9 fuel assemblies for PWR reactors and 16-25 fuel assemblies for BWR reactors.

Provisions have been made for using "Castor" containers in interim storage of spent nuclear fuel.

A two-component system of gamma and neutron shielding is typical of new container developments.

Boronated water, Bor-Silicon rubber, boronated beechwood (Permali-JN), water-ethylene glycol mixture and NS-2 are used as neutron shielding.

To reduce the cask loading time, steps are being taken for effective decontamination, e. g. making all outer surfaces of stainless steel, and isolating the ribbed surface during loading operations under water. Dry loading is not possible on most nuclear electric facilities.

To prove the safety of new containers being developed, large test stands have recently been built, enabling full-scale testing of casks for conformity to type B. For example, in Oak Ridge, Tennessee, for drop-testing 100-ton containers from a height of 46 m [Ref. 4], in West Berlin for drop-testing 100-ton containers and fire-testing, and in France for fire-testing at 1000°C for 1.5 hours.

At the present time, large amounts of spent fuel are being shipped from gas-cooled reactors to Great Britain or to France. The total volume of traffic has been nearly 1000 containers per year with capacity from 2.2 to 2.5 metric tons of uranium [Ref. 3, 9]. A total of about 15,000 metric tons of uranium in spent fuel assemblies had been shipped to Windscale up to 1979.

To handle shipments of spent fuel from light-water reactors inside Europe, the NTL company was instituted (Nuclear Transport, Limited) that transported

680 metric tons of uranium in 490 shipments using a large number of cask types up to the end of 1978 [Ref. 9]. Annual volume of shipments is approximately 150 metric tons of uranium. The Pacific Nuclear Transport Company was set up for shipment from Japan to Western Europe, which was started in 1977 with a volume of 3,200 metric tons.

In the United States, about 1000 metric tons of uranium have been shipped to intermediate storage facilities in Morris [Ref. 9].

Removal of spent fuel has not started at most nuclear electric plants.

The economics of transporting nuclear fuel is determined by capital investments and overhead. The cost of shipments in 1977 was estimated at 25-50 dollars/kg of uranium for internal shipments and 80-120 dollars/kg of uranium for intercontinental shipments [Ref. 9]. For Western Europe in 1979, the cost of shipments was estimated at 25-100 dollars/kg of uranium. The cost of making a light shipping cask is 0.5 million dollars, and for heavy casks--3 million dollars. The repective rental costs are 650 and 3,000 dollars per day [Ref. 8].

CEMA member nations are giving considerable attention to improving the reliability of packaging and ensuring guaranteed safety in transporting spent fuel.

Spent nuclear fuel has been transported from nuclear electric plants of CEMA member nations for about ten years. In the early years there were about 2-3 shipments a year. Currently, and especially in the future, plans call for a considerable increase in the volume of shipments. Experience in transporting depleted fuel shows the validity of the accepted concept:

--use of heavy shipping casks;
--use of specialized carriers;
--stress on railroad transportation.

Railroad transportation is most economic. The use of railroad transportation enables shipment of spent fuel in most cases by through freight, eliminating trans-shipment operations with containers, as well as construction of trans-shipment points, and in the final analysis improving shipping safety.

Transportation of spent fuel by special trains intended for such shipments alone improves the effectiveness of shipments and enables hauling a year's supply of spent fuel from one or two nuclear electric plant reactors in a single trip. The use of specialized trains also provides the safest conditions of transport with direct control of packaging parameters, constant guarding of freight, and timely steps where there is a deviation of set conditions and parameters of packaging facilities.

In some cases where there are no apprach tracks at the nuclear electric plant, the shipping casks with spent fuel from the power plant are delivered to the railway station by truck and water transport. Trucks and water transport facilities especially equipped just for the purpose are also used for transporting shipping casks with spent fuel.

The use of large-capacity containers reduces the traffic and the labor inputs at nuclear electric plants and the recycling plant. An increase in the dimensions of shipping casks increases the mass of the payload, i. e. the mass of the fuel carried by the containers. And the payload increases not only in absolute value, but also in the ratio of fuel mass to container mass. This ratio represents a kind of efficiency of the container, and is a criterion of its effectiveness.

CEMA member nations are doing research, developing practical recommendations and steps on transporting spent fuel within the scope of scientific-technical cooperation in the area of "Transporting Spent Fuel Elements". In recent years the specialists of CEMA member nations have been doing considerable work on solving the problem of transportation. Rules have been worked out for safe shipment of spent fuel from nuclear electric plants of CEMA member nations; these rules are the basic document for preparing, organizing and realizing transportation of spent nuclear fuel, technical specifications for transporting spent fuel by water transport in accordance with which spent fuel is being transported from Kozloduy Nuclear Electric Plant.

Research is nearing completion on processes of hydrogen accumulation when water-filled containers are used for transporting spent fuel. The technique of calculating gas mixture compositions and evaluating the explosion hazard of water-filled containers that was developed in the USSR has been experimentally confirmed.

Of considerable significance for determining optimum conditions of transporting spent fuel in gas-filled containers is research now being done in East Germany on determining the maximum temperature of fuel elements and selecting the gas coolant. Resolution of these problems will enable a sounder choice of technological conditions of packaging for the "dry" method of transporting spent fuel. Special notice should be taken of the importance of research on developing methods for thermal and strength calculations of shipment packaging, and development of a program for calculating nuclear safety of packaging.

Using the results of research done in the USSR, the TK-6 shipping cask has been developed and put into production for hauling spent fuel from nuclear electric plants with VVER-440 reactors [Ref. 10]. Development of the TK-6 specialized carrier has involved consideration of experience in developing and using shipping casks at the Novovoronezhsk and Rheinberg nuclear electric plants.

To determine the temperature conditions and shielding properties of the packaging, comprehensive thermal and radiological tests were done on the TK-6 specialized carrier. The tests confirmed that the heat-engineering and physical characteristics of the packaging basically conform to calculations.

The results of tests of the TK-6 shipping car have extended the range of applications of the container and established limiting conditions for shipping spent fuel in gas-filled and water-filled shipping casks.

Table 1 compares the characteristics of spent fuel from VVER-440 reactors that is shipped in the TK-6 container cars with those of fuel that is shipped in Rheinberg and Novovoronezhsk shipping casks.

Comparison of characteristics of spent fuel from VVER-440 reactors shipped in carriers of different types

Characteristic	TK-6 carrier		Container cars of Rheinberg and Novovoronezhsk plants
	gas-filled	water-filled	
Minimum holding time of spent fuel in pools at electric facility, years	3	3	3
Maximum burnup of fuel, GW-days per metric ton	24	40	15
Maximum rated energy release overall for spent fuel assemblies, kW	8	12	5
Maximum temperature of coolant in the cask, °C	175	85	124

Based on calculations and tests of the TK-6 packaging facility, the Competent Agency of the USSR has issued a certificate of authorization with identification symbol SU/041/B(U)F. The packaging facility loaded with spent fuel from the VVER-440 reactor in accordance with the requirements of this certificate has been classified as type B(U) packaging.

A special train has now been formed from four TK-6 container cars that enables simultaneous shipment of all spent fuel that is unloaded when reloading one VVER-440 reactor. This train has already made several runs. It is being used to haul spent fuel from the Kola Nuclear Electric Plant (USSR), the Kzoloduy plant in Bulgaria and the Bruno Leuschner facility in East Germany.

A second such train is soon to be formed up.

Development of transport facilities for organizing mass shipments of spent fuel from VVER-440 reactors has been one of the main tasks of the last five years. The development of nuclear power in CEMA member nations is bringing up new problems. Research must now be concentrated on developing effective transport facilities for shipping spent fuel from VVER-1000 reactors.

Work has started in the USSR on design of such a carrier. Characteristics of spent fuel of the VVER-1000 reactor: increased burnup, large dimensions of fuel assemblies, increased quantity of fission materials in the fuel assembly and so on, are responsible for the following differences of the packaging facility for shipping spent fuel of the VVER-1000 reactor as compared with the TK-6 shipping container:

--horizontal placement of the packaging on the railway carrier, and as a consequence the necessity of tilting the complex when loading and unloading fuel;

--separate neutron shielding in the packaging design;

--incorporation of neutron absorbers in the packaging facility.

Modifications of the packaging complex have been worked out with capacity for six, eight and ten fuel assemblies for transportation of spent fuel both with water and with gas coolant. Provisions are made for tipping the container with the crane at the nuclear electric plant (the container is shifted vertically) and a winch installed on the carrier frame (the carrier with container is displaced horizontally) without using a mobile or stationary tilter. The railway carrier for hauling the packaging complex is in principle the same as for the TK-6 container car. The container car is designed in conformity with international format 02-T with provisions for transfer from track with gage of 1520 mm to 1435 mm, enabling passage on railroads of all CEMA member nations.

In addition to developing transport facilities for hauling spent fuel from VVER-1000 reactors, another important problem to be solved during the current five-year plan is development of research on transporting spent fuel from fast reactors.

In addition it is necessary to complete research and work started in CEMA member nations during the past five years on developing methods of checking integrity of fuel elements to be transported, criteria for quality control of fuel assemblies depending on the extent of fuel element damage, development of standardized cans for shipping fuel assemblies with damaged fuel elements and working out technical specifications for transporting such fuel assemblies, research on determining permissible thermal loads and limiting temperature in the "dry" method of transportation. It is necessary to accelerate completion of work on structural materials for transport packaging, paying particular attention to research on choice of materials and development of reliable construction of neutron shielding.

It is also necessary to continue generalizing the experience of transporting spent fuel of nuclear electric plants--principles for further improvement in organizing shipments and engineering facilities.

Work should be continued on developing normative documentation and guidelines. Such work includes:

--refinement of technical specifications for transportation of spent fuel from VVER-440 reactors with consideration of accumulated shipping experience;

--development of technical specifications for transportation of spent fuel from VVER-440 reactors;

--development of methods and programs unified for CEMA member nations for claculating residual energy release of spent fuel from thermal and fast reactors, giving higher accuracy than methods now in use;

--development of unified rules for all CEMA member nations on ensuring radiation safety in transporting spent fuel to improve safety in shipment.

Of importance from the standpoint of shortening the time of turnaround of transport facilities, increasing economy and safety of spent fuel shipments is research on developing easily decontaminated polymer coatings for transport facilities, as well as developing designs of systems for sealing shipping casks that will limit the emissions of radioactive content into the environment.

Successful handling by CEMA member nations of the above-mentioned scientific and engineering aspects of the problem of transporting spent fuel will enable a still greater increase in the reliability of packaging designs and safety in shipments of spent nuclear fuel between nations. Experience in scientific and technical cooperation among CEMA member nations shows that these problems can be solved, and they must be solved.

Conclusions

1. World experience in transporting spent nuclear fuel of electric power facilities shows that type B packaging, designed and made in accordance with international rules for shipment of radioactive materials, ensures safety when using railroad, truck and waterway transport facilities. The increase in volumes of shipments, transition to transportation of spent fuel with higher and higher specific activity, with intense neutron radiation and large energy release, and the current trend toward stiffer requirements for packaging demand solution of more and more complicated technical problems.

2. Recently a worldwide trend has been observed toward the use of large shipping casks that will accommodate a considerable amount of spent fuel on each trip. The containers generally use combined gamma-neutron shielding; structural materials are stainless steel, low-alloy carbon steel, special grades of iron that have the necessary fracture strength, including resistance to brittle fracture at sub-zero operating temperatures.

3. Areas of research and development in CEMA member nations on problems of transporting spent fuel mainly coincide with the areas being studied throughout the world. However, the work being done in CEMA nations must be accelerated so that the results can be used in developing facilities for transporting spent fuel from the new generation of VVER-1000 reactors.

4. The Soviet Union, using the results of development and experience of other CEMA member nations, has solved the problem of making transport facilities for international shipments of spent fuel from VVER-440 reactors. Engineering problems have been successfully solved, and transportation of spent fuel from nuclear electric plants of CEMA member nations has been organized with the use of TK-6 container cars.

5. Future work on resolving problems of transporting spent fuel within the scope of CEMA should be aimed at:

--developing new structural materials, neutron shielding materials and stable easily decontaminated coatings,

--determining permissible thermal loads and limiting temperature of fuel elements in the dry method of shipment,

--developing designs for improved systems of sealing packaging facilities that will minimize the escape of contents from the cask,

--developing technical facilities and working out technical specifications for transportation of fuel with damaged fuel elements.

Discussion

J. Hesse: Has an analysis been made of the probability of a radiation accident based on the intensity of railroad and highway traffic, and on the basis of accident statistics of the private sector, and what are the projected consequences of a radiation accident?

V. Fisher: Statistics on transportation of spent fuel with type B containers do not include accidents with release of radioactive materials.

A. Shtolmer: How is safety evaluated when there is the possibility of a fall from a height of 19 m, as occurs in the standard project of the B-213 facility where the floor of the reactor room is about 19 m above the floor of the transport corridor?

V. Fisher: Casks are made for transporting spent fuel from the nuclear electric plant to the recycling plant, and meet all requirements of the IAEA. Operations inside the nuclear electric plant must be handled in such a way as to ensure complete safety.

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NONDESTRUCTIVE METHODS OF DETERMINING ISOTOPIC MAKEUP AND BURNUP OF SPENT VVER FUEL

Prague ISSLEDOVANIYA V OBLASTI PERERABOTKI OBLUCHENNOGO TOPLIVA I OBEZVREZHIVANIYA RADIOAKTIVNYKH OTKHODOV in Russian Vol 1, 1981 pp 114-131

[Article by B. A. Bibichev, N. M. Kazarinov. V. P. Mayorov and P. I. Fedotov]

[Text] This report examines the use of nondestructive methods of analysis (NDA) for determining burnup and composition in spent fuel elements in VVER cassettes. Results of using the gamma spectrometric method are considered. The authors note the outlook of the semiempirical method of determining burnup and isotopic composition of VVER fuel based on a combination of the data of gamma spectrometry and calculation. An examination is made of NDA based on registration of neutron self-radiation of spent fuel elements and cassettes. This method has a number of advantages over the gamma spectrometric method, and it is hoped that with further development it can be successfully used as a passive neutron method both in the thermal cycle and in the system of guarantees. In conclusion, the authors discuss prospects for developing and using nondestructive methods of analysis on spent fuel of nuclear electric plants.

1. Introduction

The selection of fuel charges of VVER reactors and prediction of fuel burnup in these reactors is completely based on computational methods [Ref. 1]. Experimental verification and correction of computational methods and programs is necessary for optimizing fuel cycles. This requires further improvement of nondestructive methods of determining burnup and content of fissile isotopes in VVER fuel elements and cassettes. Development of nondestructive methods is also needed for producing technical facilities for monitoring observance of IAEA guarantees at nuclear electric plants and for providing specifications on cassettes sent out for reprocessing.

Ref. 2 gives a survey of nondestructive methods of determining burnup and content of fissile isotopes in spent fuel of nuclear electric plants, with a discussion of the status of each method. In this paper we will examine

use of the method of gamma spectrometry of fission products and a neutron method based on registration of neutron self-radiation of fuel to determine burnup and content of fissile isotopes in spent VVER fuel.

2. Measurement of Fuel Burnup by Gamma Spectrometric Method

The method of measuring VVER fuel burnup by the concentrations of ^{137}Cs and ^{106}Ru is described in Ref. 3, 4. For several years this method has been used at Novovoronezhsk Nuclear Electric Plant for measuring fuel burnup in VVER-365 and VVER-44o cassettes. Measurements were based on selection of four VVER-365 cassettes and three VVER-440 cassettes with different enrichments and different histories of irradiation. Basic data on the history of irradiation of the cassettes are summarized in Table 1.

TABLE 1

Basic operational data of cassettes studied

Number of cassette	Initial ^{235}U enrichment, wt.%	Type of reactor	Number of runs	Total time of irradiation, eff. days
DR-3 No 80	3	VVER-365, second unit	1	311
RP-3 No 223	3	VVER-365, second unit	4	819
OI-3M-5 No 12	3	VVER-365, second unit	3	929
RI-2.4 No 03	2.4	VVER-365, second unit	4	1239
RP-3.3 No 71A	3.3	VVER-440, third unit	1	320
R-3.6 No 213	3.6	VVER-440, fourth unit	3	1032
R-3.6 No 216	3.6	VVER-440, fourth unit	3	1032

The measurements were made in the shielding chamber of Novovoronezhsk Nuclear Electric Plant. To eliminate the necessity of calculating coefficients of self-absorption of gamma radiation in the cassettes, all cassettes were broken down to individual fuel elements before measurements. In each cassette, 26 fuel elements out of 126 were studied. Several fuel elements from each row were chosen for the measurements to get information on the distribution of fission products over the cross section of the cassette. The gamma spectra of fission products were measured at ten points heightwise on these fuel elements (every 25 cm). The measurement facility was calibrated for efficiency by a standard fuel element in which the concentration of ^{137}Cs had been measured by destructive methods.

From the results of measurement of burnup in 26 fuel elements of each cassette, the average values of burnup over the cross section of the cassette were calculated at ten points over the height. The error of determining fuel burnup at a given point heightwise of the cassette was from 3 to 5% for different cassettes.

Fig. 1 and 2 show the measured and calculated distributions of fuel burnup with respect to height for VVER-365 and VVER-440 cassettes. The distributions were calculated by the BIPR-4 program [Ref. 1]. It can be seen from the curves

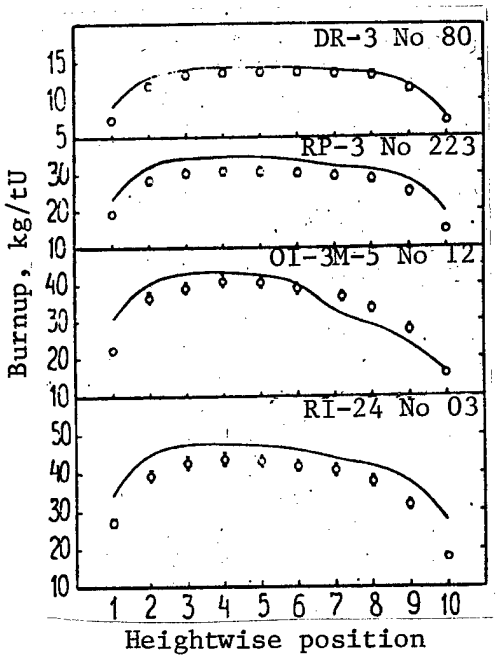


Fig. 1. Experimental and calculated distribution of burnup with respect to height of VVER-365 cassettes

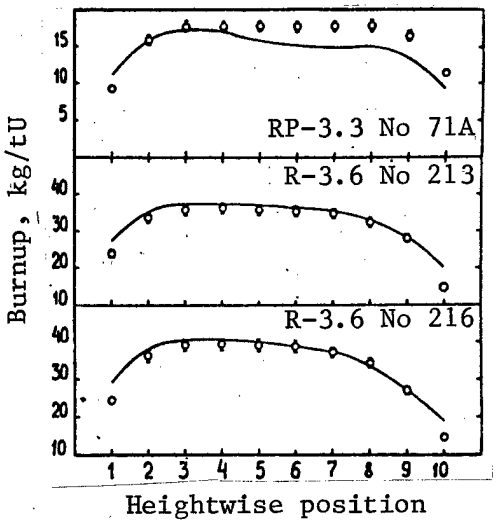


Fig. 2. Experimental and calculated distribution of burnup with respect to height of VVER-440 cassettes

TABLE 2

Calculated and measured values of average burnup of fuel in cassettes

Number of cassette	Calculated burnup, kg/tU	Measured burnup, kg/tU	Deviation of calculated burnup from measured value, %
DR-3 No 80	12.7	11.9 ± 0.6	+ 6.3
RP-3 No 223	30.2	26.7 ± 1.1	+11.5
OI-3M-5 No 12	34.1	33.1 ± 1.2	+ 3.0
RI-2.4 No 03	41.6	36.1 ± 1.5	+15.2
RP-3.3 No 71A	14.3	16.0 ± 0.6	-10.6
R-3.6 No 213	32.7	31.1 ± 1.0	+ 5.1
R-3.6 No 216	34.3	32.9 ± 1.0	+ 4.3

that on the ends of the cassettes the calculation consistently overstates the values of burnup. For example, for cassette RI-2.4 No 03, this discrepancy reaches 50% at the tenth point heightwise. The discrepancy between calculated and experimental values of burnup at the ends of the cassettes is apparently associated with the error of calculation of logarithmic derivatives of flux densities of delayed neutrons on the fuel-reflector boundary.

Table 2 summarizes calculated and experimental values of average fuel burnup in the cassettes. We can see from Table 2 that calculation gives overstated

burnups in all cassettes with the exception of RP-3.3 No 71A. This cassette was used in the first charge of the third unit; the charge was nonstandard, and calculateion of the fuel burnup in the cassette involved additional errors. The average deviation of calculated values from measured burnup is +7.6%.

It should be noted that there is as yet insufficient experimental material for the various conditions of exposure of cassettes in VVER-365 and VVER-440 reactors to enable correction of the computational programs. More experimental data need to be accumulated on the heightwise distribution of fuel burnup in cassettes.

3. Correction of Burnup and Activity Ratios of $^{137}Cs/^{134}Cs$

Determination of fuel burnup from the ratio of activities of $^{134}Cs/^{137}Cs$ (or from the ratio of concentrations) has a number of important advantages over the conventional method based on measurements of concentrations of fission products--burnup monitors [Ref. 5]. The principal advantage of this method is that it is not necessary to calibrate the measurement facility with respect to absolute efficiency, and the relative efficiency can be determined from the relative intensities of gamma lines of the fission products (^{134}Cs, ^{106}Ru and ^{154}Eu) [Ref. 6].

At the same time, theoretical and experimental studies of this correlation done in recent years for thermal reactor fuel [Ref. 7, 8] have shown how this correlation depends on the history of irradiation, fuel enrichment and neutron spectrum. Besides, when measuring the ratio of activities of $^{134}Cs/^{137}Cs$ in intact cassettes, additional errors may arise that are due to different distributions of ^{134}Cs and ^{137}Cs over the cross section of the cassette [Ref. 2].

Our paper will give experimental data on the correlation of burnup and the ratio of activities of $^{134}Cs/^{137}Cs$ for VVER-365 and VVER-440 cassettes and specimens of fuel elements. Twenty-two fuel element specimens were cut out of seven cassettes (see Table 1). These specimens were dissolved, and measurements were made on aliquot parts of the solutions to determine fuel burnup by the gamma spectrometric method with respect to ^{137}Cs with error of about 2%, and the concentration ratio of $^{134}Cs/^{137}Cs$ with error of 1.0-1.5%. The procedure for measuring burnup and concentration ratio $^{134}Cs/^{137}Cs$ in the cassettes is described in the preceding section. The error of measuring the concentration ratio $^{134}Cs/^{137}Cs$ at a given point with respect to height of the cassette was 2-4%.

The ratio of activities of $^{134}Cs/^{137}Cs$ was calculated from the following relation:

$$\eta = \frac{\lambda_{134} N_{134} C_{134}}{\lambda_{137} N_{137} C_{137}}, \qquad (1)$$

where η is the ratio of activities of $^{134}Cs/^{137}Cs$, λ_{134}, λ_{137} are the decay constants of ^{134}Cs and ^{137}Cs respectively, N_{134}, N_{137} are the concentrations of ^{134}Cs and ^{137}Cs at the given point with respect to height of the cassette or fuel element specimen at the date of reactor shutdown, C_{134}, C_{137} are coefficients that account for loss of ^{134}Cs and ^{137}Cs for the duration of exposure of the given cassette at the given point with resepct to height.

It should be noted that ^{134}Cs is not a direct product of fission, and is formed from ^{133}Cs by capture of mainly epithermal neutrons. Therefore coefficient C_{134} must depend not only on the duration of the irradiation run and the dynamics of change in neutron flux density, but also on the rigidity of the neutron spectrum. Ref. 8 describes a technique for calculating C_{134} and C_{137} for constant thermal neutron flux density in each cycle of irradiation. A similar procedure was also used in our paper to calculate C_{134} and C_{137}. The values of the thermal neutron flux densities and rigidities of neutron spectra in each irradiation run for a given cassette at ten points with respect to height were taken from calculations by the BIPR-4 and ROR-2 programs [Ref. 1].

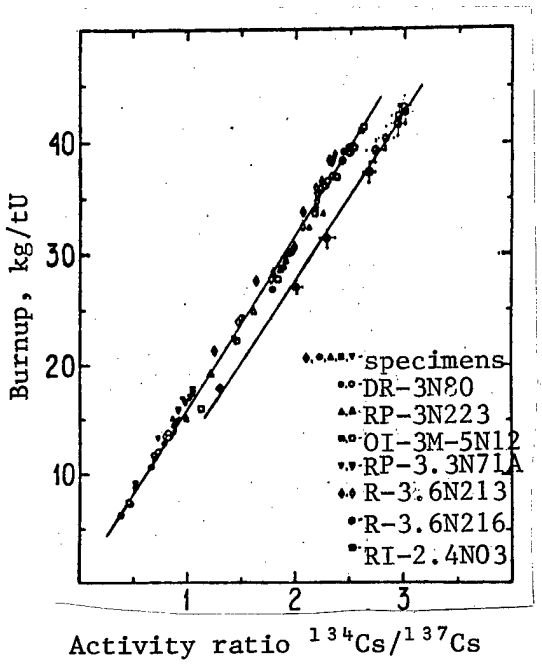


Fig. 3. Correlation of burnup and activity ratio of $^{134}Cs/^{137}Cs$

Fig. 3 shows measured values of burnup and activity ratio $^{134}Cs/^{137}Cs$ for 7 cassettes and 22 specimens. It can be seen from Fig. 3 that the experimental points for all cassettes and specimens lie on a common straight line passing approximately through the coordinate origin. An exception is cassette RI-2.4 No 03 for which the experimental points lie on a separate line. Table 3 summarizes the results of analysis of correlation lines $Y = ax + b$ [Ref. 9]. From this table we can see that the difference between parameters "a" and "b" for cassettes with enrichments of 3.0, 3.3 and 3.6% lies in the range of error in determining these quantities. Therefore for cassettes with such enrichments of fuel we can use a common correlation line. The standard deviations of experimental points for cassettes and specimens of fuel elements with such fuel enrichments are 4%. For cassette RI-2.4 No 03 this correlation is found to depend on fuel enrichment [Ref. 8]. At lower fuel enrichment (2.0, 1.6 and 0.72%) the correlation of burnup and concentration ratios $^{134}Cs/^{137}Cs$ were studied in Ref. 5 for VVER-70 fuel.

In measuring the activity ratio of $^{134}Cs/^{137}Cs$ in intact cassettes, the differing distributions of these fission products over the cross section of the cassettes may lead to additional errors of determining relative effectiveness for the 662 keV line of ^{137}Cs with respect to the relative intensities of ^{134}Cs lines. According to the estimate of Ref. 2, this error may amount to 16% for BWR cassettes.

To study this effect, the concentrations of ^{137}Cs and ^{134}Cs were measured in all 126 fuel elements at the fourth position with respect to height in cassettes R-3.6 No 213 and R-3.6 No 216. As an example, Fig. 4 shows the

TABLE 3

Results of correlation analysis of burnup and intensity ratio of $^{134}Cs/^{137}Cs$

Number of cassette	Number of experimental points	Parameter "a"	Parameter "b"	Correlation coefficient	Standard deviation, %
DR-3 No 80 RP-3 No 223 OI-3M-5 No 12 10 specimens	40	15.4(2)	0.44(16)	0.998	3.2*
RP-3.3 No 71A RP-3.6 No 213 RP-3.6 No 216 12 specimens	42	15.5(2)	1.0(2)	0.997	3.2
RI-2.4 No 03	10	15.0(3)	-2.1(8)	0.998	2.0
Weight of cassette and specimen with exception of RI-2.4 No 03	92	15.6(2)	0.54(16)	0.997	4.0

*The standard deviation is $\sqrt{\sum_{i=1}^{n} \Delta_i^2/(n-1)}$, where $\Delta_i = \frac{Y_{i(exp.)} - Y_{i(calc)}}{Y_{i(exp.)}}$

and n is the number of experimental points.

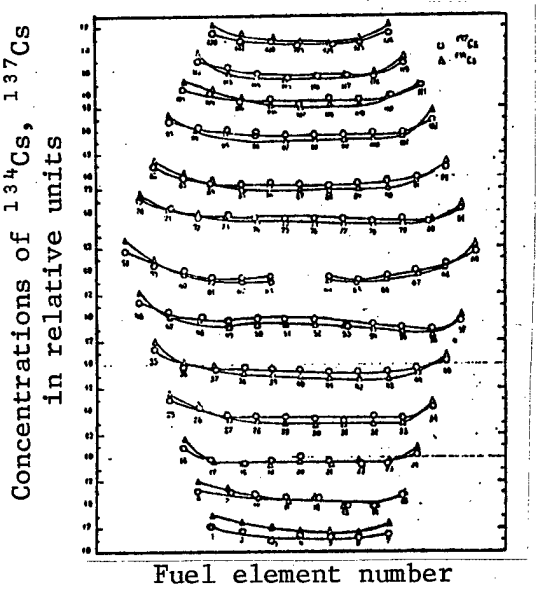


Fig. 4. Relative distribution of ^{134}Cs and ^{137}Cs concentration over cross section of Cassette R-3.6 No 213

relative distributions of these fission products over the cross section of cassette R-3.6 No 213 (values of concentrations of ^{134}Cs and ^{137}Cs in each fuel element were normalized to the average value of these quantities over the cross section of the cassette). The increase in concentration of ^{137}Cs and ^{134}Cs in the fuel elements of the outer row as compared with the central fuel elements is 20 and 25% respectively. The considerably smaller increase in ^{134}Cs concentration in fuel elements of the outer row in VVER cassettes as compared with the estimate of Ref. 2 is apparently due to softening of the neutron spectrum in these fuel elements associated with the water gaps between cassettes, which was not considered in Ref. 2. Analogous distribution of ^{134}Cs and ^{137}Cs is observed in cassette R-3.6 No 216.

Calculation of the coefficient of self-absorption by the program of Ref. 10 for the case of 20 and 25% excess of gamma

source activity in fuel elements of the outer row as compared with the center fuel elements gives a difference in the coefficient of self-absorption of less than 1.5% at an energy of 662 keV. Thus the difference in distribution of ^{137}Cs and ^{134}Cs over the cross section of VVER cassettes should not lead to an appreciable increase in the error of measuring the ratio of activities of these fission products in intact cassettes.

4. Experimental-Computational Method of Determining Burnup and Isotopic Composition of Fuel

A more general approach to the use of gamma spectrometric information for determining burnup and the content of fissile isotopes is realized in the experimental-computational method [Ref. 11, 12]. This method consists in solving equations of burnup for isotopes of uranium, plutonium and certain fission products in a two-group approximation for the neutron spectrum. The thermal neutron flux densities Φ_{th} and rigidities α of the neutron spectrum at a given height of the fuel element or cassette that are necessary for calculations are found from measurements of N_{137} and N_{134}. The possibility of determining Φ_{th} and α is based on the considerable difference between the ways that concentrations of ^{137}Cs and ^{134}Cs depend on these quantities [Ref. 11, 12].

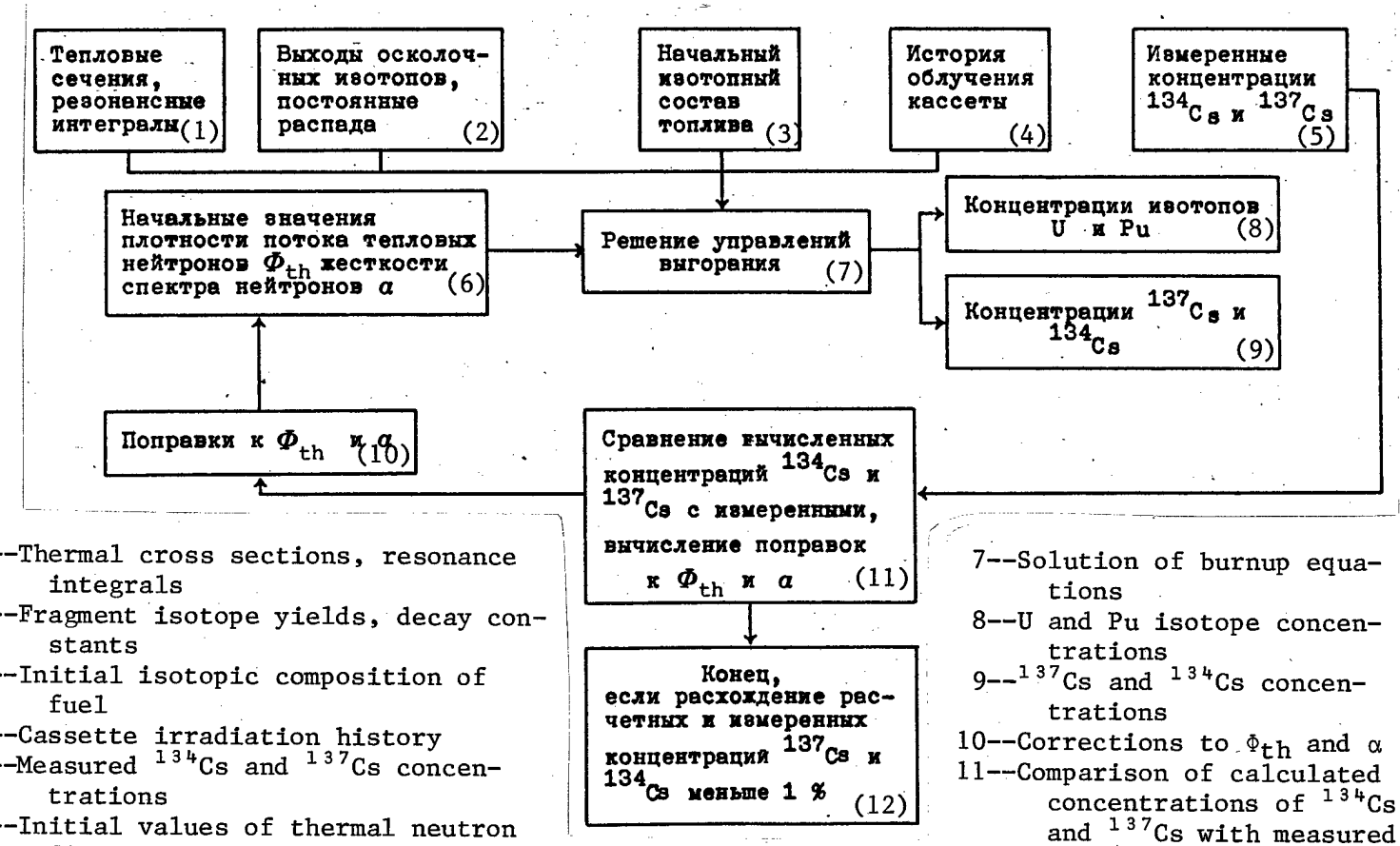
The HIC program has been developed on the basis of the experimental-computational method. Fig. 5 shows a flowchart of calculations by this program. The following nuclear data are used as input: thermal cross sections, resonant integrals, yields of fission products and their decay constants, and also specifications on initial fuel enrichment and the history of irradiation of the investigated fuel element or cassette (duration of each irradiation run and relative thermal power). Burnup equations are solved by numerical integration. The Φ_{th} and α are found by matching the calculated values of N_{137} and N_{134} to the measured values.

It should be noted that the accuracy of calculating the isotopic composition of the fuel depends both on the error of measuring the concentrations of ^{137}Cs and ^{134}Cs and on the selected values of thermal cross sections and resonance integrals. The values of these quantities calculated by the HIC program [Ref. 1] were taken as the basis in our calculations.

The error of this method was experimentally checked on fuel element specimens. To do this, concentrations of ^{134}Cs, ^{137}Cs, ^{235}U, ^{236}U, ^{238}U, ^{239}Pu, ^{240}Pu, ^{241}Pu and ^{242}Pu were measured by destructive methods with average error of 1-2% in 19 specimens of VVER-365 and VVER-440 fuel elements with initial fuel enrichment of 3.0, 3.3 and 3.6% and burnup from 6 to 40 kg/tU [Ref. 13]. The measured concentrations of uranium and plutonium isotopes in all specimens were also used to correct the four thermal cross sections: σ_f^{235}, σ_c^{235}, σ_c^{239} and σ_c^{241} and two resonance integrals: I_c^{240} and I_c^{133}. The method of correction is based on minimizing the sum of the squares of deviations of calculated and experimental concentrations of uranium and plutonium isotopes with respect to all specimens [Ref. 14].

Table 4 summarizes the deviations of calculated concentrations of uranium and plutonium isotopes in the 19 specimens of fuel elements from the

Fig. 5. Flowchart for calculating U and Pu isotope concentrations by HIC program



KEY: 1--Thermal cross sections, resonance integrals
2--Fragment isotope yields, decay constants
3--Initial isotopic composition of fuel
4--Cassette irradiation history
5--Measured ^{134}Cs and ^{137}Cs concentrations
6--Initial values of thermal neutron flux density Φ_{th} and neutron spectrum rigidity α
7--Solution of burnup equations
8--U and Pu isotope concentrations
9--^{137}Cs and ^{134}Cs concentrations
10--Corrections to Φ_{th} and α
11--Comparison of calculated concentrations of ^{134}Cs and ^{137}Cs with measured values, calculation of corrections to Φ_{th} and α
12--End, if discrepancy between calculated and measured concentrations of ^{137}Cs and ^{134}Cs is less than 1%

TABLE 4

Deviations of calculated values of U and Pu concentrations (calculation with respect to ^{137}Cs and ^{134}Cs) from the experimental values for fuel element specimens

Specimen number	Burnup kg/tU	Deviation of calculated values from experimental, %						
		^{235}U	^{236}U	^{239}Pu	^{240}Pu	^{241}Pu	^{242}Pu	Σ Pu
1	6,2	1,4	-	1,0	-6,0	-1,6	-7,0	0,3
2	10,7	-0,4	-5,4	-0,9	-6,7	-4,5	1,9	-1,9
3	12,9	0,2	9,4	-0,5	-4,1	-4,1	-2,9	-1,3
4	14,6	0,0	2,7	-2,7	-2,1	-3,9	-2,8	-2,5
5	15,0	3,8	-0,6	1,7	7,0	9,3	4,2	3,2
6	28,5	6,3	-4,0	4,3	9,4	12,5	7,3	6,7
7	33,7	-0,1	-0,5	-6,2	11,0	8,1	7,4	0,7
8	8,7	3,8	-	1,8	1,3	7,0	-	1,8
9	11,7	4,4	-2,2	5,9	2,8	9,2	3,6	5,7
10	13,4	1,6	0,1	-4,6	0,8	3,7	-1,6	-3,2
11	14,0	2,1	-6,8	3,7	-0,6	5,0	-	3,3
12	14,4	-1,3	-1,6	1,2	1,2	9,0	-	2,0
13	14,9	-2,5	2,4	-5,8	-2,0	0,1	0,6	-4,7
14	17,0	-0,2	1,1	3,6	5,3	-	-	1,1
15	21,4	1,0	3,6	-4,5	2,3	-12,6	-6,1	-4,1
16	27,8	1,0	1,0	5,4	2,2	0,7	2,3	4,1
17	38,5	0,1	0,0	-5,3	-2,4	-6,4	-6,7	-4,9
18	38,6	-4,2	0,9	-0,2	0,7	1,0	2,7	0,3
19	39,0	1,6	-1,7	1,3	0,0	1,4	0,4	1,0
Average deviations*, %		2,6	3,5	3,8	5,0	6,9	4,4	3,3

*Average deviations are calculated by the formula $\bar{\delta} = \sqrt{\sum_i \delta_i^2 / N}$, where δ_i is the deviation in the i-th specimen, N is the number of specimens.

experimental values in percent. The calculated data were obtained by the HIC program from measured concentrations of ^{134}Cs and ^{137}Cs. It can be seen from Table 4 that the average deviation of calculated and experimental concentrations of ^{235}U with respect to all specimens is 2.6%, ^{236}U--3.5%, ^{239}Pu--3.8%, ^{240}Pu--5.0%, ^{241}Pu--6.9% and ^{242}Pu--4.4%.

Such a method of determining uranium and plutonium isotope concentrations requires measurements of the absolute concentrations of ^{137}Cs and ^{134}Cs. In the experimental-computational method, the ratio N_{134}/N_{137} can also be used as the experimental information. In this case, it is advisable to fix the value of α and to find Φ_{th} from ratio N_{134}/N_{137}. The results of such a version

of calculating the concentrations of uranium and plutonium isotopes are summarized in Table 5. In these calculations, the average value of α obtained from the preceding version of calculation with respect to the measured values of N_{137} and N_{134} was used.

TABLE 5

Deviations of calculated burnups and concentrations of U and Pu (calculation with respect to concentration ratio of $^{134}Cs/^{137}Cs$) from experimental values for fuel element specimens

Specimen number	Deviation of calculated values from experimental, %							
	Burnup	^{235}U	^{236}U	^{239}Pu	^{240}Pu	^{241}Pu	^{242}Pu	Σ Pu
1	4,6	0,2	-	-0,7	-1,3	1,3	0,8	-0,7
2	6,8	-3,6	-1,0	-5,2	0,0	-2,3	13,8	-4,2
3	1,8	-0,9	5,3	-2,0	-2,3	-3,7	0,0	-2,1
4	-1,4	0,9	1,7	-1,5	-3,4	-4,1	-4,9	-2,0
5	-12,3	13,6	-9,4	13,6	-6,4	7,0	-15,5	9,2
6	-0,1	6,6	-4,1	4,6	9,3	12,6	7,1	6,8
7	-4,6	14,5	-3,0	5,0	-7,6	13,6	-0,2	6,5
8	-6,8	6,2	-	5,0	-6,1	2,7	-	3,3
9	3,7	2,6	0,5	3,2	7,0	10,6	9,9	4,2
10	-7,9	6,2	-5,9	1,6	-7,6	1,4	-13,8	0,2
11	3,9	-0,3	-4,1	0,3	3,7	5,9	-	1,2
12	3,2	-3,3	0,7	-1,5	4,7	9,7	-	0,5
13	1,5	-3,4	3,5	-7,0	-0,4	0,3	2,9	-5,4
14	-5,3	4,0	-2,8	9,2	-3,9	-	-	6,8
15	-8,8	9,8	-3,0	6,8	-6,3	-11,8	-17,8	2,1
16	0,8	-0,1	1,5	4,0	2,9	0,3	3,5	3,3
17	-0,9	2,3	-0,4	-3,4	-2,8	-5,5	-7,8	-3,8
18	0,4	-3,5	0,7	0,5	0,5	1,3	2,2	0,7
19	0,4	0,4	-1,4	0,2	0,2	0,9	1,0	0,3
Average deviations, %	5,1	6,0	3,7	5,2	4,9	6,9	8,9	4,2

We can see from Table 5 that deviations of the calculated and experimental values of concentrations of uranium and plutonium isotopes are somewhat larger than in the version of calculation with respect to N_{137} and N_{134}. This is due to the effect of a change in neutron spectrum from specimen to specimen, which was not taken into consideration in calculation with respect to ratio N_{134}/N_{137}. The average deviation of calculated and experimental concentrations of ^{239}Pu and ^{235}U with respect to all specimens is 4 and 6% respectively, which is comparable with the error of the preceding version of calculation with respect to N_{137} and N_{134}.

An advantage of the experimental-computational method over the conventional method of determining fuel burnup from the concentrations of ^{137}Cs and ^{106}Ru

or from the concentration ratio $^{134}Cs/^{137}Cs$ is that this method enables determination of both burnup and content of all uranium and plutonium isotopes. The experimental-computational method can be recommended for determining burnup and content of fissile isotopes in VVER reactor cassettes. The gamma spectrometric method is simplest for measuring the ratio N_{134}/N_{137}. In this case, the error of determining the content of ^{235}U and ^{239}Pu in cassettes may be about 5% if ratio N_{134}/N_{137} is measured with error of no more than 2%.

5. Correlation of Burnup and Neutron Flux

The method of determining burnup based on recording neutron self-radiation of the fuel (neutrons of spontaneous emission and neutrons of the (α,n) reaction on ^{18}O) is attractive for the following reasons:

1. high neutron penetrance considerably simplifies accounting for the effect of absorption as neutrons pass through the cassette material;

2. simplicity and reliability of equipment used for neutron registration;

3. neutron measurements can be made immediately after extracting the cassette from the reactor.

We have described the equipment for measuring neutron self-radiation of spent fuel in Ref. 15, 16. In our current work we have studied the correlation between burnup and neutron flux for VVER-365 and VVER-440 reactor fuel. We have studied cassettes DR-3 No 80, OI-3M-5 No 12, R-3.6 No 213 and RI-2.4 No 03. Cassette cooling times were 1384, 990, 615 and 1238 days respectively. In order to eliminate the influence of neutron multiplication and absorption in intact cassettes, the cassettes were broken down into fuel elements. The neutron fluxes were measured at ten points heightwise for 20 fuel elements from each cassette. The fuel burnup in these fuel elements was measured by the gamma spectrometric method with respect to concentrations of ^{137}Cs and ^{106}Ru. From the results of measurement of the neutron flux N and fuel burnup W in the 20 fuel elements from each cassette, the average values of N and W were calculated over the cross section of the cassette at each heightwise point.

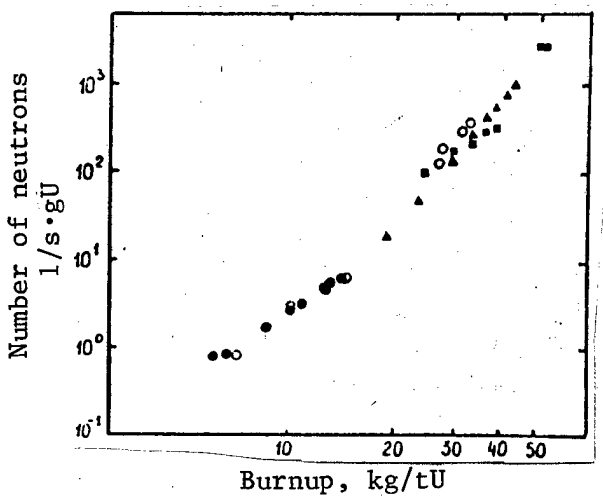


Fig. 6. Neutron flux from spent fuel as a function of burnup:
●--cassette DR-3 No 80
▲-- " OI-3M-5 No 12
■-- " R-3.6 No 213
■-- " RI-2.4 No 03
○--data of Ref. 13

TABLE 6

Relative contribution of various fissile isotopes to total neutron radiation of spent VVER reactor fuel

Burnup GW-days per metric ton	Cooling time, yr	Total neutron content n/s·kgU	^{238}U	^{238}Pu	^{239}Pu	^{240}Pu	^{242}Pu	^{241}Am	^{242}Cm	^{244}Cm
14,47	0	1,9+4	5,5-4	1,3-2	7,2-3	5,0-2	5,3-3	1,5-2	7,0-1	2,4-1
	1	9,3+3	1,1-3	2,8-2	1,5-2	1,0-1	1,1-2	3,9-2	2,9-1	5,0-1
	2	7,0+3	1,5-3	3,8-2	2,0-2	1,4-1	1,5-2	6,0-2	8,1-2	6,4-1
	3	6,4+3	1,6-3	4,1-2	2,2-2	1,5-1	1,6-2	7,4-2	1,8-2	6,7-1
	4	6,2+3	1,7-3	4,2-2	2,2-2	1,5-1	1,6-2	8,5-2	4,0-3	6,6-1
	5	6,1+3	1,7-3	4,2-2	2,3-2	1,6-1	1,7-2	9,5-2	8,7-4	6,5-1
	6	6,0+3	1,8-3	4,3-2	2,3-2	1,6-1	1,7-2	1,0-1	1,8-4	6,4-1
	7	6,0+3	1,8-3	4,25-2	2,3-2	1,6-1	1,7-2	1,3-1	3,9-5	6,1-1
32,13	0	8,9+5	1,2-5	2,4-3	2,3-4	1,2-3	1,2-3	1,2-3	5,5-1	4,3-1
	1	4,8+5	2,2-5	4,5-3	4,2-4	2,2-3	2,7-3	2,7-3	2,1-1	7,6-1
	2	3,8+5	2,8-5	5,6-3	5,3-4	2,7-3	2,7-3	4,0-3	5,7-2	9,2-1
	3	3,5+5	3,0-5	6,1-3	5,7-4	3,0-3	3,0-3	4,9-3	1,3-2	9,6-1
	4	3,4+5	3,2-5	6,3-3	6,0-4	3,1-3	3,1-3	5,7-3	2,8-3	9,7-1
	5	3,2+5	3,3-5	6,5-3	6,2-4	3,2-3	3,2-3	6,5-3	6,3-4	9,7-1
	6	3,1+5	3,4-5	6,7-3	6,5-4	3,4-3	3,4-3	7,3-3	1,3-4	9,7-1
	7	3,0+5	3,6-5	3,9-3	6,7-4	3,5-3	3,5-3	9,7-3	3,0-5	9,7-1

Fig. 6 shows the relation between N and W for the four investigated cassettes. The same figure shows the values of N and W for eight specimens of fuel elements cut from cassettes DR-3 No 80 and RP-3 No 223. The burnup in these specimens, and also the content of isotopes U, Pu, Am and Cm were measured by destructive methods [Ref. 13]. According to the data of Ref. 13, a calculation was done on the relative contribution and total flux of neutrons caused by different isotopes at different burnups and different cassette cooling times. By way of illustration, Table 6 gives the results of this calculation for two values of burnup.

The experimental points given on Fig. 6 are difficult to approximate by a power function with fixed exponent over the entire investigated burnup range. This is because the isotopes of Pu and Am make a considerable contribution to the total number of neutrons at fuel burnups of less than 15 kg/tU, as can be seen from Table 6. Accumulation of these isotopes as a function of burnup differs from the accumulation of ^{244}Cm, which almost entirely determines the neutron flux at burnup greater than 20 kg/tU and cooling time longer than three years. If the amount of ^{244}Cm in the fuel elements, and consequently the associated neutron flux, is plotted as a function of burnup, the entire set of points on Fig. 6 can be approximated by a power law with respect to MNK [expansion not given] (Fig. 7)

$$N_{244} = 4.8 \cdot 10^{-12} \, W^{4.3}. \qquad (2)$$

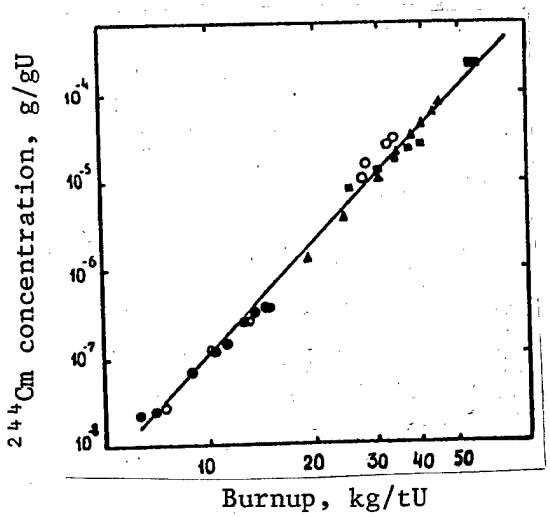


Fig. 7. Content of Cm in spent fuel as a function of burnup:
●--cassette DR-3 No 80
▲-- " OI-3M-5 No 12
■-- " R-3.6 No 213
▬-- " RI-2.4 No 03
○--data of Ref. 13

A similar relation was found by Phillips et al. [Ref. 17] for PWR cassettes. Maximum deviation of experimental points from relation (2) is 15%. It should be noted that experimental points referring only to the fuel elements of one cassette are also approximated by power law (2), but with a somewhat different exponent. And the deviation of experimental points from the resultant power law is less than for (2). As an illustration, Fig. 8 shows the ^{244}Cm concentration found by the method described above as a function of burnup for fuel elements of cassette DR-3 No 80. This fact can apparently be attributed to the difference in initial enrichment of the investigated cassettes and the difference in the neutron spectrum averaged over the irradiation run of these cassettes.

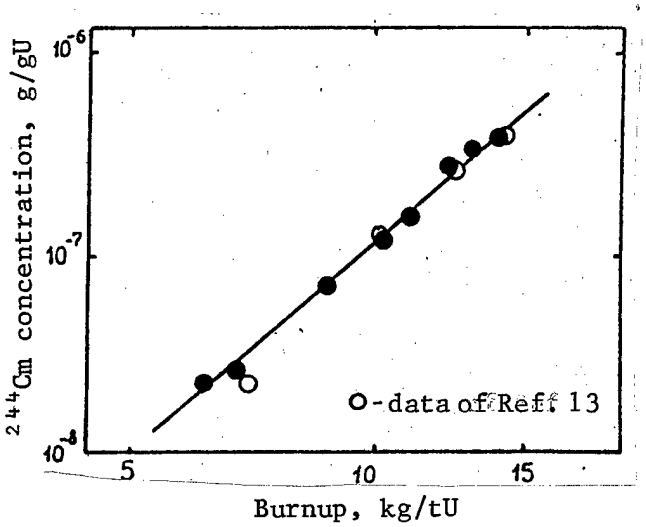


Fig. 8. Content of ^{244}Cm as a function of burnup for cassette DR-3 No 80

The observed correlation between the neutron flux and burnup may serve as a basis for developing a facility for express estimation of fuel burnup in cassettes at nuclear electric plants and recycling plants, and also for identification of spent fuel cassettes in the system of guarantees.

Discussion

F. Nietzsche: Did you also determine the energy spectrum of neutrons emitted from nuclear fuel?

V. I. Orlov: The neutron flux of fuel elements is due to neutrons of spontaneous emission of Cm-244 and neutrons of the reaction (α, n). Most of the neutron flux is due to fission neutrons.

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AUTOMATION OF LABORATORY EXTRACTIVE FACILITY INTENDED FOR RESEARCH IN AREA OF PROCESSING SPENT NUCLEAR FUEL

Prague ISSLEDOVANIYA V OBLASTI PERERABOTKI OBLUCHENNOGO TOPLIVA I OBEZVREZHIVANIYA RADIOAKTIVNYKH OTKHODOV in Russian Vol 1, 1981 pp 290-305

[Article by Ye. A. Vznuzdayev, B. Ya. Galkin, F. E. Gofman, V. S. Zhuntov, V. N. Inzhevatov, A. N. Kononov, R. I. Lyubtsev, V. I. Orlov, V. A. Popov, V. N. Romanovskiy, V. I. Ryazantsev, V. S. Samsonenkov and K. I. Shirshov]

[Text] This report describes an automated stand for solving the problem of optimum control of the extractive technological process of reprocessing spent nuclear fuel by means of an automated system of monitoring and control based on using computer facilities.

The problem of economic and safe reprocessing of spent nuclear fuel of nuclear electric plants with high burnup (25,000-40,000 MW-days per metric ton of uranium) places before technologists a number of problems whose solution requires development of pilot and consolidated laboratory facilities. Extraction is the basis of technological schemes for fuel regeneration, and in this connection the above-mentioned facilities are designed primarily for research and development of extraction technology.

In working with experimental facilities, the researcher needs analytical information on the makeup of solutions: concentrations of elements and various radionuclides, acidity, ratios of valence forms of elements and so on, as well as data on the state of such parameters of the process as expenditure of reagents, level of phases, temperature, pressure, condition of shutoff and regulating equipment. The efficacy of the experiment is directly dependent on the immediacy of this information. This goal is achieved by including on-line remote sensors in facilities, and speeding up laboratory analysis with sample-taking.

A current problem is optimum control of the technological process, which must be solved by an automated system of monitoring and control based on using computer facilities. The selection of monitoring parameters (points) on the technological scheme takes on fundamental importance for ensuring optimum control of the process.

To solve these problems, an automated stand was built in the hot chambers of the Radium Institute imeni V. G. Khlopin based on one of the experimental installations intended for research on extractive fuel reprocessing [Ref. 1, 2]. Fig. 1 shows a block diagram of the facility.

The technological equipment of the extraction facility of the stand is designed in such a way that no demounting is involved in realizing any extraction cycle of the process of regenerating nuclear fuel when using either a light or a heavy diluent.

One of the possible versions of the technological part of the stand--a diagram of the first combined extraction cycle of VVER fuel reprocessing--is shown in Fig. 2.

The major equipment of the facility -- mixer-settlers with pulsation intermixing and transportation of phases -- are assembled into modules with 14-22 stages in each.

The operation of combined extraction is done on ten stages of the first module with rinsing of the extractant out of the refinate with pure diluent (solvent) on two stages. The next eight or ten stages involve operations of acid rinses for removing zirconium, ruthenium and so forth from the combined extract.

The second extraction module is intended for re-extraction of plutonium and neptunium; the last seven stages are used for the operation of extraction of these elements. On five stages, uranium is washed out of the plutonium and neptunium re-extractate with acidulation by the return extractant. The mechanically entrained extractant is washed out of the plutonium and neptunium re-extractate with pure diluent on two stages.

The third module is a module for re-extraction of uranium on the first ten stages. On one stage, fission products are removed from the diluent by soda purification, and intermediate regeneration of the return extractant for reuse in the cycle is done by soda-alkaline treatment on the next seven stages.

The process diagram on Fig. 2 reflects the interrelation between material flows and equipment. From the standpoint of control, and consequently selection of points and methods for monitoring, it is advisable to consider the interrelation between variables [Ref. 3] that characterize the technological process. A decisive factor in such a consideration is formulation of control criteria--the criterion functionals of individual technological operations or aggregates of operations. It is advisable to divide all variables requiring monitoring into the following groups:

--vector of controlling variables $(\vec{u})$; regulating equipment acts directly on these variables in the course of the technological process;

--vector of perturbing variables $(\vec{x})$; these are variables that influence the course of the technological process, but are not subject to direct action;

--vector of output variables $(\vec{y})$; these variables characterize the state of the technological process, and are used to evaluate the quality of control.

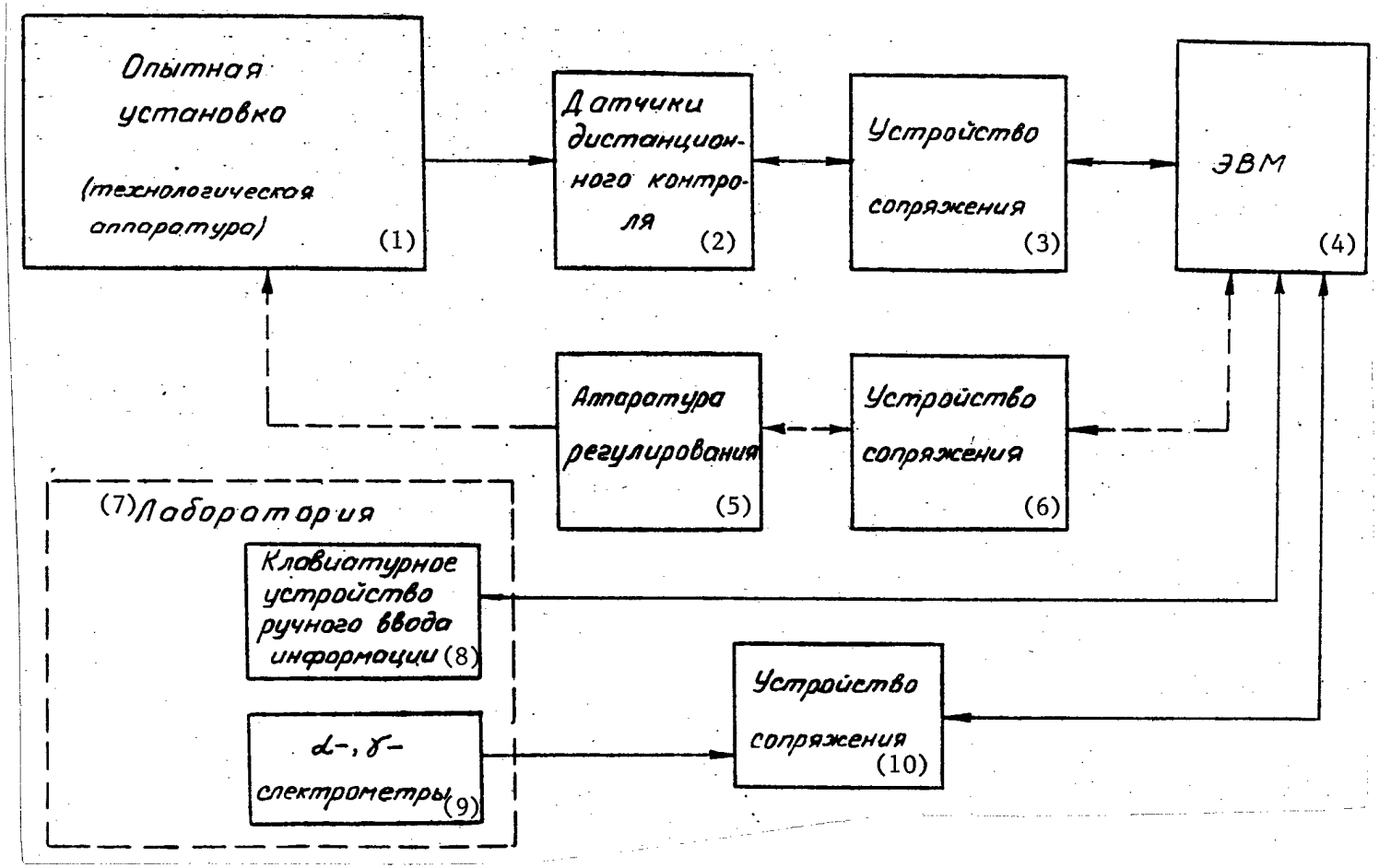

Fig. 1. Generalized block diagram of automated stand

KEY: 1--Experimental facility (technological stand
2--Remote monitoring sensors
3--Interfacing unit
4--Computer
5--Regulating equipment
6--Interfacing unit
7--Laboratory
8--Manual keyboarding input
9--Alpha and gamma spectrometers
10--Interfacing unit

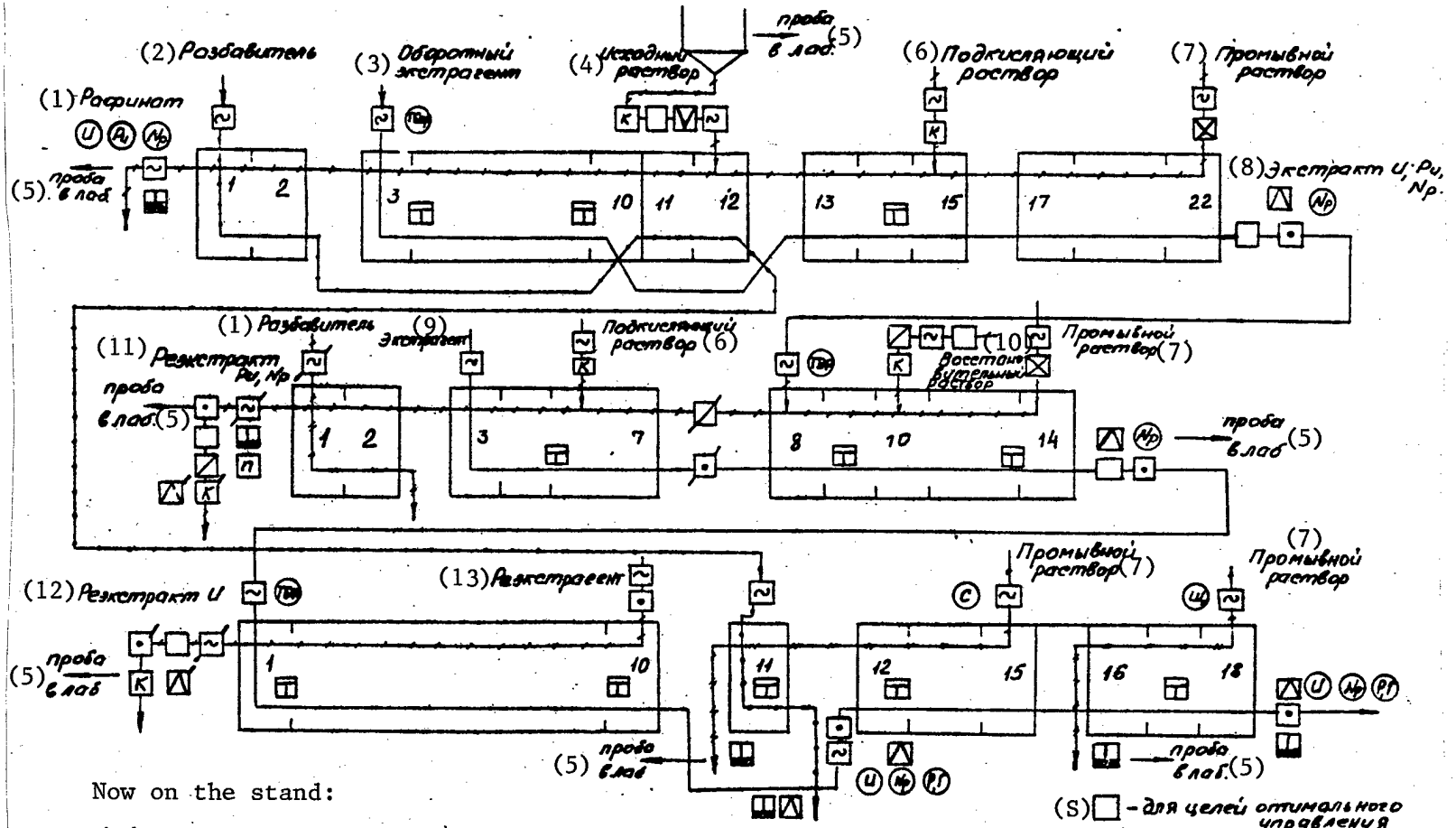

Fig. 2. Flowchart of the stand

Key to Fig. 2:

1--Refinate
2--Diluent
3--Return extractant
4--Initial solution
5--Sample to lab
6--Acidulating solution
7--Rinsing solution
8--U, Pu, Np extract
9--Extractant
10--Reducing solution
11--Pu, Np re-extractant
12--U re-extract
13--Re-extractant

A--Flowmeter
B--Level meter
C--Thermocouple
D--Gamma absorption meter
E--Alpha radiometer
F--Densimeter
G--Spectrophotometer
H--Conductometer
I--pH meter
J--EDR measurement
K--neutron flux measurement
L--Neptunium concentration meter
M--TBP concentration meter
N--Na_2CO_3 concentration meter
O--Concentration meter for products of radiolysis and hydrolysis
P--Uranium concentration meter
Q--Plutonium concentration meter
R--NaOH concentration meter
S--For optimum control purposes
T--Monitoring of limitations
U--Supplementary monitoring

The given extraction cycle can be represented by a sequence of subsystems, each of which has a certain functional designation (Fig. 3). External perturbations on the input of each subsystem come from the preceding subsystem or from the outer ambient, which is understood to mean the aggregate of technological operations, measuring devices and other equipment not directly included in the technological scheme.

The subsystem "U, Np, Pu Extraction" unifies operations of extraction proper and rinses. The functional purpose of the subsystem consists in extraction of valuable components and the removal of fission-fragment elements from these components. Requirements of extraction and purification are contradictory, and therefore the product of the degree of extraction of valuable components multiplied by the coefficient of purification with respect to fission-fragment elements can be taken as a criterion of control based on the principle of the valid compromise [Ref. 4]. The sense of optimum control consists in maximizing this criterion on the set of permissible controlling variables for any combination of perturbing variables. The controlling variables ($\vec{u}$) are the flowrates of the corresponding reagents: return extractant, initial, acidulating and rinsing solutions. The perturbing variables are the compositions of these reagents: uranium, plutonium and neptunium concentrations, gamma exposure dose rate of fission-fragment components, concentrations of nitric acid in initial, acidulating and rinsing solutions, and of TBP in the return extractant. The output variables are the characteristics of the uranium, neptunium and plutonium extract and refinate: concentrations of uranium, neptunium and plutonium in the extract and refinate, gamma exposure dose rate of fission-fragment elements in the extract, TBP concentration in the extract.

The subsystem "Np, Pu Re-extraction" unifies operations of reduction, acidulation, and also aqueous and organic rinses. The functional designation of the subsystem is the separation of neptunium, plutonium and uranium. The control criterion is the degree of separation of neptunium, plutonium and uranium, which is maximized on the set of permissible values of controlling

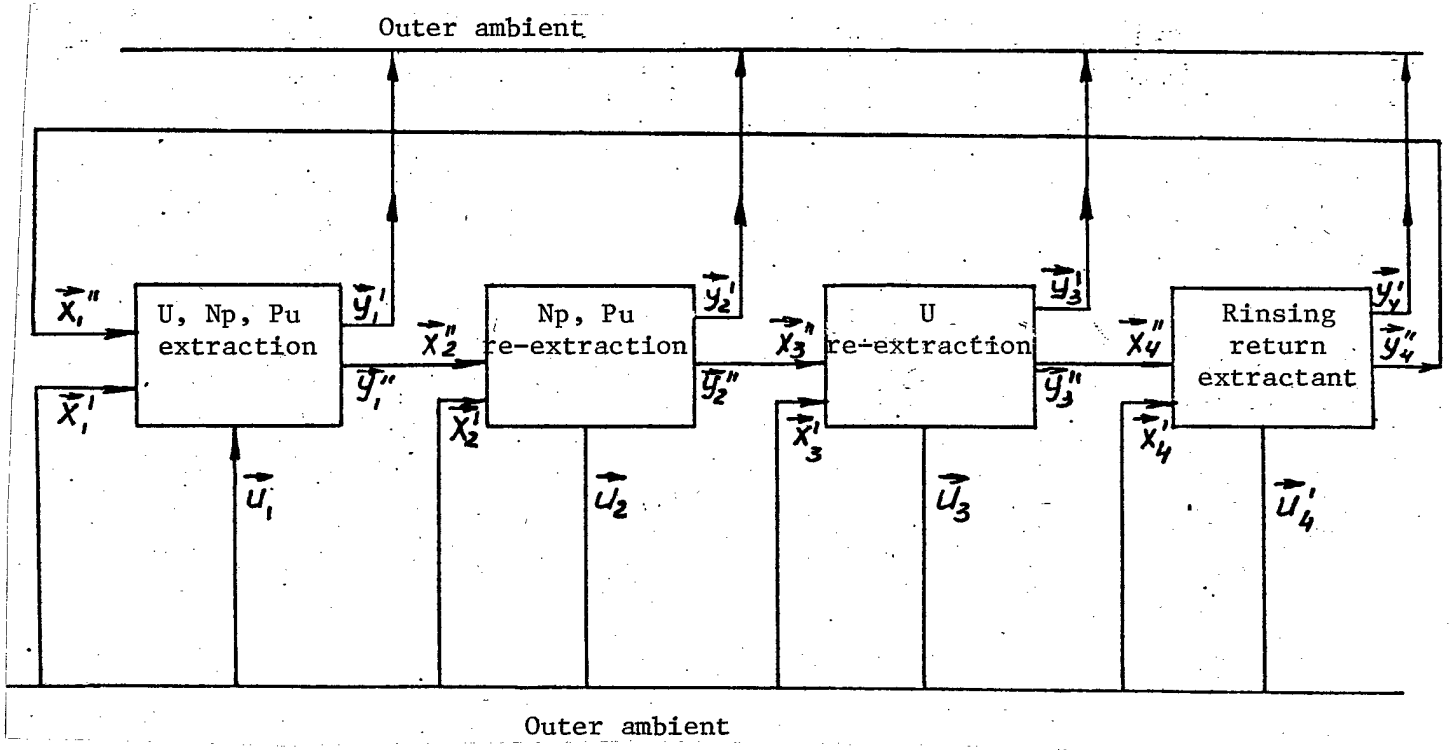

Fig. 3. Block diagram of first combined extraction cycle

variables--flowrates of U, Pu, Np extract, flowrates of acidulating, reducing and rinsing solutions. The perturbing variables are the gamma exposure dose rate from fission-fragment elements, concentrations of neptunium, plutonium, uranium and TBP in the U, Pu, Np extract, concentrations of nitric acid in the reducing, acidulating and rinsing solutions, total uranium concentration and concentration of tetravalent uranium in the reducing solution. The output variables are concentration of uranium and neptunium in the uranium extract, exposure dose rate in the uranium re-extract, exposure dose rate in the U, Pu, Np re-extract.

The subsystem "U Re-extraction" consists of only the operation of uranium re-extraction, and its functional designation is isolation of uranium. The control criterion is the coefficient of uranium concentration maximized on the set of controlling variables--flowrates of uranium extract and re-extractant. Perturbing variables are gamma exposure dose rate of fission-fragment elements, concentrations of uranium, plutonium and TBP in the uranium extract, concentration of nitric acid in the re-extractant. The output variables are concentrations of uranium and plutonium, gamma exposure dose rate of fission-fragment elements in uranium re-extract and return extractant, concentration of products of radiolysis and hydrolysis in the return extractant.

The subsystem "Rinsing Return Extractant" includes operations of soda and alkaline rinses of the return extractant. The functional designation of the subsystem is removal of impurities (uranium, plutonium, fission-fragment elements, products of radiolysis and hydrolysis) from the return extractant. The control criterion is the degree of extraction of the return extractant. The controlling variables are the flowrates of return extractant and rinsing solutions. The perturbing variables are the concentrations of impurities and TBP in the return extractant before the rinses, of soda and alkali in the rinsing solutions and return extractant after rinsing, concentration of TBP in the return extractant after the rinses.

In addition to the control criteria, the limitations on technological variables must be indicated for solving optimization problems for each subsystem. There may be three types of limitations:

--Limitations on controlling variables (reagent flowrates) are determined by design features and the required productivity of the technological complex.

--Limitations on output variables, e. g. for the subsystems of U re-extraction and Np, Pu re-extraction, are determined by the given degree of removal of extraneous elements (fission products in the Np, Pu re-extract; fission products and plutonium in the uranium re-extract).

--Limitations of the third type are determined by considerations of nuclear safety (these are insignificant for the given extraction facility).

It should be noted that optimization problems for the above-mentioned subsystems have been formulated without consideration of the mutual influence betseen the subsystems. As a matter of fact, all subsystems have a common material flow (organic phase). Therefore if the optimization problem is solved for one

of the subsystems, the number of controlling variables for the other subsystems is reduced by one. Selection of the subsystem for which the optimization problem is solved first is determined by the priority of the subsystems, and is accomplished by ranking the control criteria. If this canot be done, stage-by-stage optimization is used with consideration of future consequences.

This analysis essentially defines the optimum version of the system for monitoring the technological scheme of the first extraction cycle shown on Fig. 2, and thereby is the validation of the arrangement. The monitoring system for organizing optimum control of the process must in principle ensure measurements of all the above-mentioned variables. In addition, the controlling variables must be regulated.

Various methods of nuclear physics and physical chemistry are used for monitoring analytical parameters on the stand:

--gamma-absorptiometric,
--conductometric,
--densimetric,
--the method of neutron measurements,
--alpha-radiometric,
--the method of gamma exposure dose rate measurements,
--spectrophotometric,
--the method of pH measurements,
--the method of oxidation-reduction indicators.

1. Gamma-absorptiometric sensors are intended for measuring the concentration of uranium in a range of 5-300 g/ℓ. The sensors use sodium iodide scintillation detectors 0.3 mm thick--optimum for recording gamma radiation of the americium-241 intensifier source against the background of gamma radiation of the fission-fragment radionuclides. Sensors with gas-discharge counters are also used. The sensors are arranged to compensate for the gamma background produced by gamma self-radiation of the solutions to be monitored. The sensors with scintillation detectors can be used for measuring solutions with a high level of gamma radiation, including the initial solutions.

2. Conductometric sensors are used for measuring the electrical conductivity of aqueous solutions to determine the nitric acid concentration. Measurement range 0-4 moles/ℓ and 4-8 moles/ℓ. Low-frequency sensors are intended for monitoring the acidity of salt-free solutions (e. g. rinsing solutions). High-frequency sensors are mainly for monitoring products with a high concentration of uranium, plutonium and neptunium.

A conductometer developed and made by the Institute of Nuclear Physics of Poland is also used [Ref. 5].

3. A vibration densimeter, also developed at the Institute of Nuclear Physics of Poland [Ref. 6] is used for monitoring the density of solutions. This instrument is used to determine uranium concentration.

4. A neutron sensor that measures the neutron flux from reaction (α, n) is provided for determining plutonium in its concentrates.

5. The method of measuring exposure dose rate (EDR) is based on using a scintillation dosimeter and is intended for determining the total exposure dose rate produced by gamma radiation of process products after the main mass of fission-fragment elements has been introduced into the refinate.

6. Alpha-radiometric sensors are used to determine the specific activity of a solution in a range from minimum values of $3.7 \cdot 10^5$ Bk/ℓ to $3.7 \cdot 10^{12}$ Bk/ℓ. The sensor uses immersion semiconductor detectors based on n- and p-silicon.

7. Spectroabsorptiometer sensors are intended for determining the concentration of valence forms of elements by measuring the relative transmission of light by a solution at two wavelengths (for each valence form). Luminous flux on a given wavelength is produced by using interference filters. The facility uses sensors for determining concentrations of U(IV), Pu(III) and Np(V).

8. pH-meter sensors are used for determining the concentration of nitric acid in weakly acid solutions. Operation of the device is based on using a glass electrode immersed in the solution to be monitored. The range of pH measurements is from 1 to 14. Nitric acid concentration is determined in the region up to 2 M/ℓ.

9. Oxidation-reduction indicators are used in case of a need for measuring the redox potential of aqueous solutions.

All of these sensors are of the flow-through type. They are installed on the tubing at the inlet and outlet of the extraction modules, or between stages.

Chromel-copel thermocouples are used to measure the temperature of the process equipment. The sensors are of the immersion type, installed in the settling zone of the extraction stages and in equipment.

Thermal flowmeters are used for measuring the flowrates of phases. Flowrate is determined from the degree of cooling of preheated liquid. Range of measurable flowrates from near zero to two liters per hour.

Two types of instruments are used for determining the level of aqueous solutions in equipment: a resonant tracking level meter and a contact decade level meter that gives discrete level readings. The working principle of the decade level meter is based on closure of the decade rings of a conductometer sensor by the electrically conductive solution. The basis of the tracking level meter sensor is a coaxial waveguide that forms a tank circuit with natural frequency dependent on the level to which the waveguide is filled with liquid.

Among all the enumerated methods and means of measuring analytical parameters, the most completely developed and effective at the present time are: gamma-absorptiometric, conductometric, densimetric, methods and means of neutron measurements and measurements of gamma EDR. In addition to instruments for monitoring general technological parameters, these facilities are the basis for creating a system for monitoring the technological process.

Other methods offer additional capabilities for optimizing the process, especially in future as the corresponding measurement facilities are improved.

Note should be taken of the major significance of multiparametric monitoring methods. Simultaneous coordinated measurement of two or more parameters of the facility being monitored by using the methods considered above can give additional information on the facility--in some cases information of a fundamental nature. For example, simultaneous measurement of the density of a solution, its electrical conductivity and specific alpha-radioactivity enables us to determine the concentration of uranium, acid and plutonium for some process products, etc. Obviously the system must incorporate a computer for efficaceous use of multiparametric methods.

Fig. 2 indicates points and methods of monitoring determined in accordance with the criteria accepted above for optimum control of the process of the first extraction cycle and the selected variables $\vec{u}$, $\vec{y}$, $\vec{x}$. As can be seen from the data given on the figure, facilities are not available for some of the measurements. In this connection, we are faced with the job of developing appropriate methods and means of measurements. Obviously in many cases this may require quite considerable time and large material expenditures. Due to the increase in immediacy of laboratory control, the problem of providing efficaceous monitoring at the "missing" points can obviously be solved only in part.

All this impels the conduct of planned experiments on the stand to optimize control based on available monitoring facilities, which of course complicates solution of the problem.

Fig. 2 also indicates points and means of measurements that ensure adherence to predetermined limitations on technological parameters, as well as additional monitoring of the operation of technological equipment.

Automation of the stand is based on a small M-6000 computer. The job of interfacing remote monitoring devices is handled in two versions: by using USO modules [ustroystva svyazi s ob"yektom--devices for connection to an object] that are part of the M-6000 equipment, and by using CAMAC modules. The corresponding block diagrams of the equipment for automated data gathering, processing and presentation on the stand are shown in Fig. 4 and 5. In conjunction with software, these facilities organize remote modification of the working modes of sensors and verify their functional state. At the present time, the hardware system for direct control of the facility on the part of the computer is in the developmental stage (channel indicated by the dotted line on Fig. 1).

In accordance with the diagram of Fig. 4, devices with pulse-number signals at the output (gamma absorptiometers, alpha radiometers, tracking level meters) are interfaced by a pulse-number signal input module *(МВеЧИС)* as part of the group control module *(МГУ)*, a code control module *(МКУБ-1)* and a module for input of initiative signals *(МВеИС)*. Interfacing of instruments with discrete information (contact decade level meter) is handled by a module for input of discrete information *(МВеДИ)*. Instruments with analog output are interfaced

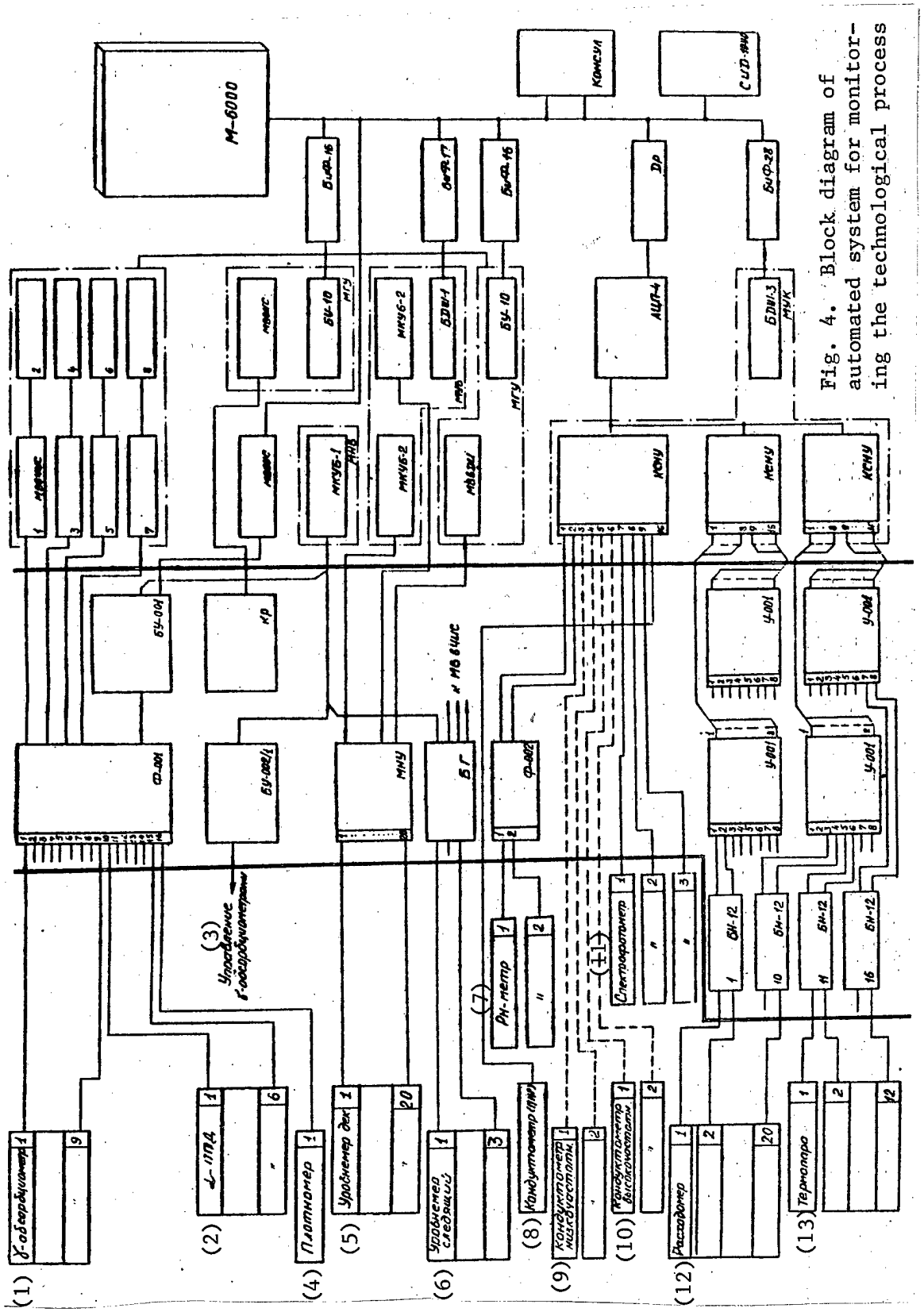

Fig. 4. Block diagram of automated system for monitoring the technological process

Key to Fig. 4:

1--Gamma absorptiometer
2--Semiconductor alpha detector
3--Gamma absorptiometer control
4--Densimeter
5--Decade level meter
6--Tracking level meter
7--pH meter
8--Conductometer (Poland)
9--Low-frequency conductometer
10--High-frequency conductometer
11--Spectrophotometer
12--Flowmeter
13--Thermocouple

by an analog-digital converter *(АЦП-4)* with a module for control of commutators *(МУК)*.

Output signals (digital and analog) of the monitoring instruments usually differ from the standard required by the USO modules. Therefore, special modules are used in addition for interfacing. Module *Ф-001* shapes standard pulses coming from the gamma absorptiometers and alpha radiometers, and modules *БУ-001* and *БУ-002* control their operation and handle commutation. Heterodyne module *БГ* is used for stepping down the frequency of the tank circuit in the tracking level meter for connection (*1с МВеЧИС*). Normalization module *МНУ* normalizes the output signals of the decade level meter and commutates them via *МКУБ-2* to the input of *МВеДИ*. Module *Ф-002* is a shaper-amplifier with high input impedance for the analog signals of the pH meter. Module *У-001* is an amplifier of analog signals of the thermocouples and flowmeters. All these modules are located outside of the confines of the hot chamber, but in direct proximity to the sensors. Also located here are the normalization modules *(БН-12)* that are part of the USO modules and are used for normalizing the signals from temperature and flowrate sensors.

When the CAMAC standard is used, data-collection from all control points is handled by a single CAMAC crate.

To eliminate intermediate interfacing arrangements between sensors and CAMAC input facilities, just as when using USO modules, it is necessary that the standard output signal be shaped either directly in the sensor, or if this is not possible, in an extension module located near the sensor.

Pulse-number signals go to counter modules controlled by a timer consisting of two units: a setting counter and a crystal-controlled oscillator. The analog information goes to multiplexers that commutate the outputs of the sensors to analog-digital converter *АЦП2048*. Discrete information from the decade level meters goes through normalization modules to the input register. The normalization module is controlled by the output registers.

The system includes service modules: a dataway display and a supply control module.

The crate is controlled by the CM1-CC crate controller. Interfacing with the M-6000 computer is organized via two duplex registers. The crate controller has an expansion bus for connecting other controllers to the same computer. This gives the capability of expanding the monitoring facility to eight crates.

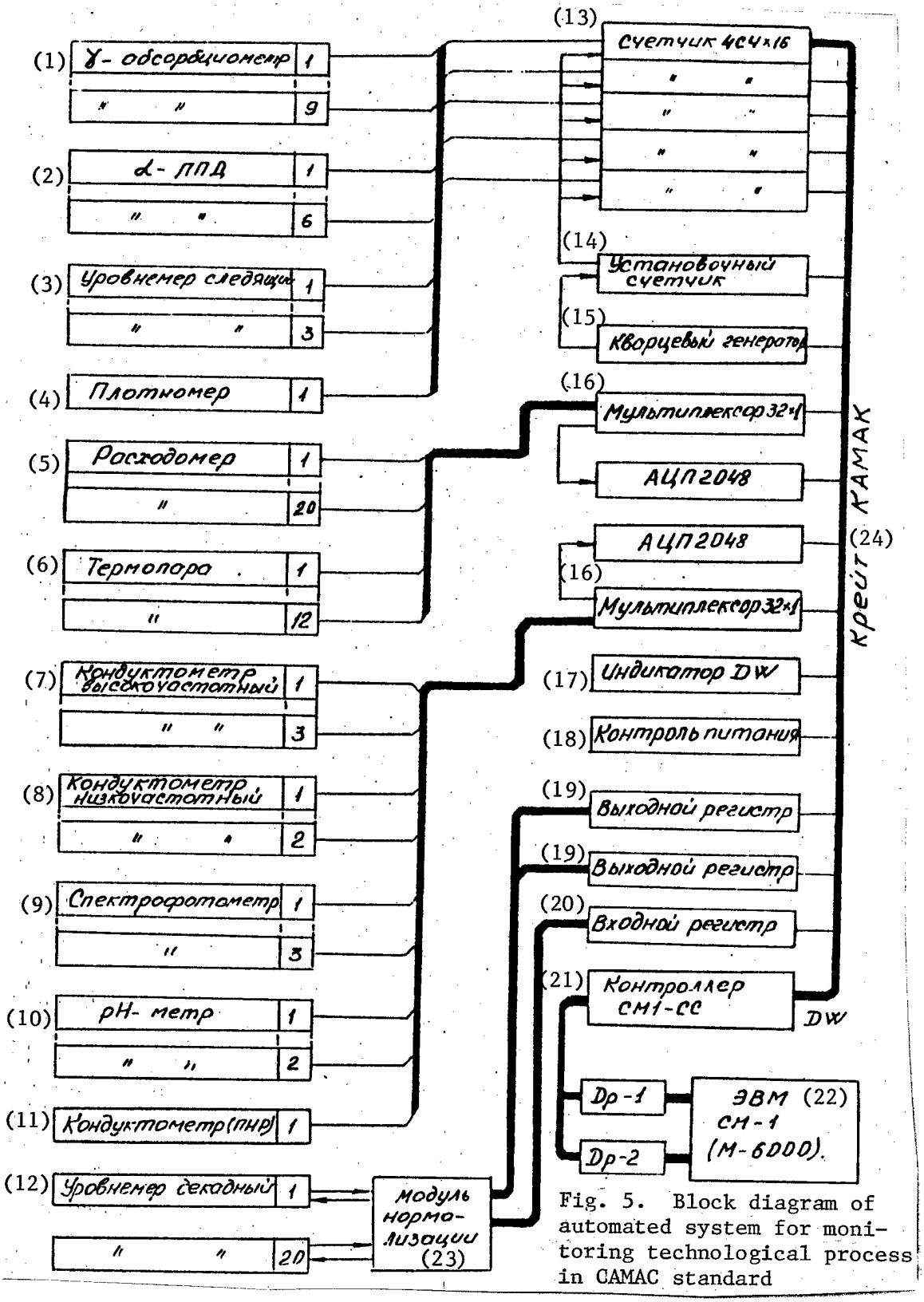

Fig. 5. Block diagram of automated system for monitoring technological process in CAMAC standard

Key to Fig. 5:

1--Gamma absorptiometer
2--Semiconductor alpha detector
3--Tracking level meter
4--Densimeter
5--Flowmeter
6--Thermocouple
7--High-frequency conductometer
8--Low-frequency conductometer
9--Spectrophotometer
10--pH meter
11--Conductometer (Poland)
12--Decade level meter
13--4c4 x 16 counter
14--Setting counter
15--Crystal-controlled oscillator
16--32 x 1 multiplexer
17--DW display
18--Supply control
19--Output register
20--Input register
21--CM1-CC controller
22--CM-1 (M-6000) computer
23--Normalization module
24--CAMAC crate

Functioning of the stand in the automated mode, interaction between experimenter, computer and controlled object is organized by the system software. The software is based on using the real-time disk operating system of the M-6000 computer. A generalized block diagram of the software is shown on Fig. 6.

All sensors that are used are non-initiating--they output information upon command of the appropriate subprogram. Information output from the sensors after computer processing is in a predetermined format on the screen of the *CUD-1000* display, with printout on an alphanumeric printer if so desired. Data on the state of the USO modules can be provided at the experimenter's request.

Based on information from the sensors about the current values of process parameters and laboratory control data, coefficients of purification and extraction are calculated in the computer that characterize the process in progress, and are displayed on the screen at given time intervals, or upon demand.

In the case of considerable deviation of any parameter from the expected value, provision is made for displaying preceding values of this parameter for any time period over the past eight hours. All information for eight hours of operation of the stand can be displayed on the screen or printed out.

Thus the main interaction between the experimenter on the one hand and the control object and computer on the other is via the display. The experimenter uses a keyboard for input of all directives.

The overall structure of the stand (Fig. 1) includes an analytical laboratory where a variety of analytical methods are used [Ref. 7]. Alpha and gamma spectrometric methods are included in the automated system for gathering and processing laboratory analytical information [Ref. 8]. The results of laboratory analyses with the use of other methods must be input manually to the computer via a teletype T-63 keyboarding device installed in the laboratory.

Alpha and gamma spectrometric analysis solves the problem of monitoring purification of uranium, plutonium and neptunium, the problem of determining the content of actinide-series nuclides and fission nuclide products, determining exposure dose rate resulting from gamma radiation in technological products.

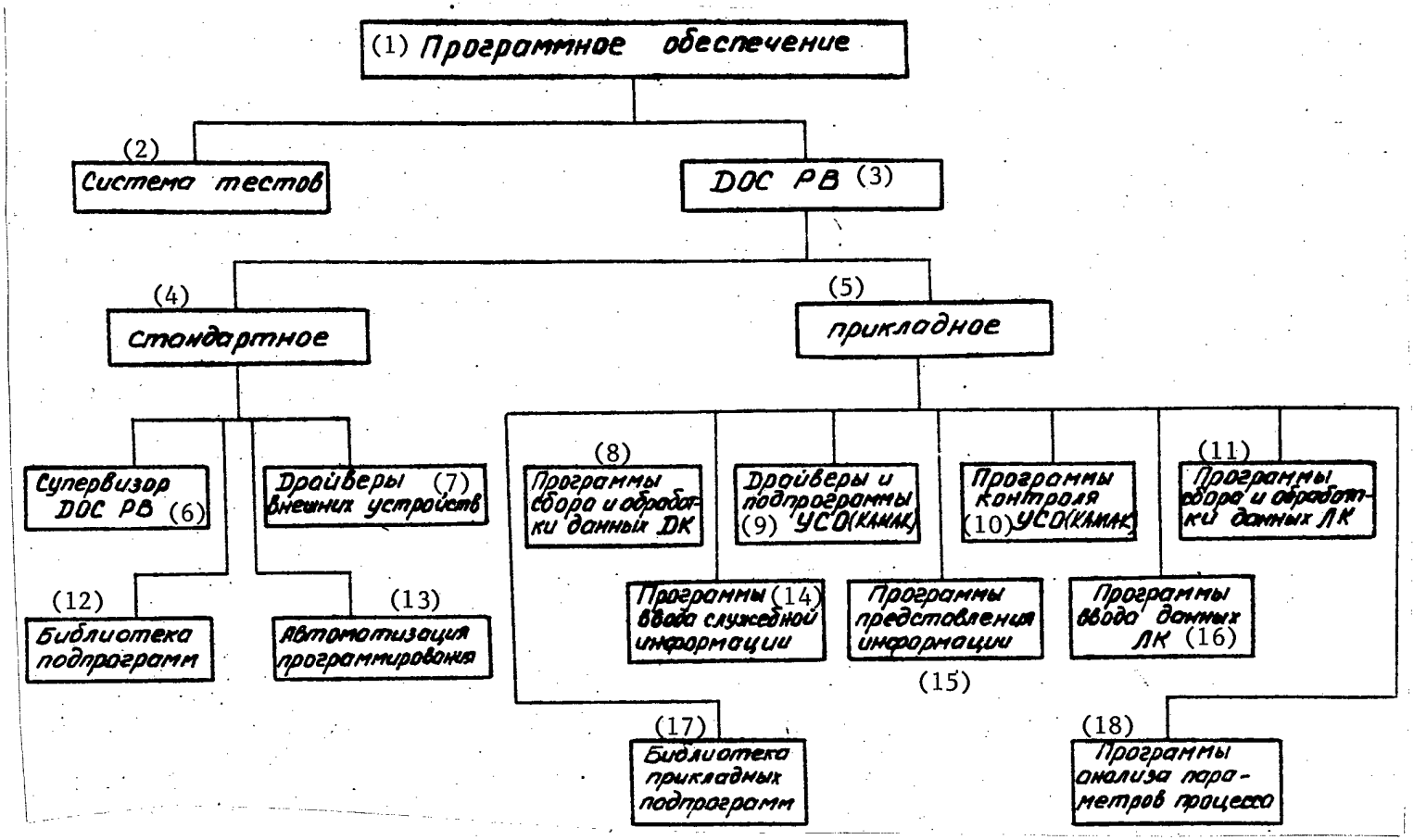

Fig. 6. Generalized block diagram of software

KEY: 1--Software
2--Test system
3--Real-time disk operating system
4--Standard
5--Applied
6--RT-DOS superviser
7--Drivers of external devices
8--DK programs for data gathering and processing
9--USO (CAMAC) drivers and subprograms
10--USO (CAMAC) monitoring programs
11--Programs for gathering and processing laboratory control data
12--Subprogram library
13--Programming automation
14--Service information input programs
15--Data representation programs
16--Laboratory control data input programs
17--Applied subprogram library
18--process parameter analysis programs

The overall productivity of the system is about 200 developed spectrometric analyses per day. A few minutes is required for doing one analysis. Analysis results are printed out.

The hardware of the system is based on spectrometric equipment produced in the USSR. Devices for interfacing the spectrometers with the computer and commutating them are custom made. Data can be transmitted over a distance of up to 500 m. The applied software includes programs for processing alpha and gamma spectra, and a number of service, auxiliary and test programs.

The algorithm for processing alpha spectra is based on digital filtration of measurement data, and is suitable over a wide range of energy resolution of the spectrum due to the thickness of the active layer of the source. The algorithm for processing gamma spectra uses a method of channel-by-channel summation. An empirical step function is used to approximate the pedestal under the photopeaks.

Preliminary experiments done on the described stand with spent VVER-440 fuel show the efficacy of the proposed automation: capability of doing technological studies in maximally compressed time periods with output of abundant information that is inaccessible in the conventional formation of this work. Research on problems of optimum control of technological operations and experimental development of the corresponding monitoring system is scheduled for the near future.

Discussion

B. Gorski: What is the response to random changes of parameters that arise in the extraction facility, such as formation of sediments, changes in TBP quality and the like?

V. I. Orlov: All monitored parameters are broken down into three groups:
--controlling,
--output,
--perturbing.

The latter characterize random effects on the technological process. The sense of optimum control consists in maximizing the control criterion on the set of permissible controlling variables for any (within limits) combination of perturbing variables.

F. Sus: You spoke of using semiconductor detectors for measuring alpha radiation. Can you tell us anything about the lifetime and memory of these detectors, and how memory is eliminated or reduced?

V. I. Orlov: The so-called memory, which is due to sorption of activity on the surface of the semiconductor detector, is the major factor that limits their use. To eliminate this effect, we use ultraviolet cleaning of the surface, or periodic rinsing with acid. However, these steps are not always sufficiently effective. Therefore the use of immersion alpha semiconductor detectors is problematical up to the present. If these effects can be eliminated, the work life is generally acceptable for practical use.

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6610
CSO: 8144/1829

UDC 621.039.546

DEVICE FOR CONTINUOUS FUEL BURNUP MONITORING

Moscow ATOMNAYA ENERGIYA in Russian Vol 53, No 1, Jul 82 (manuscript received 13 Mar 81) pp 37-38

GVERDTSITELI, I. G., KALANDARISHVILI, A. G. and KUCHIZHIDZE, V. A.

[Abstract] The feasibility of continuous in-process intrareactor fuel burnup monitoring a nuclear power plant is considered. In the device proposed for this purpose nuclides of alkali metals (^{133}Cs, ^{137}Cs, ^{85}Rb, ^{87}Rb) evolving from the fuel during irradiation serve as monitors of fuel burnup and a graphite chip selectively sensitive to these nuclides serves as sensor. Anisotropic polycrystalline graphite absorbs these nuclides in its interplanar spaces and expands, as a result, along its C-axis. The performance of such a device is evaluated here theoretically for a 3.5%-enriched ^{235}U fuel element in a VVER-440 water-moderated water-cooled reactor with thermal neutrons at a mean flux intensity of 10^{13} neutrons/($cm^2 \cdot s$). Calculations are based on the equation of diffusion for stable and long-life isotopes. The equipment includes, in addition to the graphite chip, a vapor drain channel, a baffle, bellows and a plunger, also an instrument for recording the graphite expansion. The device has a linear characteristic, the absolute elongation of graphite being proportional to the relative depletion of fuel, and its sensitivity is highest to cesium nuclides. Figures 2, references 11: 9 Russian, 2 Western. [299-2415]

UDC 620.179.(08.8)

APPARATUS FOR FLAW DETECTION ON INSIDE SURFACES OF PIPING IN AES

Moscow ATOMNAYA ENERGIYA in Russian Vol 53, No 1, Jul 82 (manuscript received 12 May 81) pp 18-21

MADOYAN, A. A., KANTSEDALOV, V. G., SAMOYLENKO, V. P. and SAMOYLENKO, P. B.

[Abstract] An experimental prototype of a DTL-500 apparatus has been developed, built and tested at the Southern branch of the Thermotechnical Institute imeni F. E. Dzerzhinskiy for flaw detection, with remote control, on the

inside surfaces of piping in AES with VVER-440 water-moderated water-cooled reactors. The equipment includes a surface preparing tool, a scan mechanism, an angle setting mechanism, fiber optics for surface inspection, an automatically controlled pneumatically driven transport mechanism, an operator's console and power, control, and data transmission lines. The transport mechanism described here consists of two cup-shaped belts facing each other bottom-to-bottom and each holding an annular chamber retained by an elastic cylindrical shell. The cups are coupled through bellows consisting of a row of rigid rings, alternately larger and smaller in diameter, joined through elastic conical membranes. The inside bellows cavity and the two annular chambers on the outside are connected to a compressor through an automatically controlled reversing switch. The transport mechanism develops a thrust of 3500 N under a pressure of 0.04 MPa inside the bellows, its maximum travel is 100 m with a payload of 250 kg, moving at a velocity of 0.005 m/s along straight lines and 0.003 m/s along curves. Figures 3, table 1.
[299-2415]

UDC 621.039.6

'ANGARA-5' MODULE FOR NUCLEAR FUSION PLANT

Moscow ATOMNAYA ENERGIYA in Russian Vol 53, No 1, Jul 82 (manuscript received 26 Feb 82) pp 14-18

BOL'SHAKOV, Ye. P., VELIKHOV, Ye. P., GLUKHIKH, V. A., GUSEV, O. A., GRABOVSKIY, Ye. V., ZAYTSEV, V. I., ISTOMIN, Yu. A., KOBA, Yu. V., LATMANIZOVA, G. M., OLEYNIK, G. M., PASECHNIKOV, A. M., PEVCHEV, V. P., PERLIN, A. S., PECHERSKIY, O. P., RUDAKOV, L. I., SMIRNOV, V. P., CHERNOBROVIN, V. I., CHETVERTKOV, V. I. and YAMPOL'SKIY, I. R.

[Abstract] The "Angara-5" module for a demonstration nuclear fusion plant with inertial plasma containment has been developed and built jointly by the Scientific Research Institute of Electrophysical Apparatus imeni D. V. Yefremov and the Institute of Atomic Energy imeni I. V. Kurchatov. The module comprises a pulse-type electron accelerator rated for 2 MeV energy, 0.8 MA current and 85 ns pulse duration. Its equipment includes a current transformer, voltage pulse generator, generator of firing pulses, pulse shaping double line with internal electrode and intermediate electrode, two thyratron amplifiers, a synchronizer, commutator, transmission line, prepulse discharger, dummy load, protective resistor, and high-voltage diaphragm. The energy transfer characteristics of this module were determined from measurements of voltage and current pulses at critical nodal points. Performance calculations based on the equivalent circuit diagram and mapping of the electric field confirm the results, which indicate stable operation at the 1.8-1.9 MV level without breakdown in the structural components and less stable operation at the 2.1 MV level. The stability margin can be increased if the generator charging voltage is increased to 180 keV and the double-line charging time is correspondingly

decreased. The authors thank the staffs of both institutes and also the staff of the Siberian Power Engineering Institute. Figures 8, table 1, references 3 Russian.
[299-2415]

UDC 669.14.018.8:621.039.5

HELIUM IN STRUCTURAL MATERIALS OF FUSION REACTOR

Moscow ATOMNAYA ENERGIYA in Russian Vol 53, No 1, Jul 82 (manuscript received 12 Jan 82) pp 3-13

NIKIFOROV, A. S., ZAKHAROV, A. P., CHUYEV, V. I., ZALUZHNYY, A. G., SOKURSKIY, Yu.N., ONUFRIYEV, V. D., TEBUS, V. N., GORODETSKIY, A. Ye., ALIMOV, V. Kh., AL'TOVSKIY, I.V. and CHEREDNICHENKO-ALCHEVSKIY, V. M.

[Abstract] An important problem in design of fusion reactors is selection of structural materials, especially for the first wall, with adequate resistance to high-temperature helium embrittlement. Series of experimental and theoretical studies have been made concerning the kinetics of helium accumulation in various materials and its effect on their mechanical characteristics. Such studies were made, for instance, at the Kharkov Physico-Technical Institute (UkSSR Academy of Sciences) based on measurements in ITR-2000, BOR-60, SM-2 fast reactors and aided by computations according to the ALICE program. Materials for which data have been obtained include iron, nickel, stainless steels, titanium and its alloys, aluminum and its alloys (Al-Li). An analysis of these data and results of microstructural examination reveal how tensile strength, yield point, percent deformation and their temperature dependence change with increasing helium concentration, also the effect of helium on the fracture mode. Studies have also been done concerning the behavior of helium in the crystal lattice of various metals (Cu, Ni, Pd, Au, α-Fe, Mo, W, V, Ta), where it dissolves poorly and forms either easily dissociable complexes or metastable solid solutions with the host metal, and concerning changes in the lattice structure and phase transformations caused by helium implantation. Figures 17, tables 6, references 57: 24 Russian, 33 Western.
[299-2415]

UDC 539.125.52:621.039.51.12

FLUX DENSITY OF FAST NEUTRONS IN REACTOR OF KOLA AES

Moscow ATOMNAYA ENERGIYA in Russian Vol 52, No 6, Jun 82 (manuscript received 20 Jul 81) pp 431-432

BONDARS, Kh. Ya., IGNATENKO, Ye. I., LAPENAS, A. A., LOBOV, V. I., LOMAKIN, S. S., MOROZOV, A. S. and TROFIMOV, B. A.

[Abstract] The flux density of fast neutrons in the water-moderated water-cooled VVER-440 reactor of the Kola AES has been determined from measurements of their space-energy distributions in the radiation shield and subsequent calculations according to the two-dimensional RADUGA program. Measurements were made in process channels with ^{115}In and ^{27}Al detectors, at the core edge, in the intrareactor control zone, and in the biological shield with ionization chambers. Both differential and integral distributions of flux density were obtained. The slope of the differential distribution curve indicates that 3-10 MeV neutrons are "harder" in the ionization chambers than at the core edge, which agrees with results of control calculations according to the SAND II program. The distributions over channel height were determined on the basis of threshold $^{115}In(n,n')$ and $^{27}Al(n,\alpha)$ reactions. Figures 2, table 1, references 2 Russian.
[300-2415]

UDC 539.172.13

MECHANISM OF TAPER OF NEUTRON YIELD FROM TRITIATED METAL TARGETS IN NG-150 NEUTRON GENERATORS

Moscow ATOMNAYA ENERGIYA in Russian Vol 52, No 6, Jun 82 (manuscript received 2 Jun 81, final edition received 27 Jan 82) pp 423-425

BARIT, I. Ya., KUZ'MIN, L. Ye., MAKAROV, S. A. and PRONMAN, I. M.

[Abstract] The taper of neutron yield from tritiated metal targets in NG-150 D-T neutron generators is attributable not only to escape of tritium from the target following decomposition of titanium tritide due to low thermal conductivity and weak bond with the cooled substrate and following buildup of radiative defects as a result of cathode sputtering, but also to replacement of tritium by implanted deuterium and subsequent escape of tritium through diffusion. This hypothesis was verified by measurement of deuterium and tritium distributions in Ti-T and Ti-D targets on copper substrates. The experiment was performed at the Institute of Nuclear Research (USSR Academy of Sciences) using an EG-2 electrostatic generator and a surface-barrier semiconductor-type detector with 38 keV resolution. Calculations were done on the basis of proton and alpha particle yields from $D(d,p)T$ and $T(d,n)^4He$ reactions as well as their energy spectra recorded at 110 and 165° angles. The results indicate that replacement of tritium by deuterium occurs outside the resonance range of D-T interaction

and that the target life can be extended by use of a purely atomic deuterium beam. The authors thank L. N. Katsaurov for helpful discussion of the results. Figures 3, table 1, references 10: 6 Russian, 4 Western.
[300-2415]

UDC 539.124.164

EXPERIMENTAL STUDY OF NEUTRON SPECTRUM IN LEAD WITH 14 MeV NEUTRON SOURCE

Moscow ATOMNAYA ENERGIYA in Russian Vol 52, No 6, Jun 82 (manuscript received 2 Fec 81) pp 422-423

ALEKSANDROV, V. D., ZAGRYADSKIY, V. A., MARKOVSKIY, D. V., NOVIKOV, V. M., CHUVILIN, D. Yu and SHATALOV, G. Ye.

[Abstract] The intensity of a neutron source and the corresponding yield of tritium can be increased appreciably (40-50%) by use of a lead layer as neutron breeder in the blanket of a fusion reactor not containing fissionable material. The neutron spectrum of such a lead layer alone, without eutectic mixture of lithium and beryllium fluorides, was measured in an experiment with a 14 MeV neutron source. The experiment was performed in the "inverse" configuration, with the neutron detector inside and the neutron source outside the spherical lead shell. The activation-type detector was a pseudospherical stack of nine disks of 0.55 mm thick foil, the center disk having the largest diameter of 5 mm and the others on both sides having successively smaller diameters. The neutron source was an NGI-5 generator ($5 \cdot 10^8$ neutrons/s) for a 30 mm thick lead shell and an NG-150 generator (10^{11} neutrons/s, 80 keV deuterons) for 50 and 70 mm thick lead shells. The parameters of the neutron spectra were determined from tests with six different such detectors: $^{103}Rh(n',n)$, $^{115}In(n,n')$, $^{27}Al(n,p)$, $^{27}Al(n,o)$, $^{65}Cu(n,2n)$, $^{19}F(n,2n)$, also with a normalizing $^{63}Cu(n,2n)$ detector having the highest reaction threshold of 11.8 MeV. The neutron source was calibrated without a lead shell. Calculations were also done theoretically, according to the BLANK program, with simulation of the neutron source as a uniform and isotropic one in a test configuration allowing a one-dimensional approximation. The values of the various constants were taken from ENDL data files. The results agree very closely for detectors with high reaction threshold, the differences are large for detectors with low reaction threshold. All calculations were referred to activity saturation of the detector at the center of the sphere in a neutron flux with a standard spectrum. Figure 1, table 1, references 4: 2 Russian, 2 Western.
[300-2415]

UDC 621.039.546:546.291

ACCUMULATION OF HELIUM IN STRUCTURAL MATERIALS WITH NICKEL CONTENT DURING IRRADIATION IN BOR-60 REACTOR

Moscow ATOMNAYA ENERGIYA in Russian Vol 52, No 6, Jun 82 (manuscript received 6 Feb 80) p 421

[Abstract] Accumulation of helium in 99.98% iron as well as in nickel steel OKh16N15M3B and in nickel alloys Kh26N6T, OKh20N40B, KhN77TYu was measured after irradiation in a BOR-60 reactor at temperatures not exceeding 500°C. The maximum allowable flux density of fast neutrons ($E > 0.1$ MeV), based on total irradiation time at nominal reactor power, ranged from $4.9 \cdot 10^{22}$ to $7.8 \cdot 10^{22}$ neutrons/cm^2 within $\pm 30\%$ accuracy. The wide discrepancy between measured and theoretically calculated helium concentrations found here is attributable not only to experimental errors and unavailability of precise data on cross sections for threshold (n, α)-reactions but also to an intricate two-stage $^{58}Ni(n, \gamma)^{59}Ni(n, \gamma)^{56}Fe$-reaction at neutron energies of less than 0.1 MeV. The results indicate nevertheless a linear relation between nickel content and accumulated helium concentration in a material. Figure 1, references 3: 1 Russian, 2 Western.
[300-2415]

UDC 621.039.546:546.291

ACCUMULATION AND BEHAVIOR OF HELIUM IN IRON IRRADIATED IN IRT-2000 RESEARCH REACTOR

Moscow ATOMNAYA ENERGIYA in Russian Vol 52, No 6, Jun 82 (manuscript received 10 Oct 80, final edition received 23 Oct 81) pp 419-421

ZALUZHNYY, A. G., SKOROV, D. M., STOROZHUK, O. M., CHEREDNICHENKO-ALCHEVSKIY, M. V. and KLYKOV, L. M.

[Abstract] The behavior of helium in 99.97% pure iron was measured, after irradiation in the IRT-2000 research reactor at temperatures not exceeding 350 K by fast neutrons ($E > 2.6$ MeV) and thermal neutrons to flux densities of $6 \cdot 10^{18}$ and $2 \cdot 10^{20}$ neutrons/cm^2 respectively. The iron specimens were 100 μm thick foils recrystallized to a 30-50 μm grain size and uniformly saturated with gas from (n, α)-reactions. The data reveal a linear relation between helium accumulation and neutron flux density. The temperature dependence of helium escape from iron upon heating is characterized by several increasingly sharp peaks (at 690, 970, 1160, 1320, . . . K). The integral helium escape from maximally irradiated iron is an exponential function of time. According to the data, helium escape proceeds in four stages: at low temperatures (720-770 K) determined by release from the crystal lattice during annealing of radiation defects, at intermediate temperatures (770-1070 K) determined by diffusion of gas atoms, at about 1160 K determined by $\alpha \rightarrow \gamma$ transformation, and above 1220 K determined by nondirectional migration of gas complexes or bubble nucleation centers toward the surface. Figures 3, references 7: 5 Russian, 2 Western.
[300-2415]

UDC 621.039.546:546.291

UNIVERSAL MASS-SPECTROMETER FOR DETERMINATION OF INERT GASES ACCUMULATING IN AND ESCAPING FROM REACTOR MATERIALS

Moscow ATOMNAYA ENERGIYA in Russian Vol 52, No 6, Jun 82 (manuscript received 10 Oct 80, final edition received 23 Oct 81) pp 418-419

ZALUZHNYY, A. G., SKOROV, D. M., STOROZHUK, O. M., CHEREDNICHENKO-ALCHEVSKIY, M. V. and KLYKOV, L. M.

[Abstract] A universal mass-spectrometer has been built for a complex of measurements pertaining to kinetics of inert gases in structural materials of a reactor, their generation, accumulation, and escape. The equipment includes an atomizing evaporation chamber, a high-vacuum annealing chamber, an oil-less suction pumping system (one GIN 05-M1 ion-getter pump, one NORD-100 magneto-discharge pump, two TsVN-1 sorption pumps), and a measuring system with an IPDO-2A instrument and an RMO-4S omegatron tube. There are also included several manometers: a VIT-2P for measuring the residual pressure and PMT, PMI vacuum-tube manometers at various critical points in the system. All components except atomizing chamber and vacuum-tube devices are made of stainless steel, vacuum seals are made of copper. The cylindrical annealing chamber contains a tungsten-wire resistance heater with cermet terminals and thermal shields around it. A shaft of magnetic material inside a thin tube of molybdenum glass carries test and reference specimens, moving them axially inside the chamber as required driven by a permanent magnet around its extension outside the chamber. Temperature of specimens inside the chamber is measured by a thermocouple. The mass spectrometer has a sensitivity of $4 \cdot 10^{-8}$ Pa, which corresponds to 10^{10} atoms of inert gas, the IPDO-2A instrument has an accuracy within 20%. The mass-spectrometer can determine, with high accuracy, the amount of any inert gas in any fissionable material with a melting point up to 2300 K annealed at any temperature up to 1500 K. Figures 2, references 2 Russian.
[300-2415]

UDC 539.172.4

MEASUREMENT OF CROSS SECTION FOR RADIATIVE CAPTURE OF 0.1-50 keV ^{236}U NEUTRONS

Moscow ATOMNAYA ENERGIYA in Russian Vol 52, No 6, Jun 82 (manuscript received 25 Sep 81) pp 406-409

BERGMAN, A. A., MEDVEDEV, A. N., SAMSONOV, A. Ye., TOLSTIKOV, V. A., KOLOSOVSKIY, A. G., MORDOVSKIY, M. V. and MALIKZHONOV, A.

[Abstract] The cross section for radiative capture of ^{236}U neutrons in industrial fast reactors was measured over the 0.1-50 keV range. Measurements were made with a neutron spectrometer, based on neutron moderation time in a lead target, the energy resolution of the instrument being nearly Guassian. A hermetic cylindrical cartridge made of Duralumin and containing U_2O_8 was the source of

99.84%-enriched ^{236}U. A special-purpose gas-filled boron counter was the detector, its efficiency of recording γ-quanta in a carbon cube and in a graphite prism being proportional to their energy. The readings were checked against those of a BaF_2-crystal scintillation counter with an efficiency of recording radiative capture dependent largely on the form of the γ-spectrum. The data were processed on the basis of the radiative capture cross section being inversely proportional to the square root of neutron energy, with a normalizing (Au in graphite prism) factor, a calibrating (count of $^{10}B(n,\alpha)$ reaction events) factor, a correction factor accounting for finite spectrometer resolution and a correction factor accounting for deviation of cross section for boron from the (1/v)-law. In addition to errors in determining these factors the authors also estimated: error due to recycling neutrons, error due to higher efficiency of recording faster neutrons from $^{10}B(n,\alpha)$ reaction, error due to presence of impurities in ^{236}U, error due to shielding of neutron flux in thermal zone of boron counter, error in determination of event recording time and error in determination of mean neutron energy. Based on these data and estimates, the energy dependence of the neutron radiation capture cross section is calculated for ^{236}U. Figure 1, table 1, references 9: 7 Russian, 2 Western.
[300-2415]

UDC 621.039.531

ELECTRON-MICROSCOPE EXAMINATION OF HELIUM DISTRIBUTION IN HIGH-NICKEL ALLOY

Moscow ATOMNAYA ENERGIYA in Russian Vol 52, No 6, Jun 82 (manuscript received 28 Jul 81) pp 401-404

GUSEVA, M. I., ZAKHAROV, A. P., KALIN, B. A., SKOROV, D. M., CHERNIKOV, V. N., CHERNOV, I. I. and SHISHKIN, G. N.

[Abstract] Radiation defects in the surface layer of 20-45 high-nickel alloy were examined under a transmission-type electron microscope, after specimens of this material had been bombarded by helium ions. Specimens for this study, 0.3 mm thick 10x10 mm square pieces of foil annealed in pure argon at 1170 K for 1 h and polished electrolytically, were bombarded by 40 keV helium ions in the ILU-3 particle accelerator at a current density of 0.3 A/m^2 ($\sim 1.9 \cdot 10^{18}$ $ions/m^2 \cdot s$) in doses from 10^{21} to 10^{22} $ions/m^2$ with the target temperature not exceeding 370 K. The fine structure of blisters was studied under EMB-100L and YeM301S,G electron microscopes with an accelerating voltage of 100 keV, by a procedure including unilateral slicing to complete removal of material under blister cap. preliminary polishing on bombarded side to approximately half depth of blister cap. and final unilateral trimming of peeled blister domes. Vacancy pores much larger than gas bubbles were found in superlattices, their clusters becoming more ordered with larger doses of bombardment. Figures 5, references 13: 4 Russian, 9 Western.
[300-2415]

UDC 621.039.512

BIFURCATIONS AND SELF-EXCITED OSCILLATIONS IN NUCLEAR REACTORS WITH LINEAR FEEDBACK

Moscow ATOMNAYA ENERGIYA in Russian Vol 52, No 6, Jun 82 (manuscript received 19 Oct 81) pp 393-398

KOLCHIN, V. A.

[Abstract] The dynamic behavior of a nuclear reactor operating at a point near the stability limit in the space of parameters is analyzed with the aid of bifurcation theory. The point model describing the reactor kinetics, with either stationary or circulating fuel, assumes a linear "neutron power - reactivity" feedback loop. Linear differential equations describe emitters of delayed neutrons and their concentrations. On this basis, the nonlinear integro-differential equation of dynamics is linearized in the vicinity of the point corresponding to zero relative change in neutron power and yields a characteristic equation subject to bifurcation with respect to any one feedback parameter changing by a sufficiently small amount while all other feedback parameters remain constant. Such a characteristic equation has one pair of purely imaginary roots and all other roots with negative real parts. A bifurcation can in this case result in self-excited oscillations. The oscillation modes are determined from the original equation of dynamics in an appropriate new time variable. For illustration, this method of analysis is applied to reactors with lagging power feedback and circulating fuel where delayed neutrons do not participate in the fission process and reactivity depends only on the temperature of the incompressible fuel. The heat exchanger is assumed to be ideal, with constant fuel and coolant temperature at core entrance. The spatial distribution of neutrons is approximated by the first harmonic. Numerical solution on a computer yields the D-fission boundary between stability and instability regions in the space of feedback parameters, also the relative change in neutron power on an oscillating function of time and its amplitude as a function of variation of a small parameter in the characteristic equation. The author thanks V. D. Goryachenko for formulating the problem and assisting in the study. Figures 3, references 12: 8 Russian, 4 Western.
[300-2415]

UDC 621.039.52

HYDRODYNAMICS IN MIXING CHAMBERS OF NUCLEAR POWER PLANTS

Moscow ATOMNAYA ENERGIYA in Russian Vol 52, No 6, Jun 82 (manuscript received 19 Oct 81) pp 385-389

OPANASENKO, A. N. and SHAN'GIN, N. N.

[Abstract] Temperature and velocity fields in the upper mixing chamber of a fast reactor are calculated on basis of model experiments and simplified mathematical models. The mathematical models include equations of natural and

forced connection. The experiments were performed with a cylindrical vessel, hot water entering through the central grid simulating the exit of the reactor core and cold water entering through the peripheral grid simulating the exit from the lateral baffles. The longitudinal profiles of radial temperature nonuniformity and of maximum relative temperature fluctuation amplitude are determined for Reynolds number $Re > 10^4$ and Froude numbers Fr=1 (nominal flow under dynamic conditions) and $Fr \ll 1$ (steady flow at reduced power level). Typical patterns of isotherms and flow lines corresponding to specific boundary conditions reveal the effect of natural convection on horizontal temperature gradients. Figures 4, references 2 Russian.
[300-2415]

UDC 621.039.522.042.48

MECHANISM OF HIGH-FREQUENCY RESONANCE-TYPE INSTABILITY IN VK-50 BOILING-WATER REACTOR

Moscow ATOMNAYA ENERGIYA in Russian Vol 52, No 6, Jun 82 (manuscript received 29 Sep 81) pp 379-382

LEPPIK, P. A., PAVLOV, S. P. and PLYUTINSKIY, V. I.

[Abstract] The mechanism of power fluctuations in a VK-50 boiling-water reactor at frequency of 1 Hz is analyzed on basis of theoretical relations and experimental data. Large-amplitude periodic fluctuations of neutron flux are the stability threshold criterion. Calculations yield the hodographs of eight complex frequency characteristics involving fundamental mode and first azimuthal mode of neutron flux as well as mass rate of water flow at core entrance and pressure drop across the core. Computations are made for plant performance levels ranging from 2.5 MPa and 65.5 MW to 5.0 MPa and 120 MW. The results include stability margins with respe t to each of the eight frequency characteristics, i. e., distances of each hodograph from the corresponding danger point. The results suggest that, with the reactor core dynamics described correctly, the resonance-type instability is not attributable to physical instability of the neutron flux alone but must also be due to strong interaction of neutron flux and all-loop circulating water. According to these results, a VK-50 reactor is sufficiently stable with respect to all-core interchannel water-neutron interaction to make it impossible to verify this type of instability in it. Figures 3, references 13 Russian.
[300-2415]

UDC 621.039.6

MATHEMATICAL MODEL OF FUSION POWER PLANT WITH HYBRID REACTOR

Moscow ATOMNAYA ENERGIYA in Russian Vol 52, No 6, Jun 82 (manuscript received 25 May 81) pp 371-374

KURBATOV, D. K. and ORLOV, V. V.

[Abstract] In future fusion power plants a hybrid reactor with natural uranium or uranium tailings in the blanket will serve mainly as producer of plutonium fuel for the fusion reactors in such a plant. The economic criteria of this operation are established on basis of the mathematical model PARIS, from the standpoint of annual costs of an exponentially growing constant-structure system. The corresponding relations for all relevant cost components give the value of produced plutonium as the ratio of cost increment to plutonium yield. The model of plant optimization according to this criterion includes equilibrium and balance equations of plasma physics in the initial approximation with corrections for the radial profiles of plasma parameters, equations describing the blanket of the hybrid reactor with natural or depleted uranium in the fuel elements, a blanket where uranium core and lithium core are both gas cooled or both water cooled, or the former gas cooled and the latter water cooled, equations describing the magnet system and equations describing the first wall. Typical calculations of energy generation cost and fuel cost as functions of the larger plasma radius and of energy generation cost as function of magnetic induction at the axis of the toroidal field indicate that at present price levels and utilization factors (0.8 for hybrid reactor, 0.6 for fission reactor) the fuel cost can be maintained at a 0.4-0.42 level of the total annual cost. Figures 3, references 4 Russian.
[300-2415]

UDC 621.039

EXPERIMENTAL CIRCULATION LOOP EQUIPMENT FOR 'MARIA' REACTOR AT INSTITUTE OF NUCLEAR RESEARCH OF POLISH ACADEMY OF SCIENCES

Minsk VESTSI AKADEMII NAVUK BSSR: SERYYA FIZIKA-ENERHETYCHNYKH NAVUK in Russian No 3, Jul-Sep 82 (manuscript received 6 Jan 81) pp 74-77

GOL'TSEV, V. P., YERMASHKEVICH, V. N., KOVALEV, S. D., KOLCHANOV, G. G., NESTERENKO, V. B. and KOZEL, Ye., Institute of Nuclear Engineering, BSSR Academy of Sciences; Institute of Nuclear Research, Polish Academy of Sciences

[Abstract] PUMA equipment for circulation of dissociating N_2O_4 coolant in the "Maria" reactor at the Institute of Nuclear Research of the Polish Academy of Sciences is being developed and built with cooperation of the Institute of Nuclear Power Engineering at the BSSR Academy of Sciences and the Kharkov Physico-Technical Institute at the UkSSR Academy of Sciences, also other Polish

enterprises and the Hungarian GAMMA combine. The equipment consists of regenerative heat exchangers, cooler, electric heater, filters and pumps. It also includes an emergency aftercooling system, an automatic control system, toxicity monitor and leak monitor. First experience with MAK-4 and KAK-4A ampoules installed for testing structural materials in this coolant has yielded favorable results. References 2 Russian.
[295-2415]

UDC 621.039.5

VG-400 PROTOTYPE HIGH-TEMPERATURE NUCLEAR FACILITY FOR COMBINED GENERATION OF POWER AND PROCESS HEAT

Minsk VESTSI AKADEMII NAVUK BSSR: SERYYA FIZIKA-ENERHETYCHNYKH NAVUK in Russian No 3, Jul-Sep 82 (manuscript received 6 Jan 82) pp 54-61

DOROFEYEV, A. M., KOSHKIN, Yu. N., KOMAROV, Ye. V., KIRYUSHIN, A. I., LYUBIVYY, A. G., MITENKOV, F. M., PONOMAREV-STEPNOY, N. N., PROTSENKO, A. N., SAMOYLOV, O. B. and STOLYAREVSKIY, A. Ya.

[Abstract] Use of high-temperature gas-cooled reactors in power technology would eliminate the need for scarce fossil fuels. Development of power plants with such reactors requires the solution of many problems in generation, transmission and utilization of high-temperature heat, in helium technology in design of new equipment and in application of new materials. A prototype VG-400 plant of this type is being built as pilot and basis for future such plants with higher power ratings. The thermal power of its reactor is 1000 MW, to allow sufficient design margin, its effective power is 300 MW. The plant includes, in addition to the reactor with graphite reflector and other appurtenances, a high-temperature intermediate heat exchanger, a steam generator with bypass loop, a turbine-generator set, a condenser, an intermediate steam superheater, two high-pressure preheaters, two feed pumps, and a deaerator. The containment housing is made of reinforced concrete capable of withstanding 0.1-0.2 MPa helium pressure and, in case of emergency due to leakage in the first stage, of localizing and retaining coolant for aftercooling. The intermediate high-temperature heat exchanger is designed with large performance margin at acceptable cost, with tubing easily manufactured, assembled, transported, installed, disassembled and replaceable for repair. The most serious problem here is selection of adequately heat resistant materials, a 60% nickel alloy being considered for this application. Figures 3, table 1, references 3 Russian.
[295-2415]

UDC 621.039.58

DYNAMIC CHARACTERISTICS OF AES EQUIPMENT AND EMERGENCY AFTERCOOLING SYSTEM

Minsk VESTSI AKADEMII NAVUK BSSR: SERYYA FIZIKA-ENERHETYCHNYKH NAVUK in Russian No 3, Jul-Sep 82 (manuscript received 16 Nov 81) pp 43-47

SHAROVAROV, G. A., BERNATSKAYA, A. M., ZENICH, T. S. and NICHIPOR, V. V., Institute of Nuclear Power Engineering, BSSR Academy of Sciences

[Abstract] The peculiarity of transient processes in AES with dissociating coolant is that they involve neutron kinetics coupled with chemical reactions and phase transformations. The emergency dynamics in such AES are analyzed here on the basis of equations describing mass, momentum and energy conservation. Calculations are done in two stages: first transients in the coolant and in components significantly affecting the coolant properties, then transients in structural components with coolant parameters serving as boundary conditions. Leakage in the main stage, cessation of coolant flow through reactor core and reactor excursion are the main three types of failure, most dangerous being ruptures in the piping of the main stage at the reactor entrance and also serious being ruptures in the steam duct at the reactor exit. The reactor control system must ensure maintenance of continuous coolant circulation, must take up residual heat generated during aftercooling and must localize the fission products when fuel elements leak. Such a system has been designed in accordance with these requirements and with performance characteristics matching the coolant dynamics during emergency. It includes two system cutoff valves, two main-stage cutoff valves, a high-speed valve, a regenerator-evaporator, a condenser, and a pump. Figures 5, references 4 Russian.
[295-2415]

UDC 621.039

TECHNICAL-ECONOMIC OPTIMIZATION OF THERMODYNAMIC CYCLE PARAMETERS AND EQUIPMENT IN AES WITH FAST REACTOR USING DISSOCIATING GAS

Minsk VESTSI AKADEMII NAVUK BSSR: SERYYA FIZIKA-ENERHETYCHENYKH NAVUK in Russian No 3, Jul-Sep 82 (manuscript received 16 Nov 81) pp 34-38

NESTERENKO, V. B., BUBNOV, V. P., BUNIN, Ye. N., BYKOV, A. I., NESTERENKO, I. E. and KURTSMAN, M. V., Institute of Nuclear Power Engineering, BSSR Academy of Sciences

[Abstract] Parameters of AES and of nuclear energy system are interdependent, inasmuch as nuclear fuel (plutonium) is both needed for AES operation and produced by it. Matching these parameters for technical-economic optimization can be done by first determining the worth of fuel and then optimizing the AES performance characteristics by successive approximations. This method requires a definition of the optimality criterion which takes into account annual changes

in costs, it involves tedious iterations, and it is not versatile enough. Another method is proposed that combines both problems into one. Here the optimality criterion is defined as a vector of independent optimizable parameters and referred to the entire design period of AES operation within the nuclear energy system. The procedure was applied to the thermodynamic cycle in a BRGD-1600 MW AES with a fast breeder reactor and dissociating gas (N_2O_4) as coolant and working fluid. The results of calculations indicate difference between initial and optimum variant, such as a 10% power total referred cost. The mathematical model for this optimization problem has been programmed on a MINSK-32 computer and allows, in its simplified form, for varying 38 parameters. Figure 1, tables 2, references 11 Russian.
[295-2415]

UDC 621.039.5

IMPROVEMENT OF FUEL BREEDING IN GAS-COOLED FAST REACTORS

Minsk VESTSI AKADEMII NAVUK BSSR: SERYYA FIZIKA-ENERHETYCHNYKH NAVUK in Russian No 3, Jul-Sep 82 (manuscript received 6 Jan 82) pp 29-33

GLUSHKOV, Ye. S., GREBENNIK, V. N., DEMIN, V. Ye., KNYAZEV, V. A., PONOMAREV-STEPNOY, N. N., PROTSENKO, A. N. and SILAYEV, Yu. V., Insiitute of Atomic Energy imeni I. V. Kurchatov

[Abstract] Improvement of fuel breeding in a helium-cooled fast reactor by replacement of oxide fuel with denser carbide or preferably nitride fuel is examined, nitride fuel being densest of all and such fuel based on the ^{15}N isotope having the smallest cross section for neutron absorption. A comparative evaluation of performance characteristics has been made according to the ROKBAR optimization program package for a BGR-16k0 MW fast reactor operating with $(U,Pu)O_2$, (U,Pu)C, (U,Pu)N and $(U,Pu)^{15}N$ fuel respectively. Free plutonium available for fueling thermal reactors builds up fastest and to the highest level with ^{15}N enrichment of uranium fuel. Use of nitride fuel based on natural nitrogen is economically feasible at this stage, but enrichment with ^{15}N isotope may become more expedient as the cost of nuclear fuel continues to rise. Figures 2, tables 2, references 9 Russian.
[295-2415]

UDC 621.311.25+621.039.52

PROBLEMS IN CONSTRUCTION OF AES WITH FAST REACTOR USING DISSOCIATING COOLANT

Minsk VESTSI AKADEMII NAVUK BSSR: SERYYA FIZIKA-ENERHETYCHNYKH NAVUK in Russian
No 3, Jul-Sep 82 (manuscript received 29 Jan 82) pp 17-25

NESTERENKO, V. B., KOLYKHAN, L. I., KOVALEV, S. D., LOMASHEV, B. I., TVERKOVKIN, B. Ye., NAUMOV, V. A., KANTEMIR, A. D., KOSYAK, Yu. F., NIKITIN, V. P., SAZONOV, A. G., OGLOBLIN, B. G., SUKHORUCHENKOV, N. V., OSOKIN, A. F., ZAVADSKIY, M. I., SINEV, N. M. and BUKHTEYEV, I. S., Institute of Nuclear Power Engineering, BSSR Academy of Sciences

[Abstract] The advantages of using N_2O_4 as coolant for fast reactors are intense transfer of chemical enthalpy in a nonisothermal stream, low viscosity, low heat of evaporation, high specific heat of gaseous phase, high stability under radiation, low induced activity, and acceptable degree of irreversible thermal breakdown. This coolant is compatible with structural components made of chromium-nickel steels. In the single-stage configuration there is a high probability of interaction with reactor combustibles in case of leakage, which requires extra safety measures such as use of composite fuel (UO_2+PuO_2+30% Cr) in the core and complex fuel (MgU_2O_4) in the breeder. These basic principles are being implemented in construction of such a pilot-industrial reactor for BRIG-300 MW and BRGD-1500 MW AES. Theoretical and experimental research is being done at BSSR, UkSSR, MSSR, LiSSR Academies of Sciences, with contributions from the Polish Academy of Sciences ("Maria" reactor) as well from Hungarian institutions (GAMMA combine) and Bulgarian institutions (Center of Applied Mathematics). Figure 1, tables 2, references 11 Russian.
[295-2415]

UDC 621.039.526

OPTIMIZATION OF THERMAL SYSTEMS IN AES WITH GAS-COOLED FAST BREEDER REACTORS

Minsk VESTSI AKADEMII NAVUK BSSR: SERYYA FIZIKA-ENERHETYCHNYKH NAVUK in Russian
No 3, Jul-Sep 82 (manuscript received 24 Nov 81) pp 25-29

PROTSENKO, A. N., KNYAZEV, V. A., GREBENNIK, V. N., SILAYEV, Yu. V., SMOLKIN, Yu. V. and SUVOROV, P. P., Scientific-Industrial Association for Power Equipment Research and Design imeni I. I. Polzunov; Institute of Atomic Energy imeni I. V. Kurchatov

[Abstract] A computer program has been developed for optimization of AES with gas-cooled fast breeder reactors, not a universal program but one tailored for individual systems and specifically the thermal system in a two-stage AES. It is based on models of steam generating equipment and of turbines. The optimization criterion is the annual economic effect, i.e., the fuel cost differential

attainable as a function of pressure and temperature. The program was used for comparative evaluation of four variants for a BGR-300 MW reactor and selecting the optimum one on this basis, with fuel cladding temperature of 800°C.
F gures 2, table 1, references 5 Russian.
[295-2415]

UDC 621.039.52

DEVELOPMENT OF FAST REACTORS WITH HELIUM COOLING

Minsk VESTSI AKADEMII NAVUK BSSR: SERYYA FIZIKA-ENERHETYCHNYKH NAVUK in Russian
No 3, Jul-Sep 82 (manuscript received 26 Jan 82) pp 10-17

YEMEL'YANOV, I. Ya., GANEV, I. Kh., KNYAZEV, V. A., PONOMAREV-STEPNOY, N. N., KRUGLOW, A. L., PROTSENKO, A. N., SMETANNIKOV, V. P., ULASEVICH, V. K. and SILAYEV, Yu. V., Institute of Atomic Energy imeni I. V. Kurchatov

[Abstract] Progress is being made in development and design of fast reactors for AES using helium instead of sodium as coolant. The main advantages of an inert gas are its very small cross section for neutron scattering and absorption and its very weak interaction with fuel elements and structural components. This makes possible operation with high-density fuel and more economical utilization of combustibles, allows higher coolant temperatures at reactor exit and in second stage, also makes higher fuel breeding ratios feasible. Two basic configurations of fast reactors with helium cooling are considered; two stages with steam-turbine cycle in the second stage (steam pressure 170 kgf/cm^2 and temperature 535-540°C, temperature at reactor exit 630-650°C) or single stage with gas turbine and regenerator (helium temperature at reactor exit 800-820°C). Construction of a pilot-industrial BGR-300 MW fast reactor with helium cooling and of the AES in which it will operate should and does provide a housing made of reinforced prestressed concrete, integrated assembly of main equipment in first stage, multiloop coolant circulation with large reserve, sectionalized heat power auxiliaries in first stage for expeditious replacement during overhaul, intermediate superheating of steam, high-speed gas injection with electric drive and speed regulation, and protective reactor envelope for containment of gas, in case of leakage from first stage, under sufficiently high pressure to allow for reactor cooldown during emergency. Figures 3, tables 2, references 6 Russian.
[295-2415]

UDC 621.362:621.383.5(088.8)

SOLAR CELLS WITH 'MIXTURE OF INDIUM AND TIN OXIDES - CADMIUM TELLURIDE' AND 'INDIUM OXIDE - CADMIUM TELLURIDE' STRUCTURES

Tashkent GELIOTEKHNIKA in Russian No 3, May-Jun 82 (manuscript received 20 Sep 81) pp 22-25

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[Abstract] The authors have produced solar cells with (99% In_2O_3+1% SnO_2)/p-CdTe and In_2O_3/p-CdTe structures by magnetron sputtering of the oxides on wafers of polycrystalline p-CdTe. The active area of these devices is 0.25-0.3 cm^2, the layer of oxide mixture having an energy gas $E_g \simeq 2.8$ V, electron mobility $\mu_n \simeq$ 20-40 $cm^2/V \cdot s$ and concentration of shallow donor centers $N=7 \cdot 10^{20}$ cm^{-3}. Their load and capacitance-voltage characteristics, spectral distribution of ph tosensitivity, and dependence of their no-load voltage and short-circuit current on the solar radiation flux density were measured, their dark current-voltage characteristics being exponential. The results indicate the feasibility of attaining 10% efficiency, with optimized geometry. The high series resistance corresponding to 0.35-0.4 surface coverage still needs to be explained. Figures 4, references 6: 2 Russian, 4 Western.
[298-2415]

UDC 621.362:621.383.5

EVAPORATOR-TYPE SYSTEMS FOR HEAT TRANSFER IN PHOTOVOLTAIC SOLAR ENERGY CONVERTERS

Tashkent GELIOTEKHNIKA in Russian No 3, May-Jun 82 (manuscript received 6 Jan 81) pp 14-18

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[Abstract] Three variants of an evaporator-type thermosiphon are described which have been designed for operation in a heat pipe at high thermodynamic efficiency and with minimum temperature drop from heat absorbing evaporator

surface to heat emitting condenser surface. These features are particularly useful for solar power plants in arid and mountainous regions with scarce water supply, where solar radiation is highly concentrated and the ambient temperature can fluctuate over a $\pm 40^{\circ}$ range. One evaporator is a finned tube with a thread at one end for screwing onto the solar cell, located at the focus of the concentrator mirror, and having a total active area of 1070 cm^2. The other two evaporators are plates with internal channels, the fin facing the sun, and have active areas of 2300 and 5000 cm^2 respectively. All three are made of aluminum, the first two produced by drawing from the melt and the third produced by diffusion welding of 0.5 mm thick sheets. Distilled ether is the coolant, with a boiling point of 304.4 K and other suitable physical properties. All three variants were tested with special equipment including an electrical simulator of thermal flux, after the ether had been degassed and the evaporator sealed against leakage. Performance characteristics were measured; temperature differential between evaporator zone and ambient medium, thermal resistance of radiators, and thermodynamic efficiency as functions of thermal input power. The results agree closely with calculations and compare favorably with the characteristics of a solid cruciform aluminum radiator. The temperature differential and the thermodynamic efficiency are highest for the first variant over the entire 10-110 W range of input power. The authors thank Zh. I. Al'ferov for interest, A. S. Kostygov for supplying the aluminum sections, and I. M. Isabekov for assisting with the measurements. Figures 3, references 5 Russian. [298-2415]

UDC 621.362:621.383.5

SILICON PHOTOVOLTAIC CELLS WITH GRAIN-ORIENTED SURFACE AND THEIR CHARACTERISTICS

Tashkent GELIOTEKHNIKA in Russian No 3, May-Jun 82 (manuscript received 26 May 81) pp 6-11

BORDINA, N. M., ZAYTSEVA, A. K., MARASANOVA, E. A. and POLISAN, A. A., Order of Labor Red Banner All-Union Scientific Research Institute of Current Sources

[Abstract] Texturizing the surface of photovoltaic cells with $\langle 100 \rangle$ -oriented silicon into a multipyramidal microrelief by selective etching in isotropic etchants increases their efficiency, because double and triple reflections decrease the resultant reflection coefficient while the long-wave sensitivity increases and the effect of hard radiation on degradation of the photocurrent weakens with a smaller depth of the light-absorbing layer. These reactions are examined theoretically on basis of the device geometry, a p-n junction along the jagged front surface and $p\text{-}p^+$ or $n\text{-}n^+$ junction along the flat rear surface. The resultant reflection coefficient and its spectral characteristic are calculated for surfaces with various angles characterizing the pyramidal pattern, also for the case of a glass cover plate on top with rubber cement filling the voids. The number of charge carriers collecting at p-n junction not deeper than 0.5 μm under the surface, particularly carriers coming from the base, is calculated assuming a maximally doped junction and taking into account changes

in the path of light due to the grain-oriented top surface. Also calculated are dependence of the resultant carrier collection factor on the length of the carrier diffusion path for various thicknesses of the base region (wavelength λ=0.95, 1 μm and absorption coefficient α=200, 80 cm^{-1}), and the spectral characteristic of sensitivity (μA/mW) of typical cells. The sensitivity is found to increase with a flat glass cover plate, especially in the shortwave range, which has been confirmed experimentally. Also the short-circuit current is higher with a flat glass cover plate. Figures 4, references 4: 3 Russian, 1 Western.
[298-2415]

UDC 62-52

OPTIMUM CORRECTION OF LASER GYROSCOPES IN ORIENTATION SYSTEMS

Leningrad IZVESTIYA VYSSHIKH UCHEBNYKH ZAVEDENIY: PRIBOROSTROYENIYE in Russian Vol 25, No 7, Jul 82 (manuscript received 1 Oct 81) pp 55-58

[Article by V. A. Besekerskiy and N. P. Pelevina, Leningrad Institute of Aviation Instrument Building]

[Text] The optimum Wiener correction of laser gyroscopes during initial setting of them and when held in a given direction is considered. The method of calculating the optimum filter is illustrated with a specific example.

In orientation systems containing laser gyroscopes [1], they must be corrected in the sense of initial setting and holding in a given direction. This is accomplished as in orientation systems containing any other gyroscopes [2] by using different direction sensors (pendulums, compasses, vertical gyroscopes and so on). The laser gyroscope itself is used in this case as a spatial memory device.

The ordinary correction circuit of a gyroscope for one of the channels is shown in Figure 1, a [2]. Here LG is a laser gyroscope, DN is the direction sensor, W. is the transfer function of the correction circuit, α and $\hat{\alpha}$ is the measured angular coordinate and estimate of it and v_1 and v_2 are noise.

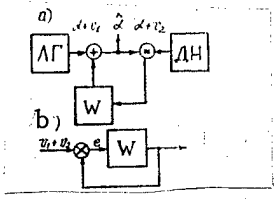

Figure 1. Correction Circuits

The transfer function for the measured coordinate is

$$H(p)=\frac{\hat{A}(p)}{A(p)}=\frac{1}{1+W}+\frac{W}{1+W}=1, \qquad (1)$$

where A(p) and $\hat{A}(p)$ are representations of the measured coordinate and estimates of it. It follows from formula (1) that the system guarantees that invariant measurement of the coordinate will be achieved.

The representation of the error $e(t) = \alpha(t) - \hat{\alpha}(t)$ at the circuit output is

$$E(p) = \frac{V_1(p)}{1+W} + \frac{WV_2(p)}{1+W}, \tag{2}$$

where $V_1(p)$ and $V_2(p)$ are representations of noise v_1 and v_2.

Formula (2) shows that the problem of optimum selection of transfer function W to minimize deviation of the error reduces to a Wiener problem. The equivalent circuit for calculation according to formula (2) is presented in Figure 1, b. Here v_1 is the analog of the useful signal at the input of the system and v_2 is noise. The error of this system is determined by formula (2).

Let us assume that the noise of a laser gyroscope is determined by three components [1, 3]:

constant deflection from start to start, which can be determined by the mean value of the rate of deflection Ω_0;

constant deflection Ω within a given start, whose value from start to start is random and is determined by the deviation of the rate of deflection D_Ω;

by the random component of the rate of deflection with spectral density $S_1(\omega)$ of the white noise type in some frequency band and by level $N\Omega$.

The first component of deflection Ω_0 can be eliminated during adjustment, but in the general case it can be compensated by using a correction circuit.

The noise of the direction sensor is determined by some spectral density $S_2(\omega)$. In most cases this noise can be reduced to white noise with level $N = S_2(0)$ in the bandpass of the correction circuit, which is very narrow (sometimes of a fraction of a hertz), but the optimization problem can also be solved by the considered method without this assumption. However, we shall limit ourselves here to this simplification.

Let us also assume that noise v_2 is centered, that is its mean value is equal to zero. If this condition is not fulfilled, the mean value of noise directly enters the error of estimate of $\hat{\alpha}$.

The spectral density of the rate of deflection of a laser gyroscope (in the effective frequency band of noise) is

$$S_\Omega(\omega) = 2\pi\Omega_0^2\delta(\omega) + 2\pi D_\Omega\delta(\omega) + N_\Omega. \tag{3}$$

The spectral density of the angle of deflection is

$$S_\alpha(\omega) = S_\Omega(\omega)/\omega^2 = \left[2\pi\left(\Omega_0^2 + D_\Omega\right)\delta(\omega) + N_\Omega\right]/\omega^2. \tag{4}$$

The total spectral density of the values of v_1 and v_2 is

$$S(\omega) = \left[2\pi\left(\Omega_0^2 + D_\Omega\right)\delta(\omega) + N_\Omega + N\omega^2\right]/\omega^2. \tag{5}$$

Let us represent it in the form

$$S(\omega) = N_{\Omega}[1 + a\delta(\omega)][1 + N\omega^2/N_{\Omega}]/\omega^2, \tag{6}$$

where

$$a = 2\pi(\Omega_0^2 + D_{\Omega})/N.$$

It is taken into account in this case that $\omega^2\delta(\omega) \equiv 0$. Further, (6) can be factored, as a result of which we find

$$[S(\omega)]^+ = \sqrt{N_{\Omega}}\left(1 + j\sqrt{a\delta(\omega)}\right)(1 + jN\omega/N_{\Omega})/j\omega. \tag{7}$$

According to the calculating formula for the optimum transfer function [4], we have

$$W(j\omega) = \frac{1}{\sqrt{N}}\{[S(\omega)]^+ - \sqrt{N}\} = \frac{K}{j\omega}\left(1 + j\sqrt{a\delta(\omega)}\right), \tag{8}$$

where the total amplification factor is $K = \sqrt{N_{\Omega}/N}$.

The resulting deviation of the error is

$$\overline{e^2} = \frac{1}{2\pi}\int_{-\infty}^{+\infty}\frac{S_{\alpha}(\omega)\,d\omega}{|1 + W(j\omega)|^2} + \frac{1}{2\pi}\int_{-\infty}^{+\infty}\left|\frac{W(j\omega)}{1 + W(j\omega)}\right|^2 N d\omega = \sqrt{N_{\Omega}N}. \tag{9}$$

It is obvious from the latter formula that the error deviation in an optimum system is not dependent on the constant components of the rate of deflection of a laser gyroscope, which indicates their total compensation, and is determined only by random components.

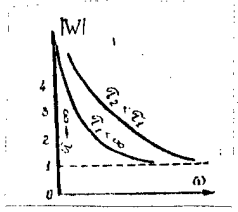

Figure 2. Amplitude-Phase Characteristics of Isodromic Device

Let us dwell on realization of the transfer function according to formula (8). The cofactor modulus in the parentheses is represented in the frequency function in Figure 2. The isodromic component with the following transfer function has this amplitude-frequency characteristic

$$W_{\text{и}}(p) = 1 + 1/(\tau p), \quad \tau \to \infty. \tag{10}$$

Characteristics at $\tau < \infty$ are also represented in this figure. It is natural that one must assume $\tau < \infty$ when realizing the correction circuit, which leads

to deviation from an optimum system, the error deviation increases, but results in a transient of finite length.

The resulting error deviation (9), upon substitution of (10) into formula (8), will be

$$\overline{e^2} = \sqrt{N_\Omega N}\left(1+\frac{1}{\tau K}\right) = \sqrt{N_\Omega N}\left(1+\frac{1}{\tau}\sqrt{\frac{N}{N_\Omega}}\right). \tag{11}$$

The inequality $\tau K >> 1$ must be observed to fulfill the condition of proximity of error deviation (11) to the optimum case (9). In this case the transient will be determined approximately by the two roots of the characteristic equation of a closed correction system: $p_1 = K$ and $p_2 = -\tau^{-1}$. The time of the transient is determined by the lesser absolute value of the root $-\tau^{-1}$:

$$t_n \approx -\tau/\ln\Delta, \tag{12}$$

where Δ is the relative value of the permissible residue after completion of the transient. At $\Delta = 0.05$ we have $t_p \approx 3$.

By varying the value of the constant time of the integrator τ, one can select the permissible time of the transient with the permissible increase of error deviation compared to the potential accuracy of (9).

Example. Let us consider the case of correction of a laser gyroscope from the vertical radar of a moving object [5]. Let us use a gyroscope of type ASLG-15 of the Sperry Company [3], which has the following input data: mean value of rate of deflection $\Omega_0 = 3$ angular s/s, deviation of rate of deflection $D_\Omega = 0.01$ (angular s/s)^2 and noise level of random component $N_\Omega = 3$ (angular s/s)^2/Hz, as the correcting laser gyroscope. Let us consider a system with two altimeters as the vertical gyro [6, 7]. The spectral density of the error of altitude measurement at velocity V = 100 m/s above the surface of a disturbed sea is shown in Figure 3, a. The spectral density of the error of measuring the inclination of the object with the altimeters located at distance R = 10 m: $S_2(\omega) = 2S_3(\omega)/R^2$, is shown in Figure 3, b.

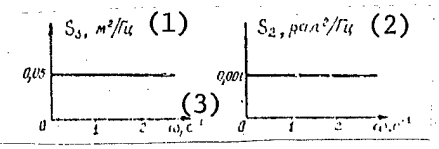

Figure 3. Graphs of Spectral Densities

Key:
1. m^2/Hz
2. rad^2/Hz
3. s^{-1}

It follows from Figure 3, b that one can assume N = 0.001 rad^2/Hz = $11.8 \cdot 10^3$ (angular second)^2/Hz.

According to the foregoing, we have the value of the general amplification factor

$$K = \sqrt{N_\Omega / N} = \sqrt{3/(11{,}8 \cdot 10^3)} = 0{,}016\,\text{s}^{-1}.$$

The potential accuracy of system (9) (the mean square error) is

$$\sigma = \sqrt{\overline{\varepsilon^2}} = \sqrt{N_\Omega N} = \sqrt{3 \cdot 11{,}8 \cdot 10^3} = 188 \text{ ang. s.} = 3.14 \text{ ang. min.}$$

If one assumes that $\tau K \sim 5$ and $\tau \sim 300$ second, then the mean square error increases to a value of

$$\sigma_1 = \sigma \sqrt{1 + 1/(\tau K)} = 3{,}14 \sqrt{1{,}2} = 3{,}44 \text{ ang. min.}$$

With automatic compensation of the rate of deflection $\Omega_0 = 3$ angular s/s to a value, for example, of 0.01 angular s/s, we have $\Delta = 0.01/3 = 0.0033$ and a time of the transient $t_p \sim -300/\ln\Delta = -300/\ln 0.0033 \sim 1{,}700$ s.

If the mean value of the rate of deflection is compensated for during adjustment, then the system should automatically compensate for the variable component of the constant rate of deflection. Having assumed its maximum value is $\Omega_{max} = 3\sqrt{D_\Omega} = 3\sqrt{0.01} = 0.3$ angular s/s, we have upon compensation of it to a value, for example, of 0.01 angular s/s, the maximum value $\Delta_{max} = 0.033$. In the worst case the maximum time of the transient is then

$$t_{n\,max} = -300/\ln 0{,}033 \approx 1000 \text{ s}.$$

If the transient must be accelerated, one must reduce the value of τ, which results in a further increase of the mean square error of orientation.

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CSO: 8144/1720

UDC 62-52

OPTIMUM CORRECTION OF LASER GYROSCOPES IN ORIENTATION SYSTEMS

Leningrad PRIBOROSTROYENIYE in Russian Vol 25, No 7, Jul 82 (manuscript received 1 Oct 81) pp 55-58

VESEKERSKIY, V. A. and PELEVINA, N. P., Leningrad Institute of Aircraft Instrument Design

[Abstract] Orientation systems with laser gyroscopes require correction of initial image display and subsequent holding in a given direction. The performance of such a system, with any type of direction sensor as corrective element and with the laser gyroscope acting as space memory, is evaluated here in terms of transfer functions for measured coordinate and for error image respectively. Interference in the laser gyroscope is due to drift which remains constant from start to start and can be determined from the mathematical expectation of drift velocity, drift which remains constant during a start but random from start to start and can be determined from the dispersion of drift velocity, and random component of drift velocity with spectral density of white noise at some intensity level within some frequency range. Interference in the direction sensor is also reducible to white noise, within the narrow correction band. The optimum transfer function of the isodromic corrective device is calculated on this basis and a practical procedure for its realization is outlined. With proper adjustment of the time integrator, it is possible to match allowable transient time with allowable error dispersion relative to attainable accuracy. This is demonstrated numerically on an ASLG-15 Sperry laser gyroscope, with two altimeters on a moving object above rough sea. Figures 3, references 7: 6 Russian, 1 Western.
[297-2415]

UDC 531.38

TIME CONSTANT OF GYROSCOPE WITH SPHERICAL HYDRODYNAMIC SUSPENSION

Leningrad PRIBOROSTROYENIYE in Russian Vol 25, No 7, Jul 82
(manuscript received 8 Jun 81) pp 59-62

ANDREYCHENKO, K. P., DANILOV, Yu. I. and SMARUN', A. B., Saratov Polytechnic Institute

[Abstract] A gyroscope with spherical suspension of the sensing element is considered, the spherical chamber and the spherical float rotating about their respective axes at constant angular velocity. The coefficient of viscous friction between float and incompressible fluid is evaluated as a function of the thickness of fluid layer between float and chamber, assuming laminar flow and any arbitrary ratio of this thickness to the "penetration depth" of a transverse wave. Theoretical evaluation involves calculating viscous forces and their moment. The results have been verified experimentally with a physical model of such a gyroscope and an inductive angle transducer generating the output signal and Lissajou figures on the screen of a cathode-ray oscilloscope indicating the angular velocity of the chamber. Figures 2, references 4: 3 Russian, 1 Western.
[297-2415]

UDC 531.383

EFFECT OF CHANGES IN GYROCOMPASS PARAMETERS ON TIME OF DETERMINING DIRECTION OF GEOGRAPHIC MERIDIAN

Leningrad PRIBOROSTROYENIYE in Russian Vol 25, No 7, Jul 82
(manuscript received 4 Jan 81) pp 62-65

NESTERENKO, T. G., Tomsk Polytechnic Institute imeni S. M. Kirov

[Abstract] The time of determining the meridian direction with a gyrocompass, i.e., the gyrocompass alignment time, which depends on the gyrocompass parameters, is determined from the equations of motion. For oscillatory motion with slight overregulation of the principal gyrocompass axis about the plane of the meridian these equations yield the misalignment angle as function of time where the misalignment contributed by the horizon indicator is negligible in the case of a gyrocompass with electromagnetic control. Expressions are derived for the sensitivity of gyrocompass alignment time to small deviations of the gyrocompass parameters (time constant and gain of horizon indicator, transfer ratios of horizontal and vertical correction channels, angular momentum) from nominal. On this basis the author calculates the resultant change in alignment time due to concurrent changes in these parameters. References 2 Russian.
[297-2415]

UDC 681.513.6:535

MULTICHANNEL NONCOHERENT ADAPTATION IN OPTICAL SYSTEMS

Leningrad IZVESTIYA VYSSHIKH UCHEBNYKH ZAVEDENIY: PRIBOROSTROYENIYE in Russian Vol 25, No 7, Jul 82 (manuscript recieved 23 Jun 80) pp 86-90

[Article by V. I. Samoylenko and I. V. Grubrin, Moscow Aviation Institute]

[Text] A new method of adaptation of optical systems that permits a considerable increase of the number of phase control channels of the light beam is proposed. The results of digital computer simulation are presented.

The phase front of the laser beam must be regulated in real time as the areas of application of laser equipment expand. The need to develop the corresponding control equipment is related to the distortions arising during propagation of laser emission in the atmosphere and during passage through the optical components of laser units [1].

It is rather easy to guarantee coherent emission in the optical band and coherent reception of a signal reflected from a target is very difficult. There are many reasons for this: atmospheric turbulence, the roughness of the reflecting surface for waves in the optical band, the absence of sensitive coherent detectors and so on. Therefore, the employed adaptation schemes most frequently utilize the index of quality of intensity of the signal being received.

The main method of adaptation during noncoherent reception is the multivibrator method [2]. A mirror, the N parameters of which may vary in such a manner that the intensity of the signal being received is maximized, is used to realize it. The simplest type of this mirror is a sectional mirror, each sector of which can be moved parallel to itself, changing the length of the beam path and thus introducing an additional phase shift. Shifting of the sectors is guaranteed so that the phase shift ϕ_i, introduced by the i-th sector, can be written in the following manner: $\varphi_i=\bar{\varphi}_i+\psi_i=\bar{\varphi}_i+\psi_0\cos\Omega_i t$, where $\bar{\phi}_i$ is the slow ("constant") component of phase variation, ψ_i is the modulation, fast component, ψ_0 is the index of phase modulation and Ω_i is the modulation frequency. As a result of this phase variation, the intensity modulation that specifically contains frequency Ω_i appears at the reception end. The value of the component of this frequency is isolated by a phase detector and is fed to the control circuit of the constant component $\bar{\phi}_i$. The value of $\bar{\phi}_i$ varies

until component Ω_i becomes close to (or equal to) zero in the signal being received. The maximum intensity with respect to variable ϕ_i corresponds to this. The quality of adaptation (the maximum intensity) is higher, the greater the number of adaptation channels N. However, the increase of the number of channels requires expansion of the range of modulation frequencies, which is limited. The lower bound of this range is determined by the spectrum of fluctuations which the adaptation system should process. Atmospheric turbulence, for example, has a spectrum that exceeds 10^3 Hz. Therefore, the lower bound of the modulation frequencies should not be less than 8-10 kHz. The upper bound is determined by the mechanical properties of the mirror sections and comprises 20-35 kHz. To process the fluctuations at frequency exceeding 1 kHz, it is necessary that the bandpass of the filters that separate the channels not be less than the spectrum of fluctuations, that is, 1 kHz. Thus, the number of channels is limited to a few dozen.

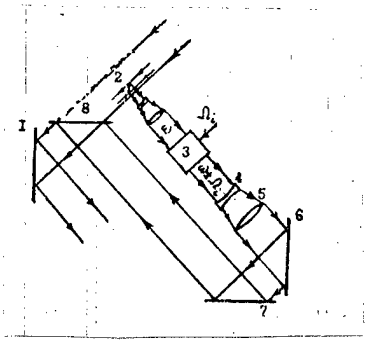

Figure 1.

Methods of expanding the modulation frequency band. Electronic phase modulators can be used at lowpower levels. The upper bound of the modulation frequency band can be increased by more than an order of magnitude in this case. However, electronically controlled phase shifters cease to function at the high outputs used in lidar [2]. The idea of splitting a small part of the output from the luminous flux, modulating the flux with electronic devices at low power level and then adding the modulated beam to the main beam of the channel occurs in this regard. A possible diagram of this modulation is shown in Figure 1. The controlled mirror 1, moving parallel to itself changes the disputed phase ϕ_i. Mirror 2 splits part of the luminous flux to a modulating device. Bragg cell 3 guarantees a frequency shift of the incident beam by $\pm\Omega_i$ (depending on polarization). The optical system 4, 5, 6 and 7 and the semitransparent mirror 8 guarantee coincidence of the beams. Thus, the beam reflected from mirror 1 consists of two components:

$$E_i(t) = a_i \exp j[\omega t - \omega\tau_i(t)] + b_i \exp j\{(\omega+\Omega_i)[t-\tau_i(t)] + \gamma_i\}, \quad (1)$$

where $E_i(t)$ is the instantaneous value of the field in a plane perpendicular to the direction of propagation, a_i is the field amplitude of the main beam, b_i is the field amplitude of the modulated beam, Ω_i is the modulation frequency, γ_i is the phase creep of the i-th modulating device and $\tau_i(t)$ is the time delay due to displacement of the i-th mirror. Having used the "brilliant"

point method [2], let us determine the intensity (square of the field amplitude) at the point of reception exact except for a coefficient. With regard to delay by propagation to the target and return $\tau_{pi}(t)$, which is dependent on time due to the motion of the target, and also due to atmospheric turbulence, the field presented to the detector is determined by substitution of the value $t - \tau_{pi}(t)$ instead of t into (1) and by summation with respect to all N channels:

$$E(t) = \sum_{i=1}^{N} \{a_i \exp -j\omega\xi_i(t) + b_i \exp -j[\omega\xi_i(t) - \Omega_i t + \Omega_i \xi i(t) - \gamma i]\} \exp j\omega t,$$

where

$$\xi_i(t) = \tau_{pi}(t) + \tau_i(t - \tau_{pi}(t)). \tag{2}$$

After simple transformations, we find the desired expression for field intensity:

$$\begin{aligned} I(t) = |E(t)|^2 = \sum_{i=1}^{N} \sum_{j=1}^{N} \{ & a_i a_j \cos \omega[\xi_i(t) - \xi_j(t)] + \\ & + 2a_i b_j \cos(\omega[\xi_i(t) - \xi_j(t)] + \Omega_j t - \Omega_j \xi_j(t) + \gamma_j)) + \\ & + b_i b_j \cos(\omega[\xi_i(t) - \xi_j(t)] - (\Omega_i - \Omega_j) t + \\ & + \Omega_i \xi_i(t) - \Omega_j \xi_j(t) - \gamma_i + \gamma_j)) \}. \end{aligned} \tag{3}$$

Such is the signal except for gain correction at the detector output. Further control should guarantee maximization of the mean value I(t), which, as can be seen from (3), is guaranteed if the following condition is fulfilled:

$$\omega\xi_i = \omega\xi_j + 2k\pi; \quad i, j = 1, 2, \ldots, N; \quad k = 0, \pm 1, \pm 2, \ldots, \tag{4}$$

where ξ_i is determined by expression (2). In this case $\overline{I}_{\max}(t) = (\sum_{i=1}^{N} a_i)^2$.

Let us separate the channels after the detector by an array of phase detectors. These detectors have a "sine" characteristic: if signals $A_j \sin \Omega_j t$ and $B_j \cos(\Omega_i t + \mu_i)$ are fed to its two inputs, respectively, then a voltage proportional to $A_j B_j \sin \mu_j$ forms at the output. All the remaining components, except Ω_j (in the ideal case), have no effect on the output signal. In this regard, the signal at the output of the j-th phase detector is determined only by the second term in (3), which contains the harmonic Ω_j:

$$v_j(t) = k_{dj} b_j \sum_{i=1}^{N} a_i \sin \psi_{ij}(t), \tag{5}$$

where k_{dj} is the transmission factor of the phase detector;

$$\psi_{ij}(t) = \omega[\xi_i(t) - \xi_j(t)] - \Omega_j \xi_j(t) + \gamma_j. \tag{6}$$

It was assumed when deriving the latter relations that the last term in (3) has no appreciable effect on the output signal of phase detectors due to the

smallness of b_i compared to a_i. Moreover, it is assumed that time variation of $\tau_i(t)$ (motion of the mirror) and of $\tau_{pi}(t)$ (motion of the target) does not contain harmonics Ω_i.

The model of an adaptive optical system can be designed on the basis of expressions (5) and (6). The input effects on this system are the vector of the time delays caused by shifting of the mirrors $T(t) = (\tau_1(t), \ldots, \tau_N(t))^T$ and the vector of time delays of signal propagation $T_p(t) = (\tau_{p1}(t) \ldots, \ldots, t_{pN}(t))^T$. The output effect is the voltage vector at the outputs of the phase detectors $V(t) = (v_1(t), \ldots, v_N(t))^T$. The relationship between T(t) and V(t) is determined by the vector differential equation:

$$\dot{x}=F(X, V), \tag{7}$$

where $X = (x_1, \ldots, x_N)^T$ is the vector of variables of state of the servo drive;

$$T=G(X). \tag{8}$$

Servo drive (7) and (8) can be designed such that the bandpass of the filters of the phase detectors is so much broader than the spectrum of input perturbations ($\sim$ 1 kHg) that these filters can be regarded as inertialess units. To do this, the modulation frequencies Ω_1 must be separated by several kilohertz one with respect to the other. The entire servo drive in each channel can then be regarded as a first-order unit. In this case T(t) can be selected as the variables of state and equations (7) and (8) can be written in linear form:

$$\dot{T}=QT+SV, \tag{9}$$

where Q is the matrix of the dynamics of the system and S is the matrix of effects.

Assuming that the interaction between the channels in the servo drive is negligible and assuming that the characteristics of the drives in each channel are identical, from (9) we find

$$\dot{T}=qT+sV, \tag{10}$$

where q and s are the parameters of the drives (the time constant and amplification).

Let us consider the simplest case of integrating drives for analysis, for which q = 0, and then

$$\dot{T}=sV. \tag{11}$$

If there exists a static mode for the case under consideration, then $\dot{T} = 0$ for it and it follows from (11) that V = 0. The following equivalent condition then ensues from expression (5):

$$\sum_{i=1}^{N} a_i \sin \psi_{ij} = 0, \quad j = 1, 2, \ldots, N.$$

Taking expression (6) into account, the latter relation can be transformed to the form

$$\sqrt{\left(\sum_{i=1}^{N} a_i \sin \omega\xi_i\right)^2 + \left(\sum_{i=1}^{N} a_i \cos \omega\xi_i\right)^2} \sin\left(-\omega\xi_j - \Omega_j\xi_j + \gamma_j + \right.$$
$$\left. + \operatorname{arctg} \sum_{i=1}^{N} a_i \sin \omega\xi_i \sum_{i=1}^{N} a \cos \omega\xi_i\right) = 0, \qquad j = 1, 2, \ldots, N.$$

The derived equality is fulfilled provided that:

$$\omega\xi_j + \Omega_j\xi_j - \gamma_j = \operatorname{arctg} \sum_{i=1}^{N} a_i \sin \omega\xi_i / \sum_{i=1}^{N} a_i \cos \omega\xi_i + k\pi,$$

where j = 1, 2, ..., N and k = 0, ± 1, ...

Since the right sides of the system differ only by the term $k\pi$, the desired condition of the state of equilibrium for the case of integrating drives under consideration is:

$$\omega\xi_j + \Omega_j\xi_j - \gamma_j = \omega\xi_i + \Omega_i\xi_i - \gamma_i + k\pi, \tag{12}$$

where i, j = 1, 2, ..., N and k = 0, ± 1, ± 2, ...

The results of digital modelling show that the static mode at k = 1, 3, ... is unstable. Let us compare system (12) with the derived condition of the maximum $\overline{I(t)}$ of (4). It is obvious that the terms $\Omega_m\xi_m$ and γ_m (m = 1, 2, ..., ..., N) in the latter expression are "spurious" terms that deflect the solution from the desired solution. The value of $\Omega_m\xi_m$ is easily compensated for if the corresponding phase is introduced into the reference signal of the phase detector since this term can be assumed known with a more or less precise knowledge of the range to the target. A precise knowledge of range is not required since $\omega >> \Omega_m$ and the term γ_m requires development of special methods of compensation.

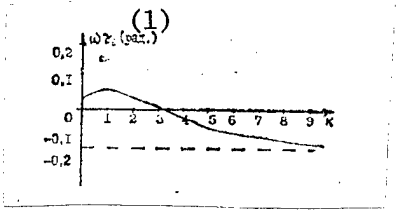

Figure 2.

Key:
1. Radians

Results of digital modelling. The work of the algorithms described at N = = 10, a_i = a and b_i = b was modelled on a digital computer. The effect of dispersion γ_i on the quality of adaptation (the field intensity at the target) was initially investigated for algorithm (11). The results of modelling are given in the table. Here σ_γ is the mean square deviation γ_i with respect to zero (identical for all channels).

(1)	1	0.7	0.5	0.3	0.1
(2)	63.8	39	21.3	8.2	1

Key:
1. Radian
2. Decrease of intensity, percent

It was found when modelling algorithm (11) that there is no steady state for $\gamma_i \neq 0$ or $\tau_i(t)$. The intensities $\tau_i(t)$ continue to increase at a constant rate after the maximum value was achieved. This forced us to turn to algorithm (10) (with an inertial drive). The similarity of the transient process to a stable state close to (12) at any initial values (γ_i and $\tau_i(0)$), but at specific values of q and s that satisfy the condition of stability, was proved as a result of modelling. The distortions introduced by the atmosphere and the range to the target were assumed constant during the adaptation time (i.e., $\tau_{pj}(t)$ = const) in all the experiments. The typical transient process under these conditions is presented in Figure 2, where k is the iteration number.

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CSO: 8144/1721

UDC 662.997:537.22(088.8)

OPTICAL CHARACTERISTICS OF SELECTIVE COATINGS ON HEAT-TYPE SOLAR RADIATION COLLECTORS

Tashkent GELIOTEKHNIKA in Russian No 3, May-Jun 82 (manuscript received 17 Nov 81) pp 35-39

KOLTUN, M. M. and GAVRILOVA, I. P., Order of Labor Red Banner All-Union Scientific Research Institute of Current Sources

[Abstract] Optical characteristics of multilayer selective coatings for metal surfaces of solar radiation collectors have been calculated on an M-4030 computer by methods yielding the spectral distribution of the reflection coefficient, the absorption coefficient under conditions in outer space and on earth, and the normal emissivity over the 27-500°C temperature range. These calculations were done for coatings produced by three most effective methods: 1) vacuum deposition of alternate dielectric (ZnS) and metal (ni) films on silver, copper, or aluminum base; 2) electrochemical deposition of black (Ni) layers on copper or nickel strike; 3) simultaneous vacuum evaporation of dielectric and metal, or their mixture, to produce cermet films. In addition to the reflection spectra the authors also calculated the ratio of absorption coefficient to emissivity as a function of coating thickness, a ratio larger than 30 being found attainable with coatings produced by any of the three methods, electrochemically deposited and cermet coatings generally having a larger ratio with larger total thickness and thus yielding better optical selectivity. Figures 4, references 12: 8 Russian, 4 Western.
[298-2415]

UDC 666.192.535.372.343

EFFECT OF REACTOR AND GAMMA RADIATION ON OPTICAL PROPERTIES OF QUARTZ GLASS

Moscow ATOMNAYA ENERGIYA in Russian Vol 53, No 1, Jul 82
(manuscript received 7 Jul 81) pp 42-44

ABDUKADYROVA, I. Kh.

[Abstract] An experimental study was done to evaluate the effect of the γ-component of mixed reactor radiation on optical properties of quartz glass, actually four types of quartz glass with different absorption bands. Measurements were made by the structurally sensitive method of electron-paramagnetic resonance, with a ^{60}Co source of γ-radiation and with recording of the ordinary axisymmetric E'-center signal. Photoluminescence spectra were recorded for determining the dependence of intensity peaks on the neutron flux density from $5 \cdot 10^{17}$ to $2 \cdot 10^{19}$ neutrons/cm^2 and on the γ-radiation dose from 10^3 to 10^8 gR. These data yield information about the kinetics of ab rption and emission, buildup of optical density in quartz glasses, buildup of radiation defects, breakup of the Si-O bond with attendant formation of Si and Si-O defects, and migration of added metal impurities. Figures 3, table 1, references 5: 2 Russian, 3 Western.
[299-2415]

UDC 662.997

COEFFICIENT OF CONVECTIVE HEAT TRANSFER AT SURFACE OF HEAT EXCHANGER WALL IN PASSIVE SOLAR HEATING SYSTEM UNDER CONDITIONS OF TURBULENCE

Tashkent GELIOTEKHNIKA in Russian No 3, May-Jun 82 (manuscript received 4 Mar 81) pp 43-46

AVEZOV, R. R. and BABAKULOV, K. B., Physico-Technical Institute imeni S. V. Starodubtsev, UzSSR Academy of Sciences, Termez State Pedagogical Institute imeni M. T. Aybek

[Abstract] The coefficient of convective heat transfer at the surface of a vertical heat exchanger wall in a passive solar heating system is determined for conditions of turbulence, this coefficient being needed for correct estimation of the heat carried away by the coolant. It is calculated according to Newton's law $\alpha = q_w/\Delta t$ (q_w - thermal flux at wall surface, Δt - temperature drop from wall to coolant) for an impermeable surface on the basis of the Reynolds analogy, with the correction factor $Pr^{-0.67}$. Next the velocity of coolant in the boundary layer is calculated from the relation

$$\delta/x = 2.96/[Re_x^{0.17}(\frac{Pr^{0.67}}{2.14+Pr^{0.67}})Pr^{0.17}]$$

(x - coordinate in direction of coolant flow). The hydraulic drag coefficient is calculated from the Darcy-Weisbach relation, with twice the thickness of the boundary layer used as the fictitious equivalent diameter. The final expression for the heat transfer coefficient is $\alpha = 1.34t^{0.33}$ and the thermal flux at the wall surface is $q_w = 1.34\Delta t^{1.33}$. The $q_w = f(\Delta t)$ relation can be approximated with a straight line, and for practicable temperature drops (0-20°C) $q_w = 3.05$ t with a ± 3.24 rms error. The resultant heat transfer coefficient is 6.3% smaller than for convective heat transfer in an unbounded volume. Figure 1, references 5 Russian.
[298-2415]

UDC 539.3

VARIATIONAL FORMULATION OF CONTACT PROBLEM FOR PHYSICALLY NONLINEAR SHALLOW SHELLS

Kiev DOKLADY AKADEMII NAUK UKRAINSKOY SSR: SERIYA A: FIZIKO-MATEMATICHESKIYE I TEKHNICHESKIYE NAUKI in Russian No 6, Jun 82 (manuscript received 29 Jun 81) pp 45-47

L'VOV, G. I., Kharkov Polytechnic Institute

[Abstract] Interaction of a perfectly rigid punch and a shallow thin shell made of material with nonlinearly elastic reinforcement is treated as a contact problem to which the strain theory of plasticity applies. The median surface of the shell is described by an equation in parametric form which maps it homeomorphically onto a plane region constituting a finite sum of bounded stellate regions and bounded by a finite number of Lyapunov Contours. Solution of the deflection problem involves determining the displacement vector-function for points of the median surface in that plane which will satisfy three systems of equilibrium equations in forces and moments, with boundary conditions of rigid clamping, with the small-displacement constraint on linearizability, and with strains and stresses determined by physical relations in the theory of shallow shells made of incompressible material. The problem is reduced to a variational one by use of the Sobolev space for the displacement vector-function. Solution of the original problem thus reduces to solution of the equivalent inequality for this vector-function, namely minimization of the corresponding convex differentiable nonlinear functional with aid of the Cauchy-Bunyakovskiy inequality. Existence and uniqueness of the solution are proved by demonstrating the coercivity of this functional. References 7 Russian.
[294-2415]

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