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Nuclear power in the Soviet Union and in Russia

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Abstract

This paper contains a brief description of nuclear power reactor designs, which gives a base planming of its development and provides the scientific and research basis of its further realization. The paper states the successive stages of these plans, their modifications and fulfilment. The article dwells upon the peculiarities of the: technical solutions for three main types of reactors and NPPs in more detail, which constitute the base for commerciial nuclear power in the Soviet Union and Russia: pressurized water vessel-type (VVER), channel graphite type wiith boiling water cooling (RBMK) and fast neutron type with sodium coolant (BN). Also stated is the development! of safety principles in Soviet nuclear power and the transformation of corresponding technical solutions in the plantcs' designs. The article describes specific traits of NPP equipment construction, which are different from Western analogues, and which resulted from the independent development of corresponding technologies and constructions. The airticle also discusses the main solutions in connection with the new generation of nuclear power plants, demonstrating the necessarily increased safety level, which are being constructed at present, as well as the scientific and techmical base which enables the realization of these solutions. The conclusion contains a discussion relating the place iof nuclear power in the Russian power industry, updated prognoses of its possible development, its long-term supply of nuclear fuel, and the economic competitiveness of nuclear energy sources compared with fossil fuel ones. © 199)7 Elsevier Science S.A.

1. The beginning and development of activities

Practical work on the creation of nuclear weapon and construction of the required nuclear industry in the Soviet Union were shortly followed by extensive studies into the possible applications of nuclear fission process for the production of useful energy (Fig. 1). The first officially documented Kurchatov instructions to study the possibility of power application for the water-cooled graphite reactor date back to 1946. By 1948 the studies had been completed and several versions had been discussed. These were:

- gas-cooled vessel-type reactor with a graphite moderator;
- high-temperature vessel-type reactcor with beryllium oxide;
- fast sodium-cooled reactor.

In 1949 Laboratory No. 2 (now the: Russian Research Centre Kurchatov Institute) issued a report on possible trends in the development of power reactors for transport and stationary power engineering.

This same year the work on designing a 'pilot nuclear power plant' began (the Government's Decree approving these activities was iissued on

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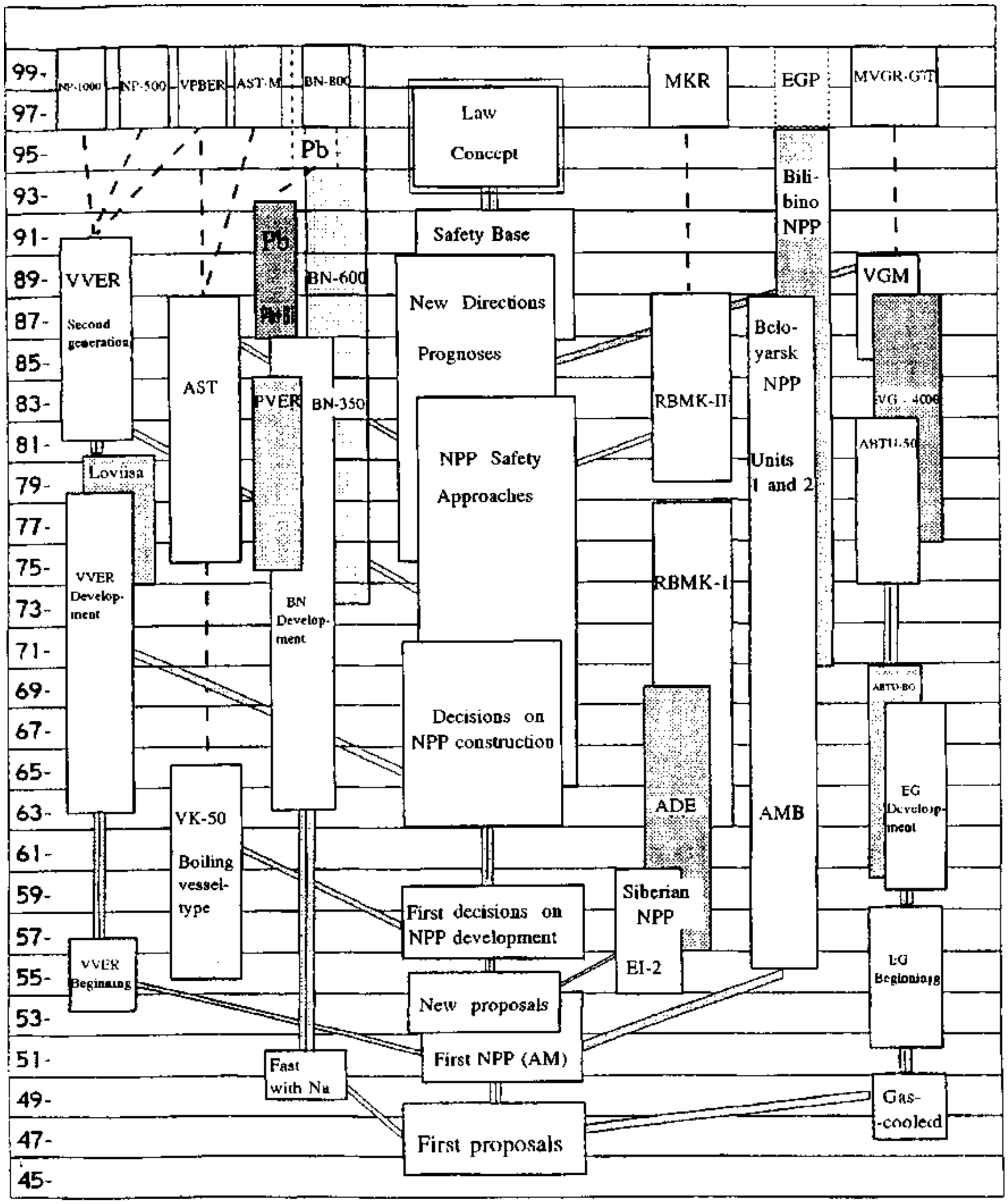


Fig. 1. Nuclear power developement in the Soviet Union and Russia.

May 16). In accordance with the Government's Decree of July 29, 1950, Laboratory B (now Institute of Physics & Power Engineering) began preparatory work on the constructiom of the first pilot NPP in Obninsk. Physical calculations were performed at the Laboratory of Measuring Instruments (the new name of the Kurchatov Institute).

In mid-1951 the project specification for the construction of the NPP had been approved and the main executors named. I.V. Kurchatov was appointed Research Supervisor, D.M. Blokhint-sev and A.K. Krasin as Deputy Supervisors, N.A. Doleazhal as Chief Reactor Designer. The State Design Institute (Moscow) was entrusted with working out the NPP design, and the OKB 'Gidropress' (Podolsk) was to design the steam generator.

The NPP was commissioned on June 27, 1954, and this was the birth of the national nuclear power industry.

Prior to giving a brief scheme of the 'nuclear power development tree[1] some estimates of the retrospective analysis results of this development should be given.

First of all the exceptionally purposeful work in the first phase of the project should be emphasized. This repeated the style of work which had been developed in the previous period of activity on nuclear weapons creation. The distinctive features of this style were rapid decision making, high development rate, acceptable depth of preliminary analyses, and methods for completion of adopted technical solutions in the course of the project realization. It should be pointed out that, unfortunately, transferring some elements of such a 'shock' style of working to the period of subsequent balanced development manifested themselves as chronic short-comings in the national nuclear power, which later required great expenditures of labor and resources to eliminate.

The first period of development is characterized by an extensive range of variant and reserve directions. As a result a rich and diverse scientific and technical potential for the nuclear industry was created.

The work was conducted under conditions of strict isolation, practically parallel with similar work carried out abroad, being subject to the system of internal political and economic factors, specific regulations for the development of nuclear technology and based on the national industrial potential.

This period constitutes about half of the 40-year history of the national nuclear power industry. The subsequent period was that of the increasing exchange with the outside world and implementation of common approaches; to the safety in this young technology.

In the development of this 'nuclear pcnwer tree' a single logical line can be traced from the beginning to the present day. But this sequeince was tragically broken by the Chernobyl accident which divided nuclear power into two ep»ochs for the whole world: the epoch 'before Chernobyl' and that 'after Chernobyl'. The present state is the result of the interaction of two processes: nuclear power development in its objective role in meeting power demands for mankind and! mitigation of the Chernobyl consequences and learning the lessons from this tragedy.

As early as 1953, a year before the stairt-up of the 'first world NPP', studies of various possible reactors for nuclear power plants were: carried out. In 1954 two dual-purpose reactor designs which could be used for simultaneous generation of electricity and production of weapoms-grade plutonium were intensely studies: the graphite-moderated water-cooled reactor with ziirconium or steel pipes which can be considered! as the prototype of the RBMK reactors and the pressurized vessel water reactor, the prototype of the VVERs.

Actually the development of dual-purpcose reactors proceeded along the path of the mo>st rapid realization of the power supply goals associated with the main problem—production of fissile materials, i.e. the way of creating the graphitic water-cooled reactors.

The first dual-purpose reactor, EI-2, was designed and constructed in the period 19!54-1958 and put into fuel power operation in 195>8-1960.

The decision on the construction of cormmercial dual-purpose reactors of the ADE-type was made between 1956 and 1957. Construction began in 1957. From 1964 to 1969 four reactors were brought to full power.

At about the same time the design of the gas-cooled graphite reactors began. The first (developments of 1947-1955 were geared to thse use of helium as the primary coolant. However tthe work was stopped after 1955 when the decision to develop the VVER was made. In 1957 the design work was restarted but geared to another coolant, carbon dioxide. The design work in this field had many directions. They were suspended several times then resumed and continued till 1968 when these reactors competed with the RBMK and VVER for use at the Kursk and Chernobyl NPPs.

Rather profound R&D work on the gas-cooled reactors was carried out until recently with the aims and directions and the main reactor plant features being substantially changed. From 1958 to 1982 the development of a complex electro-chemical plant with circulating spherical fuel elements had been conducted. It was suggested to use the fuel radiation for stimulation of chemical processes. Later other high-temperature gas-cooled reactors with a helium coolant were designed. During 1961 1970 various versions of fast gas-cooled reactors were intensely studied. By the end of the 1980s the most advanced had been the work on designing the high-temperature helium-cooled graphite reactor called VG-400. It was supposed to be used not only for electricity production but also for high-temperature technologies in the chemical and metallurgical industries. At present the aim of creating a high-temperature reactor with a gas or steam turbine still remains. The most advanced design is a modular low-capacity high-safety plant with a gas turbine.

The actual development of commercial nuclear power proceeded based on two types of thermal water-cooled reactors: the channel graphite-moderated and the vessel light-water-moderated reactors as well as on the fast sodium-cooled reactors.

Power units with graphite water-cooled reactors and improved designs of the core and fuel channels were introduced at the First (Obninsk) NPP. These were Units 1 and 2 of the Beloyarsk NPP, of 100 MW_e and 200 MW_C, respectively, with the AMB reactors, and four power units of the Bilib-ino NPP, each of 12 MW_e. In the Beloyarsk NPP (ABM) nuclear superheating of the turbine steam was practically accomplished, and in the Bilibino NPP the scheme for natural circulation of the reactor coolant through the reactor channels was realized. The Beloyarsk NPP 1 and 2 were in operation from 1964 to 1983 and from 1967 to 1990, respectively. They were decommissioned after the end of their design lifetime. The Bilibino NPP power units have been operating for the power network since 1974-1976.

2. Plans for nuclear power development

During the period of creating thee semi-industrial reactors at the first power units <of the Novo-Voronezh (using the VVER neactor) and Beloyarsk (using the AMB reactor) NPPs the first stages of governmental policy in the field of nuclear power were formed, when variious versions of development and various types of reactor plants were considered.

The first program of nuclear power development was announced by Kurchatov for the period 1956-1960. It suggested an NPP installed capacity of 2-2.5 million kW to be reached. In 1958 the plan was revised and restricted to construction of only the first power units at the No>vo-Voronezh and Beloyarsk NPPs.

In 1962 the nuclear power development program was restored and, in addition no AMB and VVER, the gas-graphite reactor andl heavy-water reactor with an organic coolant wene introduced. Subsequently the heavy-water reactor was replaced by a water-water one and the: gas-graphite reactor by the fast neutron reactor BN. The period 1963-1964 is characterized by increased attention to fuel supply for the expected nuclear power and the role of the fast breedlers.

In 1966 the Government 1966-19'75 planes for construction of NPP with a total installed capacity of 11.9 million kW was adopted, which became the basis for development of the first generation of commercial power uniits.

Further plans for NPP construction and the corresponding Governmental Decrees were announced in 1971 and 1980. In 1971 the 26.8 million kW program for 1971-1980 nvas adopted. The 1980 Governmental enactment istipulated an increase of NPP installed capacity up to 66.9 million kW from 1981 to 1990 and to reach 100 million kW in 1993. Geared to this; level of nuclear power, work on the development of nuclear fuel cycle facilities and nuclear power machine-building were started.

During the process of work on these programs an understanding of the role of nuclear power in the power supply of the country was formed, its necessary structure and, what is most important, a modern understanding of the safety aims and requirements and the means of ensuring the safety of large-scale nuclear power. The approaches to NPP safety, which had seemed acceptable as applied to single facilities or small complexes, was insufficient when a new potentially hazardous power technology began to spread.

From 1963 to 1964 the first nuclear safety regulations for power reactors were issued. In 1968 the first edition of the safety rules for nuclear power plant design was approved.

The specific features of nuclear machine-building and the high level of requirements for reliability of reactor plants gave rise to a new class of quality fixed in strength analysis standards, welding instructions and product inspection rules. This was stated in special formative documents which were brought into practice in 1972-1974. In the early 1970s radical changes were introduced to the approaches to insure NPP safety, which will be discussed shortly.

A very important factor in the formation of new approaches and a new system of NPP safety insurance was the setting up an independent governmental body for NPP safety regulation in 1983, the State Nuclear Safety Regulation Committee (Gosatomnadzor). Unfortunately, these efforts were unable to stave off the accident at Unit 4 of the Chernobyl NPP. Discussion of its causes and lessons deserves a special paper.

The fact is great efforts put into ensuring high rates of civil nuclear power growth and expanding technical, organizational and administrative structures, left practically no time for the development of the safety culture so vitally needed and indispensable in this industry.

The positive factors of a semi-military organization had been lost, and new civil forms were developing very slowly. As a result the uncontrolled monopolism of technical solutions, manifesting itself in serious deficiencies in power unit designs, combined with operation faults reflecting the lack of a safety culture at the operations stage.

During the whole period of nuclear power development, preceding the Chernobyl accident, the positive role of centralized organizationial management structures was demonstrated. It enabled the experience gained and the research a.nd engineering potential provided at the requirred level for all activities to be accumulated and usted. With the economic and organizational forms traditional to our country each case of dissociation! of the partners efforts and uncertainty of responsibility gave negative results. The most tragic of tlhem was the Chernobyl accident.

It is interesting that western experts, based on their own analysis results, noted high operational reliability of the NPP with VVER-440 (imcluding those of the first generation) and came to the conclusion that this result had been reached due to the nuclear power management structure which reflects the centralized structure of all industrial management. This factor was strengthened by the high prestige of the nuclear industry and by the possibility to apply new technologies duiring development of these designs. It is now is very important to save these factors.

3. NPP with water-water reactors VVER1

The work over the design of the watcer-water power reactor (VVER) began in 1954-11955. By May of 1995 the Institute of Atomic Energy (Kur-chatov Institute) had prepared a series off performance specifications for designing power reactors with water and gas coolants.

The neutronic features of the water-moderated reactor permit high power to be obtained! from a small-sized core and high uranium burn-mps to be reached and thus, opened wide technical amd economic prospects for application of this reactor, and as a result pushed it into the foreground in the conceptual developments of reactors for nuclear power stations. The cost of this type reactors was most encouraging.

The following reactor types were considered: •
water-water reactor VES-1 (water power plant 1) with an aluminium core for low steam parameters (two-circuit design, saturated steam at a pressure of 3 atm);

- a similar reactor VES-2, with the aluminium replaced by zirconium as more suitable for generation of steam with higher parameters (two-circuit design, saturated steam at a pressure of 29 atm);
- water reactor EVG (power water gas reactor) using a gaseous coolant for steam superheating (superheated steam at a pressure of 29 atm at a temperature of 400°C);
- in addition, the Institute of Atomic Energy had performed sufficiently detailed analysis of the possibility of using a uranium-graphite gas-cooled reactor for power engineering purposes, therefore the nuclear power station with an EG reactor (power gas reactor) was also included in the series of competing reactor designs;
- a version of a cogeneration nuclear power plant with a water-water VES-2 type reactor for saturated steam generation and a graphite-gas EG-type reactor for superheating this steam was studied separately.

Among all reactor versions for the 'first power station' preference was given to the VES-2 reactor with turbines using saturated non-radioactive steam at medium pressure. The OKB 'Gidropress' was entrusted with the development of this (as well as with all subsequent projects).

The detailed design of a 200 MW_e reactor was completed in 1956. In the same year the inter-government agreement was signed by the USSR and GDR, and in 1957 the work on erection of the Rheinsberg NPP with a capacity of 70 MW began. Various sits for the first NPP with VVER were considered, including a MOSENERGO cogeneration plant (TEC-21) in Khovrino, near Moscow, and finally the Novo-Voronezh site was chosen. There the first VVERs of all were constructed.

- Unit-1 of NVNPP (WER-1) was connected to the grid in 1964 and finished operation in 1984.
- Unit-2 of 365 MW_e worked from 1969 to 1990.
- Rheinsberg NPP was commissioned in 1966.

Construction of the first NVNPP units confirmed the technical feasibility of reliable commercial nuclear power sources. The experience of their creation and operation were of exceptional importance for further development of NPPs with VVER in our country and other countries where our designs were used.

It should be pointed out that a niumber of the basic engineering solutions worked oiut for the first VVER were original and became traditional for all subsequent VVER generations.

Among these are:

- a triangle array for the fuel assemblies in the core and the fuel elements in tthe assembly, which gives a hexagonal form to tlhe fuel assembly;
- the use of Zr-Nb alloy as a cladding material;
- the use of high-strength alloyed carbon steel suitable for performance at high meutron fluxes as the material for the reactor veissel;
- the reactor vessel is made of solidl-forged shells without longitudinal welds;
- the lower part of the reactor vessel accommodating the core is designed as a cylindrical vessel with the elliptic bottom without pipe penetrations or any other holes;
- the reactor vessel rests on a cylimdrical clamp on the lower shell of the branch pipe zone;
- control and protection systems idrive mechanisms, core systems for temperaturre control and power generation control are installed on the removable closure (head) of the reactor vessel;
- in the first units and VVER-440 the movable fuel assemblies are used as control rods;
- the original design of the horizontal steam generator with a tube plate consisting of two cylindrical headers;
- the SG heat-exchange tubes aire made of OX18H10T austenitic chromium-nickel stainless steel;
- an important factor was assurance: of rail transportability of the large sized equipment. The first generation commerciail 440 MW_e VVERs were constructed based on tlhe experience with the erection of the first two unitis of NVNPP. The premier unit of this series (NWNPP-3) was commissioned in 1971. From 1971 to 1975 six such units were put into operation in the iSoviet Union (at Novo-Voronezh, Kola and Arrmenian NPPs) and ten more units were put into serviice from 1974 to 1982 in Bulgaria, Czechoslovakiia and GDR under the intergovernmental agreements. This series of VVERs had demonstrated tthe economic competitiveness of the nuclear powetr plants.

The decisive role in forming new approaches to assurance of NPP safety was involved in the work on the VVER-440 NPP design for Finland, which began in 1969. Close contacts and detailed study of the experience of other countries developing nuclear power speeded up the formulation of new NPP safety requirements to meet international standards.

In 1969 the work on the 'General Regulations for NPP Safety' began. In 1971 the first edition was approved and in the same year the development of the design of an NPP with the second generation VVER, meeting the international approach to safety assurance, was started.

The first units of this series were erected at the Loviisa NPP in Finland (the start-ups of Unit 1 and Unit 2 were accomplished in 1977 and 1980, respectively). In the plans of NPP construction in the Soviet Union and in the CMEA countries the units of the first generation built were replaced by units of the new series. There were four in the USSR and ten in the CMEA countries.

In 1969 the design of the reactor VVER-1000 (design V-187) for the premier Unit 5 of NVNPP, taking into account the new safety requirements, began. In 1971 the detailed design was recommended for implementation. In the reactor design V-187 the containment made of prestressed reinforced concrete was adopted for the first time in our country, which was designed for full pressure occurring in the design basis accident with loss of the coolant from the 850 mm main circulation pipeline.

In designing the first VVER 'of million capacity' the large size of the reactor vessel, steam generator and main circulation pipes made it necessary to replace the materials used for VVER-440 by stronger ones which would allow the large sized equipment suitable for rail transportation to be developed, and the wall thickness to be reduced, which would facilitate welding operation at factories and sites. The work on choosing and testing new materials, carried out by CNIIKM 'Prometei' CNITMASH, Izhora Works, OKB 'Gidropress', Kurchatov IAE, had been continued for several years. The designs of the main equipment had to be corrected, and the terms of construction and commissioning changed (by about three years). Unit 5 of NVNPP with VV'ER-1000 was put into operation in 1980.

Power unit capacity was increased duie to the enhanced thermal power of the reactor amd higher efficiency of the steam power cycle provided by the increased steam pressure. The turbogenerator efficiency in the VVER-1000 was 33% comparing with 27.6% in the first generation WER (the steam pressure in the steam generator was 64 and 32 atm, respectively).

The reactor thermal power was increaised, with the limiting size of the reactor vessel remaining unchanged (in accordance with rail transport conditions) and the effective core diameter practically unaffected. This was achieved by:

- a decrease in the peaking factor;
- an increased coolant flow rate through the core;
- an increase in the total length and smrface of the fuel elements;
- a reduction in the working parameter: margins to the limiting permissible values.

In the transition from the first reactor to the VVER-440 the total length of the fuel irods was increased at the expense of reducing theiir diameter from 10.2 to 9.1 mm, while passinig to the VVER-1000 the total uranium charge amd, hence, an increase in the core height. The thermial power of the VVER-1000 is 3000 MW compared to 760 MW for the first reactor and 1375 MW for the VVER-440.

The economic indices were improved at the expense of the increased unit power of the unit and higher nuclear fuel burn-up. All VVERs operate with partial refueling during the annmal shutdown for maintenance and repair of the equipment.

The technical solutions for the VVER-1000 equipment design and reactor plant pro>cess systems were new in many aspects. For example, the core with 'soft' control rods in the forrm of absorber clusters (12 rods in each cluster) was used in the VVERs.

The VVER-1000 reactor plant for Umits 1 (V-302) and 2 of the South-Ukrainian MPP and Units 1 and 2 (V-338) of the Kalinin MPP were developed with the main design solutions of reactor V-187 remaining unchanged. In the modern-

Table 1
VVER vessel dimensions

	VVER-210	VVER-440	VVER- 1(000
Reactor vessel pressure (MPa)	10	12.5	15.7
Maximum outer diameter of the vessel (m)	4.4	4.27	4.535
Vessel height (m)	11.35	11.80	10.85
Thickness of vessel (mm):			
Cylindrical part	100	140	190
Branch pipe area	180	200	290
Flange area	380	500	460

ized reactor the number of SUZ drives was reduced from 109 to 49 in the SUNPP-1, and to 61 in the rest of the units of this group and, unlike NVNPP-5, the 'bare' fuel assemblies were used. These units were put into operation in 1982, 1985 and in 1984, 1984, 1986, respectively.

In 1978 the development of the VVER-1000 (V-320) for a large number of NPPs began. All new RU V-320 designs, with the main pressure and temperature unchanged, had to be optimized based on the experience gained in the development of the V-187, V-302 and V-338. During the period 1984-1993 fourteen such units were put into operation (two of them in Bulgaria in 1987 and 1991).

3.1. The main VVER equipment

A determining element of the VVER reactor installation is the reactor vessel. The possible maximum pressure of the reactor coolant is primarily determined by the vessel wall thickness which can be provided by the manufacturer and, in turn, must be matched with the vessel size (diameter). The reactor vessel size is chosen depending on the desired reactor power and attainable core power density (heat release per unit volume).

From the first power units on, the reactor vessel size has been chosen in accordance with the most important condition: the maximum permissible oversize degree making it transportable by railroads from the manufacturer to the NPP.

For the construction of the reactor vessels the problem is designing a special steel possessing the required strength and process properties and at the same time resistant enough to the neutron radiation of the core. For neutrom shielding a sufficiently thick layer of water and steel can be provided between the core and re:actor vessel. However, this increases the non-productive part of the vessel. It was important to fintd an optimal solution to the steel problem by imcreasing the radiation resistance of the vessel steell which could last for a reasonable economical life ([20-40 years) without unacceptable change in its sitrength characteristics. For the VVER-440 reacttors the high radiation resistant chromium -rmolybdenum-vanadium perlitic steel, 15X2M$A, was chosen, and for the VVER-1000 reaictors—steels 15X2HM<DA and 15X2HMOAA.

The adopted vessel dimensions, talking into account different working pressures im the primary circuits, are presented in Table 1.

The technology of vessel manufacturing was first implemented at the Izhora Woirks and later introduced at 'Atommash'.

The most stringent requirements; are placed upon the durability of the reactor vessel's endurance as its destruction might e:ntail serious consequences. The reactor vessel is raited in Safety Class 1 by the fact of its influence on the NPP safety and enters Group A equipment by the requirements of design, manufacture,, installation, operation and repair in accordance with the special 'Regulations on Arrangement anid Safe Operation of Nuclear Power Plant Equipment and Pipes'. The reactor vessel is the onl\y part of the VVER reactor, which is placed im this group. Group A RBMK equipment corrnponents are drum separators and fuel channels.

At this point it should be pointed out that the level of quality and reliability of the safety-related nuclear power plant equipment is qualitatively higher than the level of requirements for traditional power equipment. Therefore, the machine-building industry involved in nuclear power development had to master this increased quality level, stimulating general progress in power machine-building.

3.2. Nuclear fuel

Nuclear fuel (fuel rods) is the second determining element of the VVER reactor plant. The chosen fuel is in the form of cylindrical rods 9-10 mm in diameter, the fuel composition represents sintered uranium dioxide whose kernel is set into a sealed isolating cladding made of zirconium-niobium alloy. The material chosen (uranium dioxide and zirconium alloy) determined a number of the most important characteristics of the reactor and the NPP. Zirconium, providing economic use of neutrons in the chain reaction, requires limited working temperatures for the coolant: outlet temperatures from 273°C (VVER-210) to 322°C (VVER-1000) at the appropriate primary circuit pressures.

The maximum coolant temperature confines the possible steam pressure before the turbine from 2.9 MPa (VVER-210) to 6.0 MPa (VVER-1000).

The thermal properties of uranium dioxide as a heat generating composition (heat conductivity, melting temperature) determine the requirements of the characteristics of the primary circulation pumps, design principles and reliability of the station service power supply systems, etc.

Progress in nuclear fuel design permitted fuel utilization and nuclear cycle economy to be significantly improved in the transition from the first VVER generation to next. The main direction in this progress is the increase in the burn-up of fuel elements charged into the reactor and the rise in the specific power density.

Among the new main elements of VVER reactor plant power equipment the main circulation pumps and steam generators should be considered.

3.3. Main circulation pumps (MCP)

In the first years of commercial nuclear power development the MCP problem had be«en solved by constructing leak-tight pumps with the rotor shaft running in a sealed cavity filled w/ith water and separated from the stator with a thiin (about 0.5 mm) Ni-Cr alloy partition welde«d to the massive end parts of the stator. This system was designed to prevent release of radioactiive water circulating in the primary circuit.

The first Soviet NPPs at Beloyarsk (two units), Novo-Voronezh (four units), Kola, Rovno and Armenian were equipped with leak-tigrht pumps developed in Leningrad. Till 1978 NPPs; with the VVER-440 in the SSSR and CAMECON countries operated with the GCEN-310, lleak-tight pumps with a capacity of 6500 m^3 h ~ ' . These pumps, designed based on experience wiith developments for nuclear submarines, were fomnd to be highly reliable and simple for maintenamce at the NPP.

However the leak-tight MCP design proved to be unpromising for high-capacity NPPs; for several reasons. First, they have large electromagnetic losses on the sealing metallic partitiion of the motor. There up to 17-20% of the powrer supply is lost, and thus the efficiency of the motor drive is drastically reduced (56%), and its coioling becomes more complicated. For example, tlhe electrical power consumed by GCEN-3KO, when working with hot water, is 1650 kW, anid, hence, six MCP consume about 10 MW. Talking into account that the power unit capacity (gro)ss) is 440 MW, these are great power losses.

The design calculations showed that to create a reliable and effective design of the leak-tijght partition for MCP with a drive capacity gre;ater than 2000 kW is not practically possible. AJso, it is very difficult to increase the necessarily sluggish 'running-down' by the use of a big rotatiing mass, e.g. a flywheel on the rotor shaft.

This feature of the leak-tight pumps made the problem of reactor emergency cool-down very complicated at sudden power loss, especially within the first tens of seconds when the decay heat release from the uranium dioxide futel rods in the reactor is greatest, prior to the emergency power sources for the pumps (e.g. diesel generators) switching in.

These pumps also have some other disadvantages. Therefore in the beginning of the 1970s, in designing NPPs with the VVER-1000 and RBMK-1000, domestic power MCP with mechanically sealed shafts and minor controlled leaks provided with automatic systems of reliable locking in the radioactive water were developed. The motors for these pumps are provided with flywheels as well as with devices for electromagnetic relief of high axial forces to the thrust bearings. The MCP are designed for seismic stability at earthquakes up to force 9.

Since 1978 NPPs with the VVER-440 have been equipped with GCEN-317 pumps with mechanical shaft sealing and flywheels, instead of the leak-tight pumps. The most powerful of them in capacity (20000 m^3 h ~ ') and power consumption, designed for NPPs with the VVER-1000, is the GCEN-195 M pump. This is a vertical device 9.3 m in height and with a mass, including all auxiliary systems, of 128 t. The nominal capacity of the three-phase asynchronous motor is 6300 kW. The operating range of the pump's capacity is from 17000 to 27000 m^3 h ' depending on the total resistance of the hydraulic circuit.

The main series of MCP for the VVER were constructed at Leningrad Kirov Works.

3.4. Steam generators (SG)

When choosing an optimal thermal-power cycle for a two-circuit nuclear power plant with a water-water reactor, the temperature of the water at the reactor outlet is a determining limiting parameter. In this case the superheated steam cycles usually used in fossil-fired power are not economical, whereas a saturated steam cycle proves to be thermodynamically most efficient. The maximum possible pressure of steam required to increased efficiency is achieved through optimization of the design of the steam generators (SGs). Thus, saturated stearn SGs are used for VVERs. Among the basic design features of domestic SGs in all generations of NPPs are a horizontal shell, a bundle of heat tubes submerged below water level, vertical cylindrical tube headers, natural circulation of boiler water, and separation of steam in the drum. Variatiions of SGs were repeatedly developed to ensure ai more compact layout of equipment in the NPP reactor compartment, but up to the present tlhey have not been implemented in practice. The irmportant advantage of the horizontal SGs in operation now is a great reserve of boiler water withiin the shell, providing good lag characteristics for the entire installation under transient and acciident conditions and thereby enhancing its safet;y.

Steel OX18H10T was chosen for tHie tubes and headers of all the SGs m the first 'VVERs and VVER-440s as well as for the in-ve:ssel components and the primary-circuit pipelines and equipment. This choice has proved to be am appropriate solution.

Development of the VVER-1000 cailled for materials of higher strength for the equijpment in the primary-circuit loops. The choice ffell on steel 10rH2MO>A that has been used for S*G shells and headers, pressurizers, vessels of the emergency cooling system and piping, but the ducctility of this steel decreases when the 'cold' header operates under the complex actions of fabricaition-induced and operating stresses, actual water cHiemistry and slow deformation. For this reason, cracks were found in the tube connections betweeen the flared tubes in the coldest part of the headters. To remedy this condition fabrication stresses were reduced, water chemistry was improved and heat treatment was used. Moreover, it wais decided to make the flared parts of the headers in newly-delivered SGs from the steel OX18H10T which worked successfully in the SGs of VWER-440s.

4. RBMK-based NPPs

Channel-type graphite-moderated llight boiling water-cooled thermal reactors represeinted the second line of power reactor building im the Soviet Union. The key characteristics of the reactor core are uranium dioxide as a fuel, zirconium—niobium alloys as structural materials ifor fuel element cladding and pressure tubes, and continuous on-load refuelling.

NPPs with channel-type water-graphite reactors RBMK is the national feature of domestic nuclear power. The idea of RBMK arose in the mid 1960s with an eye to expanding the nuclear power industry without the manufacture of unique vessels and SGs. The basic characteristics of nuclear power installations were chosen in such a way as to make the most use of the experience gained in development and construction of productive (plutonium) reactors as well as the potential of machine building and construction industries. The single-circuit scheme and the boiling coolant allowed the use of proven thermome-chanical equipment with rather moderate thermophysical parameters. On the other hand, the channel-type design made it possible to increase rather simply the unit power, especially as the use of containment for safety purposes was considered to be unnecessary. The possibility of attaining higher burn-ups of low-enriched uranium with continuous on-load refuelling seemed at the time to be an advantage of the water-graphite reactor.

By the time of the Chernobyl accident, in less than 13 years, the former USSR had commis-sioned 14 units with RBMK-1000s (ten in Russia and four in the Ukraine) and one unit with the RBMK-1500 at the Ignaliana NPP, Lithuania. Their total capacity amounted to 50% of all Soviet NPPs, and in 1985 they produced 60% of the country's nuclear electricity output.

Among the units commissioned in the post-Chernobyl period are an RBMK-1500 unit at Ignalina and an RBMK-1000 unit at the Smolensk NPP. The Kursk-5 unit with a modernized RBMK-1000 reactor is in the final stage of con-struction at the Kursk NPP.

The RBMK design that has been described in sufficient detail has some unique features, such as: channel tubes, diameter 88 x 4 mm in the core, made of Zr-2.5% Nb alloy (cladding made of Zr-1% Nb alloy as in the VVERs); diffusion joints in the adapters between the zirconium and austenite parts of the channels; heat released by moderation and absorption of *y*-quanta and neutrons in the moderator (5.4% of reactor power) is removed in the channels through special graphite sprung split rings placed between the channels and the graphite columns, providing ai thermal contact and accommodating contrary rradial de-formation due to creep of the channel tube and the radiation shrinkage of the graphite column blocks; the spaces in the graphite stack oiutside the channels are filled with a slowly-circulatiing nitrogen-helium mixture which is used also Uo control the integrity of the channels by changtes in gas humidity and temperature (a KTsTK s;.ystem); a long fuel bundle consisting of two 3.5 m long fuel assemblies, each containing 18 fuel elemients 13.6 mm in diameter and suspended by a rod in a central tube on a plug sealing the chanmel head; mass channel-by-channel control of parameters (flow rates by a SHTORM system and activities by a leak-proof control system KGO) iss realized unconventionally—on the basis of baill-in-tube flowmeters SHTORM and movable radlioactivity detectors that are moved across the outlet steam-water pipelines from channel to channel!, that is, without using pulsed lines or other aidditional pressurized active coolant lines.

The on-load refuelling system using a irefuelling machine is also unique. The machine is automati-cally placed position over a desired chanmel, links up with and seals its head, equalizes the pressure, unseals the channel plug, feeds a smalll flow of pure water into the channel, withdraws tlhe burnt-up assembly up into the machine's drum,, inserts a fresh assembly , seals the channel, detaiches and ships the withdrawn assembly to a cooliing pond. The sequence and quality of execution of every operation is automatically controlled; im case of any deviation or failure the operatiions are stopped.

RBMK modifications (modes) differ rmainly in the degree of compliance of their emergeincy cooling systems, in particular accident localization systems, with standards and regulations which were in force in the final period of desigm activity.

Besides, the sixth mode has a higher capacity (RBMK-1500) and the seventh mode (thie Kursk-5 reactor now under construction) diffeirs in the physics of the reactor, because it has a nonposi-tive steam effect of reactivity. • The first, initial mode was created in 11968 with

further updating until the commissioming of a premier unit (Leningrad-1) in 1973 and back fitting based on the experience of its start-up and bringing to design power.

These and other essential alterations introduced later were taken into account in the succeeding units, first of all, in the Leningrad-2 unit commissioned in 1976; the Leningrad-1 and -2 units are almost identical and represent the first RBMK mode.

- The second mode includes units 1 and 2 at the Kursk and Chernobyl NPPs (commissioned in the period 1975-1979) and is characterized mainly by layout and construction changes to facilitate the unit mounting; the reactor installations of units 1 and 2 are rotated in the plane by an angle of 90° and brought closer together. The first and second modifications comprise the first generation of RBMKs. Construction of these units was initiated before the issue of General NPP Safety Regulations that require us to take into consideration any break in any pipe in the reactor cooling circuit. For this reason, the first and second modifications of the RBMK units had no adequate safety systems, such as an emergency core cooling system (ECCS) and an accident localization system.
- The third mode is represented by units 3 and 4 at the Leningrad NPP (commissioned 1979-1981). Unlike Lcningrad-1 and -2, provision was made for an accident localization system comprising:
  - o reinforced leak-proof compartments for the multiple forced circulation (MFC) loop (in addition to the central hall, above the reactor, communicating with the drum-separator compartments) and
  - <; an accident localization tower with bubble plates for condensation of steam and confinement of soluble volatile radioactive fission products (FPs) escaping from any break in the MFC loop; the tower adjoins the top end of the central hall;
- an ECCS was provided to feed water from cylinders by the pressure of nitrogen (an ECCS of short action) and from the accident confinement tower by the ECCS pumps (an ECCS of prolonged action) downstream of the check valves installed at the inlet of the distributing group headers (DGHs);
- the DGHs were raised up to the elevation of the core top end to prevent its dry out in the case of any leak in the common part of the MFC loop. (In all other units the IDGHs are at the elevation of the core bottom.) Besides, the power delivery schemie was somewhat changed, the emergency power supply system was improved (the numberr of diesel generators was increased), etc.
- The fourth, most numerous, mode of RBMKs includes six units commissioned ait the Kursk, Chernobyl and Smolensk NPPs im the period from 1978 to 1983.

These units, like the units of the thiird mode, are equipped with the ECCS feeding water downstream of the check valves of the IDGHs. This system has a reinforced leak-proof compartment and a pressure suppression pool unde:r the reactor to condense steam and confine volatile FPs, which somewhat reduces the volume and (cost of construction work as compared with 'the accident localization tower in the units of the: third mode. The process is facilitated and the airea of leaks from the leak-proof compartment is reduced down to a few hundred cm^2.

The single ND 600 steam pipeline was laid from each separator-drum instead of two ND 400 steam pipelines, and instead of the stteam ring all the steam pipelines were joined throiugh the tube connectors and the casings of the emergency regulating (SRK) and fast-acting reduction (BRU-K) valves of the turbines. There are als(o other non-conceptual alterations.

- The fifth mode is thus far represented by the Smolensk-3 unit alone. It has irmproved upgraded leak-proof compartments ((with a total effective leakage area of less thian 10 cm^2) because of the use of the improved mounting (but not construction) technology facilitating leak-proof maintenance and contnol. This suggests a considerable decay of sorme isotopes in volatile FPs during their residence; in the leak-proof compartment after a break in the DGH loop.

The design of the unit's pressure: suppression pool was simplified: it is single storey,, whereas the fourth mode units have two storey pools.

• The sixth mode is much different from the other modes and includes two units of the Ignalina NPP with an uprated reactor RBMK-1500 having a designed thermal capacity of 4800 MW. To increase the safety and service margins these units operate at a power of 4200 MWth (1250 MW$_C$).

The reactor, the MFC loop and its equipment as well as the entire reactor installation are essentially similar to those in the RBMK-1000. The upration of the reactor is ensured by heat removal intensifiers mounted on the fuel elements of the upper fuel assembly, by the proportional increase in the flow sections of the steam pipelines and the condensate feed circuit and in the numbers of diesel generators, feed and condensate pumps and centrifugal pumps in the shore pumping station, by new 750 MW turbo-generator sets AK-750/65-3000 and by correspondingly enhanced systems of power delivery and auxiliary power supply. The control, protection and safety systems are modernized. In particular, use is made of a 12-zone LAR-LAZ system—local automatic regulation and local emergency protection (instead of the 7-zone system used in the first generation RBMK-1000s and replaced in some units by the 12-zone one), the automated process control system is essentially changed, etc.

• The seventh mode will be presented by the Kursk-5 unit under construction now. The moderator volume is decreased by the use of octahedral graphite columns (instead of tetra-hedral ones), which will allow the void effect of reactivity to be made slightly negative. Some other changes not feasible in the units in operation now are also being introduced, in particular, a new system for release of the steam-gas mixture from the reactor space to protect it from excess pressure in a hypothetical destruction of a big group of channels.

At present, ten years after the Chernobyl accident, it is clear for unprejudiced specialists, including foreign ones, that the causes of the accident was not inherent in the uranium graphite reactors (UGRs). Therefore, they can be radically eliminated in new UGRs and with moderate economic losses in operating ones.

The thorough analysis of the frannework by various international experts suggest th;at the operation of the operating units can be continued until the end of their design life and that it is expedient (and sometimes essential) to moderately refurbish and update the operating units to match them and their operation procedures an<d training systems to present-day safety requirements.

Also under discussion is additional modernization that would allow the operation of the reactors beyond their design life until ne\w UGRs, complying fully with all present-day saifety standards, are ready to replace the existing ones.

Among the variants of such UGRs a multi-loop power reactor MKER is at the most advanced stage of development. It uses the eexperience gained with RBMKs, retains their possitive features, eliminates the negative ones and aibsorbs all the favourable potential features of inherent UGR safety.

The more radical variants of UGRs have also been developed, such as a UGR with aultonomous fuel channel modules directly generatimg motive steam. These reactors are also based rmainly on the RBMK experience and show considerable promise.

5. Fast reactors-BNs

Work on fast reactors were launched in the USSR in 1949. From the very beginning this work was focused on the use of sodium as a (coolant to extract heat from the core. As is knowm, water is unsuitable for use as a coolant in faslt reactors because of its nuclear-physical properties. Sodium possesses good thermophysical and suitable nuclear-physical characteristics. Moreover, the use of sodium provides a great gain iin thermal efficiency as compared with water-cooledi reactors. For some time the suitability of a sodiuim-potassium eutectic was also under consideratiion. However, its thermophysical characteristics ;are worse and, as experience showed, there are miore problems in handling it as compared with sodium. As a result, the development of a radically mew industrial technology using sodium as a cocolant was initiated. This task was quickly solved. In any case, it was shown that there exists no problems of a fundamental nature to the development of large-scale sodium technology.

In the early 1960s, when the development of the first commercial fast reactor BN-350 was initiated, an experimental sodium-cooled fast reactor BR-5 operated in Obninsk. Its thermal capacity was as low as 5 MW, whereas the BN-350 design thermal capacity was 1000 MW. It should be noted that the BN-350 reactor complied with the general requirements of scientific and technological progress and the demands for the intensive development of nuclear power. The BN-350 parameters were moderate. The coolant temperature as well as the temperature and pressure of the working medium (steam) were not high. This was a deliberate choice, because one of the main tasks of the reactor was to determine the effect of scaling on the operating characteristics of such systems without causing any additional problems.

Development work on a pilot reactor BOR-60 with high operating parameters and a rather low thermal power (60 MW) was started almost at the same time. For this purpose it was necessary to investigate the influence of high heat densities of fuel, high coolant temperatures and other factors on the operational characteristics of the reactor components, to examine different designs for fuel elements, to determine reasonable margins for the process parameters and to verify the engineering solutions used in the process equipment (SGs, pumps, etc.). A 12 MW_e nuclear power plant with SGs and a turbine designed to be operated at high steam parameters (9.0 MPa, 430°C) was developed and constructed at the Research Institute of Nuclear Reactors (NIIAR). The pilot NPP with the BOR-60 was commissioned in 1969.

Development of a BN-600 reactor was initiated a little earlier. The BN-600 has a higher heat density of fuel and higher coolant temperatures; the power of a unit with this type of reactor is sufficiently high to approach the basic indices for future commercial fast reactors. The BN-600 unit has an integral layout, as opposed to the loop layout of a BN-350 reactor.

The BN-350 (loop layout) began power operation in June 1973 and the BN-600 (integral layout) in April 1980. Great experience has been gained in operating these two nuclear power installations that differ considerably frrom one another in both design and the:rmophysical characteristics.

The BN-800 design represents a transition-type reactor on the way to mass construction of fast reactors. This design uses the scientiffic and engineering ideas and design features of its forerunner—the BN-600 reactor. Thte reduced thicknesses of the radial shield and trhe side blanket allowed the BN-800 core to be installed without increasing the tank diameter.

6. The new generation of nuclear planits

The nuclear plants of up-to-date design which are in operation and being built now possess an acceptable safety level. Their design amd operating procedures can be and are being updaited to bring them into compliance with the safety goals formulated by the specialists.

At the same time society, if it agnees with the option of a nuclear energy source, expects further results from the enhancement of its safety. The industry's practical actions, appropriate to these expectations, are governed by the conjuncture being formed in the development of nuclear power and concrete practical demands arising in any area of energy consumption in different countries.

Development of a new generatiom of NPPs which would ensure a qualitative step forward in the expected level of their safety is liniked in some countries to a pause in the current development of nuclear power and an expected new period of large-scale construction of nuclear pllants. In the established conjuncture under active development and discussion are the designs of iniherent- and passive-safety plants, whereas some years ago, in the first post-Chernobyl years, these directions in NPP development efforts were guiardedly received.

The desired stability, in the public: perception, of existing nuclear power dictated a moderate nature for the improvements underr discussion which were to be limited mainly by a thorough implementation of the basic safety principles and to remain in the framework of the evolutionary upgrading of designs.

An up-to-date generation of nuclear plants conforming to worldwide accepted levels was created in Soviet nuclear power in the early 1970s when new NPP safety standards were introduced. A short time later, in the middle 1970s, prerequisites arose for the next step in the development of new engineering concepts and the demonstration of a new level of safety for nuclear energy sources.

This step was associated with the use of nuclear energy in district heating systems, bearing in mind climatic conditions in most of the Russian regions, development of water heating systems and the fast deteriorating ecological situation in the large cities. The new class of nuclear installations located in the immediate vicinity of large cities and in densely populated areas was called upon to eliminate the necessity for special hard-to-realize measures for the protection of the population in accident situations.

Reactor installations developed for nuclear district heating plants (NDHPs) satisfied the above conditions. In 1978 additional standard requirements for the safety of these installations and their siting were also formulated.

In 1981 the construction of two nuclear plants of such a type with two 500 MWth installations AST-500 was started in Gorky and Voronezh.

The post-Chernobyl opposition by the population hindered the commissioning of the Gorky NDHP whose construction was completed. At the same time in these plants inherent and passive safety-related engineering features, which later gained general acceptance, were realized and particular attention was given to the confinement of radioactivity under normal and accident conditions: natural circulation during normal operation and in accidents, low power densities in the core, slow-proceeding normal and accident processes, great reserves of coolant in the primary circuit, passive emergency heat removal systems, the integral layout of the primary circuit and heat exchangers in a common vessel, the use of a second (guard) vessel designed to endure full pressure, etc.

All the design solutions were verified by calculations and experiments and approved by a nuclear safety supervision body. The international nuclear community had a chance of taking a closer look at the engineering features; of the AST-500 installation in the course of thie IAEA expert assessment of the Gorky NDHP im 1988.

It should be emphasized that, unlike work launched at the same time in Sweden, tlhe AST-500 design is based mostly on the inherrent and passive safety-related features absorbed omly from those concepts and design solutions whiich were proved in practice.

Thus, the experience in designing the AST-500 has given an advanced basis for the creattion of a new generation of enhanced-safety nucleair plants required for the new development stage of Russian nuclear power.

To overcome public acceptance problenns in the post-Chernobyl period it is necessary to demonstrate that new nuclear installations will I possess new safety-related features as well. For thiis reason the next generation of nuclear plants th;at could most likely lie on the evolutionary line miust have a maximum possible combination of new inherent and passive safety-related features. The priority NPP designs based on 640 MW_e and 10(00 MW_e water-water reactors offer the above features.

The needs of power supply development in Russia and the state of these designs leacd one to expect that a premier medium power unilt will be put into operation shortly after the year 2000.

The new generation of nuclear plamts with medium power units are being designedl in two modifications. One of them is the extension of the VVER-400 and VVER-1000 line. The ne\w power unit will have an installed electrical capacity of about 640 MW and is bound to achiieve the following objectives.

- a qualitatively new, higher safety level {provided by the use of passive safety systems ancd double containment with a simultaneous reduiction in the probability of accidents by 2 to 3 orders of magnitude;
- improvement of the specific techno-eeconomic indices by a factor of 1.5-2;
- 30-35% increase in fuel utilization;
- reduction of operational personnel by a factor of 2.5-3;

• reduction of the body of equipment and its metal content;
• extension of the plant's design life up to 50-60 years;
• diminishing the NPP impact on the environment.

The inner steel containment is intended to withstand a pressure of 0.5 MPa. The outer concrete containment is designed to endure external impacts including a blast wave and an aircraft crash.

The four-loop reactor installation has a thermal capacity of 1300 MW, an SG steam pressure of 7.0 MPa and a coolant pressure of 15.7 MPa at the reactor outlet. The power density in the core has been reduced down to 65.4 kW 1 ' at an average burn-up of 40.4 MWd kg~ ' U.

The design solutions are based mostly on the proven technologies and designs of the VVER-400 and VVER-1000 installations as well as involving the simplification of normal operational systems and the use of passive safety systems at the every stage of heat removal from the core to the atmosphere thereby reducing the possibility of human error. This project is being implemented now at an NPP near Sosnovy Bor (Leningrad Region) and on the second site of the Kola NPP.

An enhanced-safety nuclear power unit AES-92 with an electrical capacity of 1000 MW and qualitatively improved economic indices represents an extension of a Zaporozhye-type serial power unit equipped with a modernized VVER-1000 reactor installation.

The reactivity control system has 121 control mechanisms in the modernized reactor compared to 61 control mechanisms in the serial reactor. This ensures maintenance of the subcritical state with cooling down to a temperature of 100°C. In addition to the mechanical system of reactor control and protection a fast-acting boric water injection system is used to bring the reactor quickly into the subcritical state and to keep it in this state throughout the range of operational parameters.

Besides the active safety systems which perform the function of prolonged heat removal from the core, use is made of a passive heat removal system which represents a set of heat exchangers cooled by atmospheric air and connected to the SG steam spaces. Use is also made iof additional passive systems: second-stage hydraiulic reservoirs to supply low-pressure coolant to tlhe reactor installation in accidents with a leak im the primary circuit. The amount of water in thesie reservoirs is sufficient to supply the core with cocolant for 24 h after the onset of an accident in trhe absence of boric solution supply from the actrve emergency cooling systems with the low-presssure pumps. Moreover, the amount of coolant im the core can be maintained by the passive heat retmoval system which is capable of reducing the ptressure in the reactor installation for about an ruour after the onset of an accident below that in the containment interior.

In doing so, steam from the containment interior will arrive at the reactor installation and condense in the SGs, thus maintainimg the coolant reserve in the core. The structure of the safety systems in the primary and secomdary circuits allows most of the active safety systeims to be used for similar functions under normail operational conditions.

The containment is equipped willli a system of filters intended to vent and clean the medium from the containment interior in bieyond design accidents. The containment is double. Rarefaction is maintained between the inner amd outer containments; the air is pumped out amd cleaned by the filters.

The outer containment made of rmonolithic reinforced concrete protects the innerr one against external impacts (hurricanes, tornadlos, air shock waves, aircraft crash). The inner containment is made of prestressed reinforced comcrete with a steel sealing lining.

An NPP with such reactors is built on the second site of the Novovoronezh NPP.

7. Conclusions

In the coming period in many reguons of Russia nuclear power will reach the level of convincing economic expediency as compared with fossil-fuel power plants despite of greater costss than earlier plants. To develop nuclear power is economically more favourable by virtue of the folllowing:

Nuclear power is the only branch in the fuel and power engineering complex that will not need production of fuel in the near future (contrary to traditional fossil-fuel power whose most important problem is a fuel supply requiring great investment).

To implement the nuclear power programs in the considered range of capacities to be commissioned the consumption of natural uranium will have to be 8 to 15.4% of available uranium resources as from the beginning of 1993. Nuclear fuel supply problems may arise by the year 2030, providing nuclear generating capacities will be increased (to produce 30% of Russia's total electricity output by the year 2030). These problems could be solved at the cost of survey and development of new deposits in the territory of Russia, closing the nuclear fuel cycle, utilization of power-grade and weapons-grade plutonium and uranium stockpiles, and development of nuclear power based on alternative nuclear fuels. One ton of weapons-grade plutonium burnt in thermal reactors in an open fuel cycle has a calorific equivalent of 2.5 billion m^3 of gas. A rough estimation shows that, if the range of nuclear plants also includes fast reactors, the total energy potential of weapons-grade materials will correspond to an electricity capacity of 12-14 trillion kWh, that is 12-14 times as great as the annual output in 1993, and could save about 375 trillion $m^?$ of natural gas.

As already noted, the capacity for fuel manufacturing are sufficient to support the implementation of any program of nuclear development up to the year 2010 and to hold at the same time considerable production reserves.

The study of competitiveness of NPPs being designed for the energy market in Russia against fossil-fuel electricity production showed the following:

- if investments in NPPs exceed those in steam -gas TPPs by no more than 1.5 times and are greater by 15-20% than coal-fired TPPs, NPPs will be the most useful in all the Russian regions;
- to maintain the competitive position of NPPs against steam-gas TPPs, investment in the NPPs may exceed those in the TPPs by a factor of no more than 1.8 for the north westt and the centre of the country, 1.9 for the northern Caucasus, 1.7 for the middle Volga amd 1.6 for the Urals;
- to maintain the competitive position of NPPs against coal-fired TPPs, investments in the NPPs may exceed those in the TPPs *by/* a factor of no more than 1.08 for the Urals, 1.. 1 for the middle Volga, 2.1 for the north west: and the far east and 1.25 for the centre of the country. The comparison of new NPP and TPP designs shows that this condition for the cormpetitive-ness of NPPs is fulfilled in many regions of the european section and the far east. Relative changes have also occurred im the design economic indices of NPPs with theirmal and fast reactors. The construction costs for ffast reactors have become essentially little diffenent from those for thermal reactors after the hatter are equipped with the additionally safety systems. Similar systems have been already introdiuced for the most part in the designs of fast reactors. Moreover, the experience gained with fast reactor operation now allows some simplificatiion and, therefore, some reduction of cost.

In the coming years in Russia the nutelear fuel cycle will remain mostly open or partiality closed. The reprocessing of spent nuclear fuel Ifrom the VVER-440, BN-350 and BN-600 reacltors and transport plants at a RT fuel reprocessiing plant now in operation makes it possible to produce regenerated uranium with a desired uramium-235 content for the purpose of using it to fabricate fuel for RBMKs and fast reactors. Pllutonium being produced at the RT plant is stored and its stockpiles amount to about 30 t.

Spent VVER-1000 fuel assemblies arre stored thus far in a storage facility at an RT-2 plant under construction (at the Krasnoyarsk integrated mining and chemical works); their repirocessing will start only at the end of the next decade. Spent RBMK fuel is intended for long-term stiorage.

The utilization of the plutonium being accumulated, in particular bearing in mind growiing quantities of disarmament plutonium, is plamned for the beginning of the next decade, first of tall, in the BN-800 fast reactors whose construction is expected at the South Ural and Beloyarsik NPPs.

The construction of a complex for producing plutonium-based fuel should be completed at the 'Mayak' Production Association. The trial use of plutonium in VVERs during the next 5-7 years will set the stage for this option of plutonium utilization.

The problem of waste generation by nuclear power and its fuel cycle enterprises cannot be considered as a permanent obstacle to the development of nuclear power. The current acuteness of the radioactive waste problem results mainly from unresolved safety-related problems of military waste localization. These problems must be resolved by simultaneous development of a comprehensive work on management of radioactive waste from nuclear power and its fuel cycle facilities.

Besides, the development of nuclear power at a necessary or reasonable scale can be provided on the basis of developed machine-building production and construction industries.

The important role of nuclear energy in the total system of Russia's energy security must manifest itself in the following directions:

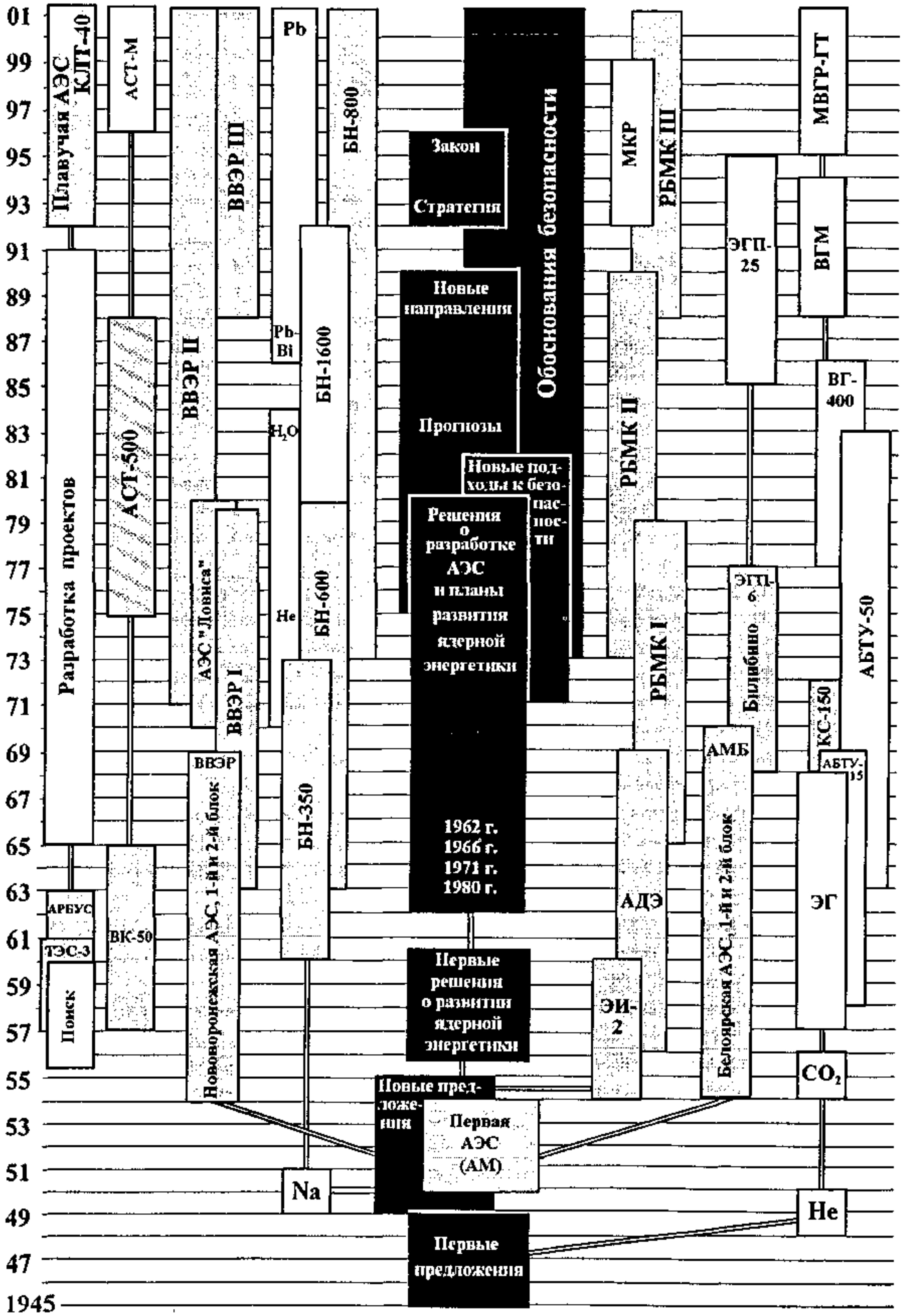
- increasing the reliability of electricity production at the expense of its diversification with savings in fossil fuel resources and simultaneously enlarging the required export of fuel. For example, with electricity output by the Russian NPPs in 1994 maintained at the level of 1993, about 22 billion kWh could be additionally generated and about 5.5 billion m^3 of natural gas (400 million dollars in current export prices) could be saved;
- creating real conditions for increasing the export of electricity by providing an excess of generating capacity in the near-border regions (investment projects based on compensation agreements could be realized);
- preparing a basis for the settlement of the ecological problems in Russia through reducing the technological impact of the fuel and power complex (today the enterprises of" this complex are responsible for about 48% of toxic releases into the atmosphere, up to 36% o>f sewage and above 30% solid waste). The policy of stabilization of 'greenhouse' gas releases imto the atmosphere is to be conducted according to international obligations and the "Energy Strategy '.

Estimations made by radioecologists and hy-gienists show that the complex impaict of nuclear power and its fuel cycle in the cases of normally operating NPPs is essentially less than the impact of the fuel cycle of thermal plants amd, primarily, coal-fired ones.

Nuclear plants allow the ecologicail situation in the regions where they are situate:d to be improved and the risk to human he;alth reduced, since any Chernobyl-type accidents are excluded by up-to-date safety engineering feattures.

Ecological expertise and the approval of local authorities are mandatory in the sitlting of NPPs to be constructed, and all the activities on the choice of an energy source, its desijgn, expertise, construction and operation must the open and publicized.

At present Russian nuclear specialists cooperate closely with the international commmnity of specialists in state institutions, companiies and international organizations. This international cooperation has become an integrall part of nuclear activities in Russia. Russian nmclear power recognizes the necessity for mutually advantageous cooperation with foreign partmers based on the high scientific and technologicall potential of the nuclear industry and its rich raw-material base. It is clear today that to keep abreast with the world community when solutions, developments and realities are mutually accessible and can be mutually validated is the tbest way for further progress. This gives additional guarantees of successful and the safe development of nuclear power in our country.



Развитие ядерной энергетики России