Russian Original Vol. 54, No. 4, April, 1983

October, 1983

JOT
JAC
RER
DS
RLH

SATEAZ 54(4) 243–318 (1983)

SOVIET ATOMIC ENERGY

АТОМНАЯ ЭНЕРГИЯ
(ATOMNAYA ÉNERGIYA)

TRANSLATED FROM RUSSIAN

CONSULTANTS BUREAU, NEW YORK

SOVIET ATOMIC ENERGY

Soviet Atomic Energy is abstracted or indexed in *Chemical Abstracts, Chemical Titles, Pollution Abstracts, Science Research Abstracts, Parts A and B, Safety Science Abstracts Journal, Current Contents, Energy Research Abstracts,* and *Engineering Index.*

Mailed in the USA by Publications Expediting, Inc., 200 Meacham Avenue, Elmont, NY 11003.

POSTMASTER: Send address changes to *Soviet Atomic Energy,* Plenum Publishing Corporation, 233 Spring Street, New York, NY 10013.

Soviet Atomic Energy is a translation of *Atomnaya Énergiya,* a publication of the Academy of Sciences of the USSR.

An agreement with the Copyright Agency of the USSR (VAAP) makes available both advance copies of the Russian journal and original glossy photographs and artwork. This serves to decrease the necessary time lag between publication of the original and publication of the translation and helps to improve the quality of the latter. The translation began with the first issue of the Russian journal.

Consultants Bureau journals appear about six months after the publication of the original Russian issue. For bibliographic accuracy, the English issue published by Consultants Bureau carries the same number and date as the original Russian from which it was translated. For example, a Russian issue published in December will appear in a Consultants Bureau English translation about the following June, but the translation issue will carry the December date. When ordering any volume or particular issue of a Consultants Bureau journal, please specify the date and, where applicable, the volume and issue numbers of the original Russian. The material you will receive will be a translation of that Russian volume or issue.

Subscription (2 volumes per year)
Vols. 52 & 53: $440 (domestic); $489 (foreign) Single Issue: $100
Vols. 54 & 55: $500 (domestic); $555 (foreign) Single Article: $7.50

CONSULTANTS BUREAU, NEW YORK AND LONDON

233 Spring Street
New York, New York 10013

Published monthly. Second-class postage paid at Jamaica, New York 11431.

SOVIET ATOMIC ENERGY

A translation of *Atomnaya Énergiya*

October, 1983

Volume 54, Number 4 April, 1983

CONTENTS

The Russian press date (podpisano k pechati) of this issue was 3/24/1983. Publication therefore did not occur prior to this date, but must be assumed to have taken place reasonably soon thereafter.

NUCLEAR POWER IN THE USSR

A. P. Aleksandrov, A. S. Kochenov, E. V. Kulov, A. G. Meshkov, V. P. Ryazantsev, and V. A. Sidorenko

UDC 621.039(47-57)

The Soviet Union has no equal as regards reserves of organic fuel. The total reserves of coal in the USSR are $5.5\text{-}6 \times 10^{12}$ tons and constitute about half of world reserves. In 1981, the volumes of organic fuel production were as follows: oil (including gas condensate) 609 million tons (first place in the world), gas 465 billion m^3 (second place in the world), and coal 704 million tons (second place in the world).

The development of power engineering in the USSR is supported by our own resources. Also, oil and gas are exported to countries in Eastern and Western Europe. On the other hand, most of the organic fuel reserves are located in the Asiatic part of the country, while four-fifths of the fuel demand lies in the European part. For this reason, the shipping of fuel from eastern regions into western ones constitutes about 40% of the total goods carried by the country's railroads.

In recent years, the Asiatic part has provided almost all the increase in the extraction of organic fuel. Therefore, if power engineering developments were to be based only on the use of organic fuel, there would be an increasing disproportion in the location of the extraction and use of fuel. There would be also an appreciable increase in the costs for transporting fuel resources to the European part of the country. This disproportion can be substantially relieved by developments in nuclear power. Also, nuclear power enables a reduction in the cost of producing electricity in the European part, while relieving railroads from transport load and improving labor productivity mainly by reducing the number of workers required in the extraction industry and in transportation, while also modifying the fuel and power balance by reducing the proportion accounted for by oil and gas.*

Oil and gas cannot remain in the basis of the world's fuel and power for long, since reserves are limited, and therefore oil and gas should be considered primarily as valuable chemical raw materials and eliminated as far as possible from the fuel balance [1].

There are also difficulties with other traditional forms of fuel in many parts of the world. On the other hand, the increasing nuclear power during the 1970s was much less than was forecast at the end of the 1960s. Major factors that hold back development at the present time are the following: inadequate development in specialized engineering, the lack of a practical solution to fuel breeding, delay in certain aspects of fuel reprocessing and in storage of high-activity wastes, and finally lack of preparation in public opinion. On the other hand, the efforts being made in certain countries lead one to believe that all these problems will be resolved in the next 10-15 years

In the Soviet Union, the first designs for nuclear power stations began to be devised at the end of the 1940s. In 1950 it was decided to construct the first nuclear power station at Obninsk on the basis of a channel uranium-graphite reactor. This was commissioned on June 27, 1954. Operating experience showed it was reliable and safe for the staff and for the surrounding population. This gave a clear demonstration that nuclear power can be used to produce electricity.

The State program for the development of nuclear power in the USSR could not be based on nuclear power stations of a single type, since this would not provide the necessary reliability and stability. On the other hand, the exploitation of each type of power reactor

*In this section the editors publish in journal form some of the papers presented by Soviet researchers at the IAEA International Conference on Experience Accumulated in Nuclear Power (Vienna, September 13-17, 1982). In addition to the papers, the editors print two surveys of the current state of foreign nuclear power engineering.

Translated from Atomnaya Énergiya, Vol. 54, No. 4, pp. 243-249, April, 1983.

0038-531X/83/5404-0243$07.50

on a commercial scale requires a considerable time interval and considerable financial and material resources.

A scientific program was drawn up by the State Committee on the Use of Atomic Energy in the USSR to define the most suitable and economical effective power reactors for this country. The program envisaged research on various power reactors: pressurized-water and boiling-water ones, channel uranium—graphite ones, reactors with organic moderators and coolants, etc. Much attention was given to fast reactors, since from the start it was clear that large-scale nuclear power is inconceivable without them.

During the research, some forms of power reactor were not brought to the stage of prototype building for various reasons. It was found, e.g., that reactors with organic moderators and coolants are suitable only for low-power stations. Research in this area led to the creation in 1963 of a block-transportable nuclear power station of electrical output 750 kW. On the other hand, research on certain types of thermal power reactor have led to the building of prototype commercial units. A water-cooled and water-moderated reactor (the VVÉR-210) was built at Novyi Voronezh nuclear power station with an electrical output of 210 MW, while a pressurized-water reactor (the VK-50) was built at Dmitrovgrad with electrical output 50 MW, and at Beloyarsk nuclear power station a channel uranium—graphite reactor was built (with nuclear steam superheating and an efficiency of 36%).

In the course of the fast-reactor program, several experimental reactors and test facilities were constructed for simulating and researching the physical and engineering characteristics. The first experimental reactor with plutonium fuel was built in 1955. In the later reactors, the fuel has been metallic plutonium, plutonium dioxide, uranium monocarbide, and uranium dioxide. The main attention has been given to designs with sodium cooling. The power outputs of the experimental reactors have gradually increased. In 1969, an experimental reactor was commissioned at Dmitrovgrad that employed fast neutrons with sodium cooling — the BOR-60 (electrical power 12 MW).

The types of power reactor optimum for this country were defined during the development, construction, and operation of these prototype units. During the second half of the 1960s, the accumulated experience led to the decision to develop nuclear power engineering on the basis of two types of thermal reactor: pressurized-water ones and channel uranium—graphite ones cooled by boiling water, since at that time several VVÉR units had been built together with the second unit for the uranium—graphite channel reactor at Beloyarsk power station. The decision provided, firstly, the fullest use of the country's engineering facilities and, secondly, a more flexible design for the fuel cycle, since in particular the RBMK reactors can utilize fuel from the VVÉR. The VVÉR-440 and RBMK-1000 units were adopted as the standard ones.

The first two units at Novyi Voronezh power station were prototypes for nuclear power station units containing VVÉR-440 [2]. The units are highly reliable and the load factors

TABLE 1. Working Parameters of Nuclear Power Stations with VVÉR Units in Individual Year

Power station	Year	Installed power, MW	Energy production, billion kW·h	Load coefficient, %
Novyi Voronezh	1979	1409	9,92	78,9
	1980	1409	11,35	82,7
	1981	2409	15,56	73,7
Kola	1979	880	5,90	76,5
	1980	880	7,22	93,5
	1981	880/1320 *	7,39	69/83 *
Armenian	1979	407,5	2,39	66,8
	1980	815	4,74	66,3
	1981	815	5,50	77,0

*The denominator gives the value without allowance for the introduction of the new unit.

are high (about 80%, and appreciably higher in some years), which provided a basis for the design of routine units. In the development of the VVÉR-440, the equipment was substantially upgraded, and major changes were made in the reactor design. The first two units containing the VVÉR-440 were also installed at Novyi Voronezh power station. Minor changes were made to the designs of the subsequent standard units. Subsequently, similar units were introduced at other nuclear power stations in the country (Kola, Armenian, and Rovensk), as well as in the German Democratic Republic, Bulgaria, Czechoslovakia, and Finland. At the Armenian power station, the equipment in the first loop was designed to withstand seismic shocks.

Table 1 gives some parameters of nuclear power stations containing VVÉR-440. The high load coefficients are due to the successful design of the main unit.

The length of a fuel cycle in the VVÉR-440 is 3 yrs, with intervals of 1 yr between partial reloads. The annual partial reload along with planned prophylactic maintenance occupies about 30 days. Periodic checks are made on the state of the main equipment in the first loop, and, in particular, the pressure-vessel metal is checked once every 4 yrs with the core and devices within the pressure vessel unloaded. The length of the shutdown necessary for checking the pressure-vessel metal is about 60 days. Some VVÉR vessels have been made of heat-resisting steel without anticorrosion stainless-steel coating. The first such vessel was installed in the second unit at Novyi Voronezh power station. The ammonia-potassium water treatment provides a satisfactory corrosion state in the pressure-vessel metal.

The work on upgrading the VVÉR-440 led to the creation of the VVÉR-1000 reactor system, of which the first unit was commissioned at Novyi Voronezh power station. The following problems were overcome in the design of this: The equipment must be transportable by railroad, the economic parameters should be improved, and the units should meet the latest safety requirements.

Transportability for the pressure vessel restricted the diameter to 450-460 cm, with the diameter of the core zone, correspondingly, 310-320 cm. The energy production density in the core is 110 kW/liter or 30% higher than that in the VVÉR-440. This requires special measures to equalize the distribution. The main circulation loop was enclosed in a protective shield of prestressed reinforced concrete designed for a pressure of 0.55 MPa. Population safety was also provided for instantaneous failure in DU-850 pipeline coinciding in time with the completely idle state of the power station.

Commissioning operations and operating experience with a fifth unit at Novyi Voronezh power station confirmed that the basic designs were correct. On the other hand, certain changes were made in the design of the VVÉR-1000 to be located at the South Ukraine, Kalinin, and Rovensk power stations. Projects for power stations containing VVÉR-1000 were devised for regions with seismicity 5-6 points on the MSK 1964 scale. No substantial changes are proposed in the VVÉR-1000 reactor system for nuclear combined heat and power stations.

Table 2 [3] gives some parameters of nuclear power stations containing RBMK-1000 units. At such power stations, the fuel is changed on load by a charging and discharging machine. Experience has shown that about 1000-1500 such operations can be performed at such a power

TABLE 2. Working Parameters of Nuclear Power Stations with RBMK-1000 Units in Individual Years

Power station	Year	Installed power, MW	Energy production, billion kW•h	Load coefficient, %
Leningrad	1979	2000	13,1	74,4
	1980	3000	18,82	71,4
	1981	4000	24,1	73,8
Kursk	1979	2000	10,35	64,1
	1980	2000	13,89	79,1
	1981	2000	13,54	77,3
Chernoby	1979	2000	12,23	69,8
	1980	2000	14,21	80,9
	1981	2000	13,44	75,2

station during a calendar year. The condition of continuous reloading enables one to approximately double the burnup by comparison with the state of simultaneous reloading for the entire core.

The experience with the RBMK-1000 confirmed preliminary theoretical conclusions that the void and temperature coefficients of reactivity would increase as the fuel is burned and absorbers are extracted from the core, which reduces the stability of the energy production distributions. Efforts to stabilize these distributions were concentrated on improving the level of automation by means of a branched controlled system for the reactor and changing the fuel isotope composition. A new local automatic-control system was devised and implemented, which showed high reliability and performance. This provided characteristic deformation times for the energy-production patterns of not less than 6-8 h, which do not cause any difficulty in managing the reactor.

In accordance with these studies, the initial fuel enrichment was raised to 2%. This not only improved the dynamic performance of the reactor, but also raised the economic parameters by increasing the extent of burnup and reducing the specific fuel consumption.

On account of elevated safety specifications, special systems were designed to provide acceptable temperature conditions in the fuel pins and to localize the escape of coolant on failure of any pipeline (including a maximum diameter one of 900 mm). Such safety systems have already been implemented at the Leningrad power station and are envisaged for all power stations that are being constructed with RBMK-1000 units.

The successful operation of the RBMK-1000 at its nominal power revealed considerable reserve margins in the design, so it was possible to incorporate heat-transfer intensifiers in the fuel-pin assemblies to increase the power in each channel by a factor 1.5 without changing the dimensions or numbers of the fuel channels. The design of the new fuel-pin assembly for the reactor, namely RBMK-1500, enables one to increase the thermal loads while maintaining a high level of unification with the RBMK-1000 fuel-pin assemblies.

The first section of the Ignala power station with two RBMK-1500 units is now being built. The commissioning of the head unit will lay the basis for the new generation of channel reactors, which are more economical and should replace the well-recommended RBMK-1000. The construction of power stations containing the RBMK-1500 will reduce the specific capital cost by comparison with the RBMK-1000 and will reduce the costs assigned to the electricity.

In the next stage of development for channel power reactors, one may go to the design of sectional block reactors with nuclear steam superheating (RBMKP) with unit powers of 1200 and 2400 MW. The gross efficiency of a nuclear power station containing RBMKP-1200 and -2400 units is expected to be about 37%.

The building of nuclear power stations containing thermal reactors has been accompanied by the construction of two commercial units containing fast reactors: the BN-350 and the BN-600 [4]. Over nine years have elapsed since the power commissioning of the BN-350. The only major defect in the equipment has been repeated failure of the sealing between loops in the steam generators. After the completion of repair in the damaged steam generators, the reactor power was raised to 520 MW (thermal) in 1975, 650 in March, 1976, and 700 in September, 1980, which provides an electrical power of 125 MW and a daily production of 85×10^3 tons of distillate. The design burnup of 5% of heavy atoms was attained in 1976. At present, the burnup is being maintained at the level of 5.8% of the heavy atoms, which is related to the acceptable shape changes in the six-faced jackets of the fuel-pin assemblies. During the operation, the fuel pins were unified on outside diameter and sheath thickness with the pins in the BN-600. The resulting increase in the gas compensating volume reduced the pressure in the sheath, which reduced the number of cases of sheath failure by an order of magnitude.

The BN-600 power unit differs from the BN-350 in having and integral (tank) style for the equipment in the first cooling loop. Steam generator of modular type also improve the reliability. The run-up to power began in April, 1980, and by December 18, 1981, the reactor was brought up to the nominal power of 1470 MW (thermal). On January 1, 1982, the unit containing the BN-600 had produced 3.7 billion kW•h, and it had operated at power for over 10^4 h. The reactor is readily controlled. Four fuel changes have been made during the operation. The maximum fuel burnup attained 7% of the heavy atoms. On the whole, the operating experience has confirmed that the actual parameters correspond to the design ones.

TABLE 3. Major Operating Nuclear Power Stations in the USSR

Power station	Electrical power of units, MW	Reactor type	Dates of major stages		
			physical commissioning	energy commissioning	reached nominal power
Novyi Voronezh	1st unit — 210	VVÉR	12.63	09.64	12.64
	2nd » — 365	»	12.69	12.69	04.70
	3rd » — 417	»	12.71	12.71	06.72
	4th » — 417	»	12.72	12.72	03.73
	5th » —1000	»	04.80	05.80	02.81
Beloyarsk	1st » — 100	UGR	09.63	04.64	09.67
	2nd » — 200	»	10.67	12.67	12.69
	3rd » — 600	BN	02.80	04.80	12.81
Kola	1st » — 440	VVÉR	06.73	06.73	12.73
	2nd » — 440	»	11.74	12.74	02.75
	3rd » — 440	»	02.81	03.81	
Leningrad	1st » —1000	RBMK	09.73	12.73	11.74
	2nd » —1000	»	05.75	07.75	01.76
	3rd » —1000	»	09.79	12.79	06.80
	4th » —1000	»	12.80	02.81	08.81
Armenian	1st » — 407,5	VVÉR	12.76	12.76	10.79
	2nd » — 407,5	»	01.80	01.80	05.80
Kursk	1st » —1000	RBMK	09.76	12.76	10.77
	2nd » —1000	»	12.78	01.79	08.79
Chernobyl'	1st » —1000	RBMK	08.77	09.77	05.78
	2nd » —1000	»	09.78	12.78	05.79
	3rd » —1000	»	06.81	12.81	06.82
Rovensk	1st » — 440	VVÉR	12.80	12.80	
	2nd » — 440	»	12.81	12.81	

The building and commissioning of nuclear power station units containing BN-350 and BN-600 is an important stage in solving the problem of fuel breeding, whose final purpose will include the design of a standard fast-reactor unit to be built on a large scale.

As of December 31, 1981, the installed power of nuclear power stations in the USSR was about 16 GW (Table 3). The production of electrical energy at nuclear power stations in 1980 was 73 billion kW•h, as against 86 in 1981.*

The safety of nuclear power stations in use and under construction in the USSR [5] is supported by a wide range of measures, of which the following are the main ones:

1) high equality in equipment manufacture and installation;

2) state monitoring for the equipment at all stages in use;

3) the definition and implementation of efficient protective measures to prevent emergencies, compensate for any faults arising, and reduce the consequences of possible emergency situations;

4) the definition and implementation of facilities for localizing radioactive materials in the case of an emergency;

5) logical execution of all engineering and organizational measures to provide for safety at all stages in the building and operation of nuclear power stations;

6) standardization of engineering and organizational aspects in the provision of safety; and

7) the Government supervision system.

Throughout the period of nuclear power station operation in this country, there have been no instances where an emergency has led to the need to take measures to protect the population, although much attention has always been given to preparing such measures. For ex-

*At the start of 1983, the installed nuclear power station capacity in the USSR was 18 GW (electrical), and the electrical energy production was about 100 billion kW•h — Editor.

TABLE 4. Releases of Gases and Aerosols Containing ^{131}I from Two Units of Nuclear Power Stations Containing VVÉR-440 and RBMK-1000 Units

Release	Year	Kola	Novyi Voronezh	Kursk	Chernobyl'
Gas, Ci/yr	1979	$2\cdot10^3$	$2.4\cdot10^3$	$67.9\cdot10^3$	$133\cdot10^3$
	1980	$2\cdot10^3$	$2.13\cdot10^3$	$88.9\cdot10^3$	$280\cdot10^3$
^{131}I, mCi/yr	1979	1	4	66	290
	1980	1	11	458	5000

ample, for each nuclear power station there is a plan of measures to protect the staff and population in the case of hypothetical emergencies going outside the framework of the design ones.

A comparison of the health risks for the staff and population in the production of electrical power by nuclear power stations and thermal ones indicates that the former are preferable. When organic fuel is used, there is considerable environmental pollution from release of ash and gases. Current coal-fired stations consume over 2×10^6 tons of coal per GW (elec.). This results in about 4×10^5 tons of ash, of which about 8×10^3 tons is released into the atmosphere. The sulfurous gases are particularly harmful, and these constitute tens of thousands of tons per GW (elec.). A difference of a nuclear power station from a coal one is that there is no such release. Also, a nuclear power station does not consume oxygen from the atmosphere.

Observations have been performed for many years on nuclear power stations containing VVÉR-440 units, not only in this country, but also in the German Democratic Republic, Bulgaria, Czechoslovakia, and Finland, as well as on stations with RBMK-1000 units, and these indicate that radiation safety for the staff and population is reliably provided. The levels of penetrating radiation at permanently staffed and largely unstaffed locations do not exceed 1.4 and 2.8 mrem/h, which have been set as the permissible levels. The average individual dose of external radiation to the staff does not exceed 15-18% of the maximum permissible for a year, which is 5 rem/yr. The release listed in Table 4 are substantially below the level permitted by the health rules of 1979 and are 183 Ci/MW (elec.). yr on gases and 3.65 mCi/MW (elec.). yr on ^{131}I. With such small releases to the environment, the radiation background in the locality is determined by natural sources of ionizing radiation as well as by the artificial radionuclides formed by nuclear weapons tests. It is impracticable to distinguish the radionuclides derived from the station against the background of the global radionuclides. The γ-ray dose outside the nuclear power station does not increase with operating time and does not vary with distance from the station.

In the Soviet Union, nuclear power is considered as a very important means of solving major problems in the fuel and energy balance over a long period. The safe and reliable operation of existing nuclear power stations goes with their minimal effects on the environment and their high economic performance, and at the 26th Congress of the CPSU a decision was therefore taken to provide for the increase in production of electricity in the European part of the country mainly by the construction of nuclear power stations and hydroelectric ones. Correspondingly, nuclear power stations are being constructed at over 20 sites and will gradually displace base-load stations employing organic fuel in the northwest, west, center, and south of the European part of the country. Nuclear power stations are being constructed along the Volga and in the Ural. The installed power of the individual stations is 4-6 GW. The rise in installed nuclear power station output in 1981-1985 is planned to be provided mainly by the introduction of RBMK-1000 and VVÉR-1000 units.

This decision substantially eases the problems in the fuel and energy balance. On the other hand, less than 25% of the energy resources go to the production of electricity, and during the next five-year period nuclear power stations can provide electrical energy only to base-load users in the European part of the country, so the contribution of nuclear power to the fuel and energy balance can hardly exceed 10-15%. This means that nuclear power stations, while substantially alleviating the problem of the power and fuel balance, cannot provide a radial solution. Solutions to the problems can come only from substantial extension of nuclear-power applications.

In the USSR, about 20% of the organic fuel used is employed in centralized heat supply, and this is mainly the scarcest form, namely gas and oil. The main users of centralized heat supply are located in the European part of the country, i.e., in regions furthest from the sources providing for an increment in organic fuel production. Therefore, the extension of nuclear power to centralized heat supply is a major task in solving the fuel and energy problems. The first steps have already been taken. The Bilibinsk combined heat and power station has been operated since 1973 and provides heat to the population, while the Shevchenko nuclear power station provides fresh water, and the heat from the Beloyarsk, Leningrad, Kursk, and Chernobyl' nuclear power stations is also utilized.

To reduce the consumption of organic fuel in centralized heat supply, two major stations have been built for domestic heat supply close to the cities of Gor'kii and Voronezh, which will supply users with hot water. These heat stations are reliably protected from accidents such as explosions, aircraft crashes, etc. It is impossible for radioactivity to reach the users, because there is an intermediate circuit in which the pressure in the coolant is less than that in the heat-user circuit. These features of the heat-supply stations make them a reasonably powerful (300-500 MW) and safe source of heat supply, which can be located in major inhabited areas. Under these conditions there is no need to lay long and expensive heat-carrying pipes.

The first major combined heat and power station is being constructed near Odessa, in which the production of heat will be accompanied by the production of electrical power. The energy source is provided by a VVÉR-1000 reactor. Studies are being made on the scope for building nuclear power stations to supply steam for industrial purposes.

The rates of introduction of nuclear heat sources are planned to increase substantially in subsequent five-year periods.

Over 15% of the organic fuel is used directly in industry, including chemistry, metallurgy, etc. The introduction of high-temperature reactors will further extend the use of nuclear power, including the production of synthetic fuel.

Of course, it is essential that the general use of nuclear power in branches of the economy using substantial amounts of energy must be reliably supported with nuclear fuel. With the existing thermal reactors, the energy yield from a ton of natural uranium is not more than 7.5×10^3 MW·day/ton, i.e., the extent of use of the natural uranium is not more than 1%. A solution to the fuel problem requires a substantial increase, namely by about an order of magnitude by comparison with the existing level. This is possible, in particular, by the introduction of fast reactors, in which the use of natural uranium can be increased by almost two orders of magnitude. Therefore, considerable attention is being given to fast-reactor development in the USSR. As developments proceed, the proportion of these in the structure of nuclear power will increase. One of the future problems is to prepare various branches of the industry for the routine introduction of fast reactors. Unfortunately, such reactors cannot take on the role of basic energy sources in many areas. It is undesirable to operate them with a variable load graph. Also, they can hardly provide the basis for centralized heat supply, since, firstly, the unit power is too large (in order to increase the economic performance and improve the neutron balance, fast reactors of unit power 800-1600 MW (elec.) are devised, while to provide heat one requires sources mainly of 300-500 MW (thermal) and less), and, secondly, for reasons of safety they have to be located at considerable distance from heat users. They also cannot produce high-potential heat (about 1000°C) efficiently.

At present, it is difficult to establish with certainty the optimum proportion of fast reactors in the structure of future nuclear power. This will be dependent on the structure of energy use, and, in particular, on the level of electrification. However, one assumes that the reasonable proportion of such reactors can hardly exceed about 0.5. The problem is that when the proportion is less than 0.5, it is still necessary to provide developing nuclear power with artificial fuel in the necessary amount.

Table 5 gives theoretical values for the rate of accumulation of excesss plutonium in fast reactors with sodium cooling for external fuel-cycle durations of 1 and 3 yrs. The less the duration of the external cycle, the higher the rate of plutonium accumulation, but also the more complicated the shipping and reprocessing of the spent fuel. Table 5 shows that with an external fuel-cycle duration of 1 yr, the BN-800 and -1600 (fuel reproduction coefficient RC = 1.3) can provide a rate of accumulation of plutonium of about 0.05 yr^{-1}, while the improved BU units (RC = 1.55) can provide about 0.08 y^{-1} [6].

TABLE 5. Rates of Accumulation of Excess Plutonium

Reactor	External fuel cycle duration, yrs	Rate of accumulation of excess plutonium, yr^{-1}
BN	3	0,028
BN	1	0,051
BU	1	0,077

TABLE 6. Equilibrium Proportion of Fast Reactors

Nuclear power structure	Development rate, yr^{-1}	Length of external fuel cycle, yrs	Input of natural uranium, ton/GW (elec.)•yr		
			0	10	20
BN + LVR	0,01	3	0,83	0,71	0,60
BN + LVRU	0,01	3	0,73	0,54	0,38
BN + LVR	0,01	1	0,79	0,68	0,57
BN + LVRU	0,01	1	0,67	0,48	0,30
BU + LVR	0,01	1	0,64	0,55	0,46
BU + LVRU	0,01	1	0,49	0,35	0,22
BU + LVR	0,03	3	*	0,90	0,77
BN + LVRU	0,03	3	*	0,84	0,63
BU + LVR	0,03	1	0,90	0,78	0,66
BN + LVRU	0,03	1	0,83	0,64	0,45
BU + LVR	0,03	1	0,75	0,65	0,55
BU + LVRU	0,03	1	0,63	0,49	0,34

*Fast reactors are not capable of providing the required rate of accumulation in nuclear fuel.

Table 6 gives the equilibrium proportion of fast reactors in relation to the rate of development of nuclear power, the duration of the external fuel cycle, and supplies of natural uranium. By LVR is meant a water-cooled reactor with the characteristics of the VVÉR-1000 (the RBMK has similar characteristics), while by the improved LVRU is meant a reactor with characteristics analogous to those of the LVR but with the oxide fuel replaced by a uranium of elevated density, while the improved fast BU reactors are ones with heterogenous structures for the core and sodium cooling [6].

Table 6 shows that the minimum proportion of fast reactors is 0.55 (if the BU is used) if these work together with the LVR, or 0.34 with the LVRU even for the comparatively low rate of development in nuclear power of 0.03 yr^{-1}, and with a consumption of natural uranium of 20 tons/GW (elec.). yr, which corresponds to an average consumption of the uranium of about 5%. The equilibrium proportion is appreciably higher if there is a lower RC and the length of the external fuel cycle is 3 yrs.

Therefore, these preliminary calculations show that a solution to the problem of reliable nuclear fuel supplies requires fast reactors with RC $\gtrsim$ 1.5, and external fuel cycles of $\lesssim$1 yr, and also improved thermal reactors with RC = 0.7–0.75.

In principle, all three conditions can be met. Research show that fast reactors with RC $\gtrsim$ 1.5 are possible, e.q., upon using carbide fuel and a heterogenous core construction with sodium cooling. A possible competitive form may be fast reactor with helium cooling. There are no theoretical obstacles to the creation of thermal reactors with RC = 0.7–0.75. If for some reasons it does not prove possible to attain such RC in LVR with uranium fuel, then upon using thorium one can attain RC $\simeq$ 1 even in reactors with light-water coolant. For example, the VVÉR-1000 can provide RC $\simeq$ 0.7 while maintaining the diameter of the fuel pins, the lattice pitch, and the reactor power if uranium dioxide is replaced by a mixture of thorium dioxide with enriched uranium.

Therefore, there is every reason to assume that the fuel problems of nuclear power will be solved, and therefore solutions will become available to the basic problems of the fuel and power balance over a long time-scale.

LITERATURE CITED

1. A. P. Aleksandrov, in: Nuclear Power in the 20th Century [in Russian], Atomizdat, Moscow (1974).
2. F. Ya. Ovchinnikov et al., (this issue).
3. N. A. Dollezhal' et al., (this issue).
4. O. D. Kazachkovskii et al., (this issue).
5. V. A. Sidorenko and O. M. Kovalevich, Papers at the IAEA International Conference on Nuclear Power and Its Fuel Cycle, Salzburg, May 2-13, 1977.
6. S. B. Bobrov et al., (this issue).

EXPERIENCE WITH THE CREATION, OPERATION, AND MEANS OF IMPROVEMENT OF NUCLEAR POWER PLANTS WITH WATER-COOLED-WATER-MODERATED REACTORS (VVÉR)

F. Ya. Ovchinnikov, Yu. V. Markov,
V. A. Sidorenko, V. A. Voznesenskii,
G. L. Lunin, V. V. Stekol'nikov,
G. G. Bessalov, and Yu. V. Vikhorev

UDC 621.039.52.44

After the start-up in 1964 of the first unit of the Novovoronezh Nuclear Power Plant (NNPP) with a gross electric capacity of 210 MW, 10 VVÉR reactors intended to produce 365-440 MW of electrical power were subsequently placed into operation at the Novovoronezh, Kol'skaya, Armyansk, and Rovensk Nuclear Power Plants. During this same period the design power has been utilized in 12 units with VVÉR in the German Democratic Republic, Bulgaria, Czechoslovakia, and Finland (Table 1). Further construction of nuclear power plants in the USSR has continued mainly with the use of a new series of water–water reactors (VVÉR-1000) with a thermal capacity of 3000 MW and a nominal value of 1000 MW for the electric capacity of the unit. The first unit with a VVÉR-1000 at NNPP was connected to the supply system on May 30, 1980 and reached its nominal power on February 20, 1981.

The range of their application in power engineering has expanded simultaneously with the increase in the individual capacity of the reactors: Designs have been created and construction has proceeded of nuclear power plants with VVÉR in seismic regions, and the operation of VVÉR has been proposed under conditions of regulation of the frequency and capacity in power systems as well as for the combined generation of electrical power and heat.

IMPROVEMENT OF THE BASIC ENGINEERING SOLUTIONS FOR NUCLEAR POWER PLANTS WITH VVÉR

The basic engineering characteristics of nuclear power plants with VVÉR are given in [1-4] and in Table 2.

Nineteen years after the start-up of the first unit of the Novovoronezh Nuclear Power Plant, the individual electric capacity of the units has increased from 210 to 1000 MW, the specific intensity of the active zone from 47 to 111 kW/liter, the pressure in the reactor (absolute) from 100 to 160 kgf/cm^2 (1 kgf/cm^2 = 9.8×10^4 Pa), and the steam pressure in the steam generators from 32 to 64 kgf/cm^2. The following solutions have remained unchanged: Six-sided heat-generating assemblies (HGA) with cylindrical fuel elements containing UO_2 and covered by an alloy of Zr + 1% Nb were used in the active zone; high-strength chrome–molybdenum steel was used for the reactor housing; steam generators of the horizontal type [5]

Translated from Atomnaya Énergiya, Vol. 54, No. 4, pp. 249-262, April, 1983.

0038-531X/83/5404-0251$07.50

TABLE 1. Sequence of Entry into Service of the Power Units of Nuclear Power Plants with VVÉR

Power plant, power unit	Reactor	Attainment of criticality	Connection to supply system	utilization of 100% of capacity
Novovoronezh, I	VVÉR-210	17.12.63	30.09.64	31.12.64
Rheinsberg	VVÉR-70	11.03.66	06.05.66	10.10.66
Novovoronezh, II	VVÉR-365	23.12.69	27.12.69	14.04.70
Novovoronezh, III	VVÉR-440	22.12.71	27.12.71	29.06.72
Novovoronezh, IV	»	25.12.72	28.12.72	24.03.73
Kol'skaya, I	»	26.06.73	29.06.73	28.12.73
Nord, I	»	02.12.73	13.12.73	11.07.74
Kozlodui, I	»	30.06.74	17.06.74	28.10.74
Kol'skaya, II	»	30.11.74	09.12.74	21.02.75
Nord, II	»	02.12.74	23.12.74	16.04.75
Kozlodui, II	»	28.08.75	26.08.75	05.11.75
Armyansk, I	»	22.12.76	28.12.76	06.10.79
Lovisa, I	»	20.01.77	08.02.77	09.05.77 *
Nord, III	»	06.10.77	03.11.77	03.05.78
Yaslovske–Bogunitse, I	»	27.11.78	17.12.78	30.03.79
Nord, IV	»	22.07.79	02.08.79	31.10.79
Armyansk, II	»	04.01.80	06.01.80	31.05.80
Yaslovske–Bogunitse, II	»	15.03.80	20.03.80	25.05.80
Novovoronezh, V	VVÉR-1000	30.04.80	30.05.80	20.02.81
Lovisa, II	VVÉR-440	17.10.80	04.11.80	22.12.80
Kozlodui, III	»	04.12.80	16.12.80	27.01.81
Rovensk, I	»	17.12.80	22.12.80	
Kol'skaya III	»	07.02.81	24.03.81	
Rovensk, II	»	19.12.81	27.12.81	

*The nominal electric capacity of the unit is utilized at a thermal capacity of 92%.

were used for production of saturated steam; and transportability of the reactor housing over the railroads of the USSR was ensured.

The layouts of nuclear power plants with VVÉR and the designs of the absorbers and the actuators of the control elements, the intrahousing mechanisms and the main circulating pumps, steam generators, and turbines have differed in their appreciable variety. For example, along with the single-unit (one reactor + one turbine) Rheinsberg nuclear power plant in the German Democratic Republic, the second unit of the Novovoronezh Nuclear Power Plant, in which there are eight circulation loops and five turbogenerators, is successfully operating with a high usage coefficient of the installed capacity (Tables 3, 4).

The main changes in the equipment and systems of nuclear power plants with VVÉR are discussed below.

Reactor and Intrahousing Mechanisms. Questions of vibration stability under the dynamic action of the coolant flow have exerted a dominant effect on selection of the design of the intrahousing mechanisms. A shift of the thermal shield in the first unit of the Novovoronezh Nuclear Power Plant in 1969 led to a reconsideration of the streamline flow conditions and the securing of all elements of the intrahousing mechanisms. The thermal shield for VVÉR-365 and the first VVÉR-440 units was mounted on the hollow shaft of the reactor with welding in the upper part around the entire perimeter (Fig. 1). For VVÉR-1000 and later modifications of the VVÉR-440, the thermal shield was eliminated as a structural element, due to thickening of the walls of the other intrahousing mechanisms.

It was repeatedly necessary in 1974-1975, when a defect in the zirconium covers of the fuel parts of individual control elements was discovered on some VVÉR-440 at the places at which they contact the steel spacer networks, to return to questions of the interaction of the coolant flow with the structural elements in the housing of VVÉR. The cause of the defects is fretting corrosion, which had arisen due to increased vibrations of the control HGA (Fig. 2) and thickening the wall of the zirconium cover from 1.5 to 2.1 mm.

TABLE 2. Basic Engineering Characteristics of Reactor Installations with VVÉR

Parameters	VVÉR-70	VVÉR-210	VVÉR-365	VVÉR-440	VVÉR-1000 (5th unit of NNPP)
Thermal capacity of reactor, MW	265	760	1320	1375	3000
Number of circulation loops	3	6	8	6	4
Pressure, kgf/cm^2:					
in reactor	100	100	105	125	160
in steam generator	32	32	33	47	64
Temperature, °C					
at reactor entrance	250	245	248	268	288
at reactor exit	266	266	274	296	317
Flow rate of coalant through reactor, m^3/h	16 000	33 000	50 000 *	45 000	88 000
Inner diam. of reactor housing, mm	2640	3560	3560	3560	4139
Equivalent diam. of active zone, cm	190	288	288	288	311
Height of active zone in operational state, cm	250	250	246	246	356
Power intensity of active zone, kW/liter	38	47	83	86	111
Number of HGA in active zone	148	343	349	349	151
Number of fuel elements in a HGA	90	90	126	126	317
Outer diam. of fuel element, m	10,2	10,2	9,1	9,1	9,1
Thickness of fuel-element covering made from alloy of Zr+1% Nb, mm	0,6	0,6	0,65	0,65	0,67
Av. linear capacity of fuel elements, W/cm	89	99	122	127	176
Fuel charge into reactor, tons of metal	17,0	40,0	41,5	41,5	66,0
Specific capacity of fuel, kW/kg of U	15,5	19	32	33	45,5
Enrichment of makeup fuel upon replacement of 1/3 HGA, %	2,0	2,0	3,0	3,5	4,4
Fuel depletion depth, MW/day	13	14	28	30	40
Number of SCR assemblies	19	37	73	73 or 37	109

*With the operation of seven loops (one is a reserve loop).

TABLE 3. Expenditures for Construction and the Average Technico-economic Indices of the Operation of Nuclear Power Plants with VVÉR in 1977-1981

Indices	Novovoronezh, units				Kol'skaya, units	Armyansk, units
	I	II	III, IV	V	I, II	I, II
Installed capacity, MW (elec.) gross	210	365	880 *	1000	880	815
Specific capital costs, rubles/kW	326	256	200	308	263	327
Electrical power output, billions of kWh	7,4	13,9	30,1	6,0 †	31,0	15,4
Average usage coefficient of installed capacity	0,80	0,87	0,81	—	0,80	0,62

*Since 1979 the total installed capacity of units III and IV has decreased to 834 MW in connection with the high average temperature of the water which cools the turbine condensers.
†Data for 1980-1981.

TABLE 4. Technicoeconomic Characteristics of Nuclear Power Plants with VVÉR during the 1977-1981 Period

Characteristics	Year	Novovoronezh, units					Kol'skaya, units			Armyansk, units	
		I	II	III	IV	V	I	II	III	I	II
Usage coefficient of installed capacity	1977	0,76	0,82	0,78	0,79	—	0,59	0,73	—	0,208	—
	1978	0,86	0,90	0,82	0,75	—	0,83	0,83	—	0,535	—
	1979	0,818	0,909	0,758	0,707	—	0,68	0,85	—	0,668	—
	1980	0,714	0,856	0,843	0,843	0,22	0,962	0,907	—	0,774	0,551
	1981	0,847	0,859	0,838	0,898	0,562	0,802	0,855	0,338	0,778	0,762
Electrical power output, millions of kWh	1977	1397	2614	3013	3057	—	2288	2823	—	834	—
	1978	1584	2892	3150	2891	—	3205	3199	—	1909	—
	1979	1505	2905	2850	2656	—	2616	3285	—	2386	—
	1980	1317	2745	3079	3088	1112	3717	3507	—	2772	1974
	1981	1558	2745	3061	3280	4918	3091	3297	1001	2779	2720

For further improvement of the hydrodynamic conditions of operation of the active zone on all VVÉR, a special perforated bottom which equalizes the flow distribution through the HGA has been installed in the lower space of the reactor, starting from the first unit of the Armyansk Nuclear Power Plant. When the new units are started up, expanded operating programs are performed for measurement of vibration and stresses in the structural elements of the reactor and the intrahousing mechanisms.

At present, a great deal of experience has been accumulated on the operation of 10 VVÉR-365 and VVÉR-440 housings made out of steel 15Kh2MFA without noncorroding planting. A satisfactory corrosion state of the inner surface of the housings is provided by the observance of an ammonia—potassium aqueous chemistry regime in the course of operation and by the creation of an increased ammonia concentration and the execution of measures for reduction of the nitrate concentration during recharging periods. Moreover, noncorring plating has again been introduced in accordance with universal practice on the VVÉR-1000 and

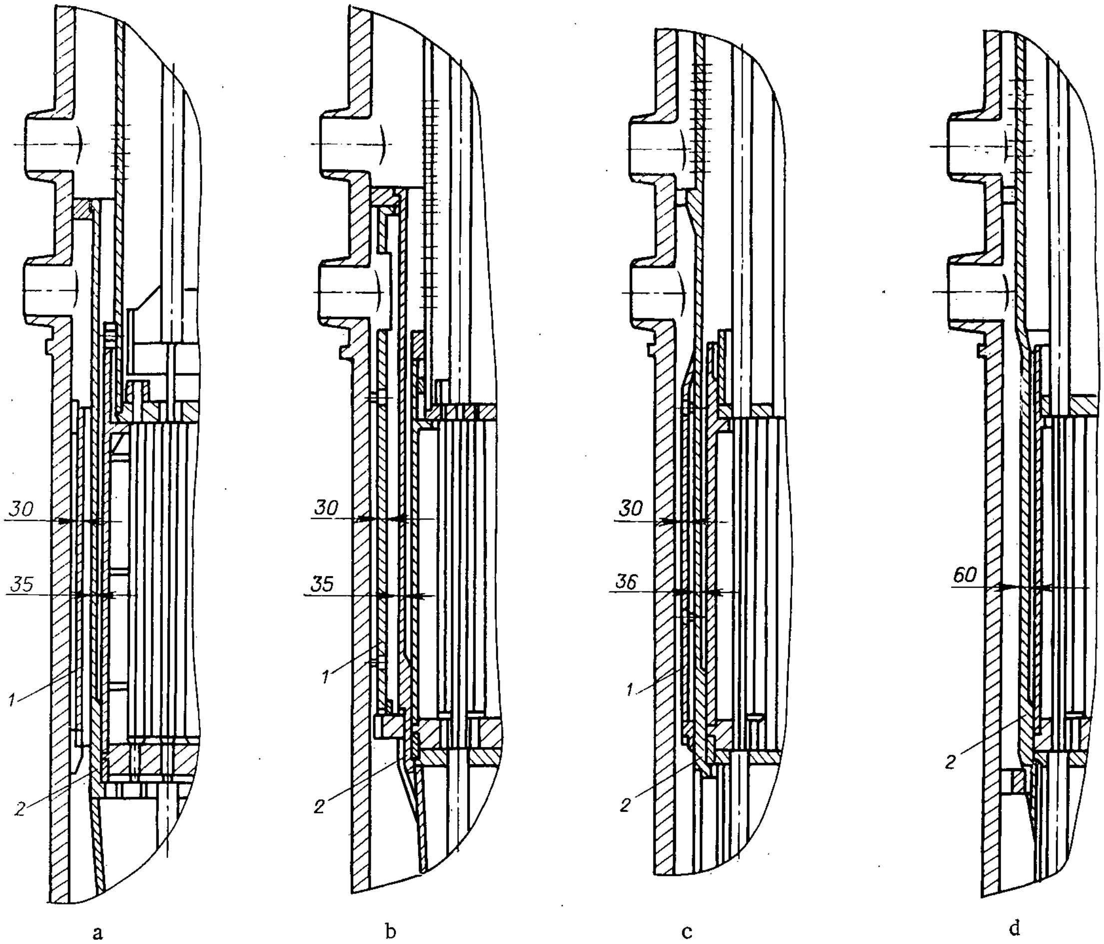

Fig. 1. Construction of the thermal shield in: a) VVÉR-210; b) VVÉR-365; c) VVÉR-440 of the third and fourth units of the NNPP; d) in the commercial VVÉR-440; 1) thermal shield; 2) shaft.

and VVÉR-440, starting from the first unit of the Lovisa Nuclear Power Plant. Maintenance of the required aqueous chemistry regime has been simplified somewhat.

It has become necessary for recent modifications of VVÉR in connection with the increase in capacity of the systems for emergency cooling of the active zone, which supply a relatively cold solution of boric acid directly to the reactor, to consider, in addition, the question of the calculated reserve of the reactor housings from the standpoint of resistance to brittle fractures. A detailed analysis of the dependence of the radiation resistance of the housing materials on the flux of fast neutrons has been performed for all VVÉR housings on the basis of an investigation of the properties of irradiated test samples. Additional measures, which provide a guaranteed calculated reserve of housing operation, were applied for several VVÉR-440 housings in connection with an increase in the amount of phosphorus and copper impurities in a welded seam in the region of the active zone. Model HGA which permit reducing the maximum irradiation of housing sections by a factor of three were mounted on the periphery of the active zones of these reactors in place of 36 HGA with fuel.

The Active Zone and the Control and Protection System. Fuel HGA with a covering of Zr + 2.5% Nb alloy were used for assembly of the active zone in all VVÉR operating on July 1, 1982. Spacing of the fuel elements in the beam was accomplished by 12-16 steel grids. The size of the "turned on" HGA for all VVÉR-1000 is 144 mm. In order to increase the specific capacity of the active zones, the outer diameter of a fuel element was reduced from 10.2 (VVÉR-70 and VVÉR-210) to 9.1 mm with a simultaneous increase in the number of fuel elements in a HGA.

The ratio of the number of hydrogen atoms to the number of uranium atoms in the operational state is 4.2-4.7, which provides a negative reactivity coefficient with respect to coolant temperature during operation of the reactors. The temperature coefficient of reac-

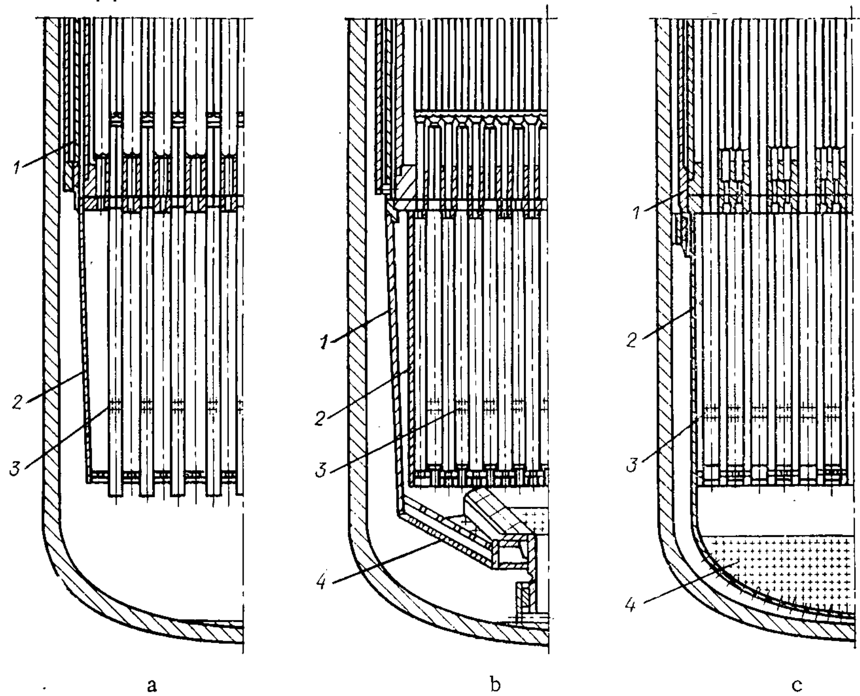

Fig. 2. Alteration of the design of the intrahousing mechanisms of the reactor to decrease the dynamic action of the coolant on the control HGA of the active zone in: a) VVÉR-365; b) VVÉR-440 of the third and fourth units of the NNPP; c) the commercial VVÉR-440; 1) shaft; 2) bottom of the shaft; 3) openings for passage of coolant into the control assemblies; 4) perforated bottom.

tivity is slightly positive only at the start of operation of the first loadings of VVÉR-440 and VVÉR-1000, which operate with boron regulation. The depletion depth of the fuel in a VVÉR-440 is ∿30 MW•days/kg of U for an average enrichment of the makeup of 3.5%. One-third of the HGA which have reached maximum depletion are unloaded each year. Fresh fuel is installed on the periphery of the active zone.

Regulating safety-and-control-rod (SCR) assemblies equipped with actuators are provided on VVÉR-70 and VVÉR-210 to act on the reactivity. Part of these assemblies, which are intended mainly for compensation of the total reactivity reserve, contain absorbers made out of boron steel in the upper part and an assembly with nuclear fuel which is similar in construction to a fuel HGA in the lower part. In addition, there are assemblies for emergency protection which are intended for rapid shutdown of the reactor. They do not contain fuel and have their own actuator construction.

All the SCR assemblies and their actuators have become identical in the subsequent development of VVÉR. The problem of increasing the efficiency of the SCR system was initially solved by increasing the number of assemblies with absorbers (the second unit of the NNPP). Starting from the third unit of the NNPP, the reactivity reserve against fuel depletion and slow changes in the reactivity have been compensated by the introduction of a boric acid solution into the coolant of the first loop.

The application of boron regulation has permitted reducing the number of regulating SCR assemblies on VVÉR-440 from 73 to 37. Control elements in the form of bunches of 12-18 fuel elements containing europium dioxide or boron carbide are used in the reactor of the fifth NNPP unit and other VVÉR-1000. The transition to a new construction of the absorbers has permitted reducing the height of the reactor housing by virtue of a decrease in the volume under the active zone in which the fuel portion of the SCR assemblies is positioned in VVÉR of smaller capacity. The larger number of control elements, including elements with absorbers one-half the length of those in the first VVÉR-1000 reactor in the USSR, permits effectively influencing the energy distribution throughout the active zone if necessary.

TABLE 5. Design Engineering Characteristics of the Main Circulation Pumps (MCP)

Characteristic	Types of reactors and pumps				
	VVÉR 210	VVÉR 365	VVÉR-440		VVÉR-1000
	MCP-138*	MCP-309	MCP-310	MCP-317	MCP-195
	Without a stuffing box				
	with pressurized electric motor			with packing of shaft and increased moment of inertia	
Flow rate, m^3/h	5250	5600	6500	7100	20 000
Coolant temperature, °C	250	250	270	270	300
Intake pressure, kgf / cm^2	100	105	125	125	156
	4	5,5	5,3	4,3	6,75
Number of rpm (synchronous)	1500	1500	1500	1500	1000
Consumable capacity (no more than), kW					
in cold water	—	—	—	1600	7000
in hot water	1650	1500	2000	1400	5300
Mass of MCP with auxiliary equipment, tons	42	36	51	55	150
including electric motors	—	—	—	15	48

*Replacement of MCP-138 by MCP-309A was performed in 1972-1975.

TABLE 6. Design Engineering Characteristics of SG

Characteristics	VVÉR-210	VVÉR-365	VVÉR-440	VVÉR-1000
Steam capacity, tons/h	230	325	452	1468
Steam pressure, kgf/cm^2	32	33	47	64
Temperature, °C; of coolant of first loop				
at entrance	273	280	301	320
at exit of	252	252	268	290
supply water	189	195	225	220
Humidity of steam at exit no more than, %	0,2	0,2	0,2	0,2
Heating surface (on outer diam. of pipes), m^2	1300	1810	2510	6115
Outer diam. and wall thickness of pipes, mm	21×1,5	16×1,4	16×1,4	16×1,5
Mass of dry steam generator, tons	104	112	155	292

Declassified and Approved For Release 2013/02/06 : CIA-RDP10-02196R000300020004-6

TABLE 7. Design Characteristics of Volume Compensators

Characteristic	VVÉR-210	VVÉR-365	VVÉR-440	VVÉR-1000
Type	Gas	Steam		
Total internal vol. vol., m^3	4×17,5	4×10,7	38 or 44	79
Inner diam. of cylindrical part, mm	1800	1500	2400	3000
Volume of nitrogen or steam at nominal capacity, m^3	52	18	16 or 18	24
Operating pressure, kgf/cm^2	100	105	125	160
Operating temperature, °C	260	313	325	346
Total capacity of heaters, kW	—	1680	1620	2520

Actuators of the "screw—nut" (VVÉR-70, VVÉR-210, and VVÉR-365) and the "rack-and-pinion," (VVÉR-440) types and several kinds of electromagnetic actuators (VVÉR-1000) were developed for movement of the SCR absorbers.

Intrareactor Control. In order to control the energy distribution in the active zone on VVÉR, systems of intrareactor control are provided. The temperature is measured on VVÉR-210 and VVÉR-365 at the exit of about two-thirds of all the HGA with fuel, and, in addition, 12 or 36 measuring channels containing 5-7 neutron-flux detectors each are mounted in the HGA on VVÉR-440. The temperature at the exit of all the HGA is controlled on the VVÉR-1000 of the fifth NNPP unit, and neutron-flux detectors are mounted in 31 HGA. Analysis of the data of the intrareactor measurements and presentation of the results to the operator is presently accomplished by a special system linked to the information—computation complex common to the unit as a whole. In the event of a breakdown of this complex, the system switches to an automatic operating mode in which information is processed by simplified algorithms.

Main Circulation Pumps (Table 5). Low-inertia pressurized pumps with a synchronous rotation frequency of the rotor of 1500 rpm are used on 17 of the 24 operating VVÉR units. In order to cool the active zone in the case of disconnection from the power system, the operation of the MCP for 100 sec after shutdown of the reactor is provided on these units by means of the energy of electromechanical coasting of the main generators or special internal-discharge generators located on the same shaft with the turbines. Pumps with the rotor of the electric motor extended beyond the confines of the first loop are used for the VVÉR-1000 and also in the new designs of nuclear power plants with VVÉR-440. A special flywheel provides for a slow decline of the flow rate when the MCP are disconnected

A test of the operation of the MCP at nuclear power plants in the USSR has shown that they are one of the most reliable pieces of equipment of a reactor facility.

Steam Generators (Table 6). Steam generators (SG) with a horizontally positioned housing and a pipe bunch are used at nuclear power plants with VVÉR, which provides for moderate loads on the surface of the evaporation mirror. Cylindrical collectors of the primary coolant are located in the surroundings of the second loop. The pipe bunch is fabricated out of OKh18N10T steel, and the housing material is carbon steel.

The principal structural change in the course of the improvement of SG is the realization for VVÉR-440 and VVÉR-1000 of access from above to the collectors of the first loop. In this case it proved to be possible to reject special subshaft access spots for disposition of equipment servicing the SG. Maintainance and a surveying of the sites where pipes are sealed into the collectors are accomplished from above directly from the central room of the nuclear power plant. Since a flanged joint is situated in the surroundings of the second loop, special attention to the behavior of the metal of the collectors of the first loop is necessary in the phase partition zone.

Pressure Compensators. A nitrogen pressure compensator is used in the first unit of the NNPP and the Rheinsberg Nuclear Power Plant. Steam pressure compensators having better weight—size characteristics (Table 7) are used in the equipment complement of the remaining VVÉR reactor facilities. The housings of the pressure compensators are fabricated out of carbon steel. On units in which the reactor housing has no noncorroding plating, it is absent on the pressure compensator.

Main Circulation Pipelines and Slide Valves. For all the reactors except VVÉR-1000, the main circulation pipelines (MCP) with an inner diameter of 500 mm (Du 500) are fabricated out of 1Kh18N9T and 1Kh18N12T steels. For VVÉR-100 the MCP are made two-layered (on the outside — carbon steel; inside — noncorroding steel), and their straight-through cross section is Du 850. The possibility of cutting off loops with the help of slide valves having an electric actuator which closes in a time no greater than 90 sec is provided for. Rapid-acting slide valves with a hydroactuator, which have been used on the MCP of the first unit of the NNPP and the Rheinsberg Nuclear Power Plant, were excluded in the subsequent designs.

Safety Systems. As VVÉR have developed, safety mechanisms intended to limit the consequences of accidents and to localize radioactivity which has leaked from the main circulation loop (MCL) have steadily been improved [6]. The increased reliability of the emergency protection systems of the reactor, emergency cooling of the active zone, diversion of heat from the steam generators, and localization of fission products has been attained both by providing the necessary emergency arrangement of these systems and by applying better engineering solutions.

If an instantaneous break in a pipeline with a diameter of about 100 mm with one-way outflow were the maximum design emergency for the first VVÉR, then the protective and localizing mechanisms for contemporary VVÉR-440 (Rovensk Nuclear Power Plant) and VVÉR-1000 provide for safety in the event of accidents right up to instantaneous fracture of the MCP coinciding in time with conditions of complete deactivation of the nuclear power plant. An increase in the temperature of the fuel element casings above 1200°C is prevented with the help of the provided systems of emeregency cooling of the active zone (water tanks connected in pairs to the entrance and exit space of the reactor; high- and low-pressure pumps). Localization of fission products which escape from the MCL is accomplished for new nuclear power plants with VVÉR-440 by the traditional method for VVÉR — with the help of a system of pressurized rooms; the reactor room remains accessible for servicing. A condenser—bubbler provides for steam condensation during the first period of a maximum design emergency. The maximum pressure in the chambers in the course of eliminating the emergency does not exceed 2.5 kgf/cm^2.

For nuclear power plants with VVÉR-1000, construction of a shell is provided which encloses all the rooms of the MCL and the reactor room and is calculated for the total pressure arising upon outflow of all the coolant (5 kgf/cm^2) with subsequent reduction in the pressure of the sprinkler system. In order to prevent the escape of activity during an accident, the installation in sequence of three rapid-acting pneumatic valves is provided on the pipelines which connect the shell to the external systems; each valve closes from its own high-pressure air system. A high degree of independence of the redundant protective and localizing systems is provided in the designs by means of placing them in different rooms, a separate electrical supply, etc.

EXPERIENCE WITH THE CONSTRUCTION AND OPERATION OF NUCLEAR POWER PLANTS WITH VVÉR

The cumulative duration of the operation of nuclear power plants with VVÉR from the time they were included in the grid to July 1, 1982 amounts to 150 reactor-yrs. Analysis of the operation of VVÉR during this period permits asserting that nuclear power plants with such reactors are capable of providing a reliable supply of electrical power to consumers, with high technicoeconomic indicators. As follows from Table 3, the specific capital expenditures for the construction of nuclear power plants decreases at first and reached a minimum sum for the third and fourth units — 200 rubles per 1 kW of installed capacity. The increase in the cost of construction for the subsequent units is explained both by factors of local significance (construction of nuclear power plants at Zapolyar'e — the Kol'skaya Nuclear Power Plant; provision of seismic-resistant buildings and equipment — the Armyansk Nuclear Power Plant) and by general tendencies: the increase in costs to provide for the safety of nuclear power plants and the rise in prices of energy resources.

In the USSR 106.6 billion kWh of electrical power was produced at nuclear power plants with VVÉR during 1977-1981, which amounts to about 2% of the total amount of electrical power produced during this period in the entire country. The average cost of the production of electrical power in 1981 is significantly lower than at thermal power plants. For units which have operated longer than a year at the design capacity, the usage coefficient of installed capacity exceeds, as a rule, the design value of 0.8 (see Table 3, 4).

A continuous effort is being made at nuclear power plants to increase the reliability and safety of the equipment. Within the framework of the system for collection of information on equipment failures which has been operating since January 1, 1977, the causes for the failures are investigated and classified, and equipment with reduced reliability is revealed. The information is directed to the factories—manufacturers and to the design organizations for adoption of the necessary measures.

Quantitative reliability indices of the first and second units of the Kol'skaya Nuclear Power Plant during the period from the start of its operation to June 30, 1980 without taking account of the failures during the shakedown period are given below in hours:

Unit as a whole		
Operating time to failure	1980	1902
Average recovery time	24	8.5
Shakedown period	3406	1490
Reactor with control and protection system		
Operating time to failure	10,000	4692
Average recovery time	2.6	4.8
Shakedown period	2923	2000
Main circulation pump		
Operating time to failure	57,230	20,020
Average recovery time	388	250
Shakedown period	0	0
Steam generator		
Operating time to failure	6804	11,190
Average recovery time	149	294
Shakedown period	3000	5000

These data indicate the stable operation both of nuclear plants as a whole and of the basic equipment. One should note that the failure of a unit as a whole is an event leading to complete degradation of the charge, but failure of the MCP or an SG is an event which results in the disruption of their work capacity, which leads to disconnection of the loop.

Among the problems which must be solved for operating units, one should note the development of measures for constant upgrading of the safety of the units in connection with the change in the operating norms and rules which regulate safety questions. In connection with the expiration in 1984 of the design term of service of the reactor housing of the first unit of the NNPP, the question of an operation extension by means of a possible annealing of the housing, with simultaneous replacement of the reactor cover, the actuators of the control elements, and part of the intrahousing mechanisms, is being considered.

METHODS OF FURTHER IMPROVEMENT OF VVÉR

The subsequent development of nuclear power plants with VVÉR will be accomplished by the application of more refined equipment, simplification of the layout of the MCL and nuclear power plants as a whole, optimization of the thermal engineering parameters and the fuel cycle, and improvement of the reliability of the systems for provision of safety. The range of possible usage of VVÉR in power engineering will be simultaneously expanded.

A modified VVÉR-1000 reactor assembly which is distinguished by an improved layout of the MCL and the absence of the main shutoff slide valves is being used in the majority of the units which will be placed into operation up to 1990. The extent of the MCL and the protective shell is reduced by more than 20%. The rejection of the use of covers for the HGA permits placing 163 HGA instead of 151 (fifth unit of the NNPP) in the active zone.

The number of control elements is decreased from 109 to 61 with a simultaneous increase in the number of fuel elements in a bunch from 12 to 18.

The design characteristics of the VVÉR-1000 (the fifth unit of the NNPP and the modernized unit, respectively) are given below:

Thermal capacity, MW	3000	3000-3200
Coolant pressure, kgf/cm^2	160	160
Average coolant temperature, °C	306	306-307
Coolant flow rate, m^3/h	80,000	80,000
Outer diameter of the reactor housing, mm	4535	4535
Height of assembled reactor, mm	22,592	19,137
Equivalent diameter of the active zone, cm	311	316
Height of the active zone in the operating state, cm	356	356
Power intensity of the active zone, kW/liter	111	107-115
Number of HGA	151	163
Shape and type of HGA	Six-sided with cover	Six-sided without cover
Size of an HGA with cover when "turned on", mm	238	234
Fuel (UO_2) charge in the active zone, tons	75.5	80
Outer diameter and spacing of the fuel elements, mm	9.1/12.75	9.1/12.75
Average thermal flux, W/cm^2	176	166-177
Operating period of the fuel, yrs	2 or 3	3
Number of rechargings per operating period	2 or 3	3
Enrichment of fresh fuel in the steady-state recharging regime, %	3.3 or 4.4	4.4
Average depletion depth of the fuel, MW•days/kg of U	27 or 40	40
Number of control elements	109	61
Number of fuel elements in a control element	12	18
Number of MCP	4	4
Number of revolutions of the MCP, rpm	1000	1000
Presence of shutoff slide valves in the MCP loops	Yes	No
Number of SG	4	4
Type of SG	Horizontal	
Steam capacity of a single SG, tons/h	1469	1469-1575
Steam pressure at the exit from the SG, kgf/cm^2	64	64
Steam temperature, °C	278.5	278.5

As the development is accomplished, new more refined equipment will be used in the VVÉR-1000. More compact SG constructions are being designed: vertical with natural circulation of the evaporator, and direct-flow. Steam generators with higher steam parameters (305-310°C, 70-74 kgf/cm^2) are being considered in a number of possible alternatives. The development of a pump assembly with a shaft rotation of 3000 rpm and with mass characteristics approximately twice as good as those of the MCP-195 is in the stage of experimental checking of the individual subassemblies.

The use of a nuclear fuel other than uranium dioxide is not anticipated for VVÉR reactors up to 1990. In order to improve the characteristics of the fuel cycle, it has been proposed to increase the average fuel depletion depth to 40 MW·days/kg of U by increasing the enrichment of the makeup fuel with simultaneous optimization of the recharging regime and a restriction in the structure of the active zone on the number of construction materials with large neutron absorption cross section (steel). The efficiency of the fuel elements of the VVÉR-1000 at such a depletion has been confirmed in an experimental microroentgenometer loop There has also been a successful test of the attainment of a depletion of ∿50 MW·days/kg of U in the regular HGA of the VVÉR-440 with them located in the active zone for 5 yrs. The gradual conversion of VVÉR to fuel of increased density or the incorporation of thorium into the fuel cycle is advisable in the future in the interests of the economy of inexpensive natural uranium. The scientific–engineering bases for such a change in the fuel cycle should be worked out in the forthcoming decade.

An increase in the safety level of VVÉR reactor installations is mainly guaranteed by means of the further development of the system of intrareactor measurements and the incorporation of operational systems for control of the state of the equipment and the metal of the MCP. At present, the USSR COMECON member-nations are working on a cooperative program which includes tests on operating reactors for working out methods of recording variations in the noise spectrum during disruptions of the normal operation of a reactor facility, including the onset of boiling in the active zone. Scientific principles and instrumentation for the detection and classification of defects in the materials of MCL equipment using the acoustic emission method are being developed.

As ordinary fuel becomes more expensive, the number of regions in the USSR in which it is economically advisable to construct nuclear power plants will increase during the succeeding decades. In this connection, as well as with possible deliveries for export taken into account, VVÉR-1000 reactor assemblies will be calculated to withstand an earthquake with a force up to a reading of 9 (maximum acceleration at ground level of 0.4 g). The possibility of using seawater for cooling of the auxiliary equipment and the placement of nuclear power plants at sites with a humid tropical climate is foreseen.

It has been proposed that nuclear power plants with VVÉR can participate in providing for the variable loading diagram of power systems. The requirements on such nuclear power plants at present provide for the possibility of daily disconnections from the grid for 5-8 h and weekly ones for 24-55 h, enhanced rates of change of the loading from 1-4% of the nominal capacity per min, and keeping the units in operation in the event of short-term decreases in the frequency right down to 46 Hz. It has become necessary to solve a number of problems and first of all to produce a design for the fuel element which is efficient under long-term cyclic loads and to verify it experimentally. It is possible that VVÉR intended for control of the power and frequency in a system will operate at a lower power intensity than baseline reactors but with higher coolant parameters.

An important aspect of the use of VVÉR is their application for a centralized heat supply. The technicoeconomic discussion carried out up to the present of the alternatives for the heat supply of a number of large cities in the European part of the USSR from sources based on nuclear and organic fuel indicates the advisability of the application of nuclear heat supply plants (NHSP) for the generation of heat and of nuclear heat–power plants (NHPP) for the combined generation of heat and electrical power in comparison with boilers and heat and electric power plants operating on organic fuel. The choice of NHPP or NHSP depends on the conditions of a given city. No significant variations of any kind are assumed in the VVÉR-1000 reactor assembly for NHPP. The heat supply for the consumers is accomplished from diversions of the steam of the TK-450/500-60 central-heating turbines, which provide a maximum heat production of 450 Gcal/h at an electrical load of about 450 MW each.

LITERATURE CITED

1. A. Ya. Kramerov et al., Third Geneva Conference, USSR Lecture R/304 [in Russian] (1964).
2. V. P. Denisov et al., At. Energ., 31, No. 4, 323 (1971).
3. V. A. Voznesenskii, At. Energ., 44, No. 4, 299 (1978).
4. Yu. V. Vikhorev et al., At. Energ., 50, No. 2, 87 (1981).
5. V. F. Titov et al., "The main trends in the development of steam generators for nuclear power plants with VVÉR in the Soviet Union," Lecture at the Soviet–Italian Seminar "Contemporary Problems of Power Engineering," November 21-24, 1977, Moscow.
6. V. A. Sidorenko et al., At. Energ., 43, No. 6, 449 (1977).

SOME CHARACTERISTICS OF AND EXPERIENCE WITH THE OPERATION OF NUCLEAR POWER PLANTS WITH RBMK-1000 HIGH-POWERED WATER-COOLED CHANNEL REACTORS (RBMK)

N. A. Dollezhal', I. Ya. Emel'yanov,
Yu. M. Cherkashov, V. P. Vasilevskii,
L. N. Podlazov, V. V. Postnikov,
A. P. Sirotkin, V. P. Kevrolev,
and A. Ya. Kramerov

UDC 621.039.56:621.039.524.2.
034.44

During the period from the end of 1973, when the first power unit with an RBMK-1000 entered into service at the Leningrad Nuclear Power Plant, to January 1, 1981, the installed capacity of nuclear power of the Soviet Union grew from 3.2 to 14.6 GW. During this period 8 GW of the 11.4-GW growth in nuclear capacities, i.e., about 70%, was due to the fraction of nuclear power plants with RBMK-1000. Now nine power units with this reactor are being operated: four at the Leningrad Nuclear Power Plant (LNPP), three at the Chernobyl Nuclear Power Plant (ChNPP), and two at the Kursk Nuclear Power Plant (KNPP). In all, nuclear power plants with RBMK-1000 have generated about 200 billion kWh of electrical power. The significant increase in the capacities of the nuclear power plants of the country based on the RBMK-1000 which has been achieved in a comparatively short period of time, the successful utilization of their nominal capacity, and the reliability and safety of their operation testify to the promising outlook for channel uranium–graphite reactors, on which the development of nuclear power in the USSR will be based in the succeeding decades. One can add the following to the list of factors in favor of channel boiling reactors of the RBMK type, which have been taken into account in the process of the design and construction development and which completely support the practice of their construction and operation:

The RBMK-1000 is manufactured at operating factories and does not require the construction of new industrial enterprises with unique equipment;

there do not exist limiting values of the individual capacity associated with the manufacture, transporting, and maintenance of the equipment used;

shattering of the main loops increases the overall safety of the reactor, since there is not complete dehydration of the active zone;

due to the good physical characteristics of the reactors and the continuous fuel recharging, the prerequisities are created for highly efficient utilization of weakly enriched fuel, the attainment of a small content of fissionable uranium isotopes in the exhausted fuel, and the production of a sufficiently large increase in the depletion due to consumption of the plutonium made in passing; and

a high thermal engineering reliability of the power units is provided by a broad range of regulation of parameters by in-channel control.

Approximately 1660 fuel channels and more than 200 special channels of the monitoring, control, and protection system are positioned in vertical openings of the graphite reactor stack in a square array with a 250-mm spacing. Two heat-generating assemblies (HGA) with 18 fuel elements in each are mounted inside the zirconium tube of the fuel channel. Pellets made out of uranium dioxide with an enrichment of 2% in ^{235}U are used as the fuel. The casings of the fuel elements with an outer diameter of 13.5 mm and a thickness of 0.9 mm are fabricated out of a zirconium–niobium alloy. Water from the distributing collectors underheated to boiling is individually supplied to each channel. The necessary flow rate is established with the help of a channel flowmeter and a regulating valve. In the active zone $\sim$15% of the water is converted into steam. The steam–water mixture from each channel is also diverted into separators through individual pipelines. Saturated steam at a pressure

Translated from Atomnaya Énergiya, Vol. 54, No. 4, pp. 257-262, April, 1983.

0038-531X/83/5404-0263$07.50

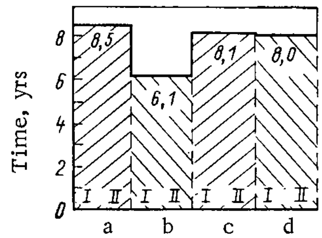

Fig. 1

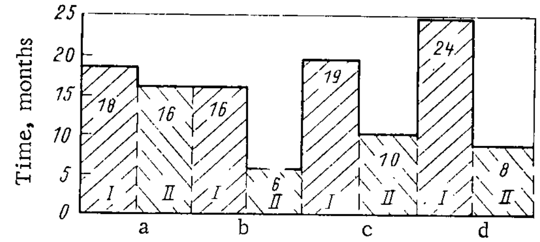

Fig. 2

Fig. 1. The duration of construction of a nuclear power plant until power-up of the second units: a, b) first and second turns of the LNPP, respectively; c, d) first turn of the KNPP and ChNPP, respectively.

Fig. 2. Duration of reactor assembly until the start of the flushings of the reactor systems (the notation is the same as in Fig. 1).

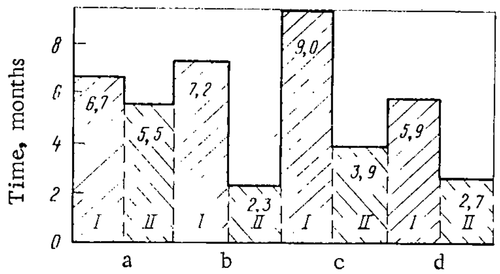

Fig. 3

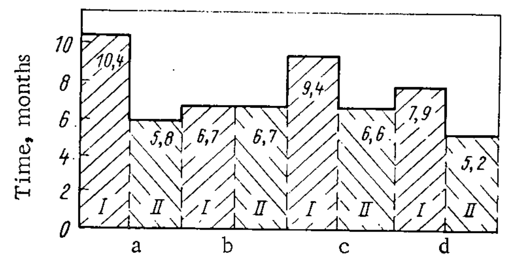

Fig. 4

Fig. 3. Duration of the start-up adjustment operations from the start of flushings of the systems until power-up of the unit (the notation is the same as in Fig. 1).

Fig. 4. Duration of utilization of the capacity from power-up of the unit attainment of nominal capacity (the notation is the same as in Fig. 1).

of 7 MPa is directed into two turbines with a capacity of 500 MW each. The separated water, mixing with the supply water, is again fed by the main circulation pumps to the entrance of the channels. The reactor has two circulation loops whose equipment is mounted symmetrically with respect to the vertical plane passing through the reactor axis in the direction of the machine room [1].

Paired layout is used in the design of nuclear power plants with RBMK-1000, i.e., two power units are located in the main building of the nuclear power plant. Each reactor, mounted in its own shaft, has an independent circulation loop. The four turbines of the two units are arranged in series on the same axis in the common machine room which is adjacent to the main building. The units do not depend on each other, but they have a series of auxiliary interchangeable systems, which creates definite advantages in the course of operation of the nuclear power plants and especially in the maintenance of the equipment. The adopted layout of a nuclear power plant provides for start of the construction of the main buildings of the first (I) and second (II) units and maintenance of the equipment practically simultaneously (Figs. 1-4). As follows from the diagrams given in Figs. 1-4, the even units are activated appreciably more rapidly. If assembly of the odd units is accomplished in 1.5-2 yrs, it has proven possible to shorten this time to 8-10 months for the even units. On the average, the time to carry out the start-up adjustments on the even units is shortened by a factor of two. The nominal capacity of the leading units was utilized in 8-10 months, and in 5-6 months on the succeeding even units.

This is explained to a significant extent by the fact that readiness of a large number of auxiliary systems which are common to both units is required for start-up of the leading units. In addition, the experience of performing the start-up adjustment on the first power units has shown that their extent can be reduced on the subsequent nuclear power plants.

TABLE 1. Some Operational Indices of Nuclear Power Plants with RBMK-1000 in 1980

Nuclear power plant	Installed capacity, MW	Production of electrical power, kWh	CUC, %
Leningrad	3000	18,82	73,0
Kursk	2000	13,89	79,2
Chernobyl	2000	14,21	80,9
Total	7000	46,92	—

The start-up adjustment operations should be determined by tests of the installed equipment and complex tests of the systems, but it is advisable to perform only those of the investigative efforts whose results obtained on the preceding units turn out for this or the other reason to be unsuitable for the subsequent ones. This has served as the basis for the creation of optimal standard procedures and programs of performing the adjustment of the systems. In the standard plot which has been developed for utilization of the capacity of a commercial unit with an RBMK-1000, six months is provided. Further reduction of the periods for utilization of the capacity of commercial units has been acknowledged as being inadvisable for the following reasons: In the first place, a specified time is required for checking the fitness of the equipment in intermediate stages which permits predicting a safe elevation of the capacity at the next stage; secondly, experience in control of the unit is acquired by the operating personnel.

In 1980, 73 billion kWh of electrical power was produced at the nuclear power plants of the country, and of this amount 47 billion kWh or 64.5% was produced at nuclear power plants with RBMK-1000. In comparison with the previous year, the production of electrical power at nuclear power plants increased by more than 35%. This has been achieved not only due to the introduction of new capacities, but also owing to the high value of the capacity utilization coefficient (CUC). The average value of the CUC for nuclear power plants with RBMK-1000 exceeds 76% (Table 1).

The data on the operational readiness of the reactor equipment of the power units is of interest. In 1979 the operational readiness coefficient (ORC) on the first turn of the Leningrad Nuclear Power Plant reached 85%, at the KNPP 84.5%, and at the ChNPP 88.7%, with CUC values of 74.4, 73.1, and 74.6%, respectively. These same high values of the ORC also characterize the operation of RBMK-1000 in 1980. For example, at the ChNPP the number of hours of operation of the reactors of the first and second units was 7522 and 7622 h, and out of these the reactors operated for 6899 and 7313 h at nominal capacity. At the KNPP the number of hours of operation of the reactors of the first and second units was 7677 and 7642 h, and the reactors operated for 6999 and 6890 h at nominal capacity. The cited operational indices of nuclear power plants with channel reactors are not inferior, judging from the published data, to the best operational indices of foreign nuclear power plants with reactor housings of equal capacity, both boiling and with water under pressure.

Deep depletion of the nuclear fuel with a low initial enrichment is characteristic of RBMK-type reactors, which is provided for by continuous fuel recharging at the operating facility. Fuel recharging at capacity is constantly accomplished at all nuclear power plants with RBMK-1000 with the help of an unloading—loading machine. A regime of continuous rechargings permits increasing by approximately a factor of two the fuel depletion depth in comparison with the regime of one-time complete recharging of the active zone. The ^{235}U concentration decreases from 18-20 to $\sim$3.7 kg/ton of U, and the amount of fissionable plutonium reaches $\sim$2.8 kg/ton of U. With such a change in the isotopic composition of the fuel, the neutron-physical characteristics of the cell are significantly altered. If in the steady-state regime of fuel recharging only the local characteristics (e.g., the power) of the channels are altered but the characteristics of the reactor as a whole remain practically constant, then the most important changes in its physical characteristics, in particular, the reactivity coefficients (steam, thermal of the graphite, thermal from heating up) occur during the initial period of operation of a reactor loaded with fresh fuel and additional absorbers. The values of these coefficients depend not only on the isotopic composition of the fuel, but also on the presence of absorbers in the active zone.

Experience with the operation of the RBMK-1000 has confirmed the theoretical conclusions that as the fuel is depleted and the absorbers are withdrawn, the reactivity coefficients increase and the stability of the energy distribution decreases. A radial—azimuthal energy distribution, for which the form of the nonsteady deformations is determined by several of the lowest harmonics, turned out to be the least stable. Measures related to stabilization of the energy distribution have been carried out in two directions:

an increase in the automation by virtue of the creation of a branched system for regulation of the reactor; and

a purposeful change in the composition of the nuclear fuel.

As a result of the development of measures of the first direction, a qualitatively new system of local automatic regulation of the energy distribution (LAR) and local emergency protection (LEP) which operates from intrazonal detectors [2] has been created and introduced into operational practice. The LAR system fulfills the function of automatic stabilization of the lowest harmonics of the radial-azimuthal energy distribution. Maintaining a specified capacity of the reactor, this system can, by virtue of auxiliary elements operating in the individual mode, automatically regulate the capacity in individual regions of the active zone. The LEP system accomplishes emergency power reduction in the case of local bursts of it, which can arise due to the failure of LAR elements or for other reasons. A structural peculiarity of the LAR and LEP consists of the use, for regulation of the capacity and protection of the reactor, of groups of (from 7 to 12) slave mechanisms with a regulating rod uniformly positioned in the active zone and surrounded by two LEP detectors and four LAR detectors. The average correction signal of the LAR detectors is used to control the rods. Triaxial chambers located in the central hermetic sleeves of the HGA serve as the detectors of the LAR—LEP system. As follows from operating experience the LAR—LEP system has exhibited high reliability and effectiveness.

Computational investigations of the effectiveness of the measures of the second direction have shown that when the initial enrichment of the fuel in ^{235}U is increased, not only do the dynamic properties of the reactor improve, but its technicoeconomic indices also increase due to an increase in the depletion depth and a decrease in the specific consumption of nuclear fuel. An important dependence of the variation of the time constant of the first azimuthal harmonic of the deformation of the energy distribution (τ_{01}) on the steam reactivity coefficient has been established. The smaller the value of the positive steam reactivity coefficient, the higher the stability of the energy distribution and the simpler the monitoring of the reactor. The most rational method for decreasing the steam coefficient is an increase of the ratio of the concentration of ^{235}U nuclei and the moderator nuclei in the active zone. A decrease in the steam coefficient due to a change to a fuel of 2% enrichement is estimated to be approximately 1.3 β, where β is the effective fraction of delayed neutrons. These conclusions have served as the basis for the adoption of the solution of increasing the enrichment of the RBMK-1000 fuel to 2% (Table 2).

The 8-yr operation of systems which provide for the control and regulation of the energy distribution in RBMK-1000 has confirmed the correctness of the engineering solutions which have been taken as the basis for their development. The combined and consistent functioning of the three systems the monitoring and protection system (MPS), which operates off lateral ionization chambers; the system for physical control of the energy distribution (SPCED) with respect to radius and height of the active zone, which uses β-emission neutron detectors of the cable type; and the SKALA system for centralized control (SCC) — has facilitated the reliable control and regulation of the energy distribution in all operating modes of the reactor. The accumulated experience of the assimilation and subsequent operation of the monitoring and control systems has permitted developing and incorporating measures directed at a further increase in the reliability of their operation. Among these measures one can count the conversion of the logic portion of the MPS to more reliable integrated circuits, which have permitted appreciably developing its functional possibilities with a reduction by several times in the dimensions of the electronic equipment, the replacement of the cable link in the slave mechanisms of the MPS by a belt link to increase their operational reserve, and the introduction of noncontact thyristor circuits for strong control of the MPS servomechanisms. The service term of the detectors for control of the energy distribution with respect to the radius of the active zone exceeds the operating time of the HGA in which they are mounted. In order to increase the reliability of operation of the detectors, soldered connections have been replaced by welded ones. The detector assemblies for control of the energy

TABLE 2. Basic Characteristics of the RBMK-1000 Fuel Cycle

Characteristic	Initial ^{235}U enrichment, %	
	1.8	2.0
Uranium depletion, MW•days/kg	18.5	22.3
Final ^{235}U content in unloaded fuel, kg/ton	3.9	3.5
Decrease in steam reactivity coefficient, β	—	−1.3
Ann. cnsmptn. of enriched uranium per priming, tons/GW•yr	50.5	41.5
Ann. cnsmptn. of fuel elements per priming of reactor, thousands/GW•yr	16.3	13.3
Ann. cnsmptn. of natural uranium per priming, tons/GW•yr	169	159
Oper. period of fuel, eff. days	1000	1350

distribution with respect to the height of the active zone preserve their effectiveness for 4 yrs.

A great deal of attention has been devoted to the perfection of thermal automation and emergency protection systems in the interests of increasing the reliability and safety of the operation of nuclear power plants with RBMK-1000. The equation of kinetics, hydrodynamics, and heat transfer and algorithms of the operation of the equipment and systems for automatic regulation of the parameters of a nuclear power plant are used in a mathematical model which has been developed for the investigation of transition and emergency conditions [3]. Upon comparison of the results of calculations with the data of the dynamic processes on operating units with RBMK-1000, it has been established that the model satisfactorily describes the dynamics of the power unit. Some emergency conditions associated mainly with the transition to natural circulation of the coolant have been studied on special test stands. In order to justify the reliability of the cooling of the active zone under conditions of natural circulation, three series of experiments have been performed under natural conditions on the first and third units of the LNPP and the second unit of the KNPP in steady-state and transitional regimes. As a result of the computational—experimental investigations, a set of measures have been developed which raise the reliability and safety of operation of the unit. One should point out the main ones:

Automatic reduction of the reactor power right down to its complete shutdown is introduced with emergency reduction of the flow rate of the supply water;

the modes of operation of the automatic steam-discharge devices and the number of main steam safety valves are optimized;

supplementary emergency protection of the reactor for a number of engineering parameters (reduction of the flow rate in the circulation loop, an increase in the pressure in the reactor space, and dehydration of the MPS channels) is introduced; and

a system of automatic regulation of the level and pressure in the separators is converted into a new element base of the KASKAD type which possesses the best characteristics, and the structure of the regulation system has been improved.

Based on the results of the start-up adjustment operations, experimental investigations, and operating experience, some changes in the construction of the individual reactor subassemblies and the equipment of the circulation loop have been introduced. A large part of further structural improvement has been accomplished not only on the nuclear power plants being designed with RBMK-type reactors, but also on the nuclear power plants which are operating and being constructed. For example, a redesign of the pipelines of the steam—water communications is being performed, a rearrangement of the steam pipes in the space of the separator rooms is being carried out, and optimal shimming of the steam-discharge fittings of the separators has been introduced for equalization of the steam loads and elimination of misalignments of the levels lengthwise and between adjacent separators.

Experience with systematic preventive and capital maintenance of the equipment of operating power units with RBMK-1000 has shown that in order to shorten the periods for carrying out the maintenance operations and to increase their quality, it is necessary to improve the

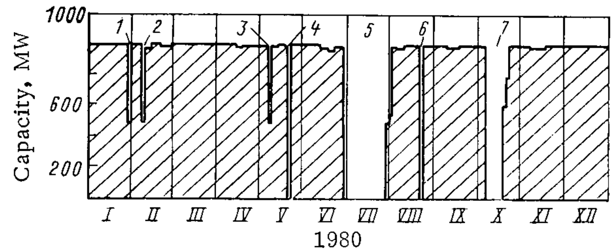

Fig. 5. Plot of the operation of the second unit of the ChNPP in 1980: 1) disconnection of the turbogenerator (TG) for the elimination of flaws on the T-junction of the separator—steam generator (SSG); 2) disconnection of the TG for replacement of a section of pipeline of warming steam condensate of the SSG; 3) disconnection of the TG due to failures of the automatic equipment of the pumps of the machine room; 4) shutdown of the unit for the elimination of leaks in the condensers; 5) planned shutdown of the unit for average maintenance; 6) shutdown of the unit for elimination of a leak in the MPS cooling loop; 7) planned shutdown of the unit for current maintenance.

maintenance technology and the methods of cooling a shutdown reactor. A method of preliminary formation of icy stoppers in the underwater communications has been developed for masssive replacement of the pressure-regulating valves and the detectors of flowmeters of the fuel channels after they have exhausted their reserve capacity. With this method of refrigerating, the necessary operations for maintenance of the indicated subassemblies is accomplished 4–5 times faster than when the design technology is used. A special system for diverting the residual heat generation with forced circulation of the coolant has been developed for the maintenance of the pipelines of the circulation loop without unloading the fuel from the active zone.

The intensive development of nuclear power has not only raised a series of immediate problems concerning providing for the safety of nuclear power plants, but has also required a tightening up of the requirements which are presented to the technical means for safety provision. First of all, this refers to emergencies associated with depressurization of the pipelines of the circulation loop. An instantaneous complete break in the pipeline with a maximum diameter of $\sim$900 mm is adopted as the maximum design emergency. The technical means of safety provision, the principal ones of which are the emergency reactor cooling system and the accident localization system [4], are calculated for this emergency. In order to determine the parameters and characteristics of these systems, a set of scientific research and experimental—structural operations is performed. As a result, systems have been designed which provide for an acceptable thermal regime of the fuel elements upon a fracture of any pipeline of the circulation loop and for localization of coolant ejections.

It is well known that the fuel temperature, the temperature of the graphite stack and the metal structures, and the margin of heat exchange until a crisis are the determining parameters which limit the power of channel uranium—graphite reactors with a boiling coolant. These parameters in an operating RBMK-1000 do not reach the limiting permissible values. Thus, the maximum power of a fuel channel at the nominal reactor power is about 2600 kW with a permissible value of 3000 kW, the maximum temperature of the graphite stack is 550°C with a permissible value of 750°C, the maximum temperature of the metal structures is 300°C with a permissible value of 350°C, and the margin of heat exchange until a crisis is no lower than 1.05–1.06. A plot of the load of the second unit of the ChNPP by months is presented in Fig. 5.

Experience with the successful operation of power units with RBMK-1000 at nominal capacity and the presence of reserves in the operation of the reactor equipment indicate that one can increase the reactor power without changing the dimensions and number of fuel channels by increasing the critical power of the channels at which a heat-exchange crisis arises [5]. This problem has been solved by means of applying heat-exchange intensifiers in the HGA. Test-stand experiments have shown that the power of an RBMK channel with intensifiers increases by approximately a factor of 1.5. The construction of a new HGA, with special mechanisms permitting an increase in the thermal loads, which has been developed for the RBMK-1500, has a high level of unification of the individual subassemblies with the HGA of the RBMK-1000.

At present, the first turn of the Ignalina Nuclear Power Plant with two RBMK-1500 units having an electric capacity of 1500 MW each is being constructed. Power-up of the leading unit will mark the start of the creation of a new generation of channel reactors, which, being more economical, should be a replacement for the well-recommended RBMK-1000. The construction of nuclear power plants with RBMK-1500 will permit reducing by 20-30% the specific capital expenditures in comparison with nuclear power plants with the RBMK-1000 and reducing the cited costs for electrical power.

Experience with the operation of channel uranium–graphite RBMK-1000 with boiling coolant confirms the validity of the adopted solution of creating in the USSR a large series of nuclear power plants with reactors of a given type. The accumulated experience in creating powerful channel power reactors is a good basis for their further refinement and development.

LITERATURE CITED

1. N. A. Dollezhal' and I. Ya. Emel'yanov, A Channel Nuclear Power Reactor [in Russian], Atomizdat, Moscow (1980).
2. I. Ya. Emel'yanov et al., At. Energ., 49, No. 6, 357 (1980).
3. I. Ya. Emel'yanov, S. P. Kuznetsov, and Yu. M. Cherkashov, At. Energ., 50, No. 4, 251 (1981).
4. I. Ya. Emel'yanov et al., At. Energ., 43, No. 6, 458 (1977).
5. N. A. Dollezhal' and I. Ya. Emel'yanov, At. Energ., 40, No. 2, 117 (1976).

DEVELOPMENT AND EXPERIENCE OF OPERATING FAST REACTORS IN THE SOVIET UNION

O. D. Kazachkovskii, A. G. Meshkov, F. M. Milenkov, V. P. Nevskii, L. A. Kochetov, V. I. Kupnyi, B. I. Lukasevich, V. M. Malyshev, V. V. Pakhomov, F. G. Reshetnikov, A. A. Samarkin, M. F. Troyanov, V. I. Shiryaev, V. A. Tsykanov, and D. S. Yurchenko

UDC 621.039.526

INTRODUCTION

The concept of the construction of fast reactors in the Soviet Union was promoted at the end of the 1940s by A. I. Leipunskii. It was based on the supposition about the more advantageous neutron balance, from the physics point of view, in a reactor with a "hard" spectrum. More than 20 yrs of intensive scientific-research and experimental-constructional work have been required in order to traverse the path from physical guesses to the first BN-350 reactor prototype. In order to consolidate the new trend in nuclear power generation, weighty arguments concerning the advantages of fast reactors by comparison with the simpler reactors were necessary, their success already having been demonstrated at this time, but such success was also due to the large efforts ground of nuclear power generation based on thermal reactors. A number of fundamental problems had to be solved in the fields of physics, heat-mass exchange, material behavior, chemistry, economics, and also time, in order to realize more clearly the acute necessity for the development of fast reactors.

Among the most important questions requiring resolution at that time, the following may be mentioned:

What realistic value of the breeding factor can be obtained in future large-sized power reactors?

What grouping (design) of the reactor will ensure the optimum value of the breeding factor?

It is possible to ensure the necessary safety of fast reactors and is control (automatic or manual) at all possible with such a reactor?

What specific power intensity must be ensured and what coolant most completely meets the demands of a fast reactor?

What structural and fuel materials can satisfy the requirements of fast reactors?

How practicable is the industrial chemical reprocessing of spent fuel and the commercial manufacture of fuel elements based on plutonium, and what are the fuel losses in this manufacture?

What are the reserves of nuclear natural fuel and how much time will be available to the community before the mass construction of fast reactors?

In order to answer these questions, the painstaking efforts of theoreticians and experimenters were necessary — it was necessary to construct a powerful experimental base. Thermohydraulic and material testing rigs were constructed, upon which research on thermophysics, hydraulics, and material behavior of liquid metal coolants (sodium, sodium-potassium, and mercury) were conducted; chemicotechnological rigs were constructed for the development of monitoring and purification of coolants; physics rigs (BR-1, BFS-1, BFS-2, and "Kobra") were constructed for the study of physics problems; and experiments on the fuel and the whole fuel cycle were developed. The joint work of the theoreticians and experimenters allowed the ini-

Translated from Atomnaya Énergiya, Vol. 54, No. 4, pp. 262-273, April, 1983.

 0038-531X/83/5404-0270$07.50

tial concepts to be refined significantly and allowed convincing proofs of the long-term outlook of the trend to be obtained. It was essential to proceed to the next stage of the work — the construction of experimental facilities and design work on prototype plants.

The construction of the research reactors BR-2 in 1956 (capacity 100 kW, coolant mercury), BR-5 in 1968 (5 MW, sodium), and the experimental power reactor BOR-60 in 1969 (60 MW, sodium) was an extremely important stage in the history of fast reactors. The development and operation of these reactors allowed the choice of the principal decisions to be confirmed finally for the BN-350 and BN-600 demonstration reactors: oxide core, sodium coolant, dense packing of the fuel elements in the fuel element assembly. Very valuable experience was obtained in the operation of sodium radioactive circuits. The feasibility emerged of proceeding to bulk radiation tests and investigations of different structural materials and types of fuel (mixed oxide, carbide, and carbide—nitride), and to tests of steam generators. The feasibility was shown of extended nuclear fuel breeding, using already tested decisions of the core with a breeding factor of 1.30-1.45. A great deal of experience was built up on transition regimes and safety; the high degree of safety and the excellent controllability of fast reactors with sodium coolant were confirmed (BN reactors). To summarize, all this allowed, with the necessary degree of justification, complete conversion to design work on demonstration facilities, and the gradual solution of problems of reliability and efficiency and of those scientific-technical questions which most of all are associated with large-scale facilities (fuel cycle, verification of different concepts of plant grouping and different plant designs, especially of the steam generators).

EXPERIENCE IN THE OPERATION OF ACTIVE FAST REACTORS IN THE SOVIET UNION

The current stage of development of fast reactors in the Soviet Union if characterized by the buildup of experience in the operation of three successfully active power reactors: the experimental BOR-60 and two large commercial reactors, the BN-350 and the BN-600. Operation of the BR-10 research reactor is continuing. In the present paper it is proposed to discuss only the results of recent years, as earlier information has been given repeatedly [1, 2].

BR-10 Reactor. From 1973 to 1979, the reactor has been operated at a power of up to 7.5 MW; plutonium dioxide was used as the fuel. On October 1, 1979 the reactor was shut down for overhaul. At this time, the maximum fuel burnup attained 14.2%.* All standard and experimental fuel element assemblies were removed from the reactor and the hermeticity of the fuel element cans was tested. According to preliminary data, the hermeticity of the cans was destroyed for ∿1% of the fuel elements. Nevertheless, this did not prevent the completion of the planned program and subsequent conducting of the repair work. After draining the sodium from the primary circuit, it was washed out three times by the steam—gas method, with a final washing with distillate. Samples were cut out at various points of the primary circuit for investigations, which confirmed the satisfactory state of the material of the circuit (steel OKh18N9T). At the instant of shutdown, the main reactor vessel was irradiated with a fluence of $8 \cdot 10^{22}$ neutrons/cm^2, which corresponds to ∿40 displacements/atom. Measurements showed that at the site of maximum fluence, the diameter of the central duct was increased by 3.10 ± 0.27 mm (swelling of the steel $\Delta V/V = 2.8\%$). The high fluence received by the material of the reactor vessel also was one of the principal reasons for shutting down the reactor for overhaul with a replacement vessel. By means of a specially developed tool and protective facilities, the vessel was cut off from the primary circuit and withdrawn from the reactor shaft. The overhaul also provided for the replacement of part of the main plant (cold trap, pump), reactor monitoring and control systems, and electrical heating and emergency cooling systems. At the end of 1981, the new vessel was installed in the shaft and completely joined to the primary circuit. The overhaul work is continuing. The next fuel charge is being prepared, based on uranium nitride.

BOR-60 Reactor. Major work has been carried out recently on the BOR-60 reactor, namely:

An extensive material testing program has been carried out, intensive investigations have been carried on a study of the radiation effect on the behavior of austenitic and ferrite—martensitic steels and fuel and moderating materials;

*Here and below, we have in mind the fraction of heavy atoms.

a micromodulular steam generator of Czechoslovakian construction has been tested (1973-1981), and investigations of a large-scale model of the BN-600 steam generator have been conducted since 1978; in 1981, tests of the so-called reverse steam generator, also developed by specialists of Czechoslovakia, were started; and a study of the monitoring and safety systems of the steam generators is continuing;

a program and experimental equipment for carrying out work on a study of sodium boiling in the core* have been developed, a study of the behavior of radioactive corrosion products and fission products in the circuit has been continued, and methods of purifying the sodium of the primary circuit from the most dangerous radionuclides have been investigated.

Recent work has been completed on the creation of a design for a trap based on graphite, which has undergone tests in the BOR-60 and then in the BN-350. By pumping the sodium through the trap in the BOR-60, ∿20 TBq (540 Ci) of cesium were removed from the circuit, and the γ background in the primary circuit compartments was reduced by a factor of ∿2.6 [3].

Among the overall achievements in the field of material behavior investigations, the assimilation of vibration technology in the manufacture of the fuel elements should be mentioned, which is interesting from the point of view of setting up an automated process for the production of fuel elements of mixed uranium–plutonium oxide fuel. This required the carrying out of an extensive complex of technological investigations, involving the following:

determination of the conditions for achieving the required density values of the mixed fuel and uniformity of distribution of plutonium oxide in the oxide mixture;

a study of the dynamics of structural changes in the initially molded filling and vibro-compacted fuel column during raising of the reactor power;

a study of the redistribution of the fuel components over the height and radius of the fuel elements during its lifetime tests;

a study of the temperature conditions and swelling of the fuel elements;

a study of gas release [4].

As a result of the investigations carried out, confirmation emerged that fuel elements prepared from a mixture of oxides by vibration technology are able to provide the same power intensity of the core and the same burnup as pelleted fuel elements. For the final verification of these preliminary conclusions an additional program was planned, according to which in 1981 a set of fuel element assemblies was prepared, based on mixed fuel, by vibration technology; these fuel element assemblies were loaded into the BOR-60 reactor.

For the purpose of determining the prospects for increasing the breeding in fast reactors, the investigations of a metallic fuel in the BOR-60 are of important value. The investigations of a metallic uranium and uranium–plutonium fuel were directed at the prevention of large swelling of the metallic fuel and its significant interaction with the cladding. As a result of many years of functioning in the BOR-60 reactor, a large number of tests of experimental fuel elements with metallic fuel have been conducted. A burnup of ∿6% was achieved in the experimental fuel elements with uranium–plutonium fuel, in conditions similar to those characteristic for the present-day fast reactors with oxide fuel.

BN-350. Since the time of the power generation startup of the BN-350, the first in the Soviet Union and the most powerful commercial fast reactor at that time, 9 yrs have elapsed. The only major plant defect which appeared during the whole process of assimilation of the power of the station was a defect of the steam generators: repeated breakdown of the inter-circuit sealing [5]. The principal cause of this was the poor quality of manufacture and welding of the lower end components of the heat transfer tubes. Because of the special features of the circulation from the direction of the tertiary circuit (natural circulation in the Field tubes), concern was caused by the primary pores and by the quality of the feed water, particularly the iron content in it (15-20 μg/kg). The overhaul of all the damaged (five of the six) steam generators was completed in 1975, and the power of the facility was raised to 520 MW (thermal); in March, 1976 it was raised to 650 MW (thermal), and in September, 1980, to 700 MW (thermal). At 700 MW the reactor provides an electrical capacity of 125 MW (elec.) and additionally generates 85,000 tons per day of distillate. In May, 1980,

*Specialists of the German Democratic Republic participated in these investigations.

assembly was completed of the first steam generator of Czechoslovakian design and it was brought to the condition of normal operation.

The time-utilization factor over the period from the instant of startup to 1977 amounted to 86%, and when operating at a power of 650 MW (thermal) and above, it was 88%, which corresponds to $\sim$7700 h of operation of the facility on power annually.

Other important economic indexes are the attained fuel burnup and the operating lifetime of the main plant. For the first time, the planned fuel burnup of 5% in the central section of the fuel element assemblies was attained in 1976. At the present time, this index is equal to 5.8% and is dependent on the permissible dimensions of the change of shape of the hexagonal sheaths of the fuel element assemblies.

The initial operating lifetime of the major part of the newly developed nonstandard plant was exceeded (with the exception of the steam-generator evaporators.). The operating lifetime of the plant as of January 1, 1982 is shown below:

Primary circuit pumps	Lifetime increased from 20,000 to 50,000 h
Secondary circuit pumps	Lifetime increased from 20,000 to 50,000 h. Maximum operating period amounts to 57,000 h
Steam-generator evaporators (after overhaul)	45,000-55,000 h
Steam-generator steam superheaters	51,000-57,000 h
Control and safety equipment	Operating up to now without replacement
Recharging systems, rotatable plugs	Ensure normal recharging cycle; observed jamming of plugs eliminated by increasing the sodium temperature from 200 to 250°C.
Sodium—sodium intermediate heat exchange	Operating without breakdown of sealing for more than 60,000 h
Primary circuit slide valves with diameter 500 and 600 mm	Operating without faults; provide absolute sealing when closed
Cold traps of primary and secondary circuits	Operating up to the present time without replacement

It can be seen that all the main plant has operated for more than 9 yrs without replacement, including all the steam-generator steam superheaters. The steam-generator evaporators after overhaul also demonstrated the considerable lifetime of accident-free operation up to 55,000 h. The evaporators of one regular steam generator operated accident-free for 56,000 h. At the present time the generator is dismantled and has been transferred to research; a second generator of Czechoslovakian design has been installed in its place.

Almost 10 years of operating experience also confirms the high degree of safety of the facility. Thus, during the whole of this time, there was not one case of sodium leakage from the primary circuit; in the secondary circuit during the same period, two leakages were recorded (in the sampling and oxide indication systems). In each case the leakage did not exceed 10 liters.

The radioactivity of the discharges into the ventilation duct is determined by ^{41}Ar and amounted to not more than $7.4 \cdot 10^{11}$ Bq/day (20 Ci/day), and the radioactivity of discharged aerosols was a factor of 10^6 less than the argon activity. One shutdown of the facility occurred during operation, in which all safety devices functioned normally.

Recently, the following systems and plants have been modernized:

The geometrical dimensions of the fuel elements have been unified with the fuel elements of the BN-600 reactor (diameter 6.9×0.4 mm); at the same time, the gas compensation space of the fuel elements has been increased, which has led to a reduction of the pressure under the cladding, and to a reduction by a factor of 10 of cases of depressurization of the fuel element cans;

the control and reactivity compensation rods have been modernized, the efficiency of the rods has been increased, and the operating time of the reactor on power between two shutdowns for recharging has been increased from 55 to 73.5 days;

Declassified and Approved For Release 2013/02/06 : CIA-RDP10-02196R000300020004-6

the steam generator feed has been converted to water with total and extreme purification (desalination), and a cycle of complex feed water processing has been introduced;

the return valves at the head of the primary circuit pumps have been redesigned.

At the present time the reactor is also being used for experimental work in physics, material behavior and sodium technology. Among the most important tasks and achievements here the following should be mentioned:

a cycle of experimental work on refining the breeding parameters (conversion). Measurements carried out have allowed the experimental value of the conversion factor 1.05 ± 0.05 to be established, which agrees quite well with that predicted by a computational method (1.03);

a cycle of work to study the changes of shape and mechanical properties of materials in conditions of irradiation with high fast neutron fluences; in individual fuel element assemblies, a burnup of 6.6% has been achieved and the maximum burnup in an experimental fuel element assembly attained 7.7%. Investigations of the change of shape of spent fuel element assemblies have shown that as a consequence of radiation swelling and radiation creep, the diameter of the hexagonal can is increased on the average from 96 to 97.2 mm, with a sag of 15-17 mm.

These and other experimental studies have been conducted in the BN-350 and not to the detriment of the planned tasks on the generation of electric power and distillate, the fulfillment of which is the main criterion for assessing the activity of the staff and the efficiency of the facility.

<u>NB-600.</u> In contrast to the BN-350, the grouping of the plant of this reactor is integral; the diameter of the vessel is 12.8 m and the height 13.0 m.

Moreover, in contrast to the facility with the BN-350, in the unit with the BN-600 straight-through modular steam generators are used, which significantly increase the thermodynamic parameters of the steam and the temperature of the sodium circuits.

The design characteristics of a unit with the BN-600 have been described in quite some detail in [6]. Therefore, we shall confine ourselves only to the most important general conclusions which can be made on the basis of the experience gained with the construction and startup of the third unit of the Beloyarsk nuclear power station.

The vessel and the intravessel structure, including the whole of the primary circuit, were assembled in an assembly area. Experience has shown that the preparation and assembly of such a unique plant could be carried out successfully, and any difficulties encountered as a result of this could be overcome.

The high saturation of the reactor with metal structures and plant caused concern because of the possible vibration of the intravessel equipment, access to which after filling the reactor with sodium, and especially after bringing the reactor to power, is extremely difficult. Therefore, design measures were taken for the installation of a large number of vibro- and strain-sensors, and special programs have been compiled for the stage-by-stage verification of the vibrocharacteristics of the reactor. Not one sensor has recorded vibration in excess of the limit of sensitivity (0.6 mm).

Among the problems associated with the sodium coolant, two have required close attention. The first was the assurance of the necessary quality (impurities, suspended matter) of the coolant; concern was caused by the narrowness of the reactor space, especially in the last stage of assembly work, and by the large volume of welding operations inside the vessel. The measures taken were successful in providing a blocking temperature of 150-155°C. The second problem was the assurance of safety during acceptance, cleansing, and pressure transfer from railroad tank cars to the primary and secondary circuit of 1800 tons of sodium. It should be mentioned with satisfaction that during these operations with sodium, and also during the startup-adjustment operations, there was only one case of sodium leakage through the sealing of the detachable section of the tank to the intake pipeline, and a few leakage droplets were noted through the sealing of the sodium slide valves of the steam generators. None of the personnel came to harm.

In order to refine the physical characteristics of the reactor, a program was prepared and executed for the physics startup and physics measurements. As a result, the critical loading was refined, the efficiency of the control rods and the fuel element assemblies and

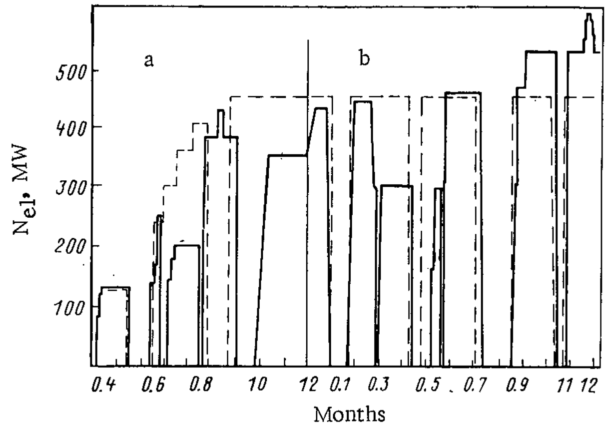

Fig. 1. Enlarged graph of the loading of the third unit of the Beloyarsk nuclear power station [----) according to plan; ——)]: a) 1980, power utilization factor = 26.8%, breeding factor = 68%; b) 1981, PUF = 50.1%, BF = 71.1%.

the power distribution over the core volume was measured, and the reactivity coefficients, essential for operation, were obtained. The measurements showed the excellent agreement of these parameters with the calculated values. The efficiency of the scram rods was found to be somewhat less than the design figure; however, it ensured completely safe operation of the reactor. A slight nonagreement of the temperature and power effects of reactivity also appeared. The series of physics measurements carried out allowed the power startup of the reactor to proceed and allowed the subsequent assimilation of the operating and power regimes of the facility.

The conclusions, made on the basis of experience of more than two decades of operation of the facility on power, reduce to the following:

The plant of the facility operates stably and reliably. The operating characteristics of the plant and circuits correspond mainly to the design values. This has allowed the power to be raised systematically up to 90% at the end of 1981 (see Fig. 1). The power restrictions of the facility were due to the permissible operating parameters of the fuel elements with fresh fuel during transition of the core to the steady state, which was achieved in the first half of 1982. On December 18, 1981, the reactor was brought to nominal power and it was operated at this level during three days for the series testing of all systems of the facility.

Investigations of the hydraulics of the primary circuit indicated excellent agreement between the design and actual characteristics in all operating regimes of the facility. As a result of measurements by means of a flow meter device, installed in the rotatable plugs, it was found that the sodium flow rate through the fuel element assemblies was close to the design figure.

During operation of the BN-600, the inadequacy of mixing of the relatively cold coolant leaving the fuel element assemblies of the radial breeding zone, with the hot coolant leaving the fuel element assemblies of the core, was verified. The observed laminar flow of coolant at a different temperature leads to temperature instability in the region of the thermocuples installed in the reactor mixing chamber. Therefore, in order to regulate the reactor, thermocouples installed above the caps of the fuel element assemblies are used.

The hydraulic resistance of the loop of the secondary circuit is somewhat lower than the design value. Nominal flow rates in the loops are ensured with a speed of rotation of the pumps of ∿720 rpm. Because of this, the operating speed range was limited to 250–720 rpm, which is achieved without difficulty with systems for regulating the number of revolutions and with a control system.

Fuel rechargings showed that the combination of mechanisms of the fuel element assembly recharging system allows the necessary number of fuel element assemblies to be replaced in the remote and automatic control regimes with small losses of time (not more than 1 h per replacement of one fuel element assembly).

The startup-adjustment operations revealed the advisability of certain changes in design of the scram rod slave mechanisms in order to increase their reliability in the case of erroneous withdrawals of the mechanism pickup into the rigid support. The design studies of the BN-800 and BN-1600, taking account of the experience in operating the BN-600, confirmed the feasibility and advantage of using, for all three of these facilities, a unified design of the scram rod pickup mechanisms.

In the initial period of operation, several cases of leakage in the steam generator tubes were noted. The leakage monitoring system detected a looseness in the early stages of their occurrence.

The separation of the steam generator into sections and modules was justified, since when individual modules were out of action, operation can be continued after cutting off the defective modules by means of the armature.

The most probable cause of leakiness in the steam generator tubes is the development of microdefects in the welded joints and in the material of the tubes in operating conditions. These defects are below the threshold of sensitivity of the instruments for monitoring the quality of the materials during manufacture, because of which the necessity arises for increasing the sensitivity of the monitoring devices and for constructional measures to reduce thermocyclic stresses in operating conditions.

For the initial period of operation, taking account of the design of the fuel element assemblies and the materials used, the maximum values of the fuel burnup were established: 5% for zones of low enrichment and 7% for zones of high enrichment. When achieving this fuel burnup, no difficulties arose due to radiation phenomena of swelling and creep of the structural materials. For the purpose of increasing the fuel burnup, jacketed tubes for the fuel element assemblies and the fuel element cans were provided, manufactured in the cold-deformed state, and also some variations of their dimensions, in order to increase the compensation power of the reactor core by comparison with the radiation change of dimensions of the jacketed tubes. The use of new, more radiation-resistant structural materials is also being considered, which will allow the fuel burnup to be additionally increased and thereby improve the economic indices of the facility.

In conclusion, we note that the new design solutions on the BN-600 reactor (integral grouping, design of the governing plant, and decisions on the main systems) were justified. The facility can operate reliably at a power of up to 100% nominal, ensuring the buildup of much practical experience for the improvement of both the BN-600 reactor itself, and the newly developed facilities BN-800 and BN-1600.

DEVELOPMENT OF THE NEXT GENERATION OF REACTORS

The successful operation of the BN-350 and BN-600 has shown the practicability, reliability, and safety of the new prospective trend in nuclear power generation. The BN-800 and BN-1600 designs being developed are intended for commercial introduction. The basis of these reactors is the experience and achievements obtained with their forerunners. The main problems of the new developments, in addition to large-scale changes, reduce to the following: a further increase of plant reliability and safety of the station as a whole, improvement of the economic indices, and increase of secondary fuel breeding.

At the present time, the technical designs of the BN-800 and BN-1600 have been developed. Their principal design parameters, with the exception of power, are similar.

The major part of the structural-grouping decisions on the BN-800 is similar to the decisions on the BN-600. The principal differences between the BN-800 and the BN-1600 reduce to the following:

The volume of the core is increased because of the increase of its height from 750 to 950 mm and the increase in the number of fuel element assemblies; the use of a three-zone scheme of power smoothing in the core (owing to fuel of different enrichment);

the gaps between the fuel element assembly jackets are increased in order to increase the fuel burnup (up to $\sim$10%);

the number of recharging mechanisms is reduced (one instead of two) due to the increase of the number of rotatable plugs (three instead of two);

the design of the bearing subassembly of the vessel has been changed, which will allow the strength and vibroresistance of the bearing collar to be increased, the stresses in the bottom of the vessel to be reduced, the dimensions of the forging to be increased, and the number of welded seams to be reduced;

an additional system of reactor cooling is provided, by means of air coolers, connected in parallel with the main coolant course of the secondary circuit;

the scheme for connecting the steam generators to the sodium circuit has been changed (the sodium intermediate steam heater has been excluded, and in place of it live steam heating in the superheaters, installed in the machine hall of the station, has been introduced; the surface area of the heat exchanger has been reduced because of the reduction of the number of modules in the steam generator section — up to 20 per steam generator instead of 24.

The increase of electric power of the BN-800 by comparison with the BN-600 is provided with approximately the same capital expenditure on the plant, which is one of the main factors for the improvement of the economic indices. Moreover, a considerable economy is achieved because of the use in the BN-800 of the major part of the plant developed for the BN-600.

The BN-1600 reactor is also being constructed for series commercial nuclear power stations with a large capacity. The thermal layout and the installation decisions are mainly similar to the decisions of the BN-600 and BN-800. The grouping of the reactor is integral. The plant of the primary circuit is disposed in the reactor vessel, with a diamter of ∿19 m. Preference is given to mounting of the reactor vessel in the upper part to a shielded strong structure, absorbing the mass of the entire plant and coolant of the primary circuit. This has necessitated the construction of screens, providing shielding of the reactor vessel roof and the upper strong ceiling from the action of the high temperature and thermal cycling.

The structural changes in the BN-1600 are related mainly with the increase of power and the corresponding increase of dimensions of the fuel element assemblies.

For the purpose of increasing the operating efficiency of the BN-800 and BN-1600, the experience of the BN-600 is being analyzed industriously and, taking it into consideration, the necessary improvements are being introduced. Work must be continued on raising the quality of manufacture of the steam generators over the whole cycle — from the choice of materials to pilot tests of the steam generators ready for delivery. Special attention should be paid to the welding to tubes with the pipe panel (improvement of both the welding technique and the procedure for subsequent inspection). Although a modular design of a straight-tube steam generator has been chosen for the BN-800 and BN-1600, other designs are nevertheless being developed. Specialists of Czechoslovakia are involved in this. Monitoring systems for leakage in the steam generators are being modernized, and the quality of the accessories is being increased. Since, in the BN-1600, it is proposed to use higher efficiency equipment (e.g., pumps), it is planned to construct the necessary experimental base of its development.

The safety of the BN-800 and BN-1600 facilities is being increased by means of the following measures:

Additional improvements in the electromechanical and electronic parts of the reactor safety system are being introduced.

Monitoring of the state of the core and the reactor protection units is being modernized. In particular, it is acknowledged as advantageous to develop a visual observation system below the sodium layer, similar to the system developed by the French and American specialists.

The fire safety system is being modernized and new methods of monitoring in the case of the occurrence and extinction of a sodium fire are being developed. In this respect there is special interest in the latest achievements in the field of passive means of fire extinction (unified means of sealing off potentially dangerous compartments are being developed; light substances which quench the sodium which has overflowed over the floor surface have been well recommended — the main advantage of which is the possibility of their disposal beforehand over the surface of the compartment floor).

The system for the removal of residual heat release, due to the introduction of autonomous contours of the natural circulation, including a third, air, circuit, is being modernized.

It is proposed once again to return to the discussion of the advisability of an additional external housing, calculated for an aircraft crash and the internal pressure of a maximum sodium fire.

In the present-day stage, until relatively inexpensive uranium is available for the fuel cycle, better economic indices than in thermal reactors can hardly be achieved successfully on BN pilot plants. This is due in the first place to the relatively high capital costs on the BN facility. Thus, the specific capital costs on power plants with BN-600 is higher than the similar costs on the fifth unit of the Novovoronezh nuclear power station with VVÉR-1000 by a factor of 1.6. Although here the difference in capacity and climatic conditions is not taken into account, nevertheless these costs are quite high. Therefore, one can understand the tendency to reduce capital costs on future facilities with BN reactors. However, this natural desire contrasts with the tendency to examine excessive means of monitoring, protection, inspection, and safety on pilot models. Therefore, the planners at the present time use only those methods of reducing costs which do not enter into contradiction with the assurance of safety. Among the decisions to reduce costs on the BN-800 and BN-1600 pilot plants may be mentioned the use of cheaper materials, where this is possible, to replace austenitic steels by low-alloy steels (shielded casing, internal neutron and thermal shields); increase of the capacity of individual plants and stations; and increase of the fuel burnup. The reduction of the construction times of the stations and transition to series construction may make a marked contribution to the solution of the economy problem.

Conjunctural considerations at present allow a low breeding coefficient to be accepted and, moreover, for the present this is even advantageous from the point of view of economy. However, in proportion to the utilization of the fuel cycle factories and the reduction of the lifetime of the nuclear fuel, the problem of breeding develops into the category of immediate and most important problems. The dynamics of the development of nuclear power generation in the Soviet Union indicates the necessity for the development of breeder reactors with a breeding factor of 1.6. It seems that this breeding factor can be obtained only with the use of a fuel which is more dense than oxide. For this, a program is being implemented to study the feasibilities of using metallic uranium in the reactivity control agents, in the end and lateral breeding zones, and in the core of a heterogeneous reactor. The possibilities of using a uranium-plutonium metallic fuel in the core are also being studied, and a study is continuing of carbide and carbide-nitride fuel. However, the primary problem is the use of a mixed oxide fuel (and its reprocessing), which will allow the economic indices of facilities with BN reactors to be improved markedly. After tests of the experimental fuel elements with mixed oxide fuel in BR-10 and BR-60, the first 10 full-scale fuel element assemblies with this fuel will be manufactured and tests will be run in the BR-350 reactor, in order to refine the breeding factor and to obtain additional data about the effect of different technological factors on the efficiency of the fuel elements. In the future, a thorough economic comparison is proposed of the different technological alternatives for the manufacture of fuel elements from mixed oxide fuel for their mass remote-controlled production.

Thus, considerable experience has been built up in the Soviet Union on the development of sodium-cooled fast reactors. This experience confirms the principal theoretical prerequisites for the new trend and indicates the feasibility of constructing reliable commercial facilities, and also shows the paths for ensuring the required nuclear fuel breeding characteristics. The purpose of this is the achievement of the necessary tempos of the economy of natural uranium and the gradual transition of nuclear power generation to self-provision with fuel.

LITERATURE CITED

1. O. D. Kazachkovskii et al., At. Energ., 43, No. 5, 343 (1977).
2. L. A. Kochetkov and Yu. E. Bagdasarov, Report at the International Symposium on Design, Construction and Experience in the Operation of Demonstration Sodium-Cooled Fast Reactors [in Russian], IAEA-SM-225/78, Bologna (1978).
3. N. V. Krasnoyarov and V. M. Polyakov, Preprint of the Scientific-Research Institute of Nuclear Reactors [in Russian], Dimitrovgrad, 6(459) (1981).
4. V. A. Tsykanov, Preprint of the Scientific-Research Institute of Nuclear Reactors [in Russian], Dimitrovgrad, 1(454) (1981).

5. D. S. Yurchenko et al., Report at the International Symposium on the Design, Construction and Experience in the Operation of Demonstration Sodium-Cooled Fast Reactors [in Russian], IAEA-SM-225/60, Bologna (1978).
6. V. M. Budov et al., The BN-600 Fast Neutron Nuclear Reactor — Facility for Power Generation of the Near Future [in Russian], Nuclex-75, Basel (1975).

PATHS FOR THE DEVELOPMENT OF FAST POWER REACTORS WITH A HIGH BREEDING FACTOR

S. B. Bobrov, A. V. Danilychev, V. A. Eliseev,
O. A. Zhukova, Yu. A. Zverkov, V. G. Ilyunin,
V. P. Matveev, A. G. Morozov, V. M. Myrogov,
A. I. Novozhilov, V. V. Orlov, I. S. Slesarev,
S. A. Subbotin, M. F. Troyanov, and B. F. Shafrygin.

UDC 621.038.526

The assurance of nuclear power generation being developed with fuel resources over a long time is possible with the presence of fast breeder reactors in the system. Estimates show that for the necessary rate of growth of this multicomponent nuclear power generation, fast reactors are required with high breeding factors (BF $\approx$1.6; T_2 $\approx$7-9 yrs). The construction of these reactors usually is associated with the use of prospective types of fuel — carbide, nitride, and metallic. At the same time, it will be very tempting to consider the possibility of the partial use of uranium metal fuel in reactors with a well-studied and sound oxide fuel, within the scope of the heterogeneous core concept. Such a core does not require high thermal loadings on the metallic fuel elements and, above all, large burnups in them. In the majority of cases, the burnup for a metallic fuel can be limited to the value $P^{max} \approx$ 1-2%.* An important question, determining the operating efficiency of metal fuel elements in these conditions, is the interaction of the metallic core with the cladding at the working temperature values of the fuel elements. However, there is a path for solving this problem, consisting in the construction of barrier blankets.

The investigations carried out in recent years on the improvement of the breeding characteristics of fast reactors have stimulated practical interest in the use of these heterogeneous cores [1-4]. Heterogeneous cores, in which fuel element assemblies are used with fuel elements containing oxide-enriched and dense-metallic fuel, operating in these conditions, may have marked advantages by comparison with the conventional homogeneous cores with oxide fuel with respect to the principal breeding characteristics [5, 6].

In the results of investigations of heterogeneous oxide—metal compositions, presented earlier, the dependence of the principal breeding parameters of the fuel on the fraction of metallic fuel ε_m and the configuration of the inner breeding zones has been analyzed. These parameters are the breeding gain (BG), the excess plutonium made r, the excess plutonium production R in the breeder reactor system, developing at a specific rate ω (%/yr), the doubling time T_2 (for a duration of the external fuel cycle T_{ex} = 1 yr), and also certain parameters determining the safety of the breeder reactor (change of reactivity due to the Doppler DER effect in the working range of variation of temperature, and the total sodium void reactivity effect ΔK^{Na}).

The investigations were conducted on different models of heterogeneous cores. In all cases the BNAB-78 library of group constants was used in the calculations [7].

INVESTIGATIONS OF THE CHARACTERISTICS OF HETEROGENEOUS CORES WITH OXIDE—METALLIC FUEL ON MODELS OF HETEROGENEOUS CELLS

The core was represented by a multiplicity of two-dimensional (r, z) cylindrical cells, containing oxide fuel in the form of "islets" surrounded by a layer of metallic uranium. The dimensions of the islets were chosen equal to the dimensions of the fuel element (quasihomo-

*Here and in the future, burnup refers to heavy atoms.

Translated from Atomnaya Énergiya, Vol. 54, No. 4, pp. 269-273, April, 1983.

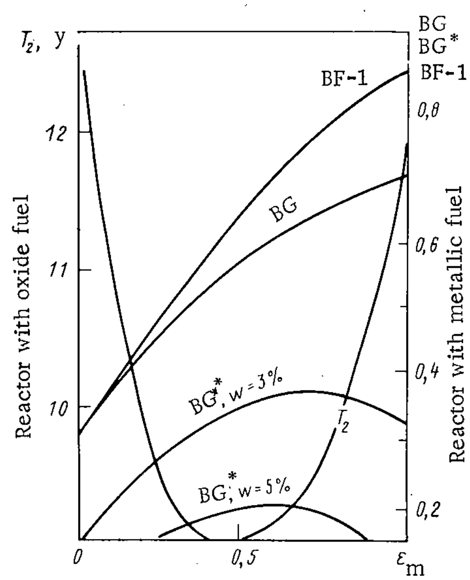

Fig. 1

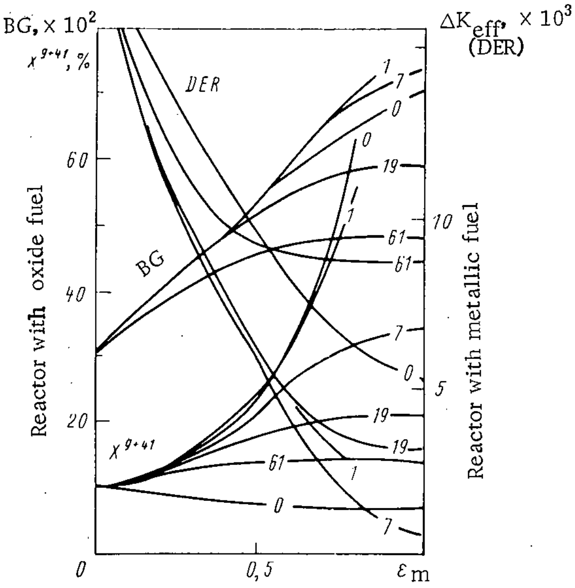

Fig. 2

Fig. 1. Dependence of T_2, BG, $BG^* = BG\left(1 - \frac{T_2\omega}{\ln 2}\right)$, and BG-1 on the fraction of metallic fuel ε_m for reactors with a fine heterogeneous structure (No. 0).

Fig. 2. Dependence of BG, DER, and X^{9+41} on the fraction of metallic fuel ε_m and the heterogeneity dimensions (values of X^{9+41} of composition No. 0 are shown for the homogenized model).

geneous composition) No. 0, one assembly (with a diameter "below the key" of 150 mm) No. 1, a symmetrical module of 7 assemblies (No. 7), 19 assemblies (No. 19), and 61 assemblies (No. 61). The latter composition simulated a small homogeneous reactor with oxide fuel and a metallic shield. The cells comprised the end shields with a thickness of up to 50 cm. The volume fractions of the core components and the end shield were: fuel 0.40, sodium 0.22. The density of the oxide fuel in the core was 8.6 g/cm³, and of the metallic fuel 15.5 g/cm³. The isotopic composition of the plutonium corresponds to that made in thermal reactors. By varying the fraction of metallic fuel, it is possible to convert from a conventional reactor with oxide fuel to a high power homogeneous reactor with metallic fuel, operating in the sparing regime. For this transition (in the calculations T_2) with composition No. 0, the enrichment of the oxide fuels in Pu did not exceed 30% (all the additional plutonium necessary or criticality in this computational model was disposed uniformly in the metallic fuel).

Figures 1 and 2 show the dependences of the fuel enriched in the isotopes ^{239}Pu and ^{241}PU (X^{9+41}), the breeding characteristics, and the safety on the fraction of metallic fuel in the composition of the reactor core. The following limitations are assumed for the oxide fuel: maximum burnup P^{max} = 10%, maximum linear power of the fuel element q_l^{max} = W/cm.

Analysis showed the following:

With the introduction, into the reactor with oxide fuel, of a small fraction of metallic fuel, a rapid increase of BG and BF is observed, but with further increase of ε_m the indications of saturation of these parameters is appreciable. This particularly concerns the BG, determining the change of r and R. The retardation of the increase of BG with a large fraction of metal ε_m in relation to the increase of BF − 1 begins because of the increased contribution of fissions in the raw isotopes (β) and the decrease of $\alpha = \left\langle \frac{\sigma_c}{\sigma_f} \right\rangle$. As a result, the specific (per unit of power) production of plutonium increases in the region of large values of ε_m more slowly than BF − 1.

In the region of small values of ε_m, the parameters BF and BG and almost independent of the geometrical heterogeneity dimensions. With relatively large fractions ε_m of metallic fuel, the physical effect of the additional increase of BG, noted earlier by S. M. Feinberg and due to "ruggedization" of the neutron spectrum is small oxide modules with increased enrichement of X^{9+41} in them, is important. As a result, an additional increase of BG occurs;

however, for the stated reasons, it is small. Moreover, in these high-enrichment "modular" reactors, the problem arises of the rapid increase of the fuel charge, which leads to a significant worsening of R and T_2.

The doubling time T_2 of compositions with a fine heterogeneity structure (No. 0) has a significant minimum for $\varepsilon_m^{opt\,T_2} = 0.3$ to 0.4, owing to the rapid increase of BF for small values of ε_m and the insignificant increase of the specific fuel charge. A homogeneous reactor with metallic fuel in the sparing regime of operation is significantly inferior with respect to T_2 than reactors with a heterogeneous core and is approximately equivalent to a homogeneous reactor with oxide fuel.

The optimum compositions with respect to R of heterogeneous reactors are slightly biased relative to $\varepsilon_m^{opt\,T_2}$ to the side of large values ($\varepsilon_m^{opt\,R} = 0.5$-$0.6$). Thus, according to the breeding indices, determined by the combination of the parameters R, r, and T_2, heterogeneous compositions with metallic fuel fractions in the range $\varepsilon_m^{opt\,R,r,T_2} = 0.3$-$0.7$ should be assumed to be optimal. In this case, by comparison with reactors with oxide fuel, an almost twofold increase of the excess fuel production and a marked (up to 30-40%) reduction of T_2 can be ensured.

The strong dependence of the Doppler reactivity effect on ε_m associated mainly with change of "hardness" of the neutron spectrum, will allow preference to be given to heterogeneous cores with a small fraction of metallic fuel.

Compositions with a fine heterogeneity structure are most desirable in power-generating breeder reactors, because of the better power uniformity and the possibility of stabilization of the power of the fuel element assemblies.

When analyzing the structure of developing nuclear power generation, the following must be taken into account: If it appears that the increased fuel charge of heterogeneous reactor breeders will limit the increase of the power of breeder reactors in the initial stage of their development, because of a shortage of plutonium, then it is economically advantageous — as shown in [3] — to introduce, into the U—Pu fuel charge of the breeder reactor, slightly enriched ^{235}U.

INVESTIGATIONS OF THE CHARACTERISTICS OF HETEROGENEOUS OXIDE—METAL CORES OF RADIAL—ANNULAR GROUPING

The groupings of heterogeneous cores are very varied: islet, symmetrical rings, modules, etc., types. However, from the point of view of the BG their differences are immaterial, as the main contribution to an increase of the BG is made by an increase of the average density of the breeder material in the core. When considering a heterogeneous core, the simplest radial—annular grouping was chosen. It supposes the presence of a central insert consisting of seven fuel element assemblies and three concentric rings, each formed with one array of breeder fuel element assemblies. The heterogeneous core of the type considered, with oxide—metallic fuel, allows the BG to be increased up to 0.46 and T_2 to be reduced to 9 yrs. It may be noted that the fuel charge of the heterogeneous core is somewhat larger by comparison with the conventionel core (by approximately 12%).

The parametric investigations of the heterogeneous core were carried out for different cases. When determining the optimum with respect to T_2, the number of breeder fuel element assemblies in the core was varied. Two volume fractions of fuel in the core were considered ($\varepsilon_f = 0.45$ and $\varepsilon_f = 0.35$) with two plutonium buildups in the breeder fuel element assemblies (2 and 4%). For comparison, versions with UO_2 in the breeder fuel element assemblies were also analyzed. The volume fraction of the breeder material in these fuel element assemblies was assumed to be equal to 0.55 for $\varepsilon_f = 0.45$ and 0.35, and the volume fraction of metallic uranium in them was the same as the volume fraction of fuel in the fuel element assemblies of the core. The results are shown in Fig. 3. In order to determine the feasibility of reducing the specific fuel charge in the heterogeneous core, the diameters of the fuel elements and the density of the fuel were varied. When varying the diameter of the fuel elements, their pitch and linear power were preserved, and when varying the density of the fuel (e.g., because of a change of the central opening in the fuel), the pitch, diameter of the fuel element, and the linear power were preserved. When determining T_2 in these investigations, the possibility was taken into account of the high fuel burnup in the heterogeneous versions, because of the reduction of the neutron fluence by comparison with the fluence in a conven-

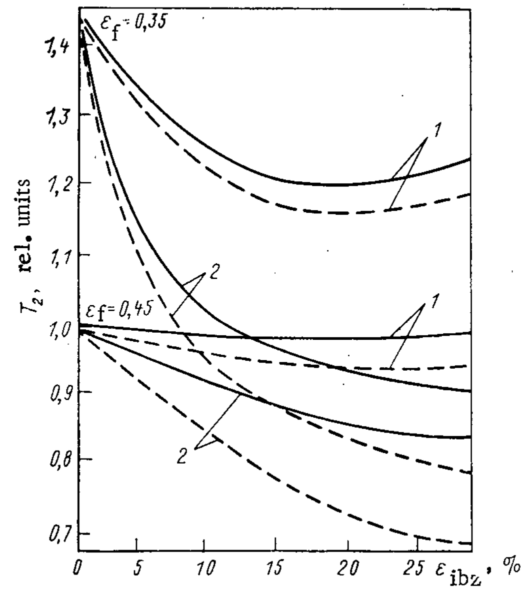

Fig. 3. Dependence of T_2 on the fraction of the inner breeding zones ε_{ibz} in a heterogeneous radial—annular core: 1) UO_2; 2) metallic uranium; ——) 2% Pu; - - - -) 4% Pu.

tional core. The safety parameters were also investigated: the sodium void reactivity effect (SVRE) and the Doppler effect (DER). A model of the BN-1600 reactor was considered, similar to the foregoing, but with two rings of breeder fuel element assemblies, and also a model with only one central breeding zone with a radius of 46 cm (34 fuel element assemblies). In the calculations the volume fractions of material were varied for both the core and the breeding zones. Versions were analyzed with uranium dioxide and metallic uranium in the breeding zones. The results of the calculations are shown in Fig. 4.

The following can be seen from the data obtained:

The improvement of the breeding index in a heterogeneous core depends on the volume fraction of the fuel. The effects of heterogeneity are expressed more strongly in less dense zones, which can be explained by the large neutron leakage from the fuel zones into the breeding zones. In the "pure" form, i.e., with identical volume fractions and identical material in the fuel and breeder fuel element assemblies, the effect of heterogeneity on the breeding parameters is revealed very weakly: the BG is increased by 0.01-0.02, but T_2 is almost unchanged. Only an increase of the volume fraction (or density) of the breeder material affects T_2. In dense cores ($\varepsilon_f \approx 0.45$) a heterogeneous grouping with oxide breeder material reduces T_2 only insignificantly, so that an appreciable effect is obtained only when metallic uranium is used. In all cases, the buildup of plutonium in the breeder fuel element assemblies has a marked effect on the reduction of T_2. The minimum of T_2 is shifted to the side of a large value of the breeder fuel element assemblies in both the case of a denser core and in the case of using a denser (metallic) breeder material. The minimum of T_2 for oxide—metal cores of fast reactors is obtained with a 30-40% fraction of breeder fuel element assemblies in the core.

In heterogeneous cores, the dependence of the breeding index on the volume fraction of the fuel is more weakly expressed by comparison with the similar dependence for a conventional core. This gives the possibility for optimizing the heterogeneous core from the point of view of the specific fuel charge. For example, in the reactor considered with an oxide—metal heterogeneous core, a reduction of the fuel density in the fuel elements from 8.6 to 7.1 g/cm^3 reduces the specific fuel charge to the level of the charge in a conventional core, when the gain in T_2 is maintained at the previous level. The gain in T_2 is also maintained with a marked reduction of the diameter of the fuel elements. In this case, together with a reduction of the specific charge, an additional effect appears due to the improvement of the hydraulic characteristics of the fuel element assemblies.

The sodium void reactivity effect depends significantly on the volume fraction of the fuel in the core. In heterogeneous cores of the annular type, the oxide breeder material

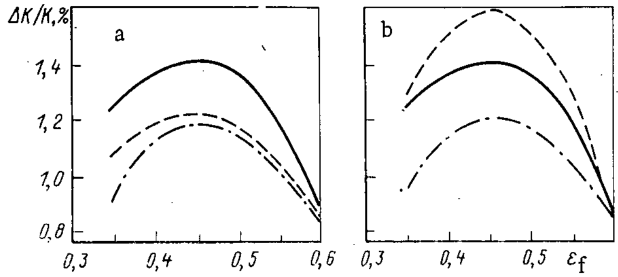

Fig. 4. Sodium void reactivity effect as a result of the removal of sodium from the whole of the reactor, as a function of the volume fraction of the fuel ε_f: a) UO_2; b) metallic fuel; ——) conventional core; -·-·-·) heterogeneous core with central breeding zone; ---) heterogeneous core with annular breeding zone.

reduces the sodium void reactivity effect (SVRE), and a metallic material increases it. Groupings of a heterogeneous core are also possible with metal breeding zones, for which the SVRE will be lower than in the conventional core. One version of this grouping corresponds to the arrangement of the breeder fuel element assemblies with uranium metal in the form of a symmetrical central zone.

The Doppler effect in heterogeneous cores is reduced because of the increased fuel enrichment. In oxide—metal heterogenous cores, it is reduced additionally because of the ruggedness of the neutron spectrum, related with the introduction of metallic uranium into the core. On the whole, the safety parameters in these cores are somewhat worse than in the conventional core. For the radial—annular type of groupings considered, the SVRE is increased by ∿15%, and the Doppler effect over the whole reactor is reduced by 25%. However, this circumstance can hardly be a serious limitation on the path of development and the introduction of such cores.

CORES WITH AN AXIAL ARRANGEMENT OF OXIDE AND METALLIC FUEL

The concept was considered of a core in which the temperature of the fuel element cans with metallic fuel can be reduced appreciably, without deterioration of the thermodynamic parameters [6, 8].

A reactor is considered where, from the direction of entry of the "cold" coolant, fuel elements with metallic fuel are distributed. The relatively low coolant temperature in this part of the reactor (400-480°C) determines favorable conditions for the operating efficiency of these fuel elements. In the region of higher coolant temperatures (500-560°C), fuel elements with oxide fuel are distributed. With the metallic fuel arranged in the low-temperature region of the core and the oxide fuel in the high-temperature region, the temperature parameters of a purely oxide reactor are maintained and, at the same time, the BG is increased by a factor of approximately 0.15. The increase of the BG is due mainly to an increase of the internal breeding coefficient of the core, which is important not only from the point of view of the rate of nuclear fuel breeding, but also optimization of the operating regime of a high-capacity fast reactor, taking account of the change of reactivity during its continuous operation.

The application of this design may be different. It can represent a single fuel element with different fuel along the height in a single cladding. Another version is the use of fuel element assemblies with two lattices of different fuel elements: with metal and oxide fuel. In this case it is advantageous to separate the different fuel elements by a filler.

In the reactor being considered with axial arrangement of the oxide and metal fuel of high enrichment, the subzone with oxide fuel, to a considerable degree, plays the role of a "seeding" subzone. As a result, despite the relatively low fuel burnup of the metal subzone (4-5%), the fraction of the depleted initial charge in it is quite high. The corresponding burnup of the oxide fuel amounts to 8-10%, with an irradiation time of the composite fuel element assemblies equal to 1.5 yrs. We note that an increase of the average burnup of the metallic fuel of more than 5% does not lead to a significant increase of the rate of nuclear fuel breeding.

A higher rate of breeding in the metal subzone, with a power fraction in it of 40-50%, and with a sufficient fraction of burnup of the initial charge for approximately the same conditions, provides a total breeding rate in the composite core that is a factor of 1.25-1.35 higher than in the reactor with oxide fuel.

It follows from the analysis presented that the use of heterogeneous oxide—metal cores is promising for the improvement of fast reactor fuel breeding.

Finally, this solution will require additional investigations on the justification of the working efficiency of the fuel elements in a heterogeneous core, and it will possibly lead to some complexity of design and technology for the manufacture of the fuel elements by comparison with the conventional oxide fuel. However, as a rule, any path for increasing the breeding rate will give rise to similar problems.

LITERATURE CITED

1. D. Thornton et al., Proceedings of an International Conference on "Optimization of Sodium Cooled Reactors," London, Vol. 11 (1977).
2. A. N. Shmelev and D. C. Yurchenko, Physics of Nuclear Reactors [in Russian], Atomizdat Moscow (1978), p. 42.
3. V. V. Orlov et al., Report IAEA-SM-244/78 at the Symposium on the Physics of Fast Reactor [in Russian] Aix-en-Provence, France, Vienna (1979).
4. V. V. Orlov et al., Report IAEA-SM-244/76 at the Symposium on the Physics of Fast Reactors [in Russian], Aix-en-Provence, France, Vienna (1979).
5. M. F. Troyanov et al., Report IAEA-SM-244/81 at the Symposium on the Physics of Fast Reactors [in Russian], Aix-en-Provence, France, Vienna (1979).
6. V. V. Orlov et al., Report IAEA-SM-244/77 at the Symposium on the Physics of Fast Reactors [in Russian], Aix-en-Provence, France, Vienna (1979).
7. L. P. Abagyan et al., Group Constants for the Calculation of Reactors and Shielding [in Russian], Énergoizdat, Moscow (1981).
8. M. F. Troyanov et al., At. Energ., 49, No. 5, 275 (1980).

STANDARDS FOR SAFETY OF ATOMIC POWER PLANTS IN THE USSR

V. A. Sidorenko, O. M. Kovalevich, and A. N. Isaev

UDC 621.039.68(47-57)

SYSTEM OF STANDARDS-TECHNICAL DOCUMENTS ON ENSURING ATOMIC POWER PLANT SAFETY

Setting up safety standards in official documents is one of the main courses taken in ensuring the reliability of atomic power plants (APP) in the USSR [1, 2]. The system of standards-technical documents (STD) on the structure and modes of introduction at various stages in the construction and operation of atomic power plants corresponds to the national conditions of the organization of the national economy and the established division of functions among state supervisory organs regulating the development of nuclear power. Naturally, the given STD system is not definitively formulated and frozen but is developing continually in keeping with the growing scales on which nuclear power develops and its range of application expands.

The safety of atomic plants is supervised by the following:

The State Committee on the Supervision of Industrial Work Safety and on Mining Supervision at the Council of Ministers of the USSR (Gosgortekhnadzor SSSR) checks that the construction of an APP and its equipment are in accordance with the technical safety standards during the design, erection, and operation.

The State Nuclear Safety Inspectorate of the USSR (Gosatomnadzor SSSR) supervises the observance of nuclear safety standards during the design, construction, and operation of APP.

The State Sanitary Inspectorate of the USSR at the Ministry of Public Health of the USSR supervises observance of the sanitary regulations and the radiation safety standards during the design, construction, and operation of APP, establishes permissible standards of irradiation of the plant personnel and the local population, as well as of environmental contamination with radioactive products, and takes the necessary measures that must be carried out in order to guarantee that these standards are met.

The system comprised of these three supervisory organs in great measure determined the structure of the set of standards-technical documents on APP safety; in it, documents that gravitate toward the problems dealt with by the aforementioned supervisory organs emerge from

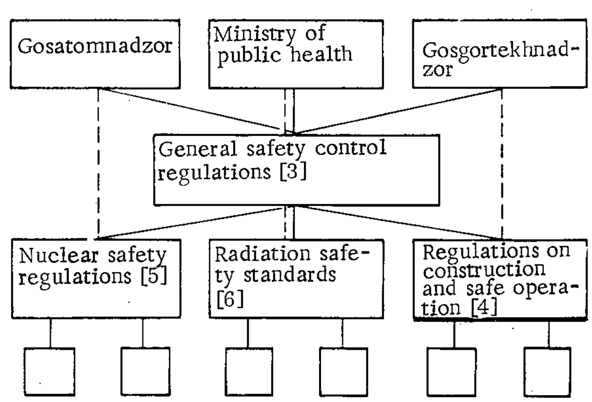

Fig. 1. Structure of standards-technical documentation of APP safety in the USSR.

Translated from Atomnaya Énergiya, Vol. 54, No. 4, pp. 273-277, April, 1983.

0038-531X/83/5404-0285$07.50

the main document in three directions. In individual cases this does not rule out horizontal links, i.e., the incorporation of lower stages of problems — pertaining to different supervisory organs — into some documents (Fig. 1).

The main standards document on APP safety in the USSR are the "General Safety Regulations for Atomic Power Plants during Design, Construction, and Operation" [3], introduced in 1973. This document formulated the main principles concerning the construction of APP and laid down the basis for coordinated execution of technical and organizational measures to ensure safety at all stages of construction and operation. The scope of this document extends to all atomic power plants with all types of reactors intended for commercial use in the nuclear power industry of the USSR in the nearest future (VVÉR, RBMK, BN, AST). This approach in great measure determined the character of the exposition of the requirement in general form without sufficient specificity. In most cases the "General Regulations" only pose the problems which must be solved in order to ensure safety (which must be done), while not dictating definite solutions (as should be done). Other standardizing documents (regulations, standards, procedures) develop the General Regulations and make them more specific in a certain direction, providing a basis for the work of designers and the pertinent supervisory organs.

One of the principal documents in the domain of Gosgortekhnadzor concerning technical safety is the document "Regulations on the Construction and Operating Safety of the Equipment of Atomic Power Plants and Experimental and Research Nuclear Reactors and Facilities" [4]. These regulations extend to reactors, steam generators, vessels, the housings of pumps, and fitting and tubing operating under pressure in the primary and secondary loops of atomic power plants with water-moderated—water-cooled and uranium—graphite reactors. The document presents the main requirements on the design of the housings, tubing, and welded joints, it discusses the requirements on the materials used to fabricate, assemble, and repair the equipment and tubing, and it indicates the characteristics of the mechanical properties that must be determined upon introducing new materials into production. Moreover, these regulations establish the requirements on the fabrication and assembly of the equipment and tubing, the methods and the scope of the methods used to check welded joints, the characteristics of the fittings, monitoring and measuring instruments, and safety devices, and determine the order of recording, the technical inspection and operation of the equipment, as well as the authorization of personnel to operate the equipment. These regulations constitute one of the main documents on which Gosgortekhnadzor bases its operations.

The document that forms the basis of the work of Gosatomnadzor is the document "Nuclear Safety Regulations for Atomic Power Plants, PBYa-04-74" [5], which was introduced in 1975 and which regulates APP safety problems associated with preventing loss of control and monitoring of the chain fission reaction in the reactor core and eliminating the possibility of a critical mass being formed during recharging, transportation, storage of fuel bundles, and assembly and repair work. It contains the main engineering and organizational requirements for ensuring nuclear safety during the design, construction, and operation of atomic power plants, as well as the requirements for the training and qualification of personnel engaged in the operation of a reactor installation. These regulations establish the main technical requirements as to the design of the reactor plant and systems that ensure nuclear safety, and in doing so they specify the minimum composition and number of channels for monitoring the capacity of the emergency protection system of the reactor, and the minimum list of signals for tripping the emergency protection system. This document describes (from the point of view of nuclear safety) the sequence in which the APP is to be put into operation and also gives a list of the documentation necessary for starting up and operating the plant.

In the realm of radiation safety, the main document, on which the organs of sanitary supervision are based, is the document "Radiation Safety Standards NRB-76" [6]. These standards have been developed on the basis of the recommendations of the International Commission on Radiological Protection and set up a system of dose limits and the principles of their application. This is the main document regulating the level of exposure to ionizing radiation. On the basis of the possible effects of ionizing radiation on the organism, these standards establish the following categories of irradiated persons: personnel, individuals from the general population, and the population as a whole, in assessing the genetically significant dose of irradiation. The standards define the maximum allowable dose of irradiation of personnel under normal and emergency conditions and also set the maximum irradiation doses for individuals from the public and for the population as a whole. The "Sanitary Regulations for the Design and Operation of Atomic Power Plants, SP-AÉS-79" [7], reflecting the specific features of APP, expand and supplement the radiation safety standards. This document presents the requirements on the siting of atomic power plants, the layout and finish of the production premises, the organization of the technological process, the biological shielding, and the sanitary and dosimetric monitoring.

This system of standards-technical documents of APP safety functions along with the system of documents of the State Committee of the USSR on Standards (Gosstandart SSSR), which is responsible on the national scale for the creation, introduction, and implementation of standards in various areas of science and technology. The system of standards (state and industrial standards, technical specifications, sanitary measures) supplement the systems of standards-technical documents in the matter of ensuring the safety of APP by guaranteeing the quality of numerous elements, materials, processes, etc., tried and tested in various branches of industry and used in the nuclear power industry. These documents play a significant role in solving the problem of ensuring the quality of APP, as this is understood in many other countries [8], and which is discussed in detail in [2].

SOME CHARACTERISTIC ASPECTS OF THE PRACTICE OF SETTING STANDARDS FOR APP SAFETY

Let us dwell in some detail on approaches to the solution of individual present-day problems of APP safety as handled in national standards documents.

Differentiation of Requirements on the Safety and Siting of APP for Different Purposes. The safety of an APP along with the quality of its construction and technical equipment is determined by the choice of its site, the main role in this choice being played by the distance of the APP from densely populated areas. The siting of APP is regulated by the "Sanitary Regulations for the Design and Operation of Atomic Power Plants, SP-AÉS-79" [7] and "Requirements on the Siting of Atomic Central Heating Plants and Atomic Heat and Power Plants with Regard to Radiation Safety" [9]. The documents give three types of atomic plants depending on their purpose: atomic power plants (APP) for the generation of electricity, atomic heat and power plants (AHPP) for the production of thermal and electrical energy, and atomic heating plants (AHP) for producing hot water for domestic purposes.

The Sanitary Regulations stipulate that an APP with a rating of 440 MW(elec.) or more should be sited no closer than 25 km to cities with a population of more than 300,000 and no closer than 40 km to cities with a population of more than 1 million. Analysis of actual sites of operating APP shows that the number of inhabitants within a 30-km radius of the plant, including the rural population and small populated localities, usually does not exceed 100,000-200,000.

The use of atomic plants for supplying heat requires that, in order to obtain acceptable economic indicators, they be put closer to the consumers, i.e., that they be built at a substantially smaller distance from large population centers. The increase in the risk to the public as a result of the atomic plants being sited closer to the cities is compensated by the imposition of additional safety requirements, ensuring protection of the APP from a broader class of internal damage and external actions. When these requirements are met, the risk for inhabitants of the city from the AHP is at least no greater than for an APP further removed from the city [2]. The document [9] allows AHP to be sited no closer than 2 km from the prospective boundary of the residential area of the city. Further development of the city should take place with allowance for the presence of the AHP.

When atomic heat and power plants (AHPP) are used as a source of heat they may be sited closer than APP to larger cities without the imposition of requirements for additional protection from internal damage and external actions, but with an additional requirement, placed on AHPP, as to the maximum irradiation of large populations under normal conditions and during failures.

Any atomic source of heat supply (AHP and AHPP) must meet requirements designed to prevent radioactive substances from reaching the consumer of the heat with coolant from the plant.

Specific safety requirements for APP are envisaged in four areas (Fig. 2):

1) In this case, during the construction of APP, AHPP, and AHP one must be guided by the requirements of the "General Regulations" [3] and SP-AÉS-79 [7];

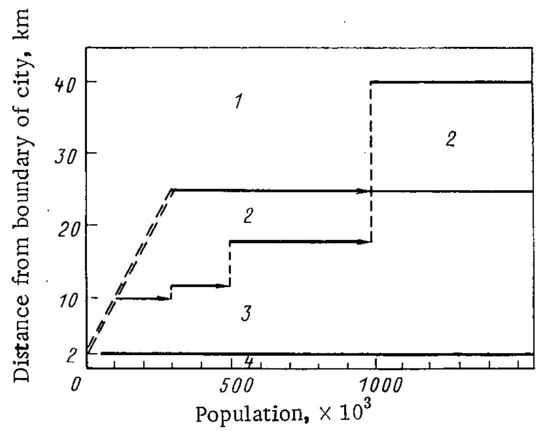

Fig. 2. Areas of specific safety requirements imposed on APP. The boundary marked by = = = is arbitrary.

TABLE 1. Limits of Dose of Irradiation of a Limited Part of the Population

Irradiation	Source of radiation	Group of critical organs*			
		1	2	1	2
		during normal operation		during maximum design failure	
Individual	Total source†	0.5 rem/yr	1.5 rem/yr	—	—
	Irradiation from APP	20 mrem/yr	60 mrem/yr	10 rem	30 rem
Collective	Irradiation from AHPP, AHP	10^4 man·rem/yr	—	10^5 man·rem ‡	

*The first group comprises the whole body and the second group is for the thyroid.
†Not including medical radiation and the natural radiation background.
‡The collective dose of irradiation of the population of a large city near the APP.

2) in this case, AHPP and AHP can be built with the additional requirements concerning the standardization of the collective dose of irradiation of the population;

3) construction of only AHP is allowed on condition that requirements of a constructional character are satisfied ("General Regulations" [3], SP-AÉS-79, and the requirements of the document [9]);

4) atomic plants cannot be sited here ("forbidden area").

Within the framework of the requirements of the "General Regulations," the maximum design failure adopted for APP is an instantaneous transverse rupture of a tube of maximum diameter, and the design should take account of the action of all natural phenomena inherent to the given site. External effects due to human activity (explosions in nearby industrial plants and in transport, possible airplane crash) are taken into account by an appropriate choice of site for the plant, making it possible to eliminate the possibility of such action on the plant. Within the framework of the additional requirements of the document [9], the maximum design failure envisaged in the AHP design is damage to any vessel of the reactor facility that leads to loss of hermeticity. Measures should be envisaged for preventing melting of fuel elements in the reactor core. Allowance must also be made for such external actions as the crash of an airplane and a shock wave in the event of explosions in the vicinity of the plant, with calculated parameters of the action as stipulated in the regulations.

Limits of Irradiation Dose for the Population. The "Sanitary Regulations" [7] establish a certain dose of irradiation of a limited part of the population because of gas—aerosol discharges during normal operation of the APP and also establish allowable doses of individual irradiation during a maximum design failure. The "Requirements on the Siting of Atomic Central Heating Plants" [9] place additional limitations on the collective dose of irradiation of the surrounding population (see Table 1).

The Role of the Quantitative-Probabilistic Approach to Setting Safety Standards. The question of what approach, deterministic or quantitative-probabilistic, is employed in the country when setting safety standards cannot be answered unambiguously, since one and the other are used at different stages. The extent to which one approach or the other is used is determined by what problems, in ensuring the safety of atomic power plants, are being solved, and when. In the light of the problem posed, it is desirable to divide the activity of ensuring APP safety into two stages: development of approaches and formulation of the requirements as to APP safety, and elaboration of a design, constructing, and operating the APP. Each stage is characterized by its own formulation of problems for a certain circle of persons or organizations and these problems are solved independently, although, naturally, not in isolation from each other, since there is a strong interrelation between these problems.

In the early period of the development of the atomic power industry, the first stage was characterized by purely intuitive and engineering approaches. At the present time the quantitative-probabilistic approach is increasingly becoming the basis. The studies being developed and expanded in the country on quantitative-probabilistic analysis are directed primarily toward these goals. The elaboration of additional safety requirements for atomic heating plants has been based in great measure on the quantitative-probabilistic approach.

For reliable application of quantitative-probabilistic analysis of safety in the design stage of APP it is necessary to have the pertinent statistical data. Such data can be obtained in sufficient number for most natural phenomena. However, statistical data about the reliability of specific equipment used in the atomic industry are limited at this time. This, in the main, is responsible for the deterministic approach in the second stage. Certain elements of the quantitative-probabilistic approach, however, do exist here and they are laid out in the standards-technical documents. We note the principal ones.

As a rule, the parameters of the natural phenomena taken into account in the design are chosen on the basis of a quantitative-probabilistic analysis. For example, the design for the construction of an APP makes provision for an earthquake with an average recurrence period of up to 100 yrs, and the maximum design earthquake is assumed to have parameters which, according to the calculations, have a probability of 10^{-4} yr^{-1} [10]. The choice of the design values for the wind, snow, and other loads when taking the meteorology into account is also based on statistical data.

There are direct indications for the use of the quantitative-probabilistic approach during designing of equipment and systems for APP. Thus, the "General Regulations" envisage a quantitative analysis of the reliability of the systems, which leads to a search for the most reliable schemes, quantitative analysis of the probability of damage to the equipment, and realization of various failure situations considered in the design stage. Special procedures have been developed for these purposes. In addition to the postulated failures, the APP design may not take account of failures of systems (elements) whose reliability is fairly high according to estimates.

As statistical data are accumulated and the pertinent methods are approved, the domain of application of the quantitative-probabilistic approach in the process of APP designing and monitoring on the part of the supervisory organs will grow.

LITERATURE CITED

1. V. A. Sidorenko et al., At. Energ., 43, No. 5, 360 (1977).
2. V. A. Sidorenko and O. M. Kovalevich, At. Energ., 50, No. 2, 93 (1981).
3. General Safety Regulations for Atomic Power Plants During Design, Construction, and Operations, Izd. Minénergo SSSR, Moscow (1973).
4. Regulations on the Construction and Operating Safety of the Equipment of Atomic Power Plantas and Experimental and Research Nuclear Reactor and Facilities, Metallurgiya, Moscow (1973).
5. Nuclear Safety Regulations for Atomic Power Plants, PBYa-04-74, Atomizdat, Moscow (1976).

6. Radiation Safety Standards, NTB-76, Atomizdat, Moscow (1978).
7. Sanitary Regulations for the Design and Operation of Atomic Power Plants, SP-AÉS-79 [in Russian], Énergoizdat, Moscow (1981).
8. Summary of IAEA Regulations 50-S-OA "Safety of Atomic Power Plants — Ensuring Quality in APP," Vienna (1979).
9. Requirements on the Siting of Atomic Central Heating Plants and Atomic Heat and Power Plants With Regard to Radiation Safety, At. Energ., 49, No. 2, 150, (1980).
10. Provisional Standards for the Designing of Atomic Power Plants for Seismic Regions, VSN-15-78, Izd. Minénergo SSSR, Moscow (1979).

RADIATION SAFETY OF ATOMIC POWER PLANTS IN THE USSR

E. I. Vorob'ev, L. A. Il'in, V. D. Turovskii, L. A. Buldakov, N. G. Gusev, O. A. Pavlovskii, and G. M. Parkhomenko

UDC 621.039.68(47-57)

At the beginning of 1982 the total power of atomic power plants (APP) in our country was as 16 GW (elec.)* and this is to be raised to 40 GW (elec.) by the end of 1985 [1]. The nuclear power industry is expected to develop at an even greater rate in subsequent periods: according to the estimates of specialists, the power of APP is to reach 90 GW (elec.) by 1990 [2] and 200 GW (elec.) by the end of the century [3].

The setting of sanitary standards for radiation factors has developed in our country in advance of the construction of large APP. The first sanitary regulations and standards were drawn up in 1953 during the period leading up to the start-up of the world's first atomic power plant at Obninsk. Later, during the period when the Beloyarsk and Novovoronezh APP were being designed, the "Sanitary Regulations for Work with Radioactive Substances and Other Sources of Ionizing Radiation, SP-333-60" were issued. In 1965, the Ministry of Public Health of the USSR set up the National Commission on Radiation Protection (NCRP), which was given the task of generalizing data on the scientific substantiation of the principles of radiation protection and developing radiation safety standards and regulations. The NCRP prepared the "Radiation Safety Standards, NRB-69," the NRB-76 standards that are in force at present, as well as the "Fundamental Sanitary Regulations with Radioactive Substances and Other Sources of Ionizing Radiation, OSP-72/80" [4]. In the course of developing the main propositions of these documents, the "Sanitary Regulations for the Design and Operation of Atomic Power Plants, SP-AÉS-79" [5] were issued, containing requirements on ensuring the radiation safety of APP personnel and the population living in the region of the APP, as well as on protection of the environment from contamination with radioactive waste and discharge of waste heat. It is necessary to point out that unlike similar international norms, our documents are legislative in character and not just recommendations.

For the protection of the population and the environment, SD-AÉS-79 sets dose limits (DL) for the dose caused by the total gas—aerosol emissions and liquid radioactive discharges from APP, which constitute 5% of the dose limit for the limited part of the population (DL_B according to NRB-76). The setting of a 5% dose limit on radioactive wastes from an APP is consistent primarily with the well known ALARA principle [6], which is particularly timely under the conditions of a nuclear power industry developing in densely populated regions of the country. Moreover, the actual dose in a locality from radioactive wastes of operating APP [7, 8] is substantially lower than the dose limits envisaged in SP-AÉS-79 for a limited part of the population (Table 1). And, finally, these limitations are in accord with the thresholdless dose-effect concept. The limits of the individual equivalent doses given in Table 1 are for inhabitants on the boundary of the sanitary-protection zone or outside it at a distance at which the highest radiation dose is expected. This refers to the limits of

*The installed capacity of APP in the USSR was 18 GW (elec.) at the beginning of 1983.

Translated from Atomnaya Énergiya, Vol. 54, No. 4, pp. 277-285, April, 1983.

 0038-531X/83/5404-0290$07.50

TABLE 1. Limits of Equivalent Dose for the Inhabitants (Category B) of the Region of an APP Site, mrem/yr

Form of radioactive wastes	Fraction of DL (NRB-76), %	Group of critical organs		
		I	II	III
Gas—aerosol emissions	4	20	60	120
Liquid discharges	1	5	15	30
Total	5	25	75	150

1 rem = 10^{-2} sievert

TABLE 2. Average Daily Permissible Gas—Aerosol Emission from an APP

Nuclide	APP power, GW (elec.)	
	< 6, Ci/GW (elec.)•day	≥ 6, Ci/APP•day
Radioactive noble gases	500	3000
^{131}I (all forms)	0,01	0,06
Mixture of long-lived nuclides	0,015	0,09
Mixture of short-lived nuclides	0,2	1,2

1 Ci = $3.7 \cdot 10^{10}$ Bq.

the annual dose reached under the conditions of radioactive equilibrium in the environment. For liquid discharges the values of DL_B are given with allowance for different ways in which water is used: fishery, fish breeding, irrigation, and drinking water supply.

Although all possible safety measures are taken when designing an APP, a failure is not ruled out theoretically. The regulations SP-AÉS-79 provide for the creation of a technical safety system that would protect the population in the event of a maximum design failure (MDF). In this case the following is the expected dose received by the population from a failure: external whole-body irradiation 10 rem and internal irradiation of the thyroid of children and of any other organs, 30 and 10 rem, respectively.

The choice of the dose of failure irradiation was based on the assumption of: first, an extremely small probability of a failure that would lead to such a dose; second, comparability with the dose limits for individual inhabitants as set in NRB-76 (0.5 rem/yr, or 30 rem in the case of chronic lifelong irradiation), which, in the opinion of the ICRP, is comparable, in regard to biological effect, to 10 rem in the case of irradiation of short duration; and, third, comparability with the maximum permissible dose (MPD) for professional workers during normal operation of the APP (5, 15, and 30 rem/yr for critical organs in groups I, II, and III, respectively).

In addition to limitations put on the main characteristics, i.e., the dose limits, derivative quantitites have also been introduced: permissible emissions (PE) and permissible discharges (PD), as well as control emissions and discharges. Strictly speaking, the DE should correspond to discharges which, under conditions where in radiological equilibrium is reached in the environment, should not exceed the DL given in Table 1. In SP-AÉS-79, however, the permissible emissions were calculated with allowance for an additional requirement: They should not significantly exceed the actual emissions of operating APP, the latter usually being lower than the calculated values of MPE (Table 2). The average monthly permissible gas—aerosol emission of ^{90}Sr from an APP with a power < 6 GW (elec.) is 1.5 mCi/GW (elec.)•

TABLE 3. Average Annual Personal Radiation Dose of APP Personnel, rem

APP	1977	1978	1979
Novovoronezh	0,78	0,61	0,6
Kol'skaya	0,84	0,59	0,68
Armyansk	0,33	0,14	0,62
Chernobyl	—	1,2	1,0
Kursk	0,23	0,33	0,38

TABLE 4. Radiation Dose APP Personnel

Reactor	Period of observation, reactor-yrs	Average dose, rem	Standardized collective dose, man•rem/MW (elec.)•yr
VVÉR-440	16	0,56	1,1
RBMK-1000	11	0,68	1,3
BN-350	6	0,01	$7\cdot10^{-3}$*

*Normalized to 1 MW (thermal).

month, and from an APP with a power ≥ 6 GW (elec.) it is 9 mCi/APP•month, and the figures for ^{137}Cs, ^{60}Co, ^{54}Mn, and ^{51}Cr are 15 mCi/GW (elec.)•month and 90 mCi/APP•month, respectively.

An inseparable part of the system of radiation protection of atomic power plant personnel and the population is that of radiation monitoring. In an APP this monitoring includes measuring the personal doses of external radiation, the equivalent dose rate of γ rays and neutrons, as well as the neutron and β-particle flux density, the concentration and composition of radioactive gases and aerosols in the air in the production premises, the radioactive contamination of working surfaces, structures, and equipment, leather covers, and working and personal clothing of the personnel, the activity and composition of gas—aerosol emissions and liquid discharges into the environment, and the content of radioactive substances in various objects in the environment.

Special service teams, provided with dosimetric and spectrometric equipment, monitor the environment for the dose rate and the annual radiation dose in a locality, and determine the concentration of radionuclides in the atmospheric air, soil, vegetation, the water of open bodies of water, foodstuffs, and animal feed produced locally.

The requirements of radiation protection of the personnel of an atomic power plant are met by organization of biological shielding of the equipment, zone layout of the premises, ventilation, organizational and technical measures to reduce the γ radiation due to the equipment of the reactor core during planned preventative maintenance, and constant radiation monitoring. Operating experience shows that γ rays are the main harmful factor for atomic power plant employees. The total annual dose of γ rays, however, does not exceed 5 rem/yr, and in the overwhelming majority of cases it is below this value (Table 3).

The average annual radiation dose of the personnel of atomic power plants equipped with reactors of different types and powers ranges from 0.14 to 1.2 rem. The data of Table 4 indicate that the conditions created in the APP in our country are such that the latest recommendations of the ICRP are fulfilled: The average radiation dose of the personnel is one-tenth the value adopted for the MPD. Comparison of the irradiation received by personnel with the materials published by the United Nations Scientific Committee on the Effects of Atomic Radiation shows that with respect to both the personal and collective doses, the irradiation in domestic atomic power plants is close to those in other countries.

The irradiation is due mainly to maintenance work. In an atomic power plant with an RBMK-1000 reactor, such work consists of inspections and maintenance of drum separators, replacement of ball-type flow regulators for water and shut-off valves, and maintenance of the main circulating pumps (all told, 32% of the collective dose; in an APP with a VVÉR-400 reactor, this work encompasses fuel recharging with the attendant maintenance work on the reactor vessel, steam generators, and main circulating pumps (62% of the collective dose)). During maintenance work, in all the APP the number of persons under radiation-hazardous conditions is increased 2 to 2.5 times. The entry of radionuclides into the human body, which is possible during fuel reloading and maintenance and repair work, is at a much lower rate than the permissible value, and the content of radioactive substances in the body of professional workers (in 90% of them) is less than 0.02 of the limiting permissible content indicated in NRB-76, and only in individual cases are ^{137}Cs and ^{60}Co found to be present in quantities of 60-180 nCi.

Analysis of the data of large-scale polyclinical examinations of the personnel (periodic medical check-ups), as well as the rate of sickness with temporary incapacitation, shows that

TABLE 5. Emissions Normalized per Unit Electricity Generated

APP, reactor type	RBG, Ci/MW (elec.)•yr			^{131}I, μCi/MW (elec.)•yr		
	1977	1978	1979	1977	1978	1979
VVÉR						
Novovoronezh*	5,6	5,8	4,3	8,4	4,6	5,5
Kol'skaya*	2,8	2,8	3,0	36	1,4	1,2
Armyansk RBMK-1000	—	4,9	6,0	—	281	441
Leningrad	288	191	145	4800	1234	2340
Chernobyl	—	60	82	—	178	180
Kursk BN-350	—	35	80	—	21	78
Shevchenkovskaya (4-yr average)		12			—	

*Only the aerosol phase of ^{131}I was measured in this period.

the health of APP employees does not exhibit any deviations attributable to the effect of ionizing radiation. No increase in the rate of sickness is observed in personnel who have been working in atomic power plants for 10-15 yrs.

One of the achievements and advantages of nuclear power from the point of view of the radiation safety of the population is that under normal operation APP do not endanger the environment with radioactive substances. Gas—aerosol emissions from domestic APP are presented in [8]. Accordingly, we give only the data concerning the main component (Table 5).

From a comparison of the actual emissions with the permissible levels established in 1979 (up until 1979 the permissible emission had been 1.3•10 Ci/yr in the case of radiobiological gases (RBG) and 36 Ci/yr in the case of ^{131}I) it is seen that the actual emissions are considerably below the limits set. This is also true of other radionuclides. Upon comparing the normalized emissions of domestic and foreign APP, we can notice that, first, the emissions of RBG and ^{131}I from VVÉR and PWR reactors as a rule are lower than those from RBMK and BWR and, second, the emissions from domestic reactors are comparable to and in some cases lower than the average from foreign reactors [7].

In our country one of the main principles of ensuring the radiation safety of the water environment in the region of an APP is that of a closed cycle of water utilization in the technological loops which may contain radionuclides. Experience gained from the operation of APP, especially those with the first, prototype units, shows, however, that during maintenance and repair work, as well as during maladjustments in the reactor operating regime, certain quantities of surplus (unbalanced) water may be produced and discharged into cooling ponds after appropriate radiation monitoring [9]. Trap (deactivation) and special washing water, subjected to special purification, is the source of this water. As shown by experience with domestic APP, in the first 2 yrs of operation of a plant, during the period of adjustment of the technological systems, including the apparatus for special purification of salt-containing water, the volume of unbalanced water is a maximum (20,000-70,000 m^3/yr per unit). At the present time a closed cycle of water utilization is ensured over a prolonged period (∿ 2 yrs) only at the Leningrad APP, but other APP are close to switching to such an organization of operation.

Normal operation of power plants ensures a low concentration of radionuclides in the unbalanced water. Constant monitoring of this water for its content of radionuclides that are most important from the health physics point of view and that make the greatest contribution to the total activity of the discharged water makes it possible to estimate the gross activity of the nuclides carried into the cooling pond by the water. As shown by the data of Table 6, the main contribution to the activity of the unbalanced water comes from tritium (tens of Ci per year for plants with VVÉR and RBMK reactors). The activity of corrosion elements, mainly ^{60}Co, is 10^{-6} to 10^{-3} Ci/yr for one unit. The content of fission fragments is also low and is determined mainly by ^{137}Cs and ^{131}I. It should be pointed out that these

Declassified and Approved For Release 2013/02/06 : CIA-RDP10-02196R000300020004-6

TABLE 6. Gross and Normalized Activity of Liquid Discharges from APP into the Water Environment

Nuclide	Kurskaya APP		Chernobyl APP		Armyansk APP	
^{3}H	50—60 *	$(2,5—3)\cdot 10^{-2}$ †	20—60 *	$(1,0—3)\cdot 10^{-2}$ †	40—60 *	0,1—0,15 †
^{54}Mn	—	—	$(1—6)\cdot 10^{-7}$	$(0,5—3)\cdot 10^{-10}$	—	—
^{58}Co	—	—	$(1—6)\cdot 10^{-7}$	$(0,5—3)\cdot 10^{-10}$	—	—
^{60}Co	$(0,3—5)\cdot 10^{-4}$	$(0,1—2,5)\cdot 10^{-7}$	$(1—3)\cdot 10^{-6}$	$(0,5—1,5)\cdot 10^{-9}$	—	—
^{89}Sr	$(1,0—4)\cdot 10^{-5}$	$(0,5—2)\cdot 10^{-8}$	—	—	—	—
^{90}Sr	$(1,0—4)\cdot 10^{-6}$	$(0,5—2)\cdot 10^{-9}$	$(1,2—3)\cdot 10^{-7}$	$(0,6—1,5)\cdot 10^{-10}$	$5\cdot 10^{-5}$	$1,1\cdot 10^{-7}$
^{131}I	$(0,1—1)\cdot 10^{-3}$	$(0,5—5)\cdot 10^{-7}$	$(0,8—6)\cdot 10^{-4}$	$(0,4—2,8)\cdot 10^{-7}$	—	
^{134}Cs	$(0,3—5)\cdot 10^{-4}$	$(0,1—2,5)\cdot 10^{-7}$	$(1,0—6)\cdot 10^{-6}$	$(0,5—3)\cdot 10^{-9}$	$4\cdot 10^{-6}$	$9\cdot 10^{-9}$
^{137}Cs	$(2,0—4)\cdot 10^{-3}$	$(1,0—2)\cdot 10^{-6}$	$(0,5—1)\cdot 10^{-4}$	$(2,5—5)\cdot 10^{-8}$	$(0,5—1,5)\cdot 10^{-3}$	$(1,1—3,3)\cdot 10^{-6}$

*Discharge rate, Ci/yr.
†Normalized discharge rate, Ci/MW (elec.)·yr.

TABLE 7. Personal γ-Ray Dose, rem/yr

Nuclide	Distance from APP, km						
	1	2	5	10	20	50	100
^{41}Ar	$6,1\cdot 10^{-7}$	$3,6\cdot 10^{-7}$	$1,2\cdot 10^{-7}$	$4,3\cdot 10^{-8}$	$1,5\cdot 10^{-8}$	$2,6\cdot 10^{-9}$	$2,9\cdot 10^{-10}$
^{85m}Kr	$4,2\cdot 10^{-7}$	$2,6\cdot 10^{-7}$	$9,0\cdot 10^{-8}$	$3,4\cdot 10^{-8}$	$1,4\cdot 10^{-8}$	$3,7\cdot 10^{-9}$	$8,5\cdot 10^{-10}$
^{85}Kr	$6,5\cdot 10^{-10}$	$4,0\cdot 10^{-10}$	$1,3\cdot 10^{-10}$	$5,8\cdot 10^{-11}$	$2,5\cdot 10^{-11}$	$9,6\cdot 10^{-12}$	$3,8\cdot 10^{-12}$
^{87}Kr	$6,5\cdot 10^{-7}$	$3,4\cdot 10^{-7}$	$1,0\cdot 10^{-7}$	$3,6\cdot 10^{-8}$	$1,1\cdot 10^{-8}$	$1,4\cdot 10^{-9}$	$8,4\cdot 10^{-11}$
^{88}Kr	$2,1\cdot 10^{-5}$	$1,2\cdot 10^{-5}$	$4,3\cdot 10^{-6}$	$1,6\cdot 10^{-6}$	$5,9\cdot 10^{-7}$	$1,3\cdot 10^{-7}$	$2,3\cdot 10^{-8}$
^{133}Xe	$7,5\cdot 10^{-6}$	$4,5\cdot 10^{-6}$	$1,8\cdot 10^{-6}$	$7,5\cdot 10^{-7}$	$3,3\cdot 10^{-7}$	$1,2\cdot 10^{-7}$	$4,8\cdot 10^{-8}$
^{135}Xe	$1,2\cdot 10^{-5}$	$6,8\cdot 10^{-6}$	$2,5\cdot 10^{-6}$	$1,0\cdot 10^{-6}$	$4,2\cdot 10^{-7}$	$1,4\cdot 10^{-7}$	$4,2\cdot 10^{-8}$
Total	$4,2\cdot 10^{-5}$	$2,4\cdot 10^{-5}$	$8,8\cdot 10^{-6}$	$3,5\cdot 10^{-6}$	$1,4\cdot 10^{-6}$	$4,0\cdot 10^{-7}$	$1,1\cdot 10^{-7}$

data pertain to the steady-state operating conditions of the APP. In the first 1 or 2 yrs after start-up the discharges of fission and induced radionuclides are roughly an order of magnitude larger. For APP with an RBMK reactor, some users of the cooling water, e.g., equipment in the machine shop, etc., can also be sources of radionuclides discharged into the cooling ponds. The activity of water emerging from the cooling system is, however, difficult to measure directly. By indirect estimates the discharges of radionuclides with the cooling water can be compared with the discharge of radionuclides contained in the unbalanced water.

Comparison of the data shows that the activity of liquid discharges of domestic APP is almost 1-2 orders of magnitude lower than that of similar foreign installations. With respect to tritium this difference is smaller, but is still a factor of 2-10.

For all APP in the country, the radiation components are monitored by dosimetric, radiometric, and spectrometric methods within a 30-50-km radius of the plant. The results of measurements of the radionuclide concentration in the atmosphere, water, soil, vegetation, and food products (milk, butter), as well as the annual dose of external γ radiation, are given in detail in [8]. All of these characteristics are fully determined by the natural background and the total radioactive fallout from nuclear explosions carried out in the 1950s and 1960s. A slight radioactive contamination of objects in the environment with fission and induced radionuclides formed during the operation of an APP is observed only on the industrial site and in individual cases in adjoining territory within the limits of the sanitary-protection zone. Thus, the inhabitants of the region in which the APP is sited are not exposed to additional radiation. This radiation cannot be measured by present-day means and can only be estimated by calculation.

As an example, we consider the radiation conditions in the region of the Novovoronezh APP, which has been operating for a long time in a steady-state regime and which is located in the densely populated European part of our country with intensive agriculture. The dose loads on the population were calculated from the formulas given in [10, 11]. In our calcula-

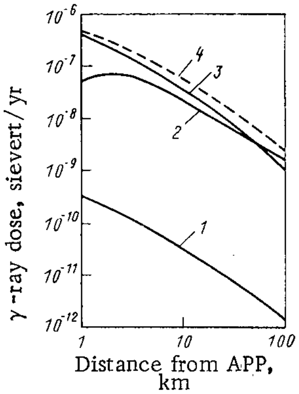

Fig. 1

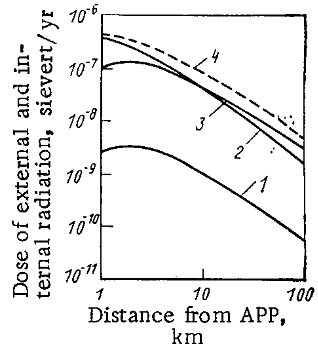

Fig. 2

Fig. 1. The γ-ray dose at a locality as the result of a cloud of emissions and radioactive fallout: 1) aerosol; 2) fallout; 3) RBG; 4) total (1 sievert = 100 rem).

Fig. 2. Dose of irradiation of the human body from gas–aerosol emissions: 1) internal (inhalation); 2) external irradiation; 3) internal (oral) ingestion; 4) total.

tions we took account of the actual recurrence of categories of weather (in the Pasquille classification), the elongation of the wind rose for various sectors of the site, as well as the data about the distribution of the rural and urban population within a 100-km radius around the power plant. The calculations were based on gas–aerosol emission in 1979, Ci/yr:

RBG		^{89}Sr	0,0008
^{41}Ar	11	^{90}Sr	0,0039
^{85m}Kr	62	^{95}Zr	0,0046
^{85}Kr	6	^{95}Nb	0,0079
^{87}Kr	17	^{103}Ru	0,0033
^{88}Kr	242	^{106}Ru	0,016
^{133}Xe	4230	^{110m}Ag	0,018
^{135}Xe	1072	^{131}I (aerosol)	0,0073
Total	5640	^{134}Cs	0,025
Long-lived aerosols		^{137}Cs	0,047
^{51}Cr	0,013	^{141}Ce	0,0005
^{54}Mn	0,040	^{144}Ce	0,0015
^{58}Co	0,016	Total	0,33
^{60}Co	0,087		

The calculated values of the γ-ray dose in an open locality from the annual emissions of both each radionuclide separately and a mixture of RBG in 1979 are presented in Table 7.

As follows from Table 7 the main contribution to the dose is made by ^{88}Kr, ^{133}Xe, and ^{135}Xe, but even if we take account of the elongation of the wind rose for the site of the Novovoronezh APP (1:15), the maximum value of the γ-ray dose from a cloud of RBG in an open locality at a distance 1 km from the APP does not exceed 63 μrem/yr, i.e., is lower than the permissible limit (20 mrem/yr) set in SP-AÉS-79. Besides RBG, a certain contribution to the γ-ray dose in the open locality can be made by the aerosol component of the emissions into the air. As follows from Fig. 1, at a short distance from the APP the largest contribution comes from the RBG cloud (up to 90% of the total dose), but with distance from the APP the role of the γ radiation from fallout radionuclides grows and at a distance of 100 km their contribution to the total dose is 60%. In our calculations we took account of the fact that for the Novovoronezh APP the fraction of ^{131}I in aerosol form comes to only about 2% of the total activity, while the remaining 98% is accounted for by its organic compounds.

The entry of radioactive products into the human body with air breathed in or with food products eaten causes internal irradiation of different organs and tissue in the body. Comparison of the dose of the external irradiation and the effective equivalent dose of internal irradiation of the human body, calculated in accordance with the ICRP recommendations, shows that (Fig. 2) even at a distance of 2-5 km from the Novovoronezh APP, the total annual dose of external and internal irradiation of the rural population is only 14-40 μrem and is due mainly to the external irradiation and oral ingestion of radionuclides into the body. The

TABLE 8. Collective Dose of Irradiation of the Population in 1979

Normalized value	Absolute value, man·rem/yr	Normalized value	
		man·rem/Ci	man·rem/MW (elec.)·yr
External irradiation	0,61	$1,1\cdot10^{-4}$	$4,6\cdot10^{-4}$
Internal irradiation:			
inhalation	$2,4\cdot10^{-2}$	$3,7\cdot10^{-6}$	$1,6\cdot10^{-5}$
oral ingestion	1,2	$2,1\cdot10^{-4}$	$9,1\cdot10^{-4}$
Total	1,8	$3,2\cdot10^{-4}$	$1,4\cdot10^{-3}$

TABLE 9. Normalized Values and Composition of Gas—Aerosol Emissions [12, 13]

Nuclide	VVÉR	RBMK	Nuclide	VVÉR	RBMK	Nuclide	VVÉR	RBMK
^{41}Ar	$2,0\cdot10^{-3}$*	$2,9\cdot10^{-1}$	^{58}Co	—	$4,0\cdot10^{-8}$	^{131}Cs	—	$5,6\cdot10^{-10}$
^{85}Kr	$6,0\cdot10^{-2}$	$4,7\cdot10^{-3}$	^{60}Co	$1,5\cdot10^{-6}$	$4,1\cdot10^{-9}$	^{137}Cs	$5,2\cdot10^{-6}$	$1,3\cdot10^{-9}$
^{85m}Kr	$5,4\cdot10^{-2}$	$4,6\cdot10^{-2}$	^{89}Sr	—	$1,6\cdot10^{-9}$	$^{140}Ba+^{140}La$	—	$1,7\cdot10^{-9}$
^{87}Kr	$1,0\cdot10^{-2}$	$9,6\cdot10^{-2}$	^{90}Sr	$1,0\cdot10^{-7}$	$3,9\cdot10^{-9}$			
^{88}Kr	$2,2\cdot10^{-2}$	$1,3\cdot10^{-1}$	^{95}Zr	$1,3\cdot10^{-6}$	—	^{144}Ce	$2,0\cdot10^{-6}$	—
^{133}Xe	$7,2\cdot10^{-1}$	$2,5\cdot10^{-1}$	^{110m}Ag	$8,6\cdot10^{-7}$	—	Activity of emission, Ci/MW (elec.)·yr	5	70
^{135}Xe	$1,3\cdot10^{-1}$	$1,8\cdot10^{-1}$	^{131}I	$9,0\cdot10^{-6}$	$5,0\cdot10^{-5}$			
^{51}Cr	$4,0\cdot10^{-6}$	$5,4\cdot10^{-8}$	^{133}I	$1,3\cdot10^{-6}$	$9,1\cdot10^{-5}$			
^{54}Mn	$5,0\cdot10^{-6}$	—	^{135}I	$4,2\cdot10^{-6}$	$6,9\cdot10^{-5}$			

*For all radionuclides we give the relative contribution in fractions of 1.

quantity of radionuclides inhaled is smaller by a factor of roughly 40, which means that this factor of the radiation effect of air emissions of an APP need not be taken into account when making estimates.

The total effective equivalent dose of irradiation of the population from liquid discharges from an APP with VVÉR-440 and RBMK-1000 reactors, with the most conservative estimates, does not exceed 0.05 μrem/MW (elec.)·yr, more than 99% of this being due to tritium. Such irradiation can occur only if the water from the APP cooling pond is used for drinking-water supplies. If radionuclides are not ingested with drinking water and there is only the "fish chain," this dose is almost two orders of magnitude lower. Estimates of the dose for operating APP in the country show that the contribution from liquid discharges to the irradiation dose of the population does not exceed 10-20%.

On the basis of data, similar to those given in Fig. 2, for various sectors of the site of the Novovoronezh APP with allowance for the actual wind rose, as well as data about the distribution of the rural and urban population in a 100-km zone around the plant, we calculated the collective irradiation dose of the population (Table 8). The first normalized value is the annual collective dose, normalized to a unit of activity of the emission. It mainly characterizes the site of the APP, since it depends primarily on the actual meteorological conditions and the distribution of the population around the APP. The second normalized value is associated with the technical indicators of the APP itself, i.e., activity of the gas—aerosol emissions per unit of electricity generated. Comparing these indicators with the data given in [7], we can see that with respect to both the first and second indicators the Novovoronezh APP is among the best in the world. Thus, the values presented above for the personal and collective doses of irradiation of the population show the absolute safety of the APP operation as far as public health is concerned.

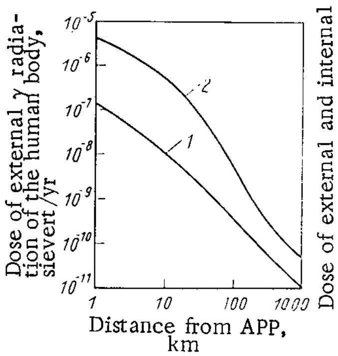

Fig. 3

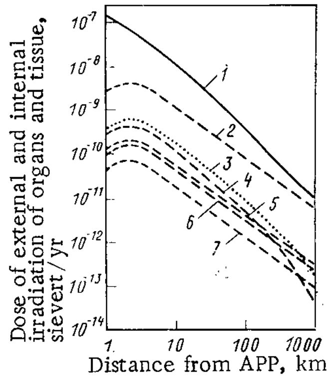

Fig. 4

Fig. 3. Dose of γ radiation of the human body from a cloud of emissions and radioactive fallout for APP with VVÉR (1) and RBMK (2) reactors.

Fig. 4. Dose of irradiation due to γ rays from an emission cloud, radioactive fallout, and inhalation of radionuclides into the human body: 1) external irradiation; 2) lungs; 3) effective dose; 4) thyroid; 5) red bone marrow; 6) gastrointestinal tract; 7) gonads.

The dose loads on the population of our country have been estimated on the basis of the same methodological principles as employed in calculating the dose around the Novovoronezh APP. It was further assumed that the average wind speed in the region of the APP site is 2.8 m/sec and the recurrence period of weather categories after Pasquille in fractions of a year is: 0.026 for category A, 0.114 for B, 0.230 for C, 0.322 for D, 0.178 for E, and 0.130 for F. The effective height of the emission from the VVÉR reactor was 140 m (H_{tr} = 100 m), while that from the RBMK reactor was 190 m (H_{tr} = 150 m). The other assumptions are in accord with the data of [11].

From the averaged characteristics of the gas—aerosol emissions of the APP, which are given in Table 9, it follows that the main contribution to the activity of the VVÉR emissions comes from ^{133}Xe, while for the RBMK reactor the role of this radionuclide in the formation of the activity of the emissions is comparable to that of ^{41}Ar that is formed in the gas loop of the reactor. Moreover, the emission of radioactive aerosol products is 4-6 orders of magnitude smaller than that of RBG and the normalized emission from channel-type reactors is an order of magnitude higher than the VVÉR emission.

The results of the calculation of the dose of external γ radiation of the human body from a cloud of emissions and radioactive fallout with allowance for the protection offered by buildings, equipment, and the life style of the population are given in Fig. 3. The values given in Fig. 3 for the dose correspond to the annual emissions of a typical 1000-MW (elec.) unit with a 75% power utilization factor (PUF). The gas—aerosol emissions of the RBMK have a substantially higher radiation significance than that of the VVÉR. This is attributed not only to the differences in the normalized emissions, but also to their isotopic composition, since the largest contribution to the activity of air emissions of the RBMK is made by the "hard" γ ray emitters ^{41}Ar, ^{87}Kr, and ^{88}Kr. It should also be pointed out that the main role in the formation of the dose of external radiation in the locality is played by RBG (the contribution of radionuclides that fall on the locality does not exceed 0.2%, while the aerosols contained in the ground layer of air contribute $\sim 10^{-3}$ %). On the basis of this, however, we cannot assert that with respect to radiation effect on the environment and man the RBMK is worse than the VVÉR. The use of boron to control the VVÉR leads to the formation of a certain quantity of tritium which enters the environment with gas—aerosol emissions and liquid discharges from the APP. Estimates show that according to normalized discharges and emissions into the environment, the RBMK is almost an order of magnitude better than the VVÉR.

In Figs. 4 and 5 the values of the dose of external irradiation of the human body are compared with the dose of internal irradiation of the organs and tissue because of inhalation

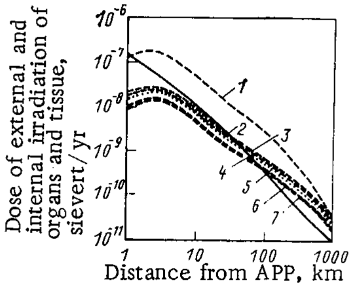

Fig. 5

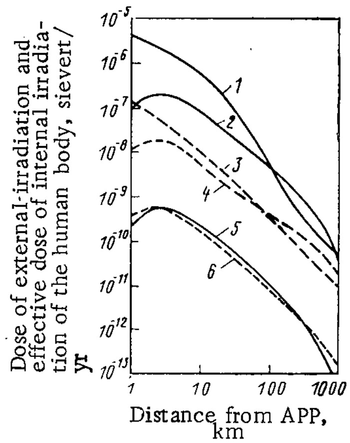

Fig. 6

Fig. 5. Dose of irradiation due to γ rays from an emission cloud, radioactive fallout, and oral ingestion of radionuclides into the human body: 1) thyroid, 2) external irradiation, 3) gastrointestinal tract, 4) red bone marrow, 5) effective dose, 6) lungs, 7) gonads.

Fig. 6. Comparative estimate of the dose loads on the population because of gas—aerosol emissions from different reactors: 1, 2, 5) external and internal oral and inhalation irradiation, respectively, for the RBMK; 3, 4, 6) external and internal oral and inhalation irradiation, respectively, for the VVÉR.

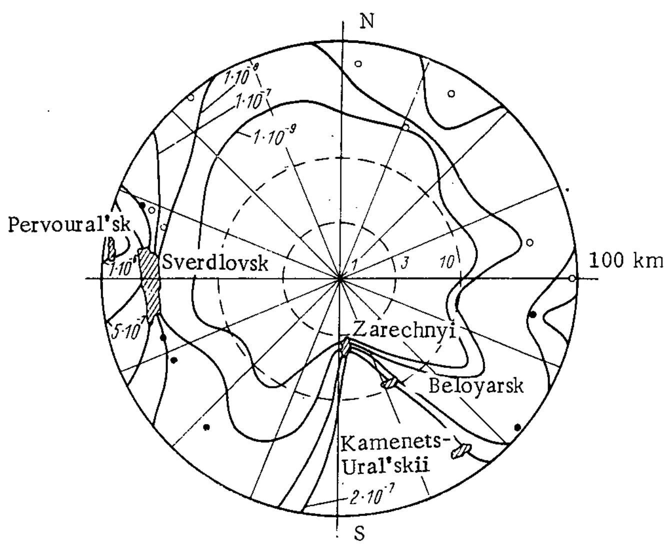

Fig. 7. Isopleths of collective dose (man·sievert/yr) of irradiation within the limits of each of 16 points of the compass for the spreading of gas—aerosol emissions from the Beloyarsk APP: ○, ●) towns with populations of 30,000-100,000 and 10,000-30,000, respectively.

and oral ingestion of radionuclides. The dose loads on the lungs, gonads, and other organs and tissues of the human body as the result of inhalation of radionuclides are shown as a function of the distance from an APP with a VVÉR reactor. The dose of internal irradiation of the organs indicated and the effective dose H_E of internal irradiation of the body are two or more orders of magnitude lower than the γ ray dose of the emission cloud. Thus, the

TABLE 10. Collective Dose of Irradiation of Population from Gas—Aerosol Emissions, man•rem/MW (elec.)

Type of reactor	Circular zone, km	External irradiation	Effective dose of internal irradiation		Total dose
			inhalation	oral ingestion	
VVÉR	1—10	$7,2\cdot10^{-5}$	$8,9\cdot10^{-7}$	$3,7\cdot10^{-5}$	$1,1\cdot10^{-4}$
	10—50	$1,6\cdot10^{-4}$	$3,0\cdot10^{-6}$	$1,4\cdot10^{-1}$	$3,0\cdot10^{-4}$
	50—100	$1,1\cdot10^{-4}$	$2,9\cdot10^{-6}$	$1,6\cdot10^{-4}$	$2,8\cdot10^{-4}$
	100—1000	$5,4\cdot10^{-4}$	$1,4\cdot10^{-5}$	$1,3\cdot10^{-3}$	$1,9\cdot10^{-3}$
	1—1000	$8,9\cdot10^{-4}$	$2,0\cdot10^{-5}$	$1,7\cdot10^{-3}$	$2,6\cdot10^{-3}$
RBMK	1—10	$3,3\cdot10^{-3}$	$1,2\cdot10^{-6}$	$4,2\cdot10^{-4}$	$3,8\cdot10^{-3}$
	10—50	$5,8\cdot10^{-3}$	$4,1\cdot10^{-6}$	$1,5\cdot10^{-3}$	$7,3\cdot10^{-3}$
	50—100	$1,5\cdot10^{-3}$	$3,4\cdot10^{-6}$	$1,5\cdot10^{-3}$	$3,0\cdot10^{-3}$
	100—1000	$2,9\cdot10^{-3}$	$1,0\cdot10^{-5}$	$6,2\cdot10^{-3}$	$9,2\cdot10^{-3}$
	1—1000	$1,4\cdot10^{-2}$	$1,9\cdot10^{-5}$	$9,7\cdot10^{-3}$	$2,3\cdot10^{-2}$

TABLE 11. Collective Dose of Irradiation of Population in 1980-2000* and the Collective Risk from Irradation

Parameter	1980	1985	2000
Collective dose, man•rem/yr			
Gas—aerosol emission			
external irradiation	96	250	1300
internal irradiation			
lungs	7	18	91
gastrointestinal tract	14	36	180
skeleton	17	44	210
red bone marrow	17	44	210
thyroid	2300	6000	29000
liver	5,5	14	72
gonads	5,8	15	76
effective dose	75	200	970
total dose	170	450	2200
Liquid discharges (total dose) †	11	25	56
Total dose	180	470	2300
Collective risk, man/yr	0,026 ‡	0,070	0,33

*It is assumed that in the period under consideration 40% of the power will come from VVÉR and 60% from RBMK.
†Calculated from the data of [9] with the assumption that 50% of the APP will have dry or wet cooling towers and 50% will discharge heat and a small quantity of radionuclides into cooling ponds.
‡The estimate was made on the basis of a risk of $1.5\cdot10^{-4}$ rem^{-1} [6].

role of inhalation of radioactive products of the gas—aerosol discharges from an APP during normal operation of the reactor is altogether negligible.

A slightly greater role is played by oral ingestion of radionuclides into the body. The dose of internal irradiation of the thyroid, especially in the case of children up to 1 yr of age, may exceed the dose of external irradiation because of an RBG cloud, but in this case as well H_E is substantially lower than the dose of external irradiation of the body. Similar results have also been obtained for RBMK emissions, as confirmed by Fig. 6. As in the construction of Fig. 3, the values of the doses here correspond to emissions from reactors with a power of 1000 MW (elec.) with a PUF of 75%. It must be pointed out that the dose values given in Fig. 6 are two or more orders of magnitude lower than the permissible levels set in SP-AÉS-79 and are equal to 20 mrem/yr for critical organs in group I.

Naturally, the actual distribution of the population around the APP with allowance for the meteorological characteristics of the locality results in an inhomogeneous collective dose of irradiation of persons in each sector to which the emissions spread. An example of such estimates is Fig. 7, which shows the position of the isopleths of the collective dose of external γ ray irradiation for a coordinate grid with 16 points of the compass relative to the Beloyarsk APP. In the calculations we assumed that the isotopic composition of the RBMK emissions corresponds to the data of Table 9, that the height of the reactor stack is 100 m, and that the emission rate is 1 Ci/yr. As is seen from Fig. 7, populated points are peculiar centers that "attract" collective-dose isopleths to themselves. In this respect, both small populated points which are close to the APP (the settlement Zarechnyi) and large towns at a considerable distance from the APP (Sverdlovsk, Pervoural'sk) are of major importance. It should also be pointed out that the collective dose of irradiation of the population for the entire 100-km zone around the Beloyarsk APP, calculated with allowance for the actual data on the meteorology and the population distribution, is $2.7 \cdot 10^{-4}$ man•rem/Ci, and when we use the average values of the meteorological data and the average population density over the provinces of Sverdlovsk and Chelyabinsk (23 inhabitants/km^2), the value obtained for the dose is substantially lower, $6 \cdot 10^{-5}$/man•rem/Ci.

The results presented here once again confirm that for a correct estimate of the collective radiation dose of the population it is necessary to use actual data about the site of the APP.

The value of the collective radiation dose of the population, normalized to 1 MW (elec.)• yr, for four characteristic zones around APP are given in Table 10. Using the data of this table we can calculate the collective dose of the population in our country from air emissions of APP in 1980, 1985, and 2000 and we can also estimate the risk. The results of the calculations, which are presented in Table 11, once again confirm the safety of nuclear power as far as the public health is concerned. Even with a total nuclear power capacity of ~ 200 GW (elec.), the expected number of cases of random consequences of irradiation from emissions of radionuclides into the environment during operation of atomic power plants is only 0.33 man/yr, i.e., a level that is absolutely undetectable against the background of the natural level of incidence of malignant tumors. One can also point out that a collective dose of $2.3 \cdot 10^3$ man•rem/yr is equal to the dose of radiation that the population of our country receives from the natural background in just 40 min.

We have not considered the radiation consequence of accidents in atomic power plants. A large number of special investigations have been devoted to this important subject. However, if we consider that the probability of major accidents in an atomic power plant is low ($\sim 10^{-6}$ to 10^{-7} yr^{-1}[14]), then despite the considerable dose loads in the region of the plant during such an accident, the collective risk of irradiation of the population of our country per year of reactor operation will not change and will be close to the values given in Table 10.

Thus, the experience from the operation of APP in our country, regulated by appropriate norms and regulations, permits us to assess the radiation conditions as being good. One can say that the working conditions, the health of the personnel, as well as the state of the environment around actual APP are more favorable than in other branches of the power industry. The present authors felt it was desirable to show this positive experience in the domain of the radiation safety of nuclear power. This is all the more important in that the view is often expressed that nuclear power is a dangerous branch of industry and a source of harmful effects on the personnel, the population, and the environment. Such unqualified statements cannot bring anything else but actual harm. One of the tasks facing the authors, therefore, is to present objective information about the true state of radiation conditions of the nuclear power industry of our country.

LITERATURE CITED

1. Proceedings of XXVI Congress of the Communist Party of the Soviet Union [in Russian], Politizdat, Moscow (1981).
2. I. S. Zheludev and L. V. Konstantinov, IAEA Bull.[in Russian], 22, No. 2, 34 (1980).
3. L. A. Il'in and O. A. Pavlovskii, in: Abstracts, Second Conference of COMECON Member-Countries on Radiation Safety of Atomic Power Plants [in Russian], Vilnius, (1982).

4. Radiation Safety Standards NRB-76 and Basic Sanitary Regulations of Work with Radioactive Substances and Other Sources of Ionizing Radiation OSP-72/80 [in Russian], Énergoizdat, Moscow (1981).
5. Sanitary Regulations for the Design and Operation of Atomic Power Plants Sp-AÉS-79 [in Russian], Énergoizdat, Moscow (1981).
6. Radiation Protection, Recommendations of the ICRP. Publication No. 26 [Russian translation], Atomizdat, Moscow (1978).
7. Sources and Effect of Ionizing Radiation. Report of the UN Scientific Committee on the Effect of Atomic Radiation [in Russian], New York (1978).
8. L. A. Buldakov, D. I. Gusev, N. G. Gusev, et al., in: Radiation Safety in the Atomic Power Industry [in Russian], A. I. Burnazyan (ed.), Atomizdat, Moscow (1981).
9. O. A. Pavlovskii et al., in: Abstracts, Second Conference of COMECON Member-Countries on Radiation Safety of Atomic Power Plants [in Russian], Vilnius (1982).
10. D. I. Gusev and O. A. Pavlovskii, in: Atomic Power Plants [in Russian], No. 2, Énergiya, Moscow (1979), p. 191.
11. O. A. Pavlovskii, in: Radiation Safety and Protection [in Russian], No. 7, Énergiya, Moscow (1982).
12. A. I. Burnazyan et al., Proceedings of Int. Conf. on Nuclear Power and Its Fuel Cycle, Salsburg, Austria, May 2-13, 1977, IAEA-CN-36/351 (1977).
13. N. S. Babaev, V. F. Demin, L. A. Il'in, et al., in: Nuclear Power, Man, and the Environment [in Russian], A. P. Aleksandrov (ed.), Énergoizdat, Moscow (1981).
14. Reactor Safety Study. Report WASH-1400, NUREC-75/014, October (1975).

EXTRACTION AND PROCESSING OF URANIUM ORE IN THE USSR

B. N. Laskorin, V. A. Mamilov, Yu. A. Koreisho,
D. I. Skorovarov, L. I. Vodolazov, I. P. Smirnov,
O. L. Kedrovskii, V. P. Shulika, B. V. Nevskii,
and V. N. Mosinets

UDC 622.775.791:
541.183.12

DEVELOPMENT OF URANIUM DEPOSITS

Uranium deposits in the USSR are situated in different locations, but genetically they belong to two categories — hydrothermal and hydrogenic. Many consist of ores which, in addition to uranium, include other elements, but extracting all of these elements is too expensive to make the enterprise worthwhile.

Development of uranium ores is a relatively recent branch of mining but it has resolved effectively a number of difficult problems in the operation of mines stemming from extraction of radioactive ores and the related peculiar features such as limited concentrations of reserves in the interior of the earth, complex morphology of most ore deposites being developed, inadmissability of large exposures of ore blocks and accumulation of considerable amounts of broken-off ore in the stoping space because of the radon release. Increasingly poorer ore becomes involved in development, and efficiency can only be achieved with the use of modern mining and processing techniques, high selectivity of extraction, field preparation of deposits, and high speed of mining, with safe and comfortable labor conditions for miners operating in high temperature stopes at deep levels (1200-1700 m) in rock beds with a high burst hazard.

Open Mining. In developing hydrogenic deposits with the open method, an effective flow process of stripping operation is practiced, using rotor equipment featuring 1000-5000 m^3/h when operating for internal and external dumps; this allows efficient uranium ore mining with stripping ratios of 50-60 m^3/ton and more. When developing strong enclosing rocks, preliminary explosive rock loosening is practiced. This increases the productivity, compared with excavation of unloosened rocks, by 1.4-1.9 times and reduces costs by 20-60%. An effective way of cutting transportation costs in driving openings and developing deep horizons in such mines was the method of transportation-dump dams built perpendicular to the stope advance front and connecting the stripping and mining benches with the inner clump stories. This reduced transportation costs by 25-40%.

Translated from Atomnaya Énergiya, Vol. 54, No. 4, pp. 286-292, April, 1983.

0038-531X/83/5404-0301$07.50

With opencast mining of hydrothermal deposits, high efficiency was achieved by using combined car-conveyor or car-track transport, where the transportation cost, compared to the cost with cars alone, was reduced 20-25%. Raising the height of drilled-off benches m and sorting the ore in excavations after blasting on subbenches and installing equipment on the blasted rock mass proved an efficient solution.

Subdivision into subbenches is important for stoping in deep mines, where uranium deposits of a complex structure are worked. Developing the blasting technologies that retain the geologic structure of ore bodies (placers) with the loosening coefficients of 1.05-1.12 has reduced the ore loss 1.5-2.0 times and curtailed ore dilution by 40-50%. This reduced the stripping coefficient, transportation amount, benefecation tails yield, and largely expanded the scope of opencast mining. A transitional area in the mining of uranium deposits is combined opencast and underground workings conducted in a number of mines with an extremely high operation density.

Underground Mining. In underground mining of hydrogenic deposits, the common practice is the shield driving of the shafts of capital and development openings during preliminary mine drainage from special drainage horizons. In stoping, powered and water supported installations have been introduced, including those with active maintenance of a "false" roof auger drilling and plow rigs with radiometric devices leading the measurement and control instruments along the bed boundary. The working of hard inclusions — carbonate and sand rocks — is done by powered roll cutters.

The high intensity of mining in the opening phase has been attained by simultaneous multistage opening of deposits and assignment of separate concentration horizons as justified economically, with the operation conducted independently in each stage; enlarging the mine fields, group opening and preparation of the stories, increasing the story height from 45-60 m to 90-120 m, and other technological solutions. The working of each stage with burial, in the worked-out space of the lower horizons of the gangue, rejected ore, and benefecation tails produced when developing the upper horizons created conditions for waste-free operation. Intensification of mining at the stage of stoping excavation was attained by enlarging the parameters of the breakage blocks, and the simultaneous panel preparation of blocks and the use of a continuous excavation procedure, multiface operation of miners' teams and the use of modern self-propelled drilling and loader-transportation units, vibration containers, breakage, and advance machines.

Because of the complex structure of hydrothermal uranium deposits, they are mostly worked with the use of a hardening filling of substory headings or crosscuts, substory warehousing, reducing the losses and ore dilution, containing the strain of enclosing rocks and ore masses, and thus reducing radon release. These systems, while ensuring a complete extraction of uranium, create conditions for extensive mechanization of the basic and auxiliary production processes, and warrant safety for the miners and high speeds of excavation advance. Intense mechanization is conducted in workings on the surface of the deposits where automated and mechanized sets of rail-car exchange are used and layouts of the mines have been improved. Radiometric sensors on hoppers of self-propelled loading and delivery machines used for sorting the flows of pay ores, substandard ore, and dead rock even in the stope at an early stage of mixing, reduce the ore loss and dilution.

Underground and Concentrated Leaching of Uranium. Underground leaching of uranium in hydrogenic deposits is the most effective method for mining of poor uranium ore found in difficult mining and geologic conditions where neither underground nor open cast methods would be efficient. Developing hydrogenic deposits with easily lixiviated poor uranium ore has changed the general estimate of uranium reserves and the very notion of raw material supplies of nuclear power engineering in the long range. With underground leaching, no large infrastructure, transport facilities, processing equipment and tails storage are required; conditions for highly efficient development of poor but large deposits are thus provided, reducing investment 2-3 times and tripling the labor productivity as assessed in terms of the end-product.

Development of the underground leaching of hydrogenic deposits in the past few years characteristically involved deep, relatively narrow and long placers concentrated in low permeability rocks. The use of powerful immersion electric pumps to extract solutions, improved hydrodynamic regimes of pumping the solution in and out, optimized network of reconnaissance and development holes, fluctuating operation schedules, creation of vertical and horizontal antipercolation shields, development of artificial cracking by using various technical facilities in low permeability rocks, increased diameters of near-filter zones, creation of new technical facilities (polyethylene and glass-reinforced plastic pipes, disk filters, and processes of pipe-free solution rise) — all have further enhanced the efficiency of underground uranium ore leaching in the USSR.

Underground leaching in hydrothermal deposits is normally practiced in combination with standard mining operations during the stage of deposit development, working of flanks and substandard ore areas simultaneous to the principal operation as well as in post-operation working of deposits. Independent underground leaching as the main method of deposit development is rare. With the standard system of opening and preparation of ore deposits, efficiency can be ensured only at the stage of cutting and preparation of blocks, when most ore is left in place instead of being raised to the surface. With improved processes of preparation of ore bodies with block-free excavation, optimization of the parameters and technology of drilling and blasting, and high degrees of ore fracturing and optimum filtrational uniformity of broken ore mass it was possible to ensure 70% uranium extraction; secondary blasting enhanced extraction to 90%, which, in current conditions, places underground leaching on a par with standard mining operation and sometimes makes it even superior.

PROCESSING OF MONOMETALLIC URANIUM ORES

Uranium Leaching. For silicate and alumosilicate ores, leaching with a solution of sulfuric acid with oxidizers (pyrolusite, sodium or potassium chlorates, and ferric ion) is the principal method. Ferric iron ions are important for oxidizing the minerals of tetravalent uranium in sulfuric acid leaching. A systematic study of the behavior of uranium minerals and accessory minerals of the gangue in sulfuric acid leaching with the use of different oxidizers has made it possible to determine the action of oxidizers, the ways of their efficient application, and potential reduction of agent consumption.

Progress in the acid leaching under atmospheric pressure followed the trend of enlarging the planned productivity to reduce the number of processing chains, spending less electricity on pulp mixing, and less steam on maintaining a rated temperature. Equipment for leaching that uses minimum amounts of liquid for pulp mixing has been developed.

Leaching of ores with a high content of acid-intensive components (carbonates) is done with carbonate and sodium bicarbonate solutions. Minerals of tetravalent uranium are oxidized in carbonate solution with the cheapest known oxidizer — the atmospheric oxygen. This process was improved by means of better air spraying, application of industrial oxygen, selection of oxidation catalysts, and efficient heat utilization and recuperation.

A promising method for extracting uranium from poorly resistant and complex ores is autoclave leaching at 100°C and higher temperatures, which raises the rate of useful reactions that evolve too slowly at lower temperatures. The use of industrial oxygen or air as the oxidizer in the autoclave made it possible to combine uranium leaching with sulfuric acid production from the pyrite present in the ore being oxidized, as well as to oxidize sulfides of other valuable metals brought into the solution. Positive features of industrial oxygen and air used as oxidizers is their availability, and absence of noxious, gaseous, or dissolved environmental pollutants in reaction products. Autoclave leaching saves chemicals and energy.

Molybdenum is a valuable component of uranium ores. Bringing molybdenum from molybdenite into sulfuric acid solution involves certain difficulties, as oxidizers commonly used for uranium ore leaching are inefficient with molybdenite. Leaching in autoclave under pressure with industrial oxygen or air used with a small admixture of nitric acid or nitrates efficiently dissolves molybdenite.

With acid autoclave leaching it is easier to create a closed process system for ore treatment, preventing formation of liquid and gaseous waste which would pollute the environment. An advantage of autoclave carbonate leaching with air is the improved extraction of valuable components and the reduced energy costs (steam and compressed air) due to process intensification and heat recuperation.

Figure 1 shows energy costs as a function of leaching duration in a standard container and an autoclave with mechanical mixing of the pulp. The cost of steam in autoclave leaching is estimated taking into account heat recuperation. If the length of leaching in a regular

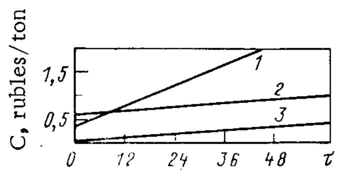

Fig. 1. Energy cost (C, ruble/ton) as a function of leaching time (τ, h): 1) steam and air in the container; 2) steam, air, and electricity in the autoclave; 3) electricity in the autoclave.

container exceeds 5 h (normally 10-12 h or longer), the process is more efficient with autoclave leaching. Carbonate leaching in an autoclave under air pressure oxidizes completely thionates and polythionates that can poison ionic exchange resins. The economic effect of the use of the autoclave ensures recovering the plant investment in 1.5-2 years. Heightened temperatures and pressures and the use of most inexpensive and pure oxidizer — air oxygen — allowed effective processing of various types of poorly resistant and complex uranium ores and the extraction of valuable components while reducing the expenditure of agents and energy. Various types of autoclaves are under serial production and in industrial use in the USSR.

Two types of autoclave are used in Soviet practice for uranium ore leaching — horizontal four-chamber units with mechanical mixers and vertical units with pneumatic pulp agitation. The agitators are two-stage turbine cantilevered mixers. Each agitator sucks in and disperses some 300 m^3 of gas per 1 m^3 of pulp. The power of the individual electric drive is 75 kW. The experience with the horizontal autoclave shows that the turbine aerator-agitator is an effective mixer, providing for a high oxygen utilization ratio. The vertical autoclave with pneumatic pulp mixing is a 100-200-m^3 container rated for operation at heightened temperatures up to 180°C and pressures up to 16 atm (1 atm = 101.325 kPa). The useful volume of the autoclave is 100 m^3; the diameter and the height being 3 and 18 m, respectively. An air lift pipe for pulp circulation is located along the central axis of the plant. The air is fed from the bottom through the distributing device (a punched plate with caps) installed beneath the air lift tube. The outer surface is insulated to reduce heat loss.

Sorptive-Extractive Removal of Uranium. In the initial stage of the development of the uranium industry, ore was processed by conventional, classic techniques, using such operations as filtration, repulping, decanting, multiple precipitation, and dissolution of uranium. For the hard-to-filtrate uranium ores of the clay type, energy- and labor-intensive operations were involved in the separation of the uranium-containing solution from the bulk of the leached ore. Multiple filtration or countercurrent decanting resulted in substantial losses of leached-out uranium in the moisture of filtered-off ore or condensed ore residue, with additional expenses on decontamination of radioactive effluents.

In 1953, a nonfiltrational sorptive-extractive method for extraction and processing of dense and viscous pulp was developed, which currently is the basic industrial process for extraction of uranium and valuable accessories from ores and concentrates. Sorption from the pulp allowed to combine separation of ionite from the ore mass with uranium concentration and purification. The introduction of the sorption of uranium from pulp reduced energy costs, enhanced productivity, and saved millions of square meters of filtering fabrics and hundreds of tons of acids, alkali, and other chemicals, while the production capacity increased severalfold. This process resulted in a review of the requirements to extracted ores and, at some mines the reduction of the rated uranium contents resulted in an increase of the effectively utilized reserves.

The process of uranium sorption from pulp is more intensive than conventional filtration and decanting methods, exceeding them by hundreds and sometimes thousands of times in efficacy. The process effectiveness increases further when sorption is combined with leaching. If ionite is introduced at the stage of leaching, uranium extraction is enhanced, the overall ore processing time is reduced, and the high sorptive capacities of such ore minerals as schist, coal, bentonite, montmorillonite, and zeolites are suppressed, so that a high concentration of leaching agent is maintained to keep the leached-out uranium in solution.

TABLE 1. Efficacy of Sorptive Absorption of Uranium from Natural Water with Salt Content of ∿6 g/liter and Uranium Concentration 60 mg/m³ at pH = 8.4-8.7 and Sorption and Desorption Durations of 48 and 6 h

	Sorbent	Granulation, mm	Uranium capacity, mg/liter of sorbent
Power (fine-granular)	Alumogel	—0,10	0,20
	Titanogel	—0,10	0,22—0,30
	Calcium phosphate gel	—0,10	0,20
	Zerolite FF	+0,20—0,40	0,21
	Amberlite IRA-410	+0,20—0,63	0,54
	Anionite AM-10Xp	+0,056—0,16	5,3
	Anionite AM-10Xp	+0,16—0,25	5,0
	Anionite AM-10Xp	+0,25—0,40	3,0—4,8
Coarse-granular	Anionite AMP	+0,63—1,6	0,45—0,8
	Varion AP	+0,63—1,6	0,5—0,7
	Anionite AM10p	+0,63—1,0	2,0—2,5
	Anionite AM-10Xp	+1,0—1,6	1,5—2,0

For uranium sorption, at the first stage, carboxyl cationite SG-1 was commonly used, which sorbed uranium through complex formation. Since 1948, studies were conducted in the USSR into the specifics of uranium sorption from sulfuric acid and carbonate solutions, as well as from nitric acid, hydrochloric acid, phosphoric acid, and fluoride media. As a result, at the second stage of development of ionic exchange technology, a wider use of anionites, ampholytes, and complexites was possible. In generalizing the results, one notes the clear connection between ionite basicity and its affinity to various uranium complexes. Normally, high-basicity, macroporous anionites, compared with gel-like anionites, have advantages in uranium sorption and desorption kinetics, which determine the optimal areas of application of the commonly known ionic exchange resins:

High-basicity anionites AM, AMP, VP-1Ap — for extraction of uranium from low-salt sulfuric acid media at pH = 1-3 and also for uranium sorption from carbonate media at pH ≤10.5 (for sorption of tricarbonate complexes of uranium phosphonic and arsonic bases are promising);

low- and medium-basicity anionites AN-2F, ÉDÉ-10p, AM-3, and BP-1p, as well as nitrogen–phosphorus containing ionites (API) of the complex type — for processing solutions with a high sulfate content and high residual acidity (30-150 g/liter of sulfuric acid). Similar data have been obtained for phosphoric–sulfuric acid solutions with phosphoric acid content of 150-350 g/liter;

specific anionite AM-10Xp — for extraction of uranium from natural lake and river waters with a content of 20-60 mg/m³ (Table 1).

In the industry, various equipment is used for sorption (at sorption and desorption stages):

standard reactors with mechanical mixing mounted as cascades of 3-6 plants;

columns with suspended ionite layer of up to 150 m³ for processing of diluted pulp of a density of 1.05-1.10 g/cm³;

units with pneumatic mixing of a volume from 0.3 to 500 m³ for processing of dense pulp up to solid-to-liquid ratio of 1:1;

columns with free moving ionite layer (KDS) for washing ionite from ooze and regeneration of uranium-containing ionite; and

columns of continuous sorption–desorption with pneumatic discharge (KNSPR), pressure sorption columns (SNK), and countercurrent ionic exchange columns (PIK-1), with the uranium desorption effectiveness ranked as follows: PIK-1, SNK > KNSPR > KDS.

Extragents such as trialkylamines (TAA), tributylphosphate (TBP), di-2-ethylhexylphosphoric acid (D2EHPA) are widely used for purifying uranium from impurities in processing of industrial eluates and in extracting uranium from ore solutions. Synergetic mixes of acid extragents with neutral organic compounds are efficiently used for these purposes.

The extractional method for purifying and producing uranium has been in use for 25 years. Extractors in most enterprises are horizontal mixer-settling tank boxes with a capacity of 1000–2000 m^3/day, with turbine agitators that have replaced column plants. By modernizing the mixer-settler, the volume of extragent in the fire-hazardous diluent (kerosene) can be reduced. A greater effect, however, can be obtained in using centrifugal extractors and clarifiers of initial solutions; industrial development of their design will allow in the nearest future to switch to processes with intensified phase separation.

Even more effective superextragent substances are under development, which include phosphinoxides, phosphoric trisamide, cyclotriphosphazotrienes, and crown ethers. Experimental production is under way in the USSR of phosphinoxides (up to several dozens of tons annually). Satisfactory results have been obtained with selective nitrogen–phosphorus containing extragents for uranium extraction from nitrate and nitric acid media. Very strong complexformants of actinoids such as phosphoric trisamide and cyclotriphosphazotriene have been found among the substances, with effective constants of uranium extraction attaining the values as high as 10^6–10^7. In recent years, there has been lively interest in a new type of absorbent–impregnated sorbents (IS), often referred to as solid extragents (SE). They are produced by injecting porous copolymers of the extragents into the granules. The resulting absorbent can be recommended for extraction from solutions of a complex salt composition.

Extragent emulsions in mineral acid solutions, such as tributylphosphate or trialkylamine in kerosene, emulsified in 3% sulfuric acid, are recommended for desorption of uranium from high-basicity anionites. The process is conducted in mixer-settler plants with immersed or outside chambers, with the volume ratios of anionite solution and extragent being 1:1:5. The mixing is conducted for 1 h at each stage; ionite is then separated from the emulsion on a sieve for 2–3 min, and a layering of the emulsion is conducted in the settling tank for 5–10 min. This completes uranium desorption from AMP anionite to residual capacity of 0.25% of initial content in 7–8 processing stages for uranium saturation of the extragent of up to 6–7 g/liter. With extractional desorption, effective mass transfer is achieved, and the desorption is completed faster than in ordinary conditions. Similarly, in four desorption stages ferric iron can be eluted from cationite SG-1 by emulsion of kerosene solution of monolaurylphosphoric acid in 30% sulfuric acid, and molybdenum can be desorbed from anionite AMP by kerosene solution of the mixture of extragents D2EHPA with TBP or TAA.

With widespread introduction of nonfiltrational sorption-extraction processes, the relative cost of raw material processing is reduced continuously, and uranium extraction from the ores is raised, while its content in raw materials is dwindling (Table 2).

COMPREHENSIVE PROCESSING OF URANIUM ORES

The accessory valuable components are usually extracted together with uranium during leaching and then separated during sorption, desorption, or extraction, or, less commonly, by chemical precipitation and crystallization.

Molybdenum. Ample industrial experience has been accumulated in the USSR in combined and separate sorption of uranium and molybdenum from acid and carbonate pulps by high-basicity anionites AM, AMP, and VP-1Ap. In acid solutions, molybdenum is readily hydrolized and polymerized, so that its sorption is sensitive to structural and stoichiometric factors.

TABLE 2. Efficiency of Hydrometallurgic Processing of Monometallic Uranium Ores

Operation year	Rel. uranium content, %	Rel. processing cost, %	Uranium extraction, % of 1952-1957 figure
1952—1957	100	100	100
1961—1965	100	80	106
1966—1970	90	70	109
1971—1972	75	70	111

Thus, the sorptive capacity of highly porous anionite VP-1Ap is 5.5 times that of gel anionite AMP. From acid pulps, the maximum separation of components is attained with two-stage sorptive processes where molybdenum is extracted at a relatively high acidity, followed by sorption of uranium from pulps neutralized to pH = 1.5-3.5.

Vanadium. Vanadium behavior during sorption greatly resembles that of molybdenum. However, the depressing action of sulfates is more pronounced. The oxidative potential is important in order to prevent formation of tetravalent vanadium ions existing in solutions with emf below 500 mV that are poorly sorbable by anionites. At emf above 700 mV, vanadium is in a heptavalent state. In real solutions, after leaching of uranium ores, vanadium concentration is 1.5-6.3 g/liter. Useful capacities in terms of vanadium for such solutions may be as high as 420, 330, and 110 mg/g for ionites VP-1Ap, VP-1p, and BPK, respectively.

Copper. In processing uranium—copper ores by acid—uranium leaching, copper passes into solution relatively easily. It can be sorbed together with uranium or separately after removal of uranium by selective ionites. The highest capacities of sorbents in terms of copper at pH = 4 are: 60-80 mg/g for ionites SG-1, AN-2F, ÉDÉ-10p, AMK, ANKB-10, and 100-140 mg/g for ampholytes VPK, ANKB-1, and ANKB-2. There is a class of specific sorbents which can extract copper from more acid solutions with the capacity of 70-100 mg/g at pH = 2. These are the ionites ÉDÉ-10p, VPK, ANKB-1, and ANKB-2.

Rhenium. For sorption of perrhenate ions, activated carbons can be used, which are ranked by their rhenium-absorption capacity as follows: AG-N, AG-3, AG-5 > KAD-iodic > AR-3, BAU, SKT. Carbon is saturated by rhenium up to a content of 1-2%. Better results can be achieved by using high polymer ionites at pH = 5-6, since raising acidity to 5-15 g/liter in the area of low- equilibrium rhenium concentration lowers the ionite capacity by 8-10 times. The efficacy of ionites in rhenium absorption decreases in the following order: VP-1p > AM-3, AN-21 > AM, amberlyte IRA-400 > carbons.

Zirconium. After sulfitization of silicates for an autoclave processing of carbonate uranium—zirconium ores, up to 3-15 g/liter zirconium passes into solution. In an aqueous solution, zirconium and hafnium ions are liable to hydrolysis and polymerization, which compounds the ionic state and diversifies the possible uses of ionic exchange materials. For instance, for solutions with zirconium contents of 0.5 g/liter and pH = 1.8-2.0, practically acceptable ionite saturation can be obtained, %:

Porous sulfocationite KU-23	8-11
Porous carboxycationite KM-2p	6.0
Porous anionite AM-2B, VP-1p	2.3-2.6
Porous ionite AE-14, VPK	3.3-4.0

By adding fluorides into sulfuric acid solutions, one can intensify zirconium absorption by anionites to the following saturation, %:

AM-p, APF-2, VP-1Ap, SF-5	10-12
AMP, VP-1p, VP-15p, VPK	4-8
Am-2B, BPG, KU-2, KU-23, ÉDÉ-10p	2-4

Tantalum and Niobium. In leaching uranium tantaloniobate by sulfitization, up to 1 and 10 g/liter of tantulum and niobium, respectively, in terms of their oxide, pass into solution. They can be extracted by porous anionites AM-p and AMP-p, with total capacity of up to 50 mg/g of sorbent. As for zirconium, tantalum and niobium absorption can be intensified three- to fourfold by introducing fluoride anions (2-8 moles per the total of tantalum and niobium moles). The following features of tantalum and niobium sorption are notable:

With identical matrices, the high-basicity anionites have the highest tantalum capacity (AMP > AM > AM-2B > AM-3 > AM-4);

for identical basicity of the functional group, the highest tantalum capacity is displayed by anionites of the styrene divinylbenzene type (AMP > BP-1Ap);

porous anionites have a higher tantalum capacity than gel anionites, while porosity does not affect niobium sorption; and

anionite AMP-p has the maximum ratio of tantalum and niobium separation.

TABLE 3. Efficacy of Ionite Saturation with Gold from Cyanide Pulps after Leaching Ore with Gold Content of 2 to 5 g/ton

Anionite	Sorbent capacity, mg/liter of sorbent					Share of gold in total absorbed components
	Au	Fe^{+3}	Zn	Ni	Cu	
AM-2B	15.3	1.3	3.1	3.5	4.7	0.55
AM-p	3.1	13.5	4.3	5.5	25.6	0.17

Gold. From gold-containing uranium ores, gold is dissolved by the method of sorptive leaching with cyanides of alkali metals before and after uranium extraction. To separate gold from the pulp, the bifunctional anionite AM-2B is used which features high capacity and enhanced selectivity (Table 3). Saturated anionite is regenerated by urea solutions. From the eluates obtained after extractive purification, crude gold ingots are obtained, which are then refined electrolytically to obtain ingots with 99.99% gold content.

Cesium. During the sulfuric acid leaching of uranium ore, up to 5 mg/liter cesium and 12 mg/liter rubidium passes into the solution from destroyed hydromica, chloride, and some other minerals. Concomitant dissolution of cesium and rubidium is enhanced when autoclave leaching of uranium ores is practiced. The dissolved rare alkali metals can be extracted from the pulp with simultaneous concentration of more than 2000 times in terms of cesium by using the mineral-organic sorbent cesiite-7. For this, countercurrent operation mode is used, and one container with sorbent load in volume of 10% and pulp stay of 1 h is sufficient. The capacity of saturated sorbent is 12.7-14.0 mg/g.

Thorium, Scandium, Rare-Earth Elements, and Phosphate Fertilizers. Clay uranium—phosphorus—rare-earth—scandium ores with considerable amounts of microscopic pyrite (melnikovite) and uranium—phosphorus—thorium ores containing some carbonates and clay are processed in the USSR. The ores are subjected to preliminary mechanical concentration because of a low content of such valuable components as phosphorus pentoxide (4-12%), thorium (0.03%), and rare-earth elements (0.2%). Two-stage disintegration and multistage classification in hydrocyclones of the class ± 0.02 mm are practiced: The underflow of hydrocyclone is the francolitic concentrate with a high content of all valuable components and high extraction (90%), while the overflow is dispersed clay and microscopic melnikovite grains. A process has been developed to separate from the tails in battery hydrocyclones, pyrite concentrate suitable for production of sulfuric acid, required for subsequent leaching of uranium—phosphorus—rare-earth concentrate. The latter is leached by a mix of sulfuric and nitric acids, bringing virtually all valuable components into solution and fixing almost half of the calcium as gypsum with which the bulk of radioactive elements (mainly ^{226}Ra and ^{230}Th) coprecipitate. The fertilizer obtained as a result contains up to 50% of water-soluble phosphorus and 50% of citrate-soluble phosphorus.

From the phosphoric acid solution, after iron reduction, collective chemical concentrates of uranium and rare earth elements and the remainder of actinium and ^{230}Th are precipitated; the concentrate is then dissolved in nitric acid and uranium is obtained by extraction and subsequently separated during reextraction stage in the form of uranyl tricarbonate, leaving rare-earth elements in the mother liquor of the extraction. Depending on the consumption of rare-earth elements, mother liquor is purified by sorption or extraction and carbonates are settled which are then annealed to obtain pure oxides of the total of rare-earth elements and yttrium. The latter are processed with separation of pure yttrium and europium oxides for luminophors, ligatures, and catalysts.

From nitric—phosphoric solutions, after separating the collective concentrate by ammonization, evaporation, and granulation, one obtains nitric—phosphoric fertilizer such as nitrophos, which contains up to 40% nutrients in the clay form. The benefication concentrate can be processed with a simplified technology without removing the rare-earth elements, by leaching in sulfuric acid, filtering on rotary filters, extracting uranium and scandium from the solution, and subsequently obtaining compounds of uranium and scandium (oxides). After ammonization, evaporation, and granulation, nitrogen—phosphate fertilizer of the ammophos type is obtained.

Uranium—phosphorus—thorium ores are processed similarly by decomposing the concentrate with sulfuric acid, uranium and thorium extraction from the solution, and separate reextraction of uranium and thorium. The uranium reextract is subjected to additional extractional purification to obtain pure crystals of ammonium uranyl-tricarbonate, as well as chemical concentrate from the thorium reextract.

From poor industrial ores of a complex composition, for the first time, uranium and many valuable accessory components thus have been extracted with positive economic characteristics and simultaneous utilization of spent chemicals and production of nitrogen—phosphate ammonium fertilizers. The sale of the by-products cuts the uranium extraction costs approximately by one half.

EXPERIENCE IN HANDLING SPENT FUEL FROM NUCLEAR POWER STATIONS IN THE SOVIET UNION, INCLUDING STORAGE AND TRANSPORTATION

V. M. Dubrovskii, V. I. Zemlyanukhin,
A. N. Kondrat'ev, Yu. A. Kosarev,
L. N. Lazarev, R. I. Lyubtsev,
E. I. Mikerin, B. V. Nikipelov,
A. S. Nikiforov, V. M. Sedov,
B. I. Snaginskii, and V. S. Shmidt

UDC 621.039.59

The development of nuclear power generation in the Soviet Union from the very start has been planned on a closed fuel cycle, i.e., with regeneration of the spent fuel and the use of plutonium produced in fast reactors. The spent fuel of nuclear power stations constructed in other countries with the cooperation of the Soviet Union will also be processed in the Soviet Union, and therefore the Soviet Union should be considered as the regional center for the reprocessing of Nuclear power station fuel.

In the designs of nuclear power stations with VVÉR, storage vaults are provided, calculated on cooling of the fuel for up to 3 years, which are attached directly to the reactors. However, this cooling is advantageous only when the mass construction of fast reactors starts, in which plutonium will be used and which represents the principal value of the spent fuel.

Taking account of the delay in the construction of commercial fast reactors, the creation of additional, individually standing storage vaults for spent fuel, calculated on an $\approx$ 10-yr operation of the nuclear power stations, is being considered.

However, the solution does not eliminate the necessity for transportation and regeneration of the spent fuel, but only reduces their pace somewhat.

<u>Storage of Spent Nuclear Fuel.</u> The storage of spent fuel is provided for directly in the case of the VVÉR-440, VVÉR-1000, RBMK-1000, and RBMK-1500 reactors, and in individually standing building in the case of nuclear power stations. The solutions with respect to these storage vaults are, in principle, approximately identical, and therefore the principal solutions are being considered by the example of a standard design of an additional, separately standing storage vault for nuclear power stations with VVÉR-440 (Fig. 1).

The additional storage vault for the spent fuel from nuclear power stations with VVÉR-440 has been developed on the basis of experience in the designing and operation of similar vaults in the Soviet Union and in other countries. This vault has been calculated on the total receipt of spent fuel from four reactors over 10 years, which amounts to $\sim$ 5000 fuel element assemblies or $\sim$ 600 tons of fuel.

The grouping, technological process, and plant have been solved by taking account of the maximum utilization of the standard type and nonstandard plant, used in similar storage vaults.

Translated from Atomnaya Énergiya, Vol. 54, No. 4, pp. 293-297, April, 1983.

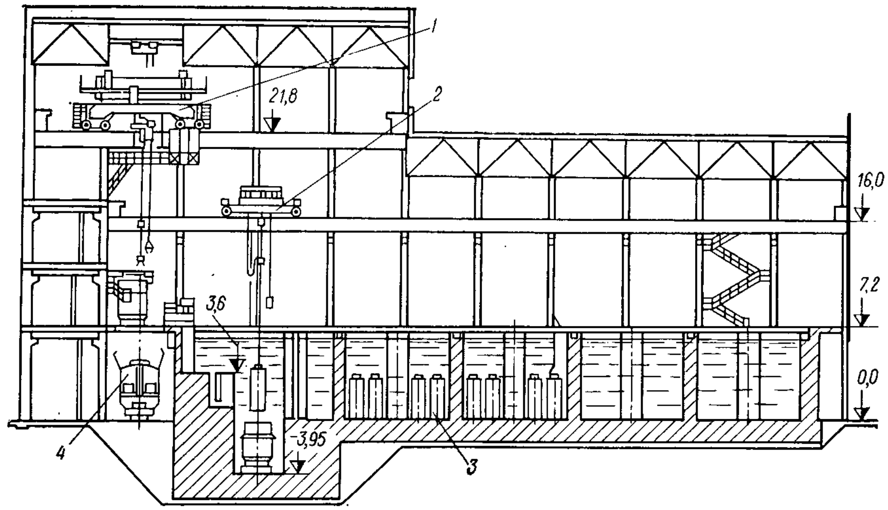

Fig. 1. Spent fuel storage vault (longitudinal section): 1) bridge crane 125/20 tons; 2) bridge crane 15 tons; 3) coffin with fuel element assemblies; 4) container wagon.

The additional storage vault consists of a section for the admission, transshipment, and distribution of the transport containers, a section for the storage of the fuel, and a section for technological systems and services for providing the fuel-storage conditions.

The section for the receipt and transshipment of the transport containers includes a transport corridor and a transshipment room. The transport corridor is located below the transshipment room and is joined to it by a shielded opening for transfer of the containers. In the transshipment room are located: the compartment for transshipment of the containers, a compartment for washing and discharging the fuel element assemblies, pits for washing out and decontamination of the containers, pits for the storage of equipment and instruments, and also a section for small repair work of technological plant. The transshipment and washing compartments are joined by a transfer corridor with storage compartments for spent fuel element assemblies.

The transshipment room is equippped with a crane with a load lifting capacity of 125/20 tons (with supplementary lifting values of speed) and other devices for operating with specialized equipment, benches for the technological monitoring of contaminated containers, and also a tool for transfer operations.

The compartment for the storage of the spent fuel consists of a tank filled with water, and a transport room. There are four sections in the tank, in which the coffins with the spent fuel element assemblies are installed for storage and a transfer corridor joining the tank compartment and the transshipment and fuel element assembly washing compartment.

In the tank compartments, a slit cover is provided, which ensures normal working conditions for the personnel. The slits are the transport paths for the conveyance of the fuel coffins and they provide the necesary orders of separation of the coffins in the compartments. Below the tank is located the transport room of the storage vault. The spent fuel is stored without the constant presence of servicing personnel. At the panel of one of the nuclear power station units, secondary monitoring signals are displayed, about the state of the principal technological parameters determining the normal storage conditions of the fuel element assemblies: temperature of the water in the tank compartment, the level of the water in the compartments, data about the functioning of the cooling and purification systems, and the conditions of the air medium of the tank. Storage is carried out under a protective layer of water (~ 3 m above the active part of the fuel element assemblies).

The compartment for the technological systems and the service for ensuring the storage conditions of the spent fuel is an annex, in which are located the cooling and water purification systems of the tank, the technological and sanitary-technical ventilation, electrical supplies, washing and decontamination, and also a desk, operator controlled, medical check point, administrative and other rooms.

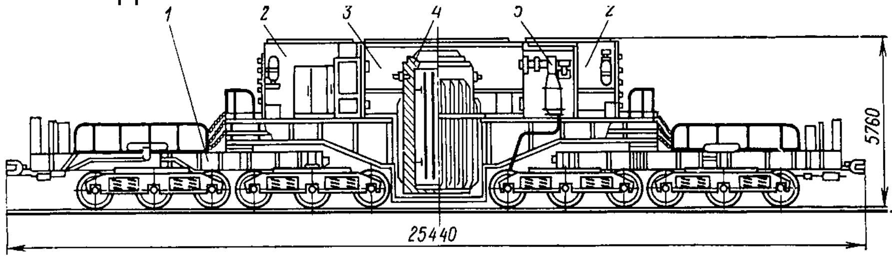

Fig. 2. Railroad container wagon for transporting spent fuel from nuclear power stations with VVÉR-440: 1) transporter; 2) auxiliary compartment; 3) load compartment; 4) container; 5) ventilation and heating system.

The fuel is conveyed from the nuclear power station units in a TK-6 transportation container, in which a transport coffin with 30 spent fuel-element assemblies are placed. After a 3-yr cooling in the reactor pond, the fuel is loaded into a container which is transferred to a specially equipped trailer in the transportation corridor of the storage vault. The container is lowered into the room with the bridge crane, and installed in the transshipment compartment which is filled with water. The transportation coffin with the spent fuel element assemblies is withdrawn from the container with the bridge crane, and transferred to the tank compartment where it is placed in storage. The container from the transshipment compartment is transferred to washing and decontamination, after which the empty coffin is placed in it. The prepared container is taken away for a new load.

Transportation of Spent Nuclear Fuel from Nuclear Power Stations. The spent fuel from nuclear power stations with VVÉR-440 is transported by train, consisting of four to eight TK-6 container-wagons (Fig. 2) and two VS-TK-3 and VS-TK-4 escort-wagons. For the transportation of fuel from nuclear power stations with RBMK, a TK-11 container wagon is being developed, and from nuclear power stations with VVÉR-1000 — a TK-10 container-wagon.

The TK-6 container-wagon is a 12-axle railroad transporter, equipped with a body inside which the shielded container is installed. The container (load) compartment of the body is provided with thermal insulation. There is a ventilation-heating facility in the body, by means of which the load compartment can be cooled or heated up. This solution ensures the necessary thermal conditions for the loaded spent fuel container under all possible transportation conditions.

The shielded container, with the coffin installed in it, comprises the transportable packing assembly. The State Committee for the Utilization of Nuclear Power in the Soviet Union issues a certificate, according to which this packing assembly loaded with fuel element assemblies with a burnup of up to 24 GW•day/ton, with the inside cavity of the container filled with gas, is defined in accordance with the IAEA classification as a type B(U) package. For this, the maximum heat release of an individual fuel element assembly is limited to 340 W, and the total heat release of the container is 8 kW. Calculations and tests of the packing assembly have shown that with the stated values of heat release the temperature of the gas coolant in the container does not exceed 175°C, and the temperature of the fuel element cans does not exceed 200°C.

When transporting the spent fuel of VVÉR-440 with a burnup of more than 24 GW•day/ton in containers filled with gas, an additional biological shielding from neutron radiation is required. It can be filled on the outside (water jacket, hydrogen-containing materials), and also it can be accomplished by filling the container with water.

Different methods of loading the spent fuel element assemblies into the container have been tested at different nuclear power stations: the "dry" loading of the coffin with 30 fuel element assemblies into the container, previously removed from the transport facility; dry loading of the coffin with the fuel element assemblies into the container without removing it from the wagon; "wet" piecemeal loading of the fuel element assemblies into the container installed in the cooling pond; and wet loading of the coffin with the spent fuel element assemblies into the container.

From the point of view of the duration of operations, the dry loading scheme must be accepted as the most suitable. Despite the fact that additionally a transfer container must be used, and time is involved in the additional operations, the overall duration of loading is found to be less than for the wet version. The surface of the container in the case of wet loading is inevitably contaminated with radioactive substances, present in the water of the storage vault tank, and the complex surface configuration (the presence of fins, accessories, etc.) makes decontamination a time-consuming and prolonged operation, the maximum duration of which amounted to about 24 h per container. As a result of this, the overall duration of the presence of the railroad echelon at the nuclear power station during loading is increased significantly.

The containers, after the fuel element assemblies are loaded, are stood until the temperature inside the container has stabilized. The accompanying documents are drawn up on finalization of the monitored measurements of the level of radiation, the temperature of the medium inside the container, and other parameters. The conveyance of the spent fuel from the nuclear power station is effected mainly by railroad transport.

In practice, different schemes of conveyance are used, in which in addition to railroad transport, truck and water transport also are used. Conveyance is carried out in accordance with "Regulations for the Safe Conveyance of Spent Nuclear Fuel from Nuclear Power Stations," and "Technical Conditions for the Assembly of Spent Fuel Elements."

On the whole, all types of trnasportation of spent fuel are conducted without incident and unforseen holdups. The parameters of the packing outfit have corresponded to the technical conditions and instruction for operation. The temperature was below the maximum permissible.

Reprocessing of Spent Nuclear Fuel. From the nuclear power station the spent fuel elements are transported to the reception department (storage vault) of the reprocessing facility, similar to their solutions for storage vaults in the case of nuclear power stations. The wagons, with the containers, are conveyed in turn into the transport corridor. The scheme for discharging the wagons without removing them from their container is adopted.

The coffins with the fuel elements are transferred from the storage vault to a room where the batches of fuel are grouped according to the fissile material content, using nondestructive monitoring. The methods for recording the γ-quanta of the fission products or the neutrons of the natural radiation are used for this.

The grouped batches of spent fuel elements in the coffin are transferred to the next room where, by means of the electric arc method and with a disk saw, the shafts of the fuel element assemblies are cut off and directed to the burial ground; the cores of the fuel element assemblies are directed to the chopping machine for size reduction. The fuel fragments from the chopping machine enter the dissolver.

A continuous action vibration dissolver and a periodic action dissolver with pneumatic charging of the fragments have been developed.

In order to catch and localize ^{129}I (~ 130 mCi/ton* of uranium) from the gas phase, two technological processes have been developed: the use of an organosilicon liquid and alkali. The latter allows the main bulk of the ^{14}C also to be caught simultaneously with the ^{129}I.

The tritium formed during operation of the reactor, due to the ternary fission of ^{235}U, ^{239}Pu, and ^{241}Pu nuclei, the content of which in the spent fuel may attain 700 Ci/ton of U, more than 95% remains in the graphite during extraction reprocessing and, during concentration of the latter, it is distributed between the regenerated nitric acid and the condensate. In this technological process, one version for the concentration of the tritium is to return the tritium-containing acid and condensate to the process with part of the condensate led away to burial ("recycle" version).

Another method of localizing the tritium is the thermochemical processing of the dissected fuel in an atmosphere of hydrogen or water, at a temperature of about 723°K. This operation, before dissolution of the fuel, allows more than 99% of the tritium to be concentrated into small volumes. In order to purify the solution of nuclear fuel obtained from solid impurities, cartridge metal-ceramic filters and centrifuges have been developed.

The extraction process lies at the basis of the nuclear fuel reprocessing technology, in which tributylphosphate (TBP) in a diluent is used as the extraction mixture. The choice

*1 Ci = $3.700 \cdot 10^{10}$ Bq.

of the diluent is determined from the results of an investigation of its physicochemical properties, radiation and chemical stability, the presence of impurities and their effect on the extraction efficiency of the TBP; the flash point and melting point, viscosity, density, and surface tension.

In the first extraction cycle, the uranium and plutonium being jointly removed separate from the main bulk of the radionuclides — fission products and impurities remaining in the aqueous-tail solution, which proceed to concentration and subsequent extraction of valuable elements. Then, from the extract by means of reducing reactions, the plutonium is reextracted, after which the uranium is reextracted with dilute nitric acid.

The aqueous flows, containing the uranium and plutonium, are guided to the corresponding cycles for final purification and subsequent separation of solid compounds. In the final plutonium purification cycles, a reflux-process can be used, which promotes an increase of the purification coefficients with a significant additional concentration of valuable elements.

The extraction process is optimized by means of mathematical modeling on a computer. The purpose of automatic control is to ensure the conditions which are maximally favorable to the purification and concentration of the valuable components with a guaranteed retention of other indices (for example, extraction) at the required level.

For extraction processes, mixer-settlers have been developed best of all; these are equipments most easily modelled and are amenable to dimensional changes during construction. They ensure stable operation, even in the case of periodic shutdowns of the process, and they allow the flows of solutions to be withdrawn from any stage of the reflux extraction. Centrifugal extractors (CE) possess these same merits, which are characterized by a short duration of contact between the extractant and the highly active aqueous solutions, and small product and extractant volumes in the cycle, which increases the safety of the process and reduces the volume of incomplete production. At the same time, the complexity of design of centrifugal extractors increases the demands on the quality of their manufacture, servicing and repair.

The purified plutonium from the solution can be converted to the form of the final solid product by precipitation of the oxalate and then, by calcining the latter, the dioxide can be obtained. The uranium can be precipitated in the form of ammonium diuranate and, in the case of necessity for producing mixed uranium—plutonium fuel, in the form of ammonium diuranate —Pu(VI) hydroxide, with subsequent calcining and reduction to $(U, Pu)O_2$. Processes for obtaining the oxides (individual and mixed) by direct thermal denitration of the reextracts after their concentration have also been investigated.

For the future use of the final regenerated products, monitoring of the isotopic content of the latter is required and, in particular, the content of nuclides — sources of the daughter elements of highly active β-emitters.

A three-zoned grouping of the process has been adopted, according to which all the plants with the radioactive solutions must be allocated to the first zone behind a biological shield. Compartments intended for maintenance work must be referred to the second zone, and operator and shielded compartments, etc. to the third zone. Ventilation of all zones is effected by different systems. In order to ensure the gravity flow of movement of solid and liquid products, a cascade arrangement of the plant is used.

All plants with a short operating lifetime must have a stand-by. Disabled monitoring-measurement devices, valves, batchers, etc. must be replaced by remote control, by means of special mechanisms and without shutting down the process.

The regeneration technology established for the fuel elements of thermal reactors, provies for their complex reprocessing and the production, together with uranium and plutonium, also of concentrates of rare-earth elements (REE) and strontium. In order to extract these elements from the aqueous tailings-solution the first cycle containing all the fission products, evaporation (in order to remove nitric acid) and partial neutralization are necessary.

In order to separate the valuable elements from the concentrated raffinate, two technological processes have been developed, in which a solution of tributylphosphate or di-2-ethylhexyl phosphoric acid (D2EHPA) in a diluent is used as the extractant. These schemes ensure more than a 90% yield of rare earths in the concentrate, the mutual separation by a factor of 20–30, and a purification coefficient from other elements (fission products) of more than 100. The use of D2EPHA as the extractant also allows a concentrate of strontium to be obtained, with a yield of more than 95% and a purification coefficient from REE of 50-100.

The necessity for reducing the acid concentration in the solution before extraction requires special distillation operations, or neutralization of the HNO_3. Therefore, a search is being conducted for extractants capable of extracting the trivalent platinoids, strontium and cesium directly from solutions of 2-3 M HNO_3. Although there are technological difficulties still limiting the feasibility of the practical application of the new extractants, it has been established that the REE are extracted well from acid solutions by phosphine dioxides and bidentate extractants, containing simultaneously the groups P=O and C=O, and cesium and strontium from 2-3 M HNO_3 by dicarbolides, products of the carboranes.

In perpective, in proportion with the emergence of sufficient demand in the national economy, certain other valuable elements can also be recovered from the raffinates of the extraction schemes for reprocessing fuel elements, for example, technetium and palladium.

The theoretical investigations in the field of the kinetics of extraction processes, mathematical modeling of counterflow extraction, forecasting the properties of extractants of different structure and diluents of different nature, and work on the preparative chemistry of solid compounds of plutonium and other actinides, should serve as the basis for the further improvement of fuel element reprocessing technology. Successes in the investigations of the scientific bases of regeneration technology for the fuel of thermal reactors, are also the basis for the creation of a technological scheme for reprocessing the fuel of fast reactors, meeting present-day demands. Research work has been directed at the creation of processes and equipments of very high speed, with a short retention time of the products and with a high unit output. These investigations are the principal reservation for increasing the labor productivity of personnel and plant, and the reduction of capital and operating costs, which is an urgent problem related with the steady growth of capacity of nuclear power generation and, correspondingly, with the increased demand for fuel regeneration.

PROBLEMS OF RADIATION SAFETY OF ATOMIC POWER PLANT PERSONNEL AND THE PUBLIC

E. I. Vorob'ev and O. A. Pavlovskii

The great deal of attention devoted to the problem of ensuring the radiation safety of the personnel and the public at various stages in the nuclear fuel cycle (NFC), which was discussed at the IAEA conference on experience acquired in nuclear power generation (September, 1982), is well illustrated by the fact that these topics were dealt with at the plenary session "Experience in the Domain of Nuclear Safety," the technical meeting "Radiation Protection in an Atomic Power Plant and Fuel Cycle Plants," as well as a special evening meeting "Nuclear Power, Energy, and the Health Physics Conditions in the Environment." These meetings heard 19 papers.

The main paper at the plenary meeting was the ICRP Report "International Recommendations on Radiation Protection: Five Years of Experience after Release of ICRP Publication No. 26." The report discussed the influence of the ICRP recommendations on the administrative aspects of radiation protection and described the experience gained from the practical application of this document during the 5 years. Total recognition of this ICRP Publication has been discussed broadly in all countries, although many of its concepts and definition such as the effective equivalent dose, collective dose, expected dose, and optimization of protection, which had at first been received with doubt and scepticism, are gradually being accepted.

The IAEA Report "Present-Day Problems of Nuclear Safety" considered the role of international organizations. It devoted much attention to the program of IAEA publications on nuclear safety such as "IAEA Safety Standards," "IAEA Safety Handbook," "IAEA Recommendations,"

Translated from Atomnaya Énergiya, Vol. 54, No. 4, pp. 303-305, April, 1983.

 0038-531X/83/5404-0314$07.50

as well as "Informational and Procedural Documents." The program encompases the preparation and publication of more than 50 publications and is known as the PDSS APP (program for the Development of Safety Standards for Atomic Power Plants). Of these, 26 publications have already been published in English and many of them in other IAEA working languages (Spanish, Russian, and French).

The paper "Principle of Designing for Safety" was devoted to the principal concepts of safety of atomic power plants in the Federal German Republic. These are the high quality of the materials, the minimum number of welded seams, optimization of the structural strength of materials, limitation of the work load on the operator, and monitoring of his working conditions, as well as monitoring and periodic inspection of the sites of coolant leaks. It is pointed out that careful design is a reliable obstacle to the emission of radioactive substances into the environment during accidents in atomic power plants.

The utmost interest was aroused by a paper by W. Marshall (Great Britain), "Major Nuclear Accidents" which pointed out that the discussion on the safety of nuclear power generation is due primarily to the accumulation of a vast number of unlikely events which might cause a major accident during the operation of a reactor, with a deadly outcome among the public. The improper use of terminology has given the public a warped picture of the safety of the nuclear power industry. The paper demonstrated that if one assumes a major accident in an atomic power plant, as a result of which the 10 million inhabitants of London will receive a dose of 1 rem, then this could subsequently cause 1250 cases of cancer with a fatal outcome. However, this same injury will be caused if every Londoner smokes only 1/20-th of a cigarette every Sunday. In the last paper of the plenary session ("Processing of Information Concerning Experience in Operation and Operator Training," France) it was noted that the training of operators and their response actions under extraordinary circumstances may have a major influence on the consequence of an accident in a reactor.

The technical meeting heard 12 papers.

The main approaches to the standardization of the dose loads on the personnel and the public during the operation of an atomic power plant were presented in an Argentinian paper. As it pointed out, a reduction in the irradiation is conditional upon improvement of the monitoring in atomic power plants and the nuclear fuel cycle. In the opinion of the ICRP, in order for operations involving the action of radiation to be legitimate it is necessary that the radiation protection be optimal and that in all cases the limits on the individual dose be observed. However, legitimacy is not a criterion that is used directly in designing the shielding. The aim of the optimization of the radiation shielding is to obtain adequate equilibrium between the cost of the shielding and the damage inflicted by the residual action of the radiation so that the total time of harmful action be reduced to a minimum.

The use of dose limits in the optimization procedure, however, poses difficulties in the case of irradiation of the population. In view of this it is proposed to use upper dose limits, established by component bodies; these upper limits would be such a small part of the corresponding dose limits so as to avoid overlapping of doses of irradiation from different sources and to ensure a certain margin for unforeseen circumstances. On the basis of this, Argentina adopted a limit on the dose of irradiation to the population from discharges and emissions from an atomic power plant according to two criteria at once: the individual dose — 30 mrem/yr and standardized collective radiation dose — 1.5 man•rem/MW(E)•yr.

Another Argentinian paper described the experience from the operation of the Atchua atomic power plant with a power of 367 MW(E) and estimated the irradiation of the public from the 600-MW(E) Cordova atomic power plant commissioned in 1983. It was emphasized that the radiation dose received by the public upon implementation of the plans for nuclear power development in the country by 40-50 GW(E) will be 200 man•rem and will be smaller by a factor of almost 20 than the radiation dose of atomic power plant personnel (3500 man•rem) and by a factor of 50 than in the production of uranium (about 10,000 man•rem).

In a paper on the technique of optimization and limitation of the dose in underground uranium mines in France attention is drawn to the necessity of ensuring optimum radiation safety. In this case an endeavor must be made to make sure that the striving for economy should not lead to a worsening of the protection. The optimization took account of the following factors: the dose-effect relation, the dosimetry, allowance for the radiometric characteristics in the uranium deposits, the mechanism of irradiation of the personnel, and protective measures.

In producing dose loads on personnel engaged in the nuclear fuel cycle in France, the individual radiation dose received by uranium miners will be in first place ($\sim$3 rem/yr) while the collective dose is 0.57 man·rem/MW(E)·yr, i.e., will be equal to $\sim$40% of the collective dose of the entire cycle.

A Soviet paper on the practice of ensuring radiation safety in atomic power plants and predicting the dose loads on the population in connection with the development of nuclear power generation presented the main premises of Soviet legislation ensuring protection of the health of personnel and the public as well as reducing the environmental pollution. It was shown that the mean annual dose received by the personnel of atomic power plants with VVER, RBMK, and BN reactors with various powers is 0.14-1.2 rem and the standardized collective dose is 1.1-1.3 man·rem/MW(E)·yr. Thus, conditions have been created in atomic power plants in the Soviet Union so as to carry out the recommendations of the ICRP that the mean radiation dose of the personnel should be one-tenth that of the maximum allowable dose of 5 rem/yr. The collective radiation dose of the population upon implementation of the plans for the development of nuclear power generation is 2300 man·rem/yr and about 98% of this will be accounted for by gas-aerosol emissions from atomic power plants. The collective risk is 0.33 man/yr, i.e., absolutely undetectable against the background of the natural level of malignant tumors. This collective dose is equal to the radiation dose received by the population from the natural background radiation in only 40 min.

Specialists from Britain characterized the conditions at several nuclear fuel cycle plants in their country. They pointed out that since 1976 cases of a dose of 5 rem/yr for employees of the Sellafield plant have become only single cases or do not occur at all whereas previously there had been 100-140 cases a year. The collective radiation dose has been stabilized at a level of 5000-5500 man·rem/yr, although the number of the personnel has increased steadily, reaching 10,000 in 1980. The mean individual dose is 0.53 rem/yr, which is in accord with the ICRP recommendation. They reported that in 1971 tests revealed that 14 workers had 50% of the maximum content of plutonium in their bones.

An analysis was made of the death rate due to cancer and other diseases among professional workers of three plants in the period 1962-1978. It was shown that the 283 deaths among workers and 202 among pensioners correlate well with the average indicators for Great Britain, thus indicating that work-related irradiation does not cause an increase in the cancer incidence among employees of the nuclear fuel cycle plants in the country.

The Central Electricity Authority of Great Britain devoted its report to 20 years of experience in the domain of radiation safety. In the fifties, certain difficulties were encountered in establishing the allowable dose limits. In 1960 generally accepted dose limits were introduced: 5 rem/yr for professional workers and 0.5 rem for the population, although a dose of 1.5 rem/yr was allowed for individual critical groups near the controlled zones. Consideration is being given to the development of views on the ICRP allowable dose limits and their reflection in standardizing documents. Although a dose of 5 rem/yr is the established dose, less than 15% of the personnel receives a dose exceeding 0.5 rem/yr and the mean dose is a mere 0.24 rem/yr. The collective dose is 0.4 man·rem/MW(E)·yr. Gas-aerosol emissions and liquid discharges from atomic power plants are carefully recorded in annual reports.

In Japan, eleven BWR and ten PWR with a total power of 16.5 GW(E) are in operation. The Japanese Nuclear Safety Commission set limits of a dose of 5 mrem/yr for the whole body and 15 mrem/yr for the thyroid of children. The limits set for emissions are $5 \cdot 10^4 - 3 \cdot 10^5$ Ci/yr and $(3-9) \cdot 10^4$ Ci/yr for PWR, which leads to a maximum calculated dose of external radiation at a level of 0.3-3.5 mrem/yr for the whole body and 0.4-13 mrem/yr for the thyroid of children with allowance for inhalation and peroral intake of radionuclides by children. The calculated collective radiation dose is 0.02 man·rem/MW(E)·yr for PWR and 0.04 for BWR. The actual emissions from atomic power plants in Japan, however, are almost two orders of magnitude lower than the maximum values. As a result, the collective dose of radiation received by the population of Japan in 1980 was only 2 man·rem/yr and with the start-up of atomic power plants with a total capacity of 51 GW(E) in Japan this will rise to only 10 man·rem/yr. Such low emissions (a few curies per year) are due to the fact that gases are held prior to discharge into the ventilating stack of the atomic power plant as well as to the low percentage of nonhermetic fuel elements.

In its paper the German Democratic Republic presented extensive material on the collective radiation dose of personnel of the Bruno Loischner Atomic Power Plant. It was shown that as in other atomic power plants, e.g., in the U.S.A., the dose increases with the operating time. For the second unit of the plant it grew from 0.8 to 2 man•sievert/reactor•yr for the period 1975-1981. The paper pointed out that the contribution of repairs to the total radiation dose of the personnel is extremely large, reaching 86-96%.

The report from the USSR generalized research done on such water systems as the Baltic Sea, the Danube basin, and the Black Sea. In the period from 1974 to 1980 the Baltic displayed a tendency toward a decreasing concentration of ^{137}Cs and ^{90}Sr. The tritium concentration in recent years has been close to 0.2 nCi/liter. It has also been found that on the whole processes reducing the concentration of artificial radionuclides predominate. The observed concentrations and dose loads are much lower than the allowable levels. In the near future radioactive contamination will not restrict the development of nuclear power on the shores of the Baltic and the Danube basin if the requirements and recommendations on limiting discharges and emissions are met.

The paper from the Federal German Republic gave data about the radioactive emissions from reactors and the radiation dose. A tendency has been observed towards reduction of discharges, especially from reactors built in recent years, which can be explained by advances in technology. The results attest to the satisfactory state of radiation protection near atomic power plants in the Federal German Republic.

Great interest was aroused by a paper from the USSR on comparative assessment of the injurity to the health of the personnel and the population during the generation of electricity in atomic and thermal power plants. It was shown that the risk due to the deleterious effect on the health of the personnel and the population during operation of a coal-fired power plant is several tens of times that from an atomic power plant with the same power. And when the chemical components of the discharges are taken into account it is less dangerous to live near an atomic power plant than near a thermal plant by a factor of 36,000.

The last paper presented in the given section emphasized that the technical features of the CANDU reactor (Canada) are characterized by a low defectiveness of the pipes in the cooling system, which was a mere 0.02% for 12 such reactors as against 2.1% for 85 LWR. It was pointed out that the radiation dose of the personnel in the Pickering-A and Bruce-A atomic power plants was 0.43 and 0.2 man•rem/MW(E)•yr, respectively, in 1981, which is below the corresponding indicators for LWR.

The evening session of the conference heard a report from the United Nations Scientific Committee on the Effects of Atomic Radiation, prepared by the General Assembly in 1982. In this document, as in the previous report in 1977, general information is presented about sources of radiation and the effects produced in the human body and animals under the action of ionizing radiation. A great deal of attention was devoted to analysis of the dose loads on the personnel and population at various stages in the nuclear fuel cycle. Another paper made a comparative assessment of nuclear and other sources of energy (especially coal and wood) from the point of view of their impact on the environment and man. In the discussion that followed much attention was paid to the contribution by a Soviet specialist concerning the health physics and ecological aspects of obtaining electricity from atomic, thermal, and hydroelectric plants as well as solar power plants. It was demonstrated convincingly that as far as minimal impact on the environment and lack of influence on public health are concerned, nuclear power is unmatched by any other developed energy source. At the same time all of the participants in the discussion pointed out that in the nuclear power industry, especially in the handling of radioactive wastes, there still are quite a few problems that require serious attention and study.

On the whole, it can be said that the papers read at the conference contained useful information about:

standards requirements on ensuring the radiation safety of the personnel and population;

practical means of ensuring the radiation protection of the personnel and population at individual stages of the nuclear fuel cycle;

the actual dose loads on the personnel and population as a result of discharges and emissions of radioactive products at various stages of the nuclear fuel cycle;

estimates of the development of nuclear power in different countries in the next few decades and the attendant dose loads on the population of these countries;

comparison of the nuclear fuel cycle with other methods of obtaining electrical energy from the point of view of protection of public health and protection of the environment. It has been shown that improvements in the technology, automation, and increase in the technical level of operation, and higher qualifications of the personnel leads to a reduction in the radioactive emissions and the radiation dose of both the personnel and the population.

The conference proceedings are of unquestionable interest for application in the domestic practice of ensuring radiation safety in atomic power plants and other enterprises of the nuclear power industry.

How To Comply With The New Copyright Law

Participation in the Copyright Clearance Center (CCC) assures you of legal photocopying at the moment of need.

Libraries everywhere have found the easy way to fill photocopy requests legally and instantly, without the need to seek permissions, from more than 3000 key publications in business, science, humanities, and social science. You can:

Fill requests for multiple copies, interlibrary loan (beyond the CONTU guidelines), and reserve desk without fear of copyright infringement.

Supply copies from CCC-registered publications simply and easily.

The Copyright Clearance Center is your one-stop place for on-the-spot clearance to photocopy for internal use.

Its flexible reporting system accepts photocopying reports and returns an itemized invoice. You send only one convenient payment. CCC distributes it to the many publishers whose works you need.

And, you need not keep any records, the CCC computer will do it for you. Register now with the CCC and you will never again have to decline a photocopy request or wonder about compliance with the law for any publication participating in the CCC.

To register or for more information, just contact:

Copyright Clearance Center
21 Congress Street
Salem, Massachusetts 01970
(617) 744-3350

a not-for-profit corporation

NAME ____________________ TITLE ____________________

ORGANIZATION ____________________

ADDRESS ____________________

CITY ____________________ STATE __________ ZIP __________

COUNTRY ____________________ TELEPHONE ____________________

CHANGING YOUR ADDRESS?

In order to receive your journal without interruption, please complete this change of address notice and forward to the Publisher, 60 days in advance, if possible.

(Please Print)

Old Address:

name

address

city

state (or country) zip code

New Address

name

address

city

state (or country) zip code

date new address effective

name of journal

THE LANGUAGE OF SCIENCE

Plenum

PUBLISHING CORPORATION

233 Spring Street, New York, New York 10013

RUSSIAN [redacted] PEEL HERE PHYSICAL
AND N IENCES

AVAILABLE IN ENGLISH TRANSLATION

ALGEBRA AND LOGIC
Algebra i Logika
Vol. 21, 1982 (6 issues) $270

ASTROPHYSICS
Astrofizika
Vol. 18, 1982 (4 issues) $320

AUTOMATION AND REMOTE CONTROL
Avtomatika i Telemekhanika
Vol. 43, 1982 (24 issues) issues) $495

COMBUSTION, EXPLOSION, AND SHOCK WAVES
Fizika Goreniya i Vzryva
Vol. 18, 1982 (6 issues) $345

COSMIC RESEARCH
Kosmicheskie Issledovaniya
Vol. 20, 1982 (6 issues) $425

CYBERNETICS
Kibernetika
Vol. 18, 1982 (6 issues) $345

DIFFERENTIAL EQUATIONS
Differentsial'nye Uravneniya
Vol. 18, 1982 (12 issues) $395

DOKLADY BIOPHYSICS
Doklady Akademii Nauk SSSR
Vols. 262–267, 1982 (2 issues) $145

FLUID DYNAMICS
Izvestiya Akademii Nauk SSSR,
Mekhanika Zhidkosti i Gaza
Vol. 17, 1982 (6 issues) $380

FUNCTIONAL ANALYSIS AND ITS APPLICATIONS
Funktsional'nyi Analiz i Ego Prilozheniya
Vol. 16, 1982 (4 issues) $320

GLASS AND CERAMICS
Steklo i Keramika
Vol. 39, 1982 (6 issues) $460

HIGH TEMPERATURE
Teplofizika Vysokikh Temperatur
Vol. 20, 1982 (6 issues) $400

HYDROTECHNICAL CONSTRUCTION
Gidrotekhnicheskoe Stroitel'stvo
Vol. 16, 1982 (12 issues) $305

INDUSTRIAL LABORATORY
Zavodskaya Laboratoriya
Vol. 48, 1982 (12 issues) $400

INSTRUMENTS AND EXPERIMENTAL TECHNIQUES
Pribory i Tekhnika Éksperimenta
Vol. 25, 1982 (12 issues) $460

JOURNAL OF APPLIED MECHANICS AND TECHNICAL PHYSICS
Zhurnal Prikladnoi Mekhaniki i Tekhnicheskoi Fiziki
Vol. 23, 1982 (6 issues) $420

JOURNAL OF APPLIED SPECTROSCOPY
Zhurnal Prikladnoi Spektroskopii
Vols. 36–37 (12 issues) $420

JOURNAL OF ENGINEERING PHYSICS
Inzhenerno-fizicheskii Zhurnal
Vols. 42–43, 1982 (12 issues) $420

JOURNAL OF SOVIET LASER RESEARCH
A translation of articles based on the best Soviet research in the field of lasers
Vol. 3, 1982 (4 issues) $95

JOURNAL OF SOVIET MATHEMATICS
A translation of Itogi Nauki i Tekhniki and *Zapiski Nauchnykh Seminarov Leningradskogo Otdeleniya Matematicheskogo Instituta im. V. A. Steklova AN SSSR*
Vols. 18–20, 1982 (18 issues) $680

LITHOLOGY AND MINERAL RESOURCES
Litologiya i Poleznye Iskopaemye
Vol. 17, 1982 (6 issues) $420

LITHUANIAN MATHEMATICAL JOURNAL
Litovskii Matematicheskii Sbornik
Vol. 22, 1982 (4 issues) $205

MAGNETOHYDRODYNAMICS
Magnitnaya Gidrodinamika
Vol. 18, 1982 (4 issues) $325

MATHEMATICAL NOTES
Matematicheskie Zametki
Vols. 31–32, 1982 (12 issues) $400

continued on inside back cover