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Liquid Metal Coolants for Fast Reactors Cooled By Sodium, Lead, and Lead-Bismuth Eutectic

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LIQUID METAL COOLANTS FOR FAST REACTORS COOLED BY SODIUM, LEAD, AND LEAD-BISMUTH EUTECTIC

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FOREWORD

One of the IAEA's statutory objectives is to "seek to accelerate and enlarge the contribution of atomic energy to peace, health and prosperity throughout the world". One way this objective is achieved is through the publication of a range of technical series. Two of these are the IAEA Nuclear Energy Series and the IAEA Safety Standards Series.

According to Article III.A.6 of the IAEA Statute, the safety standards establish "standards of safety for protection of health and minimization of danger to life and property." The safety standards include the Safety Fundamentals, Safety Requirements and Safety Guides. These standards are written primarily in a regulatory style, and are binding on the IAEA for its own programmes. The principal users are the regulatory bodies in Member States and other national authorities.

The IAEA Nuclear Energy Series comprises reports designed to encourage and assist R&D on, and application of, nuclear energy for peaceful uses. This includes practical examples to be used by owners and operators of utilities in Member States, implementing organizations, academia, and government officials, among others. This information is presented in guides, reports on technology status and advances, and best practices for peaceful uses of nuclear energy based on inputs from international experts. The IAEA Nuclear Energy Series complements the IAEA Safety Standards Series.

The major challenges facing the long term development of nuclear energy as a part of the world's energy mix are: improving its economic competitiveness, meeting increasingly stringent safety requirements, adhering to the criteria of sustainable development and achieving a high level of public acceptability.

As far as sustainability aspects are concerned, fast spectrum nuclear reactors with multiple recycling of fuel allow the energy yield from natural uranium to be increased by a factor of 60–70 compared with thermal reactors. They also provide a significant improvement with respect to nuclear waste management. A necessary condition for successful near and mid-term deployment of fast reactors and the associated fuel cycles is the understanding and assessment of technological and design options, based on past knowledge and experience, and on research and technology development efforts. Achieving the full potential of the fast neutron system and closed fuel cycle technologies with regard to both efficient utilization of the fissile resources and waste management is conditional on continued advances in research and technology development to ensure improved economics and to maintain high safety levels with simplification of the system.

The IAEA is assisting Member States in the area of advanced fast reactor technology development by providing a means for information exchange and collaborative research and development (R&D) to pool resources and expertise. The IAEA's fast reactor technology development activities are pursued within the framework of the Technical Working Group on Fast Reactors (TWG-FR). Still the only global forum for information exchange and collaborative research and technology development projects in the area of fast neutron systems, the TWG-FR assists in the implementation of IAEA activities and, through continuous consultation with Member State representatives, ensures that all technical activities performed within the framework of the IAEA project on Technology Advances in Fast Reactors and Accelerator Driven Systems are in line with expressed Member State needs.

This publication provides a comprehensive summary of the status of liquid coolant technology development for fast reactors with regard to basic data, main technological challenges and the various fast reactor concepts and designs that are being investigated, with a special emphasis on the choice of coolant.

The IAEA appreciates the efforts of the many international experts who reviewed the draft version of this report and whose comments and suggestions helped to improve the final product. The IAEA officers responsible for this publication were A. Stanculescu and S. Monti from the Division of Nuclear Power.

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1. INTRODUCTION

The present report summarizes the state of the art in liquid metal technology development for fast reactors. This report takes mainly, but not solely, into account Russian experience, activities and references.

1.1. BACKGROUND

To advance nuclear energy to meet future energy needs, the Generation IV International Forum (GIF) [1] has defined a set of priority goals: improved safety and reliability, efficiency and economic competitiveness, sustainability — i.e. efficient use of the resources, nuclear waste minimization and protection of the environment — and enhanced proliferation resistance and physical protection. To reach these goals, six preferred reactor systems have been selected, namely the very high temperature reactor (VHTR), the gas cooled fast reactor (GFR), the sodium cooled fast reactor (SFR), the lead cooled fast reactor (LFR), the super critical water reactor (SCWR) and the molten salt reactor (MSR). As seen in this list, there are three (and potentially four) fast neutron spectrum reactor systems, cooled with gas, sodium or a heavy liquid metal (lead or lead-bismuth).

A further initiative is the International Project on Innovative Nuclear Reactors and Fuel Cycles (INPRO) of the IAEA [2]. The objectives of INPRO are related to the availability and sustainability of nuclear energy to meet the energy needs of the 21st century. Another objective is to ensure information exchange in the area of innovations in nuclear reactors and fuel cycles between technology holders and users.

Besides international initiatives such as GIF and INPRO, the implementation of fast neutron spectrum reactors in closed fuel cycle is considered in several national programmes, such as the Japanese Fast Reactor Cycle Technology Development (FaCT) project [3], the Russian Federation's Federal Target Programme (FTP): Nuclear power technologies of a new generation, the French SFR R&D programme including the realization of the ASTRID (Advanced Sodium Technological Reactor for Industrial Demonstration) prototype, etc. Another significant initiative in this field is the European Sustainable Nuclear Energy Technology Platform (SNETP) [4] and, in particular, the related European Sustainable Nuclear Industrial Initiative (ESNII) carried out under the umbrella of the European Sustainable Energy Technology Plan (SET-Plan).

All these initiatives consider a fast neutron spectrum reactor as the essential tool to close the nuclear fuel cycle using innovative technologies.

In the context of innovative nuclear fuel cycles, it is considered that the optimized use of natural resources is directly linked to the sustainable development of nuclear energy. On the other hand, the partitioning and transmutation (P&T) strategies aiming at reducing the burden on a geological repository by means of waste minimization can be linked to both sustainable development of nuclear energy, and to nuclear phase out scenarios.

To summarize, three main categories of innovative fuel cycle scenarios can be envisaged [5]:

(1) The sustainable development of nuclear energy: for this scenario, the optimization of the use of natural resources is combined with waste minimization; therefore, the development of fast reactors is the reference solution;
(2) The reduction of minor actinides (MA) inventory: is an objective compatible both with a use of plutonium (Pu) as a resource in light water reactors (LWRs) for a limited period of time, in the hypothesis of a delayed deployment of fast reactors, and with a sustainable development of nuclear energy, based on the deployment of fast reactors. This objective corresponds to the so-called double strata approach, where the MA inventory is reduced either with fast reactors or with an accelerator driven system (ADS);
(3) The management of spent fuel inventories: as a legacy of previous operation of nuclear power plants. The objective in this case is to reduce the inventory of Pu and MA with the use of ADS or fast reactors (FR) having a low conversion ratio.

Although different scenarios and approaches can be identified to address innovative options of the nuclear fuels cycle, a common trunk for advanced FR and ADS technology development, together with their associated fuel cycles, can be envisaged. These common R&D areas are; e.g. coolant technologies, reactor structural materials and

innovative fuels, where the need of large research infrastructures as; e.g. fuel and material irradiation facilities, fuel treatment and fuel fabrication installations can be envisaged [6].

As far as past experience and know-how on fast reactors is concerned (see also Section 6), a considerable operation experience on sodium fast reactors has been gained in several countries; e.g. the Russian Federation by operating BOR-60 (Bystrij Opytnyj Reactor (fast experimental reactor)), BN-350 (Bystrie Neytrony (fast neutrons)) and BN-600; in France by operating Rapsodie, Phénix and Superphénix; in the USA by operating EBR-II (Experimental Breeder Reactor No. 2) and FFTF (Fast Flux Test Facility); in Japan by operating Joyo and Monju, etc. Therefore, the level of technological maturity for sodium as coolant in a nuclear system is significant.

As of today, some sodium cooled FR construction projects are underway in India, China and the Russian Federation, specifically the commercial reactor BN-800 in the Russian Federation, the small (25 MW(e)) CEFR experimental reactor in China, and the 500 MW(e) PFBR prototype in India. CEFR has been connected to the grid in 2011, whilst the startup of the other two reactors is foreseen in the very near future.

Moreover, as mentioned above, new and more challenging objectives have been defined for the future use of nuclear energy. In this context, innovative solutions for sodium cooled FRs are under consideration for instance in France and Japan.

For the planned French prototype ASTRID, the innovations and design objectives are determined by [7]:

— The development of a safe core able to transmute MA;
— A better resistance to severe accidents and external hazards;
— Optimization of the energy conversion system and reduction of the sodium risks;
— Improved conditions of operation and economic competitiveness through evaluation of reactor components design.

The innovations and design objectives of the advanced loop-type Japanese commercial fast reactor JSFR (JAEA sodium fast reactor) are enhanced economic competitiveness and safety, as well as proliferation resistance. The design challenges of the JSFR are related to a compact system design and ambitious goals in the area of the sodium system. To achieve a more compact design, a two-loop cooling system, a simplified and shortened piping configuration and a compact reactor vessel are considered. The study of a double boundary concept to reduce the sodium risk, which includes; e.g. double walled tubes for the steam generator, and the implementation of innovative in service inspection and repair methods at the design stage are two examples of design strategies to increase the reliability in the JSFR [8].

Coming to Pb and Pb-Bi eutectic (LBE), these liquid metals were proposed and investigated as coolants for fast reactor systems as early as the 1950s (e.g. in the USA). However, sodium became the preferred choice in the 1960s due to a higher power density achievable with this coolant, which resulted in a lower doubling time, an important objective at that time [9]. On the other hand, LBE was chosen as the coolant for a number of ‘alpha class’ submarine reactors in the former Soviet Union, which led to a very extensive research and development of heavy liquid metal coolant (HLMC) technology and associated materials.

During the 1990s, there has been a renewed interest in the Russian Federation for Pb and LBE as coolants for civilian fast reactors. Moreover, the potential of HLMC was considered in several projects in the emerging field of ADS. In particular, Pb cooling was associated with the proposal by European organization for nuclear research (CERN) for an energy amplifier subcritical nuclear system, and LBE has been considered both as a spallation target coolant and material [10]. In this context, important R&D projects were launched, and relevant results have already been obtained. Among the different projects, it is worth mentioning the MUSE (MUltiplication of an external Source Experiments) project aimed at the demonstration of the physics behavior of a subcritical core [11] and the MEGAPIE (MEGAwatt PIlot target Experiment) project aimed at the design, construction and safe operation of a liquid LBE neutron spallation target [12]. Both projects have successfully been conducted and completed.

Subsequent studies of ADS; e.g. in Europe, have considered both Pb and LBE as coolant and spallation material. The most prominent ADS project carried out in Europe has been the EUROTRANS (European research programme for the transmutation of high level nuclear waste in an accelerator driven system) project [13] which, in its turn, has launched many R&D activities addressing the main issues and challenges of ADS. The objective of EUROTRANS was to demonstrate the technical feasibility of transmutation of high level nuclear waste (Np, Am and Cm), present in the spent nuclear fuel as discharged; e.g. from light water reactors using ADS. With this object in mind, the design of an experimental facility (XT-ADS) cooled with LBE — aimed at demonstrating the technical

feasibilities to transmute a sizable amount of wastes and to safely operate an ADS — has been developed. In addition, a conceptual design of the European Facility for Industrial Transmutation (EFIT), cooled with Pb, has been also carried out. In addition to the design activities, in the EUROTRANS project a substantial effort has been devoted to the development and demonstration of the associated technologies, especially in the field of MA bearing fuels, heavy liquid metal (HLM) technologies, and nuclear data.

Finally, as already mentioned, GIF has identified the LFR as one of the six concepts that deserve further development. Therefore, innovative technological solutions, to meet the Generation IV goals are also considered for LFRs. Moreover, due to the properties of Pb and LBE, developments of LFR are considered for different applications as nuclear waste transmutation, hydrogen production and the development of small modular reactors with long life cores.

Several LFR design activities have been conducted in the past to address the above mentioned objectives. As an example, in Europe Pb is considered as coolant to develop an innovative and simplified LFR, named ELSY (European Lead-cooled SYstem) [14]. The innovative solutions under consideration in ELSY are related to the compactness of the system [14]. In the Russian Federation, LFR concepts are being developed, considering both LBE and Pb as coolant. The Russian LBE cooled system is the SVBR (lead-bismuth fast reactor) which has 75-100 MW power and is a small scale modular system. As for Pb cooled systems, the reference design is the BREST-OD-300 system.

For both the ADS and LFR development, common developments can be envisaged in the different technological areas; e.g. the HLM technology and structural materials, the integration of in service inspection and repair features in the design of these systems, etc.

Several items related to the development of HLM technologies and materials have been already addressed in different international R&D programmes in the Russian Federation, Europe, USA, Korea, Republic of and Japan. A good overview of these activities and related results is provided in the Handbook on lead-bismuth eutectic alloy and lead properties, materials compatibility, thermal-hydraulics and technologies published by the OECD/NEA (Nuclear Energy Agency) [15].

1.2. SCOPE AND OBJECTIVES

This publication is intended to provide a comprehensive summary of the status of the liquid metal coolant technology development for fast reactors, with regard to basic data and main technological challenges. It also covers various fast reactor concepts and designs that are being investigated, with special emphasis on the choice of the coolant.

The choice of the coolant is one of the main technical issues concerning fast reactors design, since it determines design approach as well as technical and economic characteristics of the system. Moreover, several aspects concerning coolants are related to the reliability and safe operation of liquid metal cooled fast reactors (LMFRs). These aspects are in the field of coolant quality control, materials compatibility, thermal-hydraulics behaviour in different operation regimes, and innovative and robust instrumentation and measurement techniques development.

It is worth noticing that several R&D areas might be common to both types of LMFRs — i.e. sodium and HLM cooled FR — as, for example, the development of ODS steels for the fuel cladding to increase the burnup, or the development of MA bearing fuels.

Significant experience in liquid metals technology has been gathering thanks to a number of R&D activities, as well as the design, construction and operation of several experimental and prototype fast reactors; this know-how represents a key factor for future developments, in particular as far as the so called Generation IV nuclear systems. The need to preserve knowledge and to exchange information is a major recommendation by the IAEA Technical Working Group on Fast Reactors (TWG-FR); therefore this report represents the joint effort of various Member States with an active fast reactor programme to collect, review and document the available information about liquid metal coolants — sodium, lead and lead-bismuth — for fast neutron systems.

1.3. STRUCTURE

After historical remarks on the nuclear power development given in Section 2, a complete survey of physical and chemical properties of liquid metals is provided in Section 3. Among these properties, important parameters to be considered are their associated chemical and radiological hazards. For these issues the report underlines what are the potential drawbacks of using liquid metals as coolant, since e.g. the chemical hazards drive the design option selection of the secondary systems and the radiological hazards drive both the design of coolant and cover gas quality control systems and the procedures for maintenances.

The coolant quality control, discussed in Section 4, is of primary importance since it has a direct impact on the structural materials compatibility performance, on the thermal-hydraulics performance of the core, the primary system and the heat exchanger and, to some extent, on maintenance and decommissioning procedures. The study of the physical and chemical properties of the liquid metal and the evaluation of contamination sources which can occur in the reactor system are essential data for the development of technologies to control the coolant quality.

Thermal-hydraulics studies are discussed in Section 5 for both sodium and lead alloys systems. They are related to the evaluation of heat transfer relations and pressure drop to assess the heat transfer characteristics of a liquid metal cooled nuclear system. The heat transfer coefficient and pressure drop in the core are relevant parameters to be assessed for core design layout as well as for safety analysis, where relevant transients are taken into consideration. The objective of the thermal hydraulic studies is to reduce the data dispersion obtained with the different heat transfer relations available.

As far as the radioactivity of coolants is concerned, a complete discussion is provided in Section 6.

Section 7 deals with liquid metal cooled fast reactors technology development. It provides remarks on the past experience as well as on currents projects, and underlines the key issues for the future achievement of the technical and economic objectives.

Finally, in line with the general purposes of data retrieval and knowledge preservation of this report, the Russian experience and current activities with heavy liquid metal cooled fast reactors are treated in Sections 8 and 9. Particular emphasis is given on the two concepts BREST and SBVR-75/100, cooled by pure liquid lead and eutectic lead-bismuth, respectively.

2. SHORT HISTORY OF NUCLEAR POWER DEVELOPMENT

Nuclear energy, whose concentration and resources of fuel are millions of times greater than those of chemical energy, has inspired researchers with hopes since the early 20th century. In the early years of that century, E. Rutherford had reasons for questioning its practical utilization, considering the large energy needed to accelerate charged particles, and the stellar temperatures required for fusion of light nuclei. However, the discovery of neutrons (1932) and neutron induced uranium fission (1938–39) opened the door to self-sustaining nuclear ‘burning’, which proved to be not only feasible but also remarkably simple in terms of its physics and engineering principles. It must be stressed that the use of this energy source was possible only due to a (first) ‘miracle’: the only long lived fissile nuclide (^{235}U) had remained in natural uranium in precisely the right proportion (~0.7%) needed for thermal, natural uranium fuelled, graphite (or heavy water moderated) reactors (the first of such a reactor was started by E. Fermi in Chicago in December 1942).

The key inherent resource of fission reactors in terms of physics, which allows resolving practical problems and choosing adequate technologies, is neutron excess NE (larger than unity for sustaining the chain reaction). For one ‘up’ fissile nucleus (^{235}U, ^{239}Pu, ^{233}U)

$$NE = \frac{\nu + f(\nu' - 1)}{1 + \alpha} - 1 \qquad (1)$$

where ν, ν' stand for the number of neutrons per fission of fissile and fertile (^{238}U, ^{232}Th) nuclides, α represents losses due to absorption by a fissile nucleus, and f is the contribution from threshold fission of a fertile nuclide. Thermal reactors have NE~1 (1.3 in case of Th^{233}U), which is sufficient to maintain the chain reaction. This NE is also sufficient for the generation of weapons plutonium in natural uranium fuelled, graphite or heavy water moderated thermal reactors designed accordingly. It is also sufficient for small submarine propulsion light water reactors fuelled with enriched uranium. Both the USA and the Soviet Union developed nuclear weapons as early as the 1940s, followed by nuclear submarines in the 1950s.

In the 1950s, the Soviet Union, UK, USA and Canada built the first nuclear power plants derived from the design of thermal neutron military reactors (graphite, heavy water, and light water moderated). The USA, Soviet Union and UK also built experimental fast reactors designed with the objective of high plutonium breeding, in view of establishing — on a relatively short time scale (within the 20th century) — a large nuclear power programme of thousands of GW.

Due to the scale of the nuclear power industry and to its various specific problems (economics, radioactive waste, safety, and proliferation resistance), achieving the objective of nuclear energy supply is much more difficult than the military effort. However, the generally favorable, even if limited, operating record of military reactors encouraged the belief that it was possible to ensure the safety of nuclear power plants by relatively simple engineering means, and by high equipment quality as well as personnel qualification. While recognizing that such special features linked to safety concerns would result in slightly higher nuclear power plant construction and operation costs, it appeared reasonable to expect that the cost of nuclear electricity would be lower, considering the much lower prices of the nuclear fuel (even if less than 1% of the uranium was utilized in thermal neutron reactors). These general ideas are reflected in the considerable nuclear power capacity (in the range of hundreds of GW) designed and constructed in the 1960s. On the other hand, the satisfactory management of the radioactive wastes and of the proliferation issues was not fully achieved until the 1970s–1980s. As far as the nuclear fuel resources are concerned, it must be highlighted that as of January 2009, the total undiscovered resources (prognosticated resources and speculative resources) amounted to more than 10.4×10^6 tU [16]. These resources, if utilized in thermal reactors, are smaller in energy terms than the oil and gas resources, to say nothing of coal. Hence, nuclear energy based on thermal reactors alone is capable of meeting only limited energy demands, and its capability would be limited in time. Assuming no shortage of conventional fuels, and with no environmental concerns (greenhouse gas emissions) in mind, nuclear energy development based on thermal neutron reactors alone would be much like ‘cracking nuts with a sledgehammer’.

Hence, the fuel balance of sustainable large scale nuclear energy programmes must include the 99.3% ^{238}U available in natural uranium. This implies reactors with plutonium breeding ratios at least equal to unity (self-sustaining mode). Thus, NE > 1 is required, with allowance made for inevitable losses in the neutron balance. Reactors with breeding ratios larger than unity (i.e. fast neutron breeder reactors) allow increasing the energy yield from natural uranium by a factor of 60–100, granting realization of nuclear power programmes in the TW capacity range for thousands of years. As a matter of fact, the fast neutron reactor represents the second ‘miracle’ of nuclear fission (after the ^{235}U): the combination of high ν_{239Pu} (around 2.9) and f_{238U} (up to 0.4), with small α_{239Pu} (around 0.1) leads to (see equation 1) NE_{theor} of approximately 2.3 (neutron moderation in a real fast reactor reduces f and increases α, but still leaves NE>>1). The first to appreciate these unique neutron balance characteristics of fast reactors fuelled with ^{238}U-^{239}Pu (NE>>1) was Enrico Fermi. In 1944, a couple of years after the startup of the first reactor, having realized that inclusion of fast reactors allowed sustainable large scale nuclear energy utilization, he came to speak in favor of developing peaceful applications of nuclear energy. After the war, E. Fermi returned to fundamental physics, having left the fast reactor concept to his followers at ANL. As early as 1951, the first nuclear electricity by a fast reactor was produced by EBR-I in Idaho Falls, Idaho (1.2 MW, U_{met}, NaK as coolant), followed by EBR-II and FFTF (in Idaho Falls, Idaho, and Richland, Washington, respectively). EBR-I suffered an accident, which occurred while experiments on reactor stability were conducted. This accident was analyzed and lessons were learned on how to properly mechanically restrain a fast reactor core. In the 1980s, the USA, due to proliferation concerns, wound up its fast reactor construction and development programme.

As for the Soviet Union, one of its major nuclear physicists, A.I. Leipunsky, who in 1947 was investigating neutron irradiation as a means of defense against nuclear weapons, also evaluated the neutron balance of fast reactors and recognized their potential for the nuclear power industry. He led the fast reactor R&D efforts in the USSR, the construction in 1959 of the experimental BR-5 (5 MW, PuO_2, sodium as coolant) and of the

experimental BOR-60 reactor, the design of the first prototype fast reactor BN-350, and of the commercial BN-600 fast reactor. He died in November 1972, three months before the first criticality of BN-350.

Subsequently, France, Germany, Japan and the UK also built uranium-plutonium mixed oxide fuelled fast reactors cooled with sodium. Currently, India and China have important fast reactor programmes, with India operating the Fast Breeder Test Reactor (FBTR) at Kalpakkam and building, at the same site, the 500 MW(e) Prototype Fast Breeder Reactor (PFBR); and China commissioning the China Experimental Fast Reactor (CEFR) at the CIAE site near Beijing (first criticality achieved on 21 July 2010).

However, it is to Fermi and Leipunsky, more than anyone else, that the world owes this great idea of 20th century physics, which has important implications for sustainable nuclear energy supply. Picked up by others and thanks to the efforts of many scientists and engineers in all the countries with fast neutron reactor research, technology development, and deployment programmes, the fast reactor did not remain a mere 'Fermi's dream'.

The nuclear power plants with thermal neutron reactors were considered to be the first stage in nuclear power deployment, and a source of plutonium needed to fuel future fast reactors. With plutonium shortage and high post-war rates of energy growth (for example, ~12% per year in the former Soviet Union), fast reactor design efforts were aimed at high breeding ratios, hence to designs with fertile blankets and high power densities. Nuclear power had a good start, indeed, in the 1950s–60s, and was expected to reach the level of 850–1400 GW (according to the US Atomic Energy Commission) and 600 GW (according to the State Committee for Atomic Energy of the Soviet Union) towards the end of the 20th century. In reality, the US LWR-based electricity generation attained a capacity of approximately 100 GW, and the Soviet one of about 34 GW. Globally, nuclear electricity generating capacities reached approximately 375 GW [17] (with LWR contributing about 91%), providing 14% of world electricity and accounting for about 6% of overall energy consumption. Nuclear's share in the energy supply has been declining since the 1990s. The life of the first fast reactor plants was cut short, and the original plan for nuclear power development failed in the past century.

With hindsight, the 20th century may be termed the 'prehistory' of the nuclear era. However, the hard lessons learned, as well as R&D advances, do offer the promise that the energy sector may actually see the full realization of the nuclear era in the 21st century.

3. PHYSICAL AND CHEMICAL PROPERTIES OF COOLANTS (SODIUM, LEAD AND LEAD-BISMUTH ALLOY)

3.1. LIQUID METAL DEVELOPMENT (SHORT HISTORY)

Liquid metal (mercury) was utilized for the first time in engineering applications in the 1930–1940s in the USSR, Germany and the USA, in connection with the attempt of using more than one working fluid in order to increase the efficiency of thermal power plants by 1.5 times and more. After several experiments that displayed a number of obstacles and practical difficulties, these first tries were abandoned.

Based on a limited experience on liquid metals in the chemical industry and non-ferrous metallurgy in 1949–1953, the erection of experimental liquid metal rigs was first performed in the USA and USSR. Liquid metal R&D work started by the trial-and-error method due to a lack of knowledge of their technological properties. There was not any experience on equipment production (pumps, devices, etc.), cleaning methods, safety technique and the like. Therefore, the choice of coolant for designing the elements of fast breeder reactors had to be made in the absence of necessary information [18–27].

Prior to the construction of reactors cooled by liquid metals, the three principal areas related to thermal physical problems had to be settled.

In hydro-dynamics, the matter was whether the flow of liquid metal obeys the same laws as for ordinary liquids and gases.

As far as the heat transfer characteristics are concerned, applying the known water heat transfer correlations to liquid metals yielded a very conservative increase, as compared to water (by factor of 6), of the liquid metals heat

transfer coefficients. Obviously, this result was deemed to be implausible, since the thermal conductivity of metals is 10–100 times higher than that of water.

The liquid metals treatment technology had to be developed and experimentally mastered. That concerned the equipment production (pumps, heat exchangers, control instrumentation, and so forth), decontamination of metals — an issue that many underestimated in the beginning. While obvious for the chemically active alkaline metals (Na, K, NaK), this last problem was not considered critical for lead and bismuth. As experience on liquid metals became available, various articles were published in the 1960s–1980s introducing liquid metal cooled reactor concepts. As an example, a reactor concept cooled by NaK alloy was designed with an inlet temperature of 700°C. The NaK alloy heated the hot joints of a battery cascade (silicon germanium). A second NaK alloy loop cooled the cold joints, which removed the heat in the radiator-cooler to 300–400°C.

At the Institute of Physics and Power Engineering (IPPE), the development of thermo-emission nuclear facilities started in 1958 [28]. These facilities used NaK alloy as coolant. The reactor was rated at 120–150 kW thermal power, zirconium hydride moderated, resulting in an intermediate neutron energy spectrum. The reactor was equipped with 7 kW(e) electro-generating channels (elements).

Towards the end of the 1950s, pre-design work was started to study the feasibility of aircraft nuclear power facilities (ANPF), using Na as coolant (900°C). In a Na-air heat exchanger, the reactor heat was removed to air, which was then directed to the turbine engine. Eventually, after a series of calculations and experiments, these activities were abandoned [29–36].

Another project, in which a liquid metal cooled reactor served as energy aimed at the realization of an air-ship gliding on an air cushion produced by compressors (turboprop engines). This project was also abandoned in the 1960s.

In spite of the comprehensive liquid metals R&D investigations carried out over the past years, some issues are still unresolved. These problems surfaced gradually during the process of gaining experience with liquid metals. An example for such problems is the issue of mass transfer over a long time operation, in non-isothermal conditions, of circuits made of heterogeneous structural materials (circuits that are typical for power facilities).

The tendency to use intense heat transfer processes, and consequently the necessity to increase the flow velocity in circuit elements, lead to other difficulties, namely to dangerous vibrations, elements wear, and fatigue breakdowns.

As of today, all prototype, demonstration and commercial liquid metal cooled fast reactors (LMFRs) have used liquid sodium as coolant (see Section 7). Sodium cooled systems, operating at low pressure, are characterized by large thermal margins relative to the coolant boiling temperature and a very low structural material corrosion rate. In spite of the negligible thermal energy stored in the liquid sodium available for release in case of leakage, there is some safety concern because of its chemical reactivity with respect to air and water. Lead, lead-bismuth or other lead alloys, appear to eliminate these concerns because the chemical reactivity of these coolants with respect to air and water is very low. These systems could be attractive, provided high corrosion activity inherent in lead, bismuth and their eutectic alloy lead-bismuth (LBE) on unprotected materials and other problems is resolved. Extensive research and development work is required to meet this goal [37–38].

Preliminary studies on LBE and lead cooled fast reactors and ADS have been initiated in France, Germany, India, Italy, Korea, Republic of, Japan, USA, and other countries. Considerable experience on LBE was gained in the Russian Federation during research activities performed on liquid metals in different domains of science and technology [25, 39–42]. It is also worthwhile mentioning, that USSR submarine reactors used LBE as a coolant in [43–44].

In the process of developing liquid metal facilities, several novel equipment and components were designed and fabricated, that were never used before in power engineering or in any other domain [26, 27, 40, 45]. Examples for this are metal water and metal gas heat exchangers and steam generators for liquid metal facilities [46]. Various types of electromagnetic pumps and valves for liquid metal application have been designed as well. All these devices may be applied in metallurgy. The principally new liquid metal decontamination devices (‘cold’ and ‘hot’ traps) implemented in nuclear facility may be used in chemical industry. Sensors based on new methods of measuring the level, flow rate, pressure of liquid metals at high temperatures provided reliable monitoring systems for nuclear facility.

Liquid metals are being considered for space nuclear applications, which can involve either machine for conversion from thermal to electric energy, or the direct conversion of thermal energy (thermoelectric and thermo-emission types) [29–35, 46]. Such metals as NaK alloys and Li [36] were assumed to be used as coolants, whilst

potassium vapor was considered as working fluid in turbine. The eutectic NaK alloy appeared to be the most suitable coolant for space applications since it does not require pipe heating systems and is liquid at room temperature (Tmelt = –12°C).

In 1956, the use of vapors of mercury and potassium for energy conversion machine was supposed to be the most ready for deployment. Experience on binary cycles with mercury were gathered. The draft design was fulfilled with nuclear facility of 5 kW(e) power with mercury vapor and some projects of greater power with potassium vapor (1–100 kW(e)) as well. A number of difficulties encountered that time give preference to thermo-electrical scheme of direct conversion (as the most simple) — the 'BUK' facility (5 kW(e)), development and operation of reactors cooled with lead-bismuth eutectic, in particular, propulsion reactors. Studies on lead cooled fast reactors are also under way in this country.

The need to exchange information on alternative fast reactor coolants was a major consideration in the recommendation by the IAEA TWG-FR to collect, review and document the information on lead and lead-bismuth alloy coolants: technology, thermohydraulics, physical and chemical properties, as well as to make an assessment and comparison with respective sodium characteristics.

3.2. PHYSICAL PROPERTIES

The highest purity extent of commercial grade sodium is 99.8%, this value corresponding to the purity of sodium that passed through the cold or hot traps [47, 48]. However, sodium content of 99.995% can be achieved using distillation.

Sodium has one stable isotope: ^{23}Na. Characteristics of other isotopes of sodium are presented in Table 1. It is just isotope ^{24}Na that determines the main proportion of radioactivity in the coolant flowing in the circuit.

Crude lead contains 93–99% of the basic metal. Among the basic impurities are Cu (1–5%), Sb, As, Su (0.5–3%), Bi(0.05–0.4%), Al and Au. The highest purity of lead is 99.992%. Lead has four stable isotopes: 204, 206, 207 and 208 (see Table 2).

The last three are the products of U, Ac and Th decay. In Table 3, the Bi isotopes are reported.

TABLE 1. SODIUM ISOTOPES [49, 50]

Isotopes	Atomic mass	$T_{1/2}$	Energy, MeV
20	19.99	0.4 s	β^+ (5.5; 7.15; 11.2)
21	20.99	23 s	β^- (2.5)
22	21.99	2.6 a	90% β^+ (0.54; 2.8;); 10% (sec) γ (1.275)
23	22.99	—	—
24	23.99	14.97 h	β^-(5.5); γ (1.38; 2.76)
25	24.99	60 s	β^-(3.7; 4.0); γ (0.37; 0.58; 0.98)

TABLE 2. LEAD ISOTOPES [49, 50]

Isotopes	Atomic mass	$T_{1/2}$	Natural abundance for stable istopes, Radiation and energy (MeV) for unstable isotopes
204	203.973		1.4%
205	204.874	3×10^7 a	β^- (2.5)
206	205.974		24.1%
207	206.976		22.1%
208	207.981		52.4%
209	208.984	3.25 h	β^- (0.644)
210RaD	209.984	22.3 a	β^- (0.656); γ (0.008-0.046)
210RaF	209.984	5 day	β^-(1.17)-(100%); α (4.93) –(10^{-4}–10^{-5})%

TABLE 3. BISMUTH ISOTOPES [49, 50]

Isotopes	Atomic mass	$T_{1/2}$	Energy, MeV
209	208.9804		
210	209.9841	5.013 day	β^- α (1.08, 1.165,β^- 5.0): γ (0.08)
210m	209.9841	3×10^6 a	α (4.93); γ (0.26)
211	210.9873	2.14 m	β^- (0.276)
212	211.9913	66.5 m	β^-; α(0.036)
213	212.9944	45.6 m	β^-; α(5.98)
214	213.9987	19.9 m	β^-; α(5.62)
215	215.0018	7.4 m	β^-
216	216.0062	3.6 m	β^- (4.0)

TABLE 4. THERMO-PHYSICAL PROPERTIES OF Na, Pb, Bi AND Pb-Bi (44.5/55.5%) [22]

Properties		Units	Na	Pb	Bi	Pb-Bi
Atomic number		—	11	82	83	—
Atomic mass		—	22.99	207.2	208.98	—
Melting temperature		°C	98	327.4	271.4	125
		K	371	600.85	544	398
Boiling temperature		°C	883	1 745	1 552	1670
		K	1 156	2 018	1825	1943
Heat of melting		kJ/kg	114.8	24.7	54.7	38.8
		kJ/mole	2.6	5.1	11.4	8.07
Heat of vaporization		kJ/kg	3871	856.8	852	852
		kJ/mole	89.04	178	178	178
Density	sol. 20°C	kg/m^3	966	11 340	9 780	10 474
	liq. 450°C		845	10 520	9 854	10 150
Heat capacity	sol. 20°C	kJ/kgK	1.23	0.127	0.129	0.128
	liq. 450°C		1.269	147.3	150	146
Thermal conductivity	sol. 20°C	W/mK	130	35	8.4	12.6
	liq. 450°C		68.8	17.1	14.2	14.2
Kinematic viscosity (450°C)		m^2/s	3×10^{-7}	1.9×10^{-7}	1.3×10^{-7}	1.4×10^{-7}
Prandtl number (450°C)		—	0.0048	0.0174	0.0135	0.0147
Surface tension (450°C)		mN/m	163	480	370	392
Volume change with melting		%	+2.6500	+3.6	–3.3	~ +0.5

The main thermo-physical properties of sodium, lead, bismuth and lead-bismuth eutectic alloy (44.5% Pb–55.5% Bi) are presented in the Table 4.

3.3. CHEMICAL PROPERTIES

3.3.1. Chemical properties of sodium

Due to its low ionization energy, sodium is one of the most electropositive metals, well ahead of hydrogen in the electrochemical series. It therefore reacts exothermically with water yielding sodium hydroxide (NaOH) and hydrogen gas. The reaction depends on temperature and impurities present in sodium.

Sodium interacts with dry hydrogen and forms sodium hydride (NaH) which is soluble in sodium. At the temperature of 420°C NaH is decomposed with release of hydrogen. In the molten sodium, only sodium oxide (Na_2O) is stable, while sodium peroxide (Na_2O_2) dissociates.

Solubility of chemical elements in sodium. The saturation concentration (Cs) of the element in solution follows the Arrhenius equation:

$$\lg Cs = A - B/T \quad (2)$$

where A and B are experimentally determined constants (A — solution entropy, kJ/mole; B-evaporation heat, kJ/mole), Cs – wt%. The values of constants are given in the Table 5.

3.3.2. Chemical properties of lead

In dry air, solid lead is practically not oxidized, whereas in humid air it becomes coated with an oxide film (PbO). This film under air exposure is transformed into the basic carbonate: 3 $PbCO_3$.Pb(OH). Air causes molten lead oxidation initially to Pb_2O and then to PbO oxide. At the temperature of 450°C, PbO is transformed to Pb_2O_3 and then at 450–470°C to Pb_3O_4. Being unstable all these compounds dissociate into PbO and O_2. Lead interacts with water producing hydroxide $Pb(OH)_2$ in the temperature range 400–500°C, and in not isothermal circuits also with steam-hydrogen mixture in a pressure ratio range of P_{H20}/P_{H2} = 5–50.

3.3.3. Solubility of chemical elements in molten lead-bismuth alloy

The solubility of chemical elements in molten lead-bismuth alloy follows the equation (1). Data on solubility make it possible to compare element resistance to liquid metal. Values of the A and B coefficients of equation (1) are given in Tables 6–8.

4. COMPARISON OF DIFFERENT SODIUM AND LEAD COOLANT TECHNOLOGIES

4.1. BACKGROUND

Coolant technology, as applied to nuclear reactors, encompasses the material data and properties, the performance of various coolant materials and cooling systems (under both normal operating and abnormal conditions), and the methodologies applied to the design of the systems having cooling functions.

TABLE 5. A AND B COEFFICIENTS IN EQUATION (1) FOR SODIUM [47]

Element	A	B	Temperature °C
H_2	2.467	3023	100–400
O_2	2.257	2444	100–600
C	3.2	5465	600–950
Fe	1.16	4310	300–850
Ni	–1.93	1570	300–850
Cu	1.45	3055	300–700
Mo	–0.73	3962	800–1000
Na_2O	–2.8	1777	100–600

TABLE 6. A AND B COEFFICIENTS IN EQUATION (1) FOR LEAD [51]

Element	A	B	Temperature, °C
H_2	–1.946	2360	500–900
N_2	no solubility		
O_2	3.44	5240	350–850
C	1.026	3850	350–1000
Fe	0.34	3450	330–910
Cu	2.72	2360	327–1000
Cr	3.74	6750	908–1210
Co	2.60	4400	350–1650
Ni	2.78	1000	330–1300
Nb	solubility at 1000°C		$<10^{-5}$ wt%
Mn	2.02	1825	327–1200
Mo	solubility at 1000°C		$<10^{-3}$ wt%
Ti	solubility at 500°C		$\sim 5.6 \cdot 10^{-4}$ wt%
Si	3.886	7180	1050–1250
U	3.921	5121	400–800
Zr	solubility at 500°C		$\sim 1.2 \cdot 10^{-9}$ wt%

TABLE 7. A AND B COEFFICIENTS IN EQUATION (1) FOR BISMUTH [51]

Element	A	B	Temperature, °C
C*	–3.17	360	300–750
Fe	1.687	3 490	350–750
Nf	2.098	2 000	300–700
Mo	2.061	22 300	700–1030
Nb*)	0.150	1 911	500–1200
V	–0.072	2 520	500–900
Pt	2.957	1 654	300–1000
O_2	2.00	3 692	400–700

* The new data; old data A = -0.012; B = 1672.

TABLE 8. A AND B COEFFICIENTS IN EQUATION (1) FOR Pb-Bi ALLOY [51, 52]

Element	A	B	Temperature, °C
O_2	1.2	3400	327–1000
C	–1.36	1870	—
Fe	2.01	4380	—
Ni	1.70	1000	450–550
Cr	7.68	6959	370–540
Zr	0.15	3172	350–750
Si	2.99	6000	

(1) Development of coolant quality standards;
(2) Analysis of impurities condition, their sources, and accumulation rate in the circuit;
(3) Analysis of corrosion and mass transfer;
(4) Development of methods and equipment for keeping impurity content in coolant within acceptable limits;
(5) Analysis of different operating procedures and evaluation of related impurities input;
(6) Control of coolant quality during operation.

Composition of coolants (Na, Pb, etc.) supplied by the industry does not always meet technological requirements. Therefore, additional procedures are performed prior to filling circuits with the coolant in order to bring it to the required condition. Maintenance and repair works resulting in pollution of the coolant can always occur during facility life. Another cause of changes in coolant composition is corrosion, which can also cause deterioration of mechanical properties of structural materials. Corrosion products are transferred along the circuit, depositions are formed that can affect hydrodynamics and heat transfer, and hence the reliability of the facility. All circumstances mentioned above lead to the necessity of continuous control of the impurities content in the liquid metal and the corrosion processes in the circuit.

The main elementary corrosion processes in liquid metal are as follows:

(1) Interaction of coolant with passive or oxide films — passivation or activation of materials;
(2) Dissolving of steel components and their chemical interaction with non-metal impurities (oxygen, hydrogen, etc.);
(3) Penetration of liquid metal into solid materials causing uniform and inter-granular corrosion;
(4) Transport of dissolved structural material elements along the circuit.

In order to predict the direction of corrosion and mass transfer it is essential to have data on thermochemical properties of the steel elements as a function of temperature. If liquid metal is flowing at high velocity, the material is subject to erosion. Formation of the film (consisting of both steel and liquid metal coolant elements) on the structural metal surface is another type of corrosion, since this is not the protective film. Due to the difference in chemical activity between sodium and lead, technologies of these coolants are quite different although some methods share a number of common features.

4.2. SODIUM TECHNOLOGY

4.2.1. Impurity sources

Typical composition of commercial-grade sodium is presented in Table 9.

Among permanent contributors of impurities under normal operating conditions are corrosion related hydrogen from steam generators, tritium from reactor core, oxygen and hydrogen from cover gas and corrosion products (see Table 10).

TABLE 9. STANDARD SODIUM ANALYSIS

Component	Content (mass)
Total, (Na), %	99.9
Metal, (Na), %	99.8
Potassium, (K), ppm	200
Calcium, (Ca), ppm	250
Barium, (Ba), ppm	<5
Iron, (Fe), ppm	<20
Chlorine, (Cl), ppm	<10

TABLE 10. SOURCES OF IMPURITIES IN SODIUM [53]

Sources	Oxygen	Water kg/year	Hydrogen	Corrosion products from reactor kg/year	Tritium from the core g/hour
Cover gas	1 kg/year*	0.1–0.5	$(3\div6) \times 10^{-2}$ g/h**	20**	6.3×10^{-5}*
Repair works	6 kg/year*	0.6	0.5–1g/h**	60*	—
Steel (Cr18Ni10Ti)	0.01 g/m^2	—	4.4×10^{-3} g/kg	—	—
Steel (1Cr2Mo)	0.01 g/m^2	—	6.4×10^{-3} g/kg	—	—

* BN-600
** BN-350

The characteristics of impurity sources as evaluated from BN-350 and BN-600 reactors operating experience are:

(1) Initial impurity content due to oxygen absorption on the circuit surface is 3.4 g/m^2 (I circuit) and 2.2 g/m^2 (II circuit), that is in a good agreement with the experimental data: 1.4–2.4 g/m^2. Total amount of oxygen absorbed is about 30 kg;
(2) Impurities introduced during repair operation: Na_2O, NaOH and Na_2CO_3. Assessments have shown that during operation (20 years) about 200 kg of substances aforementioned is introduced [53];
(3) Impurities added during SA loading. Assuming 2 g/m^2 specific content, total amount is equal to about 200 kg;
(4) Impurities caused by diffusion from other sources: diffusion rate is evaluated to be about 50 g O_2/day rate. Total amount of impurities is about 720 kg;
(5) Impurities added due to structural material corrosion. Total amount of oxygen impurities accumulated for 20 years is estimated to be 1400 kg of O_2 or 5.5 t of oxides (in the primary circuit).

4.2.2. Purification of sodium and cover gas

At present, the most wide practical application of sodium purification is the cold trap technique, which is based on decreasing the solubility of the major part of impurities in sodium by reducing the temperature [54, 55]. In the cold trap, sodium is cooled within the settling tank and the section upstream the filter. This results in reduction of oxygen and hydrogen concentrations respectively to 1 and 0.05 ppm. Cold trap is capable of retaining impurities in amount up to ~30% of its volume [56].

Carbon is confined in the cold trap as suspension, with the purification time being 10 times longer than that for oxygen. In order to catch caesium, graphite based purification technique was developed [57, 58]. After 10 years of BN-350 operation, the capacity of secondary traps was exhausted and their hydrogenation was carried out. The basic amount of sodium transformed to caustic phase was removed at 420°C. Thus, the normal operation of traps was recovered.

As regards cover gas, it is necessary to purify argon eliminating impermissible release to the environment of $^{133, 135}$Xe and $^{85, 87, 88}$Kr, short term total activity of which in the BN-600 reactor reached 6.7×10^9 Bq/L value. This problem was solved using special filters and absorption on activated coal [54].

4.2.3. Control of impurities content in the coolant and cover gas

Control of the content of non-metal impurities in the coolant (O_2, H_2, C) and cover gas (H_2O, CH_4, N_2), as well as ^{90}Sr, ^{131}I, ^{137}Cs, ^{54}Mn, ^{58,60}Co in the primary circuit is the most important from the safety and reliability point of view.

4.2.3.1. Oxygen control

In order to make measurement of oxygen activity in sodium the electrochemical control technique based on galvanic cell has been achieved. The sodium flows over the electrolytic pellet of thorium and yttrium sealed into the

metal tube and the reference electrode is located inside the tube. E.m.f. generated depends on temperature and oxygen concentration: e.m.f. = f(T, CO_2). Service life of such device is more than 10^4 hours.

4.2.3.2. Hydrogen control

The necessity of hydrogen control was due to the need of detection of water leak into sodium. Diffusion technique has found the widest application. This method uses a metal membrane permeable to hydrogen in combination with different secondary devices (mass-spectrometer, magnetic discharge pump, etc.). In this way, flow rate of hydrogen passing through the nickel membrane into vacuum cavity is measured by the system equipped with magnetic discharge pump, where gas ionization takes place. Automatic hydrogen detector is capable of detecting 10–30 g water leak in 100 t of secondary sodium. Description of other techniques can be found in Ref. [54]

4.2.3.3. Carbon control

Carbon needs to be controlled for evaluating carbonization of structural materials because of possible impact to their mechanical properties. Diffusion and electrochemical cells similar to those mentioned above are used for carbon control, with the salt mixtures (Na_2CO_3-Li_2CO_3 and CaC_2-LiCl) used as electrolytes.

4.2.3.4. Control of impurity crystallization starting temperature (plugging indicator)

The operational principle of plugging indicator is based on impurity deposition from supersaturated solution in narrow gaps. Super-saturation is achieved by decreasing temperature of the coolant flowing through the indicator. As the temperature becomes below the saturation point, the flow rate through the indicator decreases, the decrease is registered by the indicator. Actually, indicator measures the temperature at which rate of impurity deposition or solubility is sufficient to detect change in the flow rate.

4.2.3.5. Radioactive impurity control

Prompt control of radioactive impurities in the primary sodium is made using three methods:

(i) Measurement of γ-radiation from piping 10–12 days after reactor shutdown;
(ii) Measurement of activity of fission and corrosion products on special-purpose bypass section 10 days after its disconnection from the primary circuit;
(iii) Concentration of selected products (Cs, I) in the graphite sorbent.

4.2.4. Corrosion processes in sodium

Comprehensive studies have been performed and reliable industrial experience has been gained on material corrosion in sodium. Corrosion rate in sodium is significantly lower than that in water or lead-based coolants [59]. In sodium, as well as in the other liquid metals, corrosion rate depends on many factors (temperature, velocity, impurity content, temperature difference, time, etc.). When evaluating corrosion rate, the major part of researchers took into account only the most contributing factors. Empirical equations for corrosion rate were most commonly derived for 316 steel at the velocity >4 m/s and oxygen content ≤ 10 ppm. The most reliable results were obtained in [60, 61].

It was shown that austenitic and low chromium steels (2.5Cr1Mo) have high corrosion resistance. The comparison of two steels HT-9 and Fe9Cr1Mo tested at 600–650°C, 6 m/s of flow velocity and 1 ppm oxygen content gave the following results (after 4000 hours): the corrosion rate of the HT-9 is two times lower than that for 316 steel. Content of Ni in ferritic materials is less than 1%, with even lower Cr content. Such alloys can be used as structural material of fuel subassembly wrapper tube and fuel pin cladding up to 650°C.

It has been found that corrosion rate in the leakage area is several times higher than corrosion rate in sodium under normal conditions [62]. For this reason, secondary failures of steam generator tubes could be possible and, as a result, pollution of circuit and coolant can occur through the products formed from the water-sodium interaction. In the early stage of industrial scale facilities operation, these problems required additional efforts by researchers, designers and operating personnel. In particular, the issue of spontaneous leak development in sodium heated steam

generators had become of high importance. For instance, the accident on the BN-350 steam generator resulted in penetration of about 900 kg of water into the sodium circuit. This did not cause any serious consequences, however the steam generator was put out of operation.

The proceedings of the International Working Group on Fast Reactors (IWG-FR) contained new information on the research and development programmes for the commercial deployment of fast reactors [63–65].

4.3. LEAD AND LEAD-BISMUTH TECHNOLOGY

4.3.1. Basic issues of technology

Basic technological challenges of using lead as reactor coolant is the assurance of the quality control of the coolant (as well as that of the surfaces contacting coolant), such that the following conditions are realized [66, 67]:

— Sufficient corrosion resistance of structural materials;
— Stable hydrodynamics and heat transfer during life time.

Presence of impurities in lead coolant is harmful for at least two reasons, namely:

— Possible partial or full plugging of the coolant flow cross-section area can occur, that disturbs hydrodynamics and hence heat transfer;
— Deposits on heat transfer surfaces (especially on the fuel elements of the reactor core) can cause an increase of cladding temperature.

4.3.2. Impurity sources

The following impurities appear on the initial stage of circuit filling:

— Residual oxygen remaining after evacuation and water vapours;
— Gas adsorbed on the inner surfaces;
— Steel corrosion products;
— Casual impurities (chip, welding hail, etc.).

When under operating conditions, the basic contributors of impurities in lead-coolant circuit are:

— Structural material corrosion;
— Erosion and abrasion of materials;
— Grease from pump sealing and bearings;
— Cover gas entrainment by the coolant;
— Admixtures to the coolant aimed at forming protective films.

Impurities can be located in the following parts of circuit:

— In the cover gas plenum;
— On the molten lead free surface (since density of practically all impurities is lower than that of lead);
— In the stagnant sections of circuit;
— On the surface of structures — as deposits;
— During facility operation the impurities are transported along the circuit.

4.3.3. Cleaning from slag

Oxide-based slag (PbO) can be removed from the coolant by settling and reducing with hydrogen. Slag originating dispersed impurities due to lead interaction with structural materials (Fe, Cr, Ni) are removed by

mechanical filter or by flowing settling. Oil and pyrolysis products can be removed using organic solvents or water vapor injected into the circuit [68–70].

4.3.4. Corrosion processes in lead

The choice of structural materials is probably the main problem in lead cooled reactors. Lead exhibits strong erosion-corrosion effect on structural materials, such as material dissolving, embrittlement, thermal transport of mass and inter-granular penetration of lead [63–65, 71–80].

The most resistant materials to lead are refractory metals followed by chromium steels, austenitic steels being less resistant because of high solubility of nickel, which is a typical alloying element. Stabilization of austenitic steels by Ti, Nb, Mo enhances their resistance to lead.

Oxide films formed on the steel surface prevent it from interaction with liquid lead. Since breakdown of oxide films is possible during operation, precautions must be taken for resuming and maintaining their thickness and density.

Thus, steel corrosion in molten lead can be significantly slowed down by the oxide film formation on the steel surface. The main technological problem is maintaining such oxygen content in the coolant which, on the one hand, would provide stability of oxide film (Fe_3O_4) on the steel surfaces, but, on the other hand, would preclude generation of lead oxide (PbO) in the coolant, that could result in the circuit slugging.

There are ranges of content of oxygen dissolved in lead meeting these two conditions, for instance (~5 × 10^{-6}–10^{-3} wt%) range. Oxygen content in lead can be controlled by injecting gaseous oxygen or dissolving solid PbO.

Required oxygen content in lead can be maintained in two ways:

— Bubbling of argon, hydrogen and water vapor mixture or gaseous oxygen through molten lead [81];
— Lead oxide filling through which molten lead is pumped.

In order to change oxygen content and remove surplus PbO, reactions with water vapor or hydrogen can be used. To determine oxygen content in molten lead (similarly to Pb-Bi technology development) galvanic cell can be used.

In reaction of PbO reduction, water vapors are efficiently removed from the circuit. Small amount of moisture acts as diluted oxidizer preventing from achieving reduction conditions for oxide films on the steel surface.

Procedure and parameters of all these approaches have to be developed with the necessary control of hydrogen content in cover gas and oxygen activity in liquid lead.

Preliminary studies showed that principal possibility exists to develop the technology mentioned, but additional experimental activities are needed to exploit this technology.

It was found in the tests performed on the experimental and industrial facilities, that corrosion rate for chromium steels in Pb-Bi alloy are 6–60 mg/m^{2}h at 450–500°C. It can vary with temperature, coolant velocity, oxygen content and other parameters. In lead, this value is about 0.026 mg/m^{2}h at 600°C with no mass transfer [82].

4.3.5. Possibility of application of lead-bismuth technology data to lead technology

It is not easy to transfer data gained on Pb-Bi cooled submarine facilities to nuclear power plants cooled by lead for the following reasons:

— The area of the circuit inner surface in contact with liquid metal in fast reactors is by a factor of 10 higher that the one of submarine facilities;
— Nuclear power plant (NPP) operation period is about 6000–7000 h/year that is at least one order higher than that of submarines;
— Inlet temperature in lead circuit is ~200°C higher than that for Pb-Bi alloy;
— Additional corrosion effects can occur in NPP such as iron diffusion through passivating protective oxide film, high steam generator pressure, and so on.

Contamination of the inner surface of the secondary circuit gives rise to danger in case of equipment repair, since the alloy in the secondary circuit is constantly in contact with sources coming from water. Water replacement and steam generator (SG) inner surface decontamination without any alloy removal would not give the desired

result. Secondary circuit decontamination turns out possible only after complete removal of the alloy from the secondary circuit.

Polonium activity is one of the important problems for lead-bismuth as a coolant (see also Sub-section 5.2). To eliminate this problem, the use of lead instead of Pb-Bi can be assumed. However, in this case the minimum temperature in the circuit should be above 400°C. This has the following consequences:

— Problems related to the control and maintenance of coolant quality, compatibility with structural materials, corrosion strength, etc. should be studied to a great extent at higher temperatures. Although lead is similar, to some extent, to lead-bismuth, it has different properties;
— Some engineering problems are caused by thermal insulation to heat up loops to the temperature over 400°C. Corresponding system should have high reliability and ensure necessary conditions during both operation and repair work;
— There are problems related to coolant freezing under any accident condition (quick pressure decrease in the secondary circuit, etc.), or it's following unfreezing without structure damage;
— Realization of two-circuit heat removal system with steam water cycle can be difficult since supercritical water parameters can be required because of high primary coolant temperature. Further, it should be noted that the problem of polonium radioactivity is not completely eliminated if lead is used. Moreover, because of higher lead temperature polonium volatility would be higher than that for lead-bismuth, while polonium activity in the Pb circuit cover gas would be lower than that in lead-bismuth circuit at normal operation. Probably this relationship would be also in the room aerosol activity after coolant leakage. The structures of the designed HLMC reactors, in case of SG leak (except for the SG tube tightness failure) would reduce the probability of penetration of radioactive coolant into the secondary circuit. This is due to the fact that the SG tube plates are placed at a higher level than the coolant free surface. Moreover, in the event of coolant penetration into the secondary circuit, it would be easier to remove LBE than Pb.

In any case, it seems obvious that there is a need to develop special-purpose mass transfer equipment for catching corrosion products with permanent passivation of the surface contacting with liquid metal.

4.4. CONCLUSION

— Sodium technology developed to the industrial scale for fast reactors has demonstrated the capability of existing methods and technologies to provide solutions of the majority of problems concerning future nuclear power. The possibility of continuous improvement of the technology already brought into commercial use ensures that such work will not require large additional expenses;
— Scientific and engineering approaches used in lead coolant technology are essentially clear, but their implementation calls for time and financial resources;
— Direct application of data gained on lead-bismuth alloy to lead coolant is difficult, and it should be noted, that further long term studies are required;
— Through the IAEA's TWG-FR, forums for exchange of information on national programmes, collaborative assessment, knowledge preservation, and cooperative research in areas agreed by Member States with fast reactor and ADS programmes are provided [83].

5. THERMOHYDRAULICS OF REACTOR CORE

High power rating and high temperature impose high requirements to justification of fast reactor thermohydraulics. Analysis of thermohydraulic issues assumes working out reliable hydraulic and heat transfer relationships. As a result, distributions of flow rate, velocity, and finally, fuel and core structure temperatures would be obtained. These data are required for evaluation of core integrity and mechanical behavior.

5.1. HYDRODYNAMICS

As a rule, hydrodynamics of Na and Pb is quite similar, since both sodium and lead are Newton liquids. Therefore, available data on hydraulic resistance of the reactor core incorporating pin bundles can be used for adequate evaluation of hydraulic friction loss in the core [84–87].

Local pressure drop in subassemblies (for instance due to spacing elements) can make significant contribution to the total pressure drop [88, 89]. It is unlikely that spacing elements used in fuel subassemblies (SAs) of sodium cooled reactors (wire, etc.) could be applied for lead cooled reactors. There are two reasons: the first one consists in different stagnation zones and various deposits in different liquid metals, the second is related to different vibration characteristics. Therefore, the available data cannot be applied for the new spacers under design. This would require special purpose experiments for not only determining hydrodynamics but also to test structure for vibration and friction/fretting.

The two most important points for the development of heavy liquid metal cooled reactors are:

— When studying hydrodynamics of lead-bismuth and sodium cooled reactors, it was discovered that coolant flows entering core from different loops do not mix with each other in the core. Similar effects are observed in LWRs. But designers and researchers had decided to consider the mixing to be full for reserve [90]. Until now, it has been impossible to equalize outlet temperature of the reactor core consisting of ductless Sas and even that of the adjacent Sas with wrappers (using local mixing devices); When carrying out studies on the models of lead-bismuth cooled reactor flow path (including full-scale model), it was found that in parallel with the toroidal eddies appearing at the core diagrid inlet, sufficiently stable horizontal eddies may arise near the bottom of the diagrid [90, 92]. These eddies result in reduction of local inlet flow rates and flow stability, thus facilitating local concentration of particles suspended in the core diagrid area and, hence, offering risk of their penetration into the core. These effects are essential in liquid metal cooled reactors as compared to water cooled reactors.

5.2. HEAT TRANSFER

The specific feature of liquid metal heat transfer is high temperature rise along the channel as compared to wall-fluid temperature difference value. Because of this, SA temperature behavior is not determined by heat transfer coefficient, but mainly by local temperature rise of liquid metal coolant depending in particular on local flow rate distribution. For instance, if non-uniform temperature distribution over the model bundle cross-section is not taken into account, significant deviations arise in heat transfer coefficients evaluation for the ‘infinite’ bundle.

In general, comprehensive studies have been made on liquid metal heat transfer of different metals (Hg, Na, NaK, Pb-Bi, Li, etc) [93]. Thermal contact resistance at the wall-liquid boundary causes the low heat transfer coefficient. Thermal resistance is influenced by surface wetting with coolant, presence of oxide films on metal surface, as well as deposits of oxides and other impurities. Thermal contact resistance can hardly be evaluated, only a possible upper limit can be specified [94].

Purification of liquid metal leads to almost complete elimination of thermal contact resistance, while saturation with impurities results in 1.5–2 times the reduction of heat transfer coefficient.

Thus, common a conclusion cannot be reached on the value of thermal contact resistance. As for lead, the available data must be checked.

It is clear that coolant must be purified. While this is not a problem for sodium, in case of lead one should bear in mind permanent oxide presence in the circuit taking into consideration corrosion activity of lead and lead-bismuth eutectic and the necessity of protective oxide films on the metal surface.

Lead cooled fast reactor (LFR) design assumes SA with fuel pins arranged in square lattice with large pitch-to-diameter ratio (s/d~1.4–1.5). In the SFR, more tight arrangement of fuel pins is adopted (s/d~1.1–1.18):

Reactor	BR-10	BOR-60	BN-350	BN-600	BN-800	Phénix	Superphénix
x=s/d	1.10	1.10	1.16	1.16	1.17	1.178	1.15

Both experimental and analytical studies of heat transfer in Hg, Pb-Bi, NaK and Na coolants have been carried out [93, 96]. Relationships for evaluation of heat transfer in triangular lattice bundle were obtained experimentally.

Heat transfer in square bundles is roughly estimated, therefore, special experiments are needed for verification. By now, there are no convincing methods to extrapolate to the square bundle of the data on temperature non-uniformity around fuel pin. As a first approximation, it can be assumed that values increase in the same way as in case of laminar flow. Unfortunately, this cannot be adopted for the peripheral pins whose non-uniformity is significant. In-pile and out of pile test results show that boiling occurring behind the blockage does not lead to fast propagation of failure in the fuel pin bundles, while the gas release taking place as a result of pin failure behind the blockage may cause more pin damages within the SA. Studies of various factors influencing temperature profile (flow rate, power rating, Reynolds and Peclet numbers, blocked fraction of flow cross-section area) call for carrying out additional comprehensive experimental and analytical studies [95, 96]. Special attention should be given to studies of thermal hydraulics of SA with porous heat generating blockages, initial dry-out conditions with boiling occurring behind the blockage and SA cooling limits.

As for local thermohydraulic characteristics, such as hot spot factors, temperature non-uniformity in the irregular arrangement of pins etc., the processes of inter-channel exchange are of crucial importance. Comprehensive studies of inter-channel heat and mass transfer in the bundles of smooth pins with ribs or wires provided a basis for development of calculation techniques [97, 98].

Mixing factors for wire-to-pin contact in the bundle of wire spaced fuel elements turned out to be higher with respect to the wire-to-wire contact. Counter-direction wires providing lateral coolant flows would be most effective [97].

The issue of spacing elements development for lead cooled SA is rather complicated because of high dynamic head, possible vibrations, etc. This is the subject of further detailed studies.

The item that rather high thermal conductivity would smooth temperature non-uniformity has not been confirmed. It is only true for stagnant liquid metals that are equivalent to the solid body. Experiments and evaluations indicate on the necessity of considering conjugate tasks of heat removal from fuel elements, i.e. taking into account properties of fuel elements.

As it was demonstrated by experimental studies, considerable temperature fluctuations take place in both coolant and channel walls. Preliminary estimations showed that such pulsation would not affect structural strength. However, this does not relate to the situation when two flows of different temperature are mixed. In this case, thermal stress can result in structure failure, which happened at Phénix intermediate heat exchanger (IHX) [99]. This problem cannot be resolved without implementing a comprehensive study programme, including analysis of thermal striping, experimental studies on computer code verification, development of codes and techniques for measurements of the fluctuations of the wall surface temperature, etc.

5.3. CONCLUSIONS

— Analysis of experimental results on sodium and sodium-potassium alloy (as modelling liquid for Pb-Bi) has served as the basis for an approximate evaluation of the main characteristics of velocity and temperature profiles in the fuel subassemblies with the blockage of single-phase flow, i.e. recirculation zone length, coolant flow distribution in SA, coolant temperature rise in the wake, etc. On the other hand, it is clear that currently available data provide only rough estimations of SA thermo-hydraulic characteristics in the case of blockage depending on the Reynolds number and blocked flow cross-section area;
— Results of in-pile and out of pile tests demonstrate that the coolant boiling occurring behind the blockage does not lead to fast propagation of failure within the pin bundle. Gas release taking place as a result of pin failure behind blockage may be the cause of more pin damages in the SA;
— Studies of various factors influencing temperature profile (such as flow rate, power rating, Reynolds and Peclet numbers and blocked fraction of flow cross-section area) calls for carrying out additional comprehensive experimental and analytical studies. Special attention should be given to the investigation of thermal-hydraulics of SA with porous power generating blockages, as well as to the initial dry-out conditions with boiling occurring behind the blockage and SA cooling limits;
— Pressure drop in the pin bundles can be evaluated using relationships available;

— Evaluation of local pressure drop in LFR core (spacing grids and other structural elements of SA) requires additional studies to be carried out on each modification stage;
— Hot spot factors associated with:
  (a) Uncertainty of coolant parameters distribution at the SA inlet;
  (b) Probable distortion of SA configuration during operation, that may impact LFR parameters, particularly in case of high values of pin bundle relative pitch;

must be taken into consideration.

6. COOLANT RADIOACTIVITY

6.1. SODIUM RADIOACTIVITY

The natural isotope of sodium is ^{23}Na (100%). Sodium neutron capture processes generate ^{24}Na with half-life time of 15 hours. There is (n, 2n) threshold reaction producing ^{22}Na with 2.6 years half-life time. ^{24}Na is the main isotope giving rise to requirement of protection against γ-radiation. Activation of sodium reaches equilibrium state in about ten years of the first cycle of its use and will never exceed this level.

Specific activity of the primary ^{24}Na (770 t)[1] and ^{22}Na is respectively about (8-10) × 10^{11} Bq/kg and 2 × 10^{8} Bq/kg. As estimated, upon 30 year-operation specific activity of ^{22}Na is (4÷10) × 10^{7} Bq/kg.

Cover gas is another source of SFR radioactivity. Primary gas activity is to considerable extent determined by impurities in sodium and activation of ^{40}Ar and ^{41}Ar. As a result of (n, p) reaction, radioactive ^{23}Ne with short half-life of 38 s is produced from ^{23}Na [100]. After reactor shutdown the primary circuit activity is mainly determined by ^{22}Na ($T_{1/2}$ = 2.6 years), ^{60}Co ($T_{1/2}$ = 5.3 years), ^{137}Cs ($T_{1/2}$ = 30 years).

6.1.1. Cleaning caesium radionuclide from sodium

The sodium cooled BN-350 fast breeder reactor started operation in 1972 in the city of Aktau, Kazakhstan, to provide heat and electricity to the local population. The reactor was shut down in 1999 and has been under decommissioning since then.

According to the sodium sampling data taken in 2001 [101], 2 years after reactor shutdown, the specific activity of ^{22}Na and ^{134}Cs radionuclides in the BN-350 primary coolant accounted for < 3% of the amount of ^{137}Cs radioactivity in the system. This means that ~259 000 GBq (7 000 Ci) of radioactive ^{137}Cs in the primary coolant sodium, was the dominant radiation source for subsequent shutdown and decommissioning activities. The ^{137}Cs concentration in the primary sodium was ~296 MBq/kg (8 000 μCi/kg) at that time. This level of caesium in the primary sodium would significantly increase the hazards and costs involved with placing the reactor facility in an industrially and radiological safe condition. This amount of caesium could also create risks to the public through potential system upsets and accidents, so it was decided to remove as much caesium as possible from BN-350 prior to removing sodium from the primary system. The ^{137}Cs removal would both reduce personnel radiation exposure and reduce radioactive waste after sodium processing. Thus, one of the first steps of the BN-350 decommissioning was cleaning ^{137}Cs from the primary sodium to decrease the radioactive dose load on personnel during draining and processing of the primary sodium.

A system for removing ^{137}Cs from the BN-350 primary sodium was designed, fabricated, installed, and operated. The caesium trap system, had been previously studied by several organizations [101–105]. In these systems, the adsorption increases with decreasing temperature and is greater on reticulated vitreous carbon (RVC) than on other forms of carbon. The essential function of the caesium-trapping operation was to remove caesium from the primary sodium and internal reactor surfaces and collect it in pressure vessels for interim storage and later

[1] Primary circuit BN-600.

disposal. Data and experience from EBR-II were used to size and design the trapping system. Operational measurements in BN-350 validated the design data and the modeling approach.

Caesium trapping was accomplished by pumping sodium from the primary circuit through a block of RVC within each trap and returning the cleaned sodium to the primary circuit. Caesium has a strong affinity for adsorption onto the surface of carbon of any form, and RVC has proven to be an effective caesium adsorption material in sodium at several sodium cooled reactors worldwide [102–105]. Seven traps containing RVC adsorbent were fabricated and installed one at a time for the BN-350 cleaning operations. About 255 300 GBq (6 900 Ci) (>98% of the system total) of caesium was trapped, and the ^{137}Cs specific activity was decreased by a factor of ~800 to a final concentration of 0.37 MBq/kg (10 μCi/kg).

Reference [101] describes how this project was realized, provides information about the BN-350 reactor, briefly describes the caesium trapping system design, reports the experimental results obtained during caesium trapping, and demonstrates how modeling was performed to describe the caesium trapping results.

6.2. LEAD AND LEAD-BISMUTH RADIOACTIVITY

Some researchers suppose that the use of Pb-Bi or Pb [106–111] as a coolant in a fast reactor could improve NPP reliability due to the elimination of fire hazard. However, when using Pb or Pb-Bi, attention should be paid to α-radioactivity polonium contamination, produced during reactor operation [112–114].

The bismuth (^{209}Bi$_{83}$, 100% abundance in natural bismuth) is the most important impurity for the production of short-term induced activity in the lead and Pb-Bi alloy. The activation chain is as follows:

$$^{209}Bi_{83} + \mathrm{n} \rightarrow \begin{matrix} ^{210m}Bi_{83} + \gamma, \\ ^{210}Bi_{83} + \gamma, \end{matrix} \quad (3)\ (4)$$

$$^{209}Bi_{83} \xrightarrow[5.012\ \mathrm{days}]{\beta^-} {}^{210}Po_{84}, \quad (5)$$

and

$$^{209}Po_{84} \xrightarrow[138.4\ \mathrm{days}]{\alpha^+} {}^{206}Pb_{82}\ \text{(stable)}. \quad (6)$$

The isomer ^{210m}Bi has a very long half-life of 3.3×10^6 a. This isomer emits an alpha particle to form ^{206}Tl, which in turn emits a β^- to form the stable isotope ^{206}Pb. The cross-section of reaction (1) is about one-half of reaction (2). However, even in this case, the cross-section of reaction (2) is rather large in all energy ranges.

The ^{210m}Po$_{84}$ is formed by the activation of bismuth impurity in lead. However, even if bismuth-free lead is used, the ^{210}Po$_{84}$ isotope will be formed after an extended operation time of a reactor because of the bismuth generation by the (n, γ) reaction in the ^{208}Pb$_{82}$ (52.3% abundance) isotope. The activation chain of polonium generation in lead is as follows:

$$^{208}\mathrm{Pb}_{82} + \mathrm{n} \rightarrow {}^{209}\mathrm{Pb}_{82} + \gamma, \quad (7)$$

and

$$^{209}\mathrm{Pb}_{82} \xrightarrow[3.25\mathrm{h}]{\beta^-} {}^{209}\mathrm{Bi}_{83}. \quad (8)$$

Thus, the ^{210}Bi$_{83}$ and the ^{210}Po$_{84}$ isotopes can be generated from ^{209}Bi in this case as well following reactions (2) and (3). Direct polonium generation in lead is also possible, namely, in the ^{209}Pb$_{82}$ isotope generated in reaction (5). The activation chain is as follows:

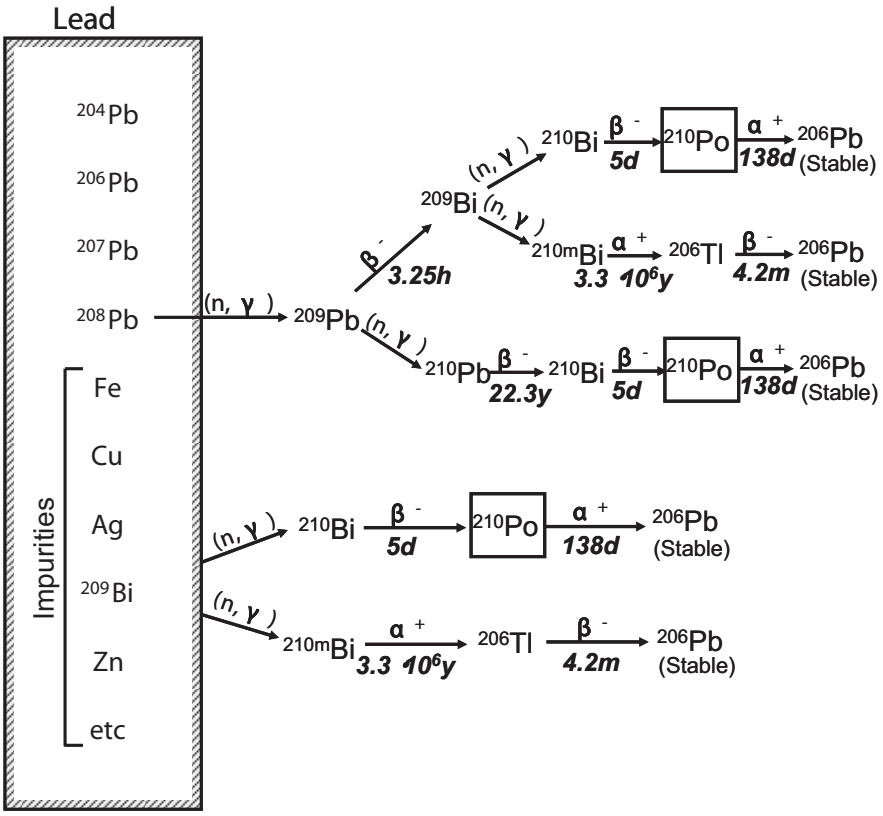


FIG. 1. The scheme of polonium buildup in lead.

$$^{209}Bi_{83} + \mathrm{n} \rightarrow {}^{210}Pb_{82} + \gamma, \tag{9}$$

$$^{210}\mathrm{Pb}_{82} \xrightarrow[21.3\ \mathrm{yr}]{\beta^-} {}^{210}\mathrm{Bi}_{83}, \tag{10}$$

and

$$^{210}\mathrm{Bi}_{83} \xrightarrow[5.012\ \mathrm{days}]{\beta^-} {}^{210}\mathrm{Po}_{84}, \tag{11}$$

However, the influence of activation chain reactions (7), (8), and (9) on the polonium buildup in the lead is negligible compared with that by activation chain reactions (5) and (6) because of the following reasons:

— The short half-life of the $^{209}Pb_{82}$ isotope, namely, 3.25 h.
— The long half-life of the $^{210}Pb_{82}$ isotope, namely, 21.3 a.

This half-life time is even comparable with the reactor, operation time of say 15 to 25 a. Thus, according to rough estimations of this study, the probability of reaction (7) is expected to be 10^{-5} to 10^{-7} times that of reaction (6). Therefore, obviously only a tiny fraction of $^{210}Po_{84}$ comes via $^{210}Pb_{82}$, and it seems reasonable to neglect activation chain reactions (7), (8), and (9) as the source of polonium activity in Pb. The scheme of polonium buildup in both the lead and bismuth is summarized in Fig. 1.

The main gamma-emitters in the coolant with operating nuclear reactor are radionuclides of lead: ^{207m} Pb, ^{204m}Pb and ^{203}Pb generating because of threshold nuclear reactions. Lead-204m is of practical importance with activity level (1–2) × 10^7 Bq/cm^3 and ^{203}Pb with activity (2-5) × 10^6 Bq/cm^3. A few days after reactor shutdown gamma activity of the coolant is caused, mainly, by ^{110m}Ag, ^{124}Sb, ^{60}Co being the products of activation of

impurities and corrosion products. Because of the large coolant density (~10.2 g/cm^3), it shows sufficiently high ability for self-absorption of gamma-radiation and the coolant circuit requires relatively modest radiation protection. During NPP operation, mainly ^{210}Bi gives the beta-activity of lead-bismuth coolant.

Levels of beta activity in the circuits of nuclear power reactors with Pb-Bi liquid metal coolant reach (2–5) × 10^9 Bq/cm^3 [114]. The characteristic feature of radioactive Pb-Bi liquid metal coolant is generation of radiotoxic α-activity (polonium-210) that requires the specific safety measures in performing different repairs and other works associated with opening of the primary circuit on the plants with this coolant as well as in the case of emergency leaks of coolant. For more than thirty-year period of operation of nuclear reactors with Pb-Bi coolant considerable experience in insurance of radiation safety during this coolant handling has been accumulated. Physical and chemical mechanisms of forming polonium gas-aerosol activity above melt, contaminations of equipment surfaces and working rooms, metabolism ^{210}Po in the human organism, a degree of radiobiological hazard of this radionuclide were studied.

Metallic polonium shows high volatility depending greatly upon temperature. During evaporation in air, polonium forms aerosols at the expense of condensation of sublimating polonium or its volatile compounds on the smallest hard particles, dusts and moisture drops presenting in air. Aerosols being formed show multi-dispersion character with a wide spectrum of particle dimensions. Their precipitation is accompanied by surface contamination of adjacent areas of rooms.

Although the rate of production of ^{210}Po by this two-step process is much lower (1000 times), the fraction of polonium migration out of the coolant is high (by 100 times) because of higher temperature of coolant [113–114].

In Pb-Bi cooled reactor, Po is produced basically in the reaction on Bi nuclei: Its α-activity (E = 5.3 MeV) in the alloy can be equal to (4–40) × 10^{10} Bq/kg.

In normal operation, when primary circuit is gas tight, Po is not dangerous. The hazard is caused by depressurization of the primary circuit during repair, refueling or coolant leaks.

Due to the short half-life (138 days) of polonium, α-activity reaches saturation, its decay after reactor shutdown being relatively rapid. Presence of α-activity of high energy (5.3 MeV) and Po toxicity may cause problems related to reactor maintenance.

The main reasons of hazard are α-activity of Po aerosols present in the air and surface contamination due to aerosol deposits. In order to exclude unacceptable contamination by Po, it is necessary to improve technological culture of Pb-Bi coolant handling and take radiation safety measures.

A number of engineering and organizational measures ensures normalization of a radiation situation under the conditions of contact with Pb-Bi coolant containing polonium. These measures include: baffles installation with local air suction in the areas where works are made; application of individual protective means; gas-electric welding minimizing in the contaminated areas; decontamination of the contaminated surfaces and putting special protective polymeric coatings on it; application of special filters in ventilation systems.

Based on Russian operating experience with Pb-Bi cooled reactors, it is concluded that the combination of these measures reliably excludes polonium dispersal avoids any over-exposure of the operating personnel.

6.3. CONCLUSIONS

- Radioactivity of Na and cover gas in SFR is well known on the basis of large operation experience gained in different countries. There are reports about successful design and operation of a system to remove highly radioactivity caesium from sodium coolant;
- Coolant radioactivity in LFR is mainly determined by α-activity of ^{210}Po. Intrinsic α-activity of Pb, determined by the ^{204}Pb isotope with half-life time of ~67 min, is about 3.7 × 10^{10} Bq/L;
- In order to use lead in FR, purification technique should be developed for antimony removal from lead, operating experience on lead cooled facilities should be gained, and methods of lead purification from Polonium and radioactive corrosion products should be studied and developed;
- Comparative assessment of accidents in SFR and LFR are difficult to be performed because of different initial conditions.

7. LIQUID METAL COOLED FAST REACTOR TECHNOLOGY DEVELOPMENT

7.1. INTRODUCTION

LMFRs have been under development for more than 60 years. Twenty LMFRs have been constructed and operated. Five prototype and demonstration LMFRs — BN-350 in Kazakhstan, Phenix and Superphénix in France, prototype fast reactor in the UK, BN-600 in the Russian Federation — with electrical output ranging from 250 to 1200 MW(e) and large scale (400 MW(th)) experimental fast flux test reactor FFTF in the USA have gained nearly 110 reactor-years. In total, LMFRs have gained nearly 310 reactor-years of operation. In many cases the overall experience with fast reactors has been extremely good, the reactors themselves and, more frequently, particular components, showing good performance well in excess of design expectations. They have as well very attractive safety characteristics, resulting largely from being low pressure system with large thermal inertia and negative power and temperature reactivity coefficients.

The development of fast power reactors has been delayed in countries characterized by advanced market economics, relatively low primary energy consumption growth and unlimited availability of oil and gas. Moreover, alternative LMFR application is being developed in Europe, Japan and the USA, namely transmutation of long lived nuclear waste and utilization of surplus plutonium.

In Western Europe, with France leading role, the European Fast Reactor (EFR) design has been completed. This synthesizes the extensive experience form France, Germany and the UK of large pool type oxide fuelled reactors. One of the outstanding achievements of the EFR programme has been the firm and reliable cost estimates. Construction of an EFR reactor type may not be possible in the near future, but a well validated way forward to commercial utilization of fast reactors has been established. This way is generally consistent with other studies, and indicates that the goal of competitive fast reactors may accessible.

In Japan, the fast reactor development programme demonstrates a national nuclear fuel recycling programme, as it is stated in the national long term plan. The experimental fast reactor Joyo has shown excellent performance over 20 years. The prototype reactor Monju (280 MW(e)) was stopped due to the leak in the non-radioactive secondary circuit, which was restarted on 6 May 2010.

Russian experience gained on operation of experimental, prototype and semi-commercial LMFRs (BR-10, BOR-60, BN-350 and BN-600) is rather good. The BN-350 LMFR was shut down in 1999 after a long time of reliably supplying fresh water and energy to Aktau, Kazakhstan (a large city and industrial centre in the desert that had quickly grown following the construction of the reactor) and the neighboring region. Stable operation of semi-commercial reactor BN-600 with nominal power output of 600 MW(e) (commissioned in 1980), as well as construction of the largest fast reactor SPX in France, and have been relevant milestones in mastering LMFR technology. Current efforts with regard to LMFRs in the Russian Federation are directed towards improving safety margins and economics. All these efforts will take some time, where it is foreseen to use the fast reactors for energy production, as well as Pu and MA utilization. In the Russian Federation, detailed design of commercial fast reactor BN-800 (800 MW(e)) was completed, and license was issued for its construction on Belyarskaya NPP sites.

India is now in a position to confidently embark on the construction of a 500 MW(e) prototype LMFR in this decade. Development of technology for manufacture of key components and of low doubling time fuels and structural materials capable of sustaining high neutron fluence has already been initiated and work is going on satisfactorily. The central facility in the unit is the test LMFR of 40 MW(th)/13 MW(e), which is operational from 1985. Presently per capita electrical energy consumption in India is about 420 KW·h/a, which is roughly one fifth of world average. Therefore, to reach a level of world average it would be required that the electricity generation capacity should be raised from 100 GW(e) (present) to at least 500 GW(e) considering the population increase. Coal and hydropower are the sources of electricity now in India and hydro may have reached saturation for all practical purposes because of remoteness of untapped major source and environmental issues. The potential availability of coal is estimated to be about 200 billion t (bt) and at a capacity of 500 GW(e), it may not sustain for ~70 a. The only major alternative for India is nuclear energy.

The potential availability of natural uranium in India is estimated to be around 50 000 t, which can be considered negligible (about 1 bt of coal equivalent (btce)) if utilized in an once through cycle and the capacity also

will be limited to about 10 GW(e). If the same U along with the Pu generation in pressurized heavy water reactor (PHWR) is invested in FBR the potential resource would increase to 180 btce and the capacity also can be increased to 250 GW(e). FBRs thus form the second stage of the nuclear power for India. FBRs in India will be deployed on U-Pu cycle for rapid growth of nuclear power capacity and generate enough nuclear fuel simultaneously for deployment of Th-U cycle in the third stage of the programme. Thorium is abundantly available in India and the resource level of 320 000 t is estimated to be equivalent of 1000 btce. It should be noted that in other South and East Asia countries with few indigenous fossil fuel and limited indigenous uranium ore reserves there is the same situation concerning effective nuclear fuel breeding by LMFR.

In China, an experimental fast reactor 25 MW(e) (65 MW(t)) LMFR is under construction.

The sodium cooled integral fast reactor (IFR) project at the Argonne National Laboratory (US) has been based on the use of a ternary alloy U-Pu-Zr fuel for its core loading. General Electric integrated the IFR concept into a full plant design of an advanced liquid metal reactor (ALMR) of 300 MW(e). The plutonium is not separated from the higher radioactive actinides; these are recycled together in the reactor and never leave the reactor site. All IFR designs are based on full actinide recycling using a pyro-chemical processing and fuel fabrication plant co-located with the reactor complex. In fact, by using a U-Pu-10%Zr alloy and ferritic-martensitic HT9 cladding as duct, a burnup of about 20% has been achieved in the USA LMFR EBR-II. All irradiation results in the EBR-II and FFTF have demonstrated reliable performance of metallic fuel and the potential to achieve high burnup in prototypical fuel elements cooled by liquid sodium.

The ALMR was designed to provide high reliability for the key safety functions, including reactor shutdown, heat removal, and containment. These functions can be achieved by passive means (thermal expansion, temperature effects on neutron absorption, natural circulation of the sodium coolant, and natural air circulation).

Significant experience of mixed oxide (MOX) fuel elements development for LMFRs has been gained in many countries over the last years.

The first fuel assemblies containing plutonium recycled into new MOX fuel were loaded in the Phénix reactor core in January 1980 and in the PFR (prototype fast reactor) reactor core in June 1982, thereby closing the reactor's fuel cycle. The cumulative amounts of FR fuel reprocessed are about 30 t in France and about 25 t in the UK. In March 1994, the Dounrey reprocessing plant had treated a total of over 23 t of MOX spent fuel with the highest burnup in the fuel (about 18%), providing the technical feasibility of MOX fuel reprocessing via a Purex-cycle, with recovery of over 99.5% of the plutonium. This high recovery was also reflected in the low amounts of plutonium in the liquid and solid waste streams from the plant. The amount of radioactivity discharged to the environment was always about an order of magnitude less than the licensed limits. The early development of experimental and prototype LMFRs was largely conducted by individual nations. However, for advanced LMFRs, international cooperation plays an important role and the IAEA promotes this objective for development. For R&D incorporating innovative features, international cooperation allows pooling of resources and expertise in areas of common interest.

7.2. OBJECTIVES AND CHALLENGES FOR THE DEVELOPMENT OF ADVANCED LIQUID METAL COOLED FAST REACTOR

The objectives are to achieve better economics than with alternative systems and to find optimal solutions for the back end fuel cycle and achieve a high degree of safety and reliability. Achieving a very high degree of safety requires the design of a reactor system, which excludes any radiological impact that would require evacuation of the public, not only during normal operation but also in the case of an accident. The following features are important among others:

— The stability of the reactor core under all modes of normal and abnormal operating conditions must be assured, while minimizing excess reactivity and sodium void effects as the result of a strong negative power and reactivity feedback with increased temperature. Large margins of reactor coolant boiling at operating temperatures must be maintained. A low pressure system with large thermal inertia and sufficient safety margins are key requirements;

— Reasonably long operator grace periods must be assured, allowing keeping the coolant temperatures below boiling, and fuel pin cooling temperatures below prescribed limits without the need of rapid operator action. Heat removal from the core and the reactor under all abnormal conditions must be assured;
— Take advantage of a passive system to provide safety-related functions without reliance on operator action or on external mechanical and/or electrical power signals or forces;
— Minimize the burden on the operator of a nuclear power plant. Extensive consideration is being given to human–machine interfaces.

Advanced LMFRs are designed on the basis of previous experience but taking into account the aim of increasing safety and reliability margins. Notable examples of innovative safety characteristics are passive decay heat removal systems, inherently designed reactor shutdown, and stabilization, as the result of thermal and reactivity response characteristics of the reactor even under extremely unlikely accident conditions.

As to the argument that some LMFRs faced reliability issues, it should be noted that this would also be true for any other reactor line at the initial stage of development. Full industrial development of fast reactors is not yet complete. It is too early at the present prototype stage of development to assess LMFR technology and economic characteristics properly, particularly in view of the current situation with antinuclear environmentalist goals to discourage investors from entering this field with required resources. Other reactor technologies, including water cooled reactors, achieved high reliability and low generating costs when their large scale introduction had taken place. Arguably, this will also happen in the case of LMFRs.

The current view is that the technologies of sodium coolant and mixed oxide fuels are largely mastered and large prototypes and demonstration LMFRs have been built. They have clearly demonstrated that a LMFR is capable of sustained reliable contribution to an electricity supply system. The worldwide investment already made in the development and demonstration of LMFR technology exceeds $50 billion. With regard to LMFR, design approaches, the accumulated knowledge on materials, thermohydraulics and mechanical science indicates, that a substantial decrease in investment costs together with better assurance that safety margins are effectively maintained might be provided by an accurate analysis of actual transients, as well as by an optimum reactor and auxiliary systems design. Perhaps there is only one disadvantage inherent in the LMFR coolant liquid sodium, namely that it interacts chemically with air and water/steam. Therefore, providing integrity of the sodium circuits is the most important requirement of the LMFR design, construction and operation tests. The solution of the problem of the reliable elimination of coolant leaks is determined by the application of experimentally and highest quality control of all stages of LMFR components manufacture.

Experience has shown, design, construction and comprehensive testing of LMFR plants are essential to establish efficient criteria and rules and effective LMFR technology. This stage has been reached in the world. For example, in the Russian Federation, an experimental reactor BR-10, an experimental 15 MW(e) NPP; BOR 60, a prototype NPP; BN-350 (presently in Kazakhstan) and a semi-commercial reactor BN-600 have been operating respectively for more than 40, 30, 25 and 20 years, providing invaluable information on LMFR technology. That is why the BN-600 plant has been running successfully for 25 years with an overall lifetime load factor of ~75%. This success was achieved because in the design and construction of the plant, manufacture of the equipment and operation of the plant past errors were not repeated and good design solutions were incorporated.

The comprehensive operational experience with LMFRs BN-350, Phénix, PFR, BN-600, Superphénix and Monju has shown that, if plant components have been designed and manufactured without errors and representative specimens or models have been tested prior to installation, reliable operation can be ensured during the whole operational life [115–116]. The corrosion inertness of sodium to stainless steel, the near atmospheric operation pressure, the use of ductile structural materials, and the reliable heat removal by a coolant having no phase change, imply that there should be nothing to provoke loss of the sodium system integrity in a LMFR.

Table 11 shows the summary of location, key dates and major features of all the thermal power producing fast reactors constructed worldwide.

TABLE 11. SUMMARY OF LOCATION, KEY DATES AND MAJOR FEATURES OF ALL THE THERMAL POWER PRODUCING FAST REACTORS CONSTRUCTED WORLDWIDE

Reactor name	Country	Location	Criticality	MW(th)	MW(e)	Fuel	Cooling system
EBR-I	USA	Arco-Idaho	1951	1.2	0.2	U	1 loop
BR5	Russian Federation	Obninsk	1958	5.9	—	PuO_2/UC	2 loops
DFR	UK	Dounreay	1959	60	15	U-Mo	24 loops
Enrico Fermi	USA	Detroit	1963	200	61	U-Mo	3 loops
EBR-II	USA	Arco-Idaho	1963	62.5	20	U- Zr	pool
KNK-II	Germany	Karlsruhe	1977	58	20	PuO_2-UO_2	2 loops
Rapsodie	France	Cadarache	1972	40	—	PuO_2-UO_2	2 loops
SEFOR	USA	Arkansas	1969	20	—	PuO_2-UO_2	1 loop
BOR-60	Russian Federation	Melekess	1968	55	12	PuO_2-UO2	2 loops
BN-350	Kazakhstan	Chevchenko	1972	750	150	UO_2	6 loops
Joyo	Japan	Oarai	1977	*	—	PuO_2-UO_2	2 loops
Phénix	France	Marcoule	1973	563	255	PuO_2-UO_2	pool
PFR	UK	Dounreay	1974	650	250	PuO_2-UO_2	pool
FFTF	USA	Hanford	1980	400	—	PuO_2-UO_2	3 loops
BN-600	Russian Federation	Beloyarsk	1980	1470	600	UO_2	pool
Superphénix	France	Creys-Malville	1985	2990	1242	PuO_2-UO_2	pool
SNR-300	Germany	Kalkar	—	762	327	PuO_2-UO_2	3 loops
Monju	Japan	Tsuruga	1994	714	280	PuO_2-UO_2	3 loops

* 50 and 75 for Mk-I, 100 for Mk-II, 140 for Mk-III

7.3. FEATURES OF LIQUID METAL COOLED FAST REACTORS

7.3.1. Physics

Sodium cooled fast reactor design (LMFR) with the internal breeding ratio ~1.0 features:

— Near zero burnup reactivity swing;
— Strong negative temperature reactivity effects;
— Small power reactivity effect (owing fuel high heat conductivity);
— Sufficiently small hot to cold reactivity swing;
— Possibility of coolant temperature rise changes in wide range.

LMFRs provide a potential to design a nuclear system in which the power would eventually passively adjust itself due to self-regulate heat production and heat removal. In the correct reactor core design, reactor plant would be passively protected against nearly all control system failures and operator errors. Low moderation and absorption of neutrons in lead gives benefit in the breeding ratio. Partially owing to this, core breeding ratio value equal to one can be achieved in lead cooled reactors with nitride fuel. However, use of lead coolant implies further research and development activities.

In the case of Pb, pressure drop in the core is much higher (about seven times) than for sodium with other conditions being equal (reactor power, coolant flow cross-section area in the core, coolant temperature rise and fuel element length). This pressure drop increase is caused by lower thermal capacity of lead as compared to that of sodium, and cannot be compensated by its higher density. A countermeasure to decrease core hydraulic resistance, flow cross-section area in the lead cooling core should be increased by means of both reduction of volumetric fuel fraction and increase of the core diameter. That is why overall core dimensions of 300 MW(e) BREST-300 lead

cooled reactor are rather large: D/H = 2.3/1.1 m with less fuel volumetric fraction (0.23–0.32) while in the sodium cooled 800 MW(e) BN-800 reactor (LMFR), this ration is D/H = 2.5/0.88 m, fuel volumetric fraction being equal to ~0.4. Simple extrapolation made for BREST type lead cooled reactor of 800 MW(e) power shows that overall dimensions of its core would be D/H = 3.7/1.1 m. Analysis has shown that owing to increase of the BN-800 type reactor core diameter from 2.5 m to 3.7 m the following benefit can be achieved:

— Coolant velocity in the core becomes about two times lower causing almost five time decrease of nominal pressure drop value (down to ~1 atm), thus providing significant (up to 10%) increase of natural circulation flow rate of the coolant required for passive removal of the reactor decay heat under abnormal operating conditions;
— Fuel element diameter and fuel volumetric fraction in the core can be increased, assuring BR_{core} ~1 even in sodium cooled reactor (on condition that higher density nitride fuel is used), in spite of elimination of both radial and axial fertile blankets.

Studies made in the Russian Federation have confirmed the possibility of achieving BR = $BR_{core} \cong 1$ with volumetric fraction of nitride fuel in the core of sodium cooled 800–1600 MW(e) reactor typical of LMFRs, its core overall dimensions being much lower than those of lead cooled reactor. The above considerations confirm the fact that comparison of physical and economical characteristics of the reactors with different coolants should be made based on identical input data, such as power rating, as well as the fuel type. Careful comparison does not reveal any advantages of lead cooled reactors as compared to LMFRs from the standpoint of achievement of $BR_{core} \sim 1.0$ value and assurance of the reactor decay heat removal using passive means under abnormal operating conditions.

Comparative evaluation of sodium and lead coolant natural circulation flow rates in the reactors of similar design has shown that sodium flow rate is higher as compared to that of lead because of almost three times higher thermal expansion coefficient and over ten times lower density of sodium. Therefore, the problem of emergency heat removal by natural circulation in BREST type reactors can only be solved by decreasing power densities and increasing coolant flow cross-section area in the core that results in the increase of fuel elements pitch and core diameter. As it has been shown by the analysis, even partial application of these measures (for instance, increasing fuel elements pitch with BR_{core} being kept at the level of ~1.0 in fast reactors with sodium coolant and high density fuel) would assure more efficient natural flow as compared to that in BREST type reactors. Sodium coolant is obviously advantageous from the standpoint of safety assurance in the mode of emergency heat removal from the reactor. Drawbacks of lead coolant used by its thermophysical characteristics are only partially compensated by its low neutron moderation and absorption, but cannot be deleted. In addition, it should be noted that sodium cooled reactor is nuclear technology system, having maximum flexibility extent; i.e. it is capable of assuring the following operation modes in accordance with user requests:

— Fuel breeding (BR > 1 with T_2 maintained at the acceptable level);
— Fuel self-supply (BR ~ 1.0);
— Controlled plutonium buildup (BR < 1).

Thus, sodium cooled reactors as well would fully come up to the concept of development of nuclear technology systems having no enriched uranium, declared at the UN Summit (Initiative of the Russian President, announced at the UN Millennium Summit on 6 September 2000). Moreover, no plutonium stocks will exist outside LMFR with attached plant for fuel reprocessing and fabrication (e.g. similar to that developed for the IFR).

7.3.2. Economics and experience

Along with the physical and thermo-hydraulic characteristics, the economic criterion is the major ones in the comparison of reactor concepts. The electricity generating costs including investment, fuel and operational and maintenance (O&M) costs of WWER (water cooled water moderated power reactor), LMFR and lead cooled reactor BREST have been compared for normalized power of 600 MW(e) (Table 12).

Taking into account the status of components cost, the energy generating costs can be calculated from practical data of LWR and LMFR, and theoretical/computational of BREST. Moreover, the possibility to increase the thermal efficiency of compared LMFRs to the level of BREST, the conclusion can be as follows: all three types

TABLE 12. COMPARABLE PARAMETERS OF DIFFERENT POWER PLANTS

Name and dimensions of the parameter	SVBR-75/100	WWER-1500 [116]	VER-1000	BN-1800 [83]	HEPP* with PGU**-325 Steam-gas installation	BREST-1200
Set up power of the power-unit, MW(e)	1625	1550	1068	1780	325	1296
The number of the units at the plant	2	2	2	2	10	2
Electric power necessary for plant's own needs, %	4.5	5.7	6.43	4.6	4.5	5.15
Efficiency of the net plant (power unit), %	34.6	34.4	33.3	43.6	44.4	43.9
Capital investment in the industrial construction of the plant, $/kW (price of 1991)	661.5	680	819.3	860	600	690
Design cost of produced electricity, cent/kW·h (price of 1991)	1.46	1.62	2.02	1.6	1.75	1.23

* Heat electric power plant
** Steam-gas installation

of reactor are comparable in investment costs. Of course, a convincing economic analysis of any new reactor type can be made only on the basic of profound level in R&D and construction of a prototype. It is unlikely, that the costs of generating power from lead cooled fast reactors can be lower than from a comparable water and sodium — cooled reactors. Many lessons have been learnt from construction experience of LMFRs, the most recent being Superphenix in France and BN-600 in the Russian Federation, which have provided a wealth of information allowing the simplification and optimization of future plant design. As a result, the design work on BN-800, and BN-1800 (Russian Federation) has achieved substantial investment cost reduction. It should be pointed out that the drawbacks inherent in the lead and lead-bismuth reactor coolants give rise to the following possible problems:

— Damage of components and fuel elements caused by high corrosivity of heavy metals;
— Freezing of lead in the steam generator in case of failure of high pressure feed water heaters in the turbine plant and supply of low temperature feed water (150–170°C below the melting point of lead) from low pressure feedwater tank (LPFT) to SGs (it is for the first time in the power engineering practice, that melting point of proposed coolant is higher than the temperature of water in LPFT);
— Problems concerning repair and maintenance work owing polonium hazard and remotely controlled refueling operation in lead cooled reactor carried out at rather high lead temperature (over 400°C);
— Blockage of coolant flow cross-section in the fuel subassembly caused by the products of water/steam interaction with coolant in SG, as well as solid inclusions and structural material corrosion products suspended in the heavy coolant;
— Problems with production by heavy metals of long lived isotopes.

The elimination of the drawbacks listed above can be obtained by selecting/developing appropriate structural materials, by innovative design and technological approaches and by demonstration of the operational reliability of two-circuit supercritical SG. If this will occur, than the most important advantage of lead and lead-bismuth cooled reactors will be the possibility to eliminate the safety concern of LMFR caused by radioactive sodium leaks and its chemical reactivity with air and water. All other supposed advantages of fast reactor with heavy metal coolant, namely:

— Minimum reactivity excess (BR_{core} ~1);
— Transmutation of actinides and long living fission products;

— Proliferation resistance by elimination of fertile blankets and on-site fuel reprocessing and fabricating;
— Safely termination of accidents on the basis of passive mechanism;
— Economical competitiveness.

have been confirmed in practice and justified within the framework of advanced designs of sodium cooled fast reactors (BN-800 (Russian Federation), ALMR (USA), and EFR (Western Europe)). In the desire to do something immediately about deterministic safety, there may be immediate calls for construction of prototype. Prototype construction is expensive. R&D is what is now necessary to select a nuclear system that promises the necessary characteristics.

Techniques to counter the lead and lead-bismuth coolant disadvantages are being investigated, but in spite of this work and some disadvantages of sodium, the consensus in favor of sodium remains strong. This is demonstrated by fact that even the Russian Federation, which possesses a comprehensive experience in heavy coolant technology has announced, “before lead cooled fast reactor BREST-300 is built, MINATOM (Ministry for Atomic Energy of Russian Federation) will first build a sodium cooled LMFR BN-800” (E. Adamov, NW 23 September 1999). Moreover, in the last few years sodium has been chosen in India, China, and Japan for the respective fast reactor development project. This is significant endorsement for sodium as a fast reactor coolant. Nuclear power has involved numerous groups of people in a few tens of countries in the process of creative engineering work. In this kind of activity, intellect is as important as technical potential. So all countries (the highly developed and the developing ones), contribute in addressing and finding solution of some problems with decisive contribution.

7.4. CONCLUSION

The purpose of advanced nuclear power systems development is to satisfy the world’s significant need for environmentally clean and sustainable energy for the future. The LMFR is still the only proven technology capable of providing nearly unlimited energy supplies from the world’s ample resources of depleted uranium, low-grade natural uranium and thorium. An additional promising mission for the LMFR is transmutation of long lived nuclear wastes. Advance LMFRs are being developed to meet the needs of increasingly demanding safety requirements, economic competitiveness and public acceptance.

Knowledge and experience of the 60 years R&D, design, construction and operation more than 30 experimental, prototype and full-scale fast reactors in 10 countries, experiences of NPP with more than 320 fast-reactor years in operation (and with worldwide investment near $50 billion): all these are very essential factors and basis for further development of liquid cooled fast reactors.

These unique experiences and knowledge should be taken into account when discussing and analyzing the shorter experiences of heavy metal cooled systems development, especially when we try to predict and to plan future needs for R&D of Pb-Bi and Pb technology and new advanced fast reactor development.

Because of the high cost of development of advanced reactors, Member States, which have relevant ongoing programmes, may find it attractive to cooperate on advanced reactor technology development. The IAEA’s programme on LMFR technology encourages international cooperation through technical information exchange and joint research. To make certain that its activities are desirable and useful, the IAEA’s development programme on advanced reactors is coordinated by the proper international working groups whose members are leaders on national programmes. Cooperation on fast reactors conducted under the aegis of the IW-GFR allows pooling of efforts in areas of common interest, thereby benefiting all participants. Excellent international cooperation for dealing with nuclear technology has been achieved in LMFR studies. Efforts are directed at finding optimal solutions for saving fuel resources and for radioactive wastes burning. Mixed oxide fuels and sodium coolant technology are now largely in hand. Further, R&D is needed to achieve competitive commercialization. All countries, not only highly developed but developing countries as well, should work together under the aegis of the IAEA.

8. MODULAR LEAD-BISMUTH COOLED SMALL SIZE FAST REACTORS

8.1. INTRODUCTION

As early as 1950, A.I. Leypunski investigated the possibility to design a lead-bismuth eutectic (LBE) cooled fast breeder reactor [117]. His analysis showed that the heat transfer capabilities of this heavy liquid metal eutectic were not suitable to allow designing cores with sufficiently high power densities needed to ensure very high breeding gain and very short plutonium doubling time which were the objectives of fast breeder reactor development at that time. This analysis resulted in the selection of sodium for the development of fast breeder reactors.

For the reasons outlined in Section 1, the rationale for fast breeder reactor development in the Russian Federation has shifted, with extremely short plutonium doubling time not being a design objective anymore. Therefore, LBE cooled fast reactors concepts are being revisited, especially in light of the Russian Federation experience made in this field with the design and operation of nuclear submarine reactors.

Nuclear power plants based on LBE cooled fast reactors are able to operate both in an open and closed nuclear fuel cycle (in break-even mode or with a small breeding gain).

It is worthwhile mentioning that quite a few projects aimed at developing and realizing LBE cooled fast spectrum systems are being implemented worldwide; e.g. in Belgium, China, Japan, the Republic of Korea and the USA. Some of the concepts developed in the past or under development nowadays are the following:

— In the Russian Federation, the small 75–100 MW(e) LBE cooled power fast reactor SVBR-75/100 [118–119];
— In Belgium, the 100 MW(th) multipurpose fast neutron spectrum MYRRHA facility, being designed to operate in both critical and subcritical mode [120];
— In Japan, a small power reactor cooled by lead-bismuth and fuelled with metallic and nitride fuel featuring extra long life time [118]; a 150 MW(e) lead-bismuth cooled fast reactor concept Pb-Bi cooled direct boiling water fast reactor (PBWFR)) featuring direct contact steam generators (‘steam-lift effect’ of lead-bismuth coolants) [119]; and a medium sized lead-bismuth cooled fast reactor [120], for which the rationale (lower breeding ratios in a Japanese scenario from 2030–2050 on) is presented in [121];
— In the USA, the modular lead-bismuth cooled STAR-LM (Secure, Transportable, Autonomous Reactor-Liquid Metal variant) concept featuring natural circulation [122–123] and the lead or lead-bismuth cooled Small, Sealed, Transportable, Autonomous Reactor(SSTAR) concept rated 10–100 MW(e) [124];
— In Japan and the USA, the lead-bismuth cooled encupsulated nuclear heat source (ENHS) concept, featuring natural circulation in both primary and intermediate circuits [125, 126];
— In China, a lead-bismuth cooled and thorium fuelled fast reactor concept [127];
— In the Republic of Korea, a lead cooled fast reactor dedicated to utilization and transmutation of long lived isotopes in the spent fuel [128].

Above mentioned studies cover, among other topics, also experimental material behavior investigations, especially corrosion resistance of various steels in the lead-bismuth eutectic (LBE) environment.

8.2. EXPERIENCE AND OPPORTUNITY TO USE LEAD-BISMUTH EUTECTIC ALLOY AS A FAST REACTORS’ COOLANT

Due to lead and bismuth properties, it is possible to vastly reduce deterministically the probability that certain severe accidents might occur.

A high coolant boiling point increases the reliability of heat removal from the core. Low pressure in the primary circuit reduces the risk of losing its tightness and allows reducing the thickness of the reactor vessel walls and the limitations imposed on the rate of temperature changes in compliance with thermo-cycling strength conditions.

Lead-bismuth eutectic coolant (LBE) reacts very slightly with water and air. Progress of the processes caused by loss of primary circuit tightness failure and steam generator inter-circuit leaks occur without hydrogen release and without any exothermic reactions. There are no materials within the core and reactor installation (RI) that release hydrogen because of thermal and radiation effects and chemical reactions with coolant. Therefore, the likelihood of chemical explosions and fires as internal events is virtually eliminated.

The use of LBE cooled reactor installations in nuclear power (NP) makes it necessary to consider at least two specific issues. One issue concerns the alpha-active ^{210}Po productions during irradiation and the second issue is related to the small scales of bismuth production and insufficiently explored bismuth resources.

With regard to the bismuth resources, it is worthwhile to mention that the available information on explored bismuth resources has not allowed use of LBE on a large scale. However, just recently the specialized MINATOM enterprises — OAO 'Atomredmedzoloto' and VNIPI of industrial technology — have carried out technical and economical investigations into the opportunity to organize large scale bismuth production in the Russian Federation and estimations of bismuth resources in the Commonwealth of Independent States (CIS). Only on the basis of the explored bismuth mines of the Chita region in the Russian Federation, it is possible to produce bismuth in quantities sufficient enough to put into operation ~70 GW(e) of LBE cooled FRs, at a rate of 1 GW(e) per year [129]. In addition, there are large bismuth resources in the North Caucasus and it is possible to put into operation ~300 GW(e) by using the bismuth mines of Kazakhstan. In compliance with an assessment made by Japanese experts, the available bismuth resources are ~5 million t [130]. It should also be highlighted that, according to a general geological and economical law, the quantity of the mineral ore increases as the squared cost that the consumer would be ready to pay for the resource. At the current world costs of bismuth, its contribution to the capital investment for construction of a large NPP based on FRs is ~1% [129]. For that reason, the technical and economical parameters of the NPP will not get noticeably worse even in the case of considerable cost increases of bismuth.

In the future, with diminishing bismuth resources, it will be possible to utilize as coolant non-eutectic lead-bismuth alloy with reduced bismuth content and higher boiling point. For example, when bismuth content in the alloy is reduced to 10% (5.5 times), the melting point is increasing from 125–250°C, still considerably lower than the melting point of pure lead (~327°C). Utilizing non-eutectic lead-bismuth alloy would not create any difficulties for reactor operation.

As development of LBE cooled FRs is based on the experience of its use in the nuclear submarine (NS) reactors, the gained experience needs brief description.

8.3. BRIEF DESCRIPTION OF EXPERIENCE WITH LEAD-BISMUTH EUTECTIC APPLICATIONS

In the early 1950s, nearly at the same time, the USA and the USSR launched their nuclear submarine reactor development programmes. Both countries developed two types of nuclear submarine reactors; pressurized water reactors and reactors cooled by liquid metal coolants (LMC).

In the USA, sodium was selected as LMC because of its thermo-physical characteristics in comparison with those of LBE. The ground based test facility prototype of the nuclear power submarine 'Seawolf' was constructed. However, operating experience accumulated from this prototype facility was not satisfactory. After several reactor installation accidents, it was decommissioned together with its compartment and replaced by a pressurized water reactor. R&D works on LBE mastering were also carried out in the USA. However, a selected approach to find the solution to the problem of structural materials corrosion resistance, as well as control and quality maintenance of the coolant did not give any positive results, and the activities were stopped.

In the Russian Federation, lead-bismuth eutectic (LBE) alloy was selected as LMC from the very beginning. For fifteen years, various organizations carried out works on mastering the LBE technology under SSC RF-IPPE scientific supervision. The LBE technology involves methods of providing corrosion resistance for structure materials, as well as control and maintenance of the coolant quality during the operation. The result of these works has revealed that to provide reliable reactor operation it is necessary to measure and maintain the value of a single parameter, namely of the concentration of oxygen dissolved in LBE within the indicated interval. This can be realized automatically. This allows to successfully mastering the lead-bismuth technology and was substantiated by many-year experience of nuclear power submarines [131].

The problem of ensuring radiation safety due to the formation of ^{210}Po was solved in the same way. The experience from operating the nuclear submarines has revealed that the measures implemented to provide radiation safety ensured that personnel present in the reactor compartment when accidental LBE spillage occurred, have not received not allowable doses. The same is true for personnel who took part in repair-reconditioning works [132]. Having carried out the researches and having analyzed the experience gained, the US experts made a conclusion that formation of polonium in LBE cannot hamper its use in NP in the future [133].

The paper published in the USA [134] summarizes the data of the retrospective analysis on mortality among the personnel (about 4500 men) involved in works with ^{210}Po from 1944-1972, and whose internal intakes of ^{210}Po were examined. The authors concluded that there was no correlation between the doses of internal intake caused by ~1 Sv (100 rem) of incorporated polonium and the death rate caused by cancer. For the examined personnel almost all trends characterizing the death rate caused by various cancer diseases were negative; i.e. the death rate was even less than that for the control representative groups of people who did not come into contact with polonium.

Along with the other problems, the problem of multiple 'freezing-unfreezing' of LBE was also solved [135]. When liquid LBE is transformed to solid state, and then it is cooled down to room temperature, elimination of damage to the reactor core is prevented by the contraction of the solidifying LBE and by its plasticity that is high enough at low strength in the solid state. For the safe 'unfreezing' of the reactor core, a special schedule for the temperature-time heating mode was developed and tested in large scale models and in an operating reactor installation. Altogether eight nuclear submarines with LBE cooled reactor core were constructed. The first experimental nuclear submarine of the Project 645 had two reactors. Each of the other seven nuclear submarines of the Project 705 (in terms of NATO – 'Alpha') had one reactor. Due to its speed parameters, this nuclear submarine was entered into the Guinness Book of Records. In addition, two full-scale ground reactor facilities-prototypes were constructed and operated in IPPE (Obninsk) and NITI (Science and Research Technological Institute) (Sosnovy Bor). The total operating time of the LBE cooled reactors was ~80 reactor-years. Arguably, it can be concluded that the LBE cooled reactor technology was demonstrated at industrial scale.

8.4. REACTOR INSTALLATION SVBR-75/100

The concept considered here is the small 75–100 MW(e) LBE cooled power fast reactor SVBR-75/100 [136–137].

8.4.1. Basic parameters of SVBR-75/100

The basic SVBR-75/100 parameters are shown in Table 13.

8.4.2. Description of SVBR-75/100

An overview of the SVBR-75/100 system arrangement is given in Fig. 2.

The primary circuit's system includes a core, SG modules, main circulation pumps (MCP), in-vessel radiation shielding and it is installed in the vessel of the reactor monobloc (RMB).

The secondary circuit's system includes SG modules, feeding water and steam pipelines, separators and autonomous cooling condensers.

The protection gas (argon) system includes gas system condensers, a membrane-protection device, a bubble device and pipelines.

The purpose of the heating system is to heat up the RMB prior to filling it with coolant and to maintain it in the hot state. It includes a system of pipelines installed between the RMB main vessel and the safeguard casing, along which the heating steam is conveyed.

Core cooling system includes mass exchangers, gas mixtures ejectors, and sensors of oxygen activity in LBE. The purpose of these systems is maintaining the LBE quality and inhibiting structural materials corrosion. The safety systems include a reactor emergency protection (EP) system, a system for localizing leaks in the SG, autonomous cooling system (ACS), and a passive heat removal system (PHRS). However, as it has been determined, the safety system is only a system of emergency protection of the reactor (i.e. EP). The remaining systems, namely the system of localizing leaks in the SG, ACS, and PHRS combine the functions of normal operation and accidents preventing systems.

TABLE 13. BASIC SVBR-75/100 PARAMETERS

Parameter		Value
Thermal power (nominal), MW		280[a]
Electric power, MW		101.5[a]
Steam production, t/h		580[a]
Steam parameters:	pressure, Mpa	9.5[a]
	temperature, °C	307[a]
Feeding water temperature, °C		241[a]
LBE temperature, °C:	at the core outlet	482[a]
	at the core inlet	320[a]
Average power density of the core, kW/L		140[a]
Average linear load of the fuel element, kW/m		~24.3[a]
Fuel (UO_2):	U-235 loading, kg	~1470[a]
	U-235 enrichment, %	16.1[a]
Number of control rods		37
The core lifetime, thousands of full power hours		~53
The time interval between threshold, years		~7–8
The number of SGs		2
The number of SG modules		2 × 6
The number of main circulation pumps		2
LBE volume in the primary circuit, m^3		18
Dimensions of the reactor vessel: D × H (diameter × height), m		4.53 × 6.92
Specific material consumption, kG/kW€		
	Steel	4.25
	Boron carbide	0.3
	Lead	0.88
	Bismuth	1.1

[a] The characteristics presented are those of SVBR-75/100 being a component of the modular NPP with two units of 1600 MW each [129]. The reactor installation SVBR-75/100 is able to generate steam with different characteristics. Under low steam parameters of the existing turbines at the Novovoronezh NPP, the effectiveness of the thermodynamical cycle is lower, and electric power output is reduced. If SVBR-75/100 is used as a component of other NPPs, these characteristics may be changed.

8.4.3. Arrangement of SVBR-75/100 equipment

The basic equipment of SVBR-75/100 is installed in a tight box-confinement of 11.5 m in height (see Fig. 3).

In the lower part of each box, a concrete vault for the PHRS tank is installed. The reactor monobloc is installed inside the PHRS tank and is fastened on the head ring of the tank roof. In the PHRS tank are also installed 12 immersed vertical heat exchangers in which heat from the PHRS tank water is transferred to the intermediate cooling water circuits.

The reactor core is not part of the reactor mono-bloc and is installed above the PHRS tank in the upper part of the box. It includes two steam separators and two ‘hung’ on them cooling condensers. The vertical position of the steam separators is chosen in such a way that the necessary natural circulation level in the secondary circuit be ensured at any time. The condensers of the gas systems are installed in the upper part of the box in the separate concrete compartment.

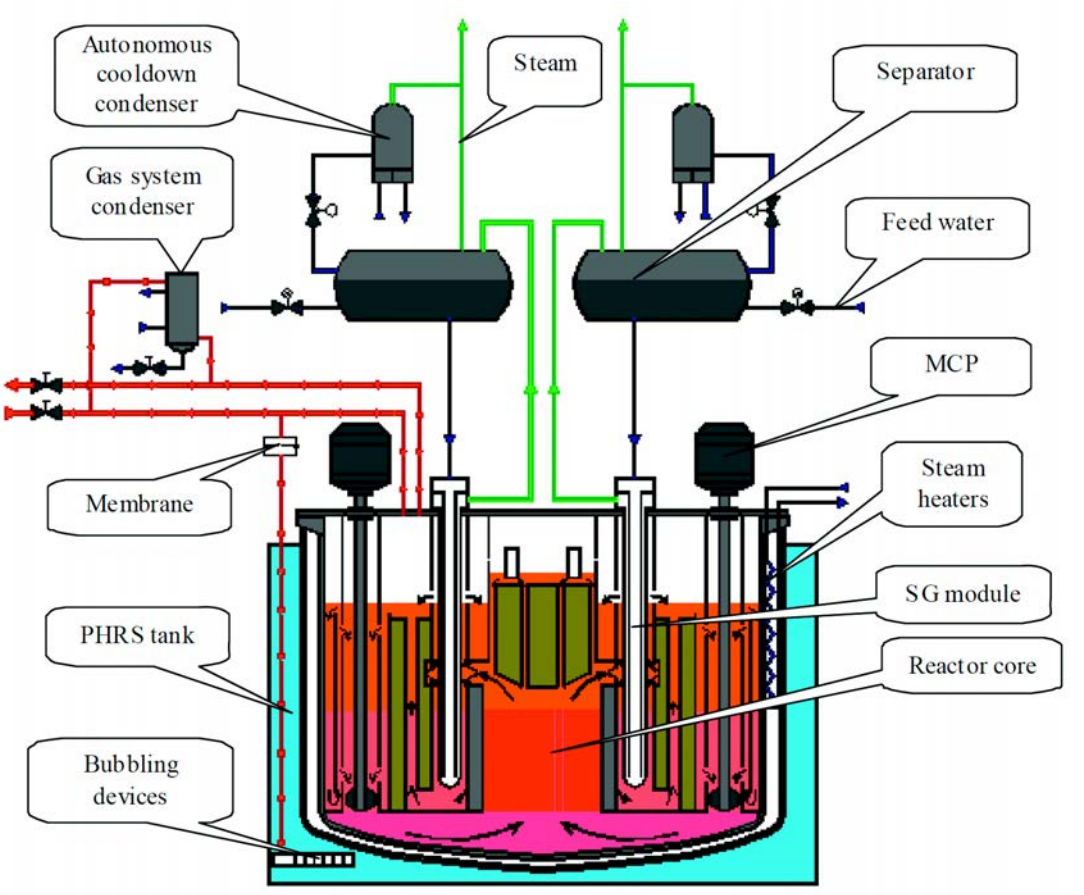


FIG. 2. SVBR-75/100 system arrangement.

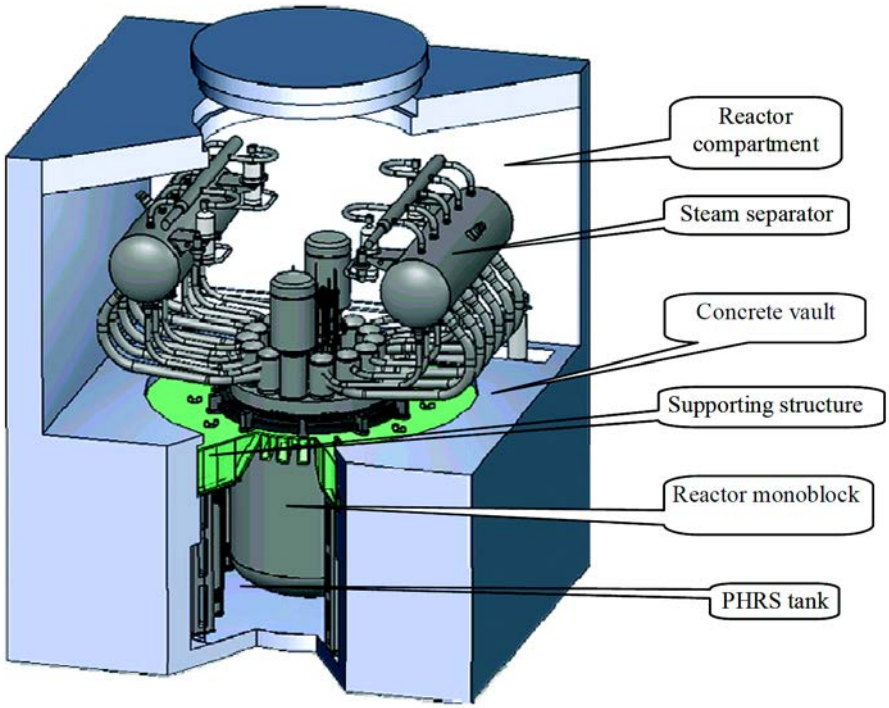


FIG. 3. SVBR-75/100 equipment arrangement.

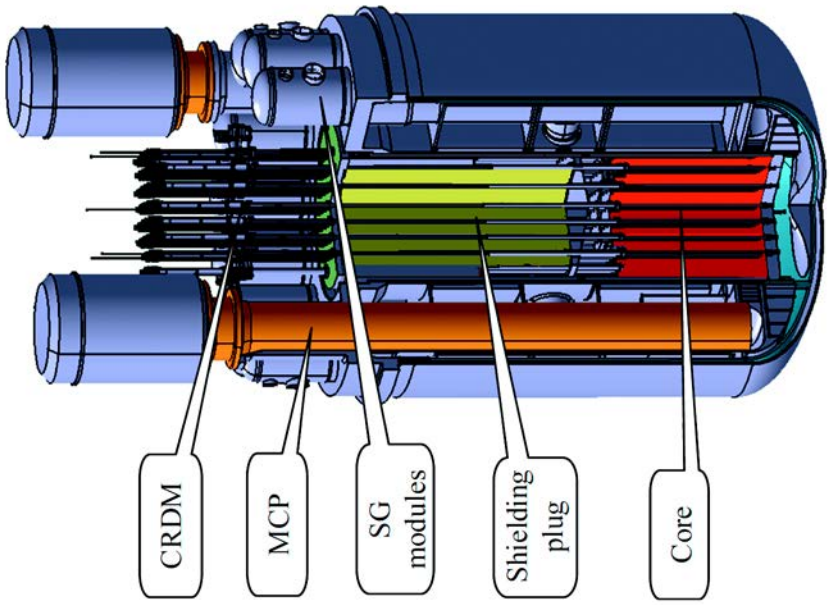


FIG. 4. Equipment arrangement in the RMB vessel.

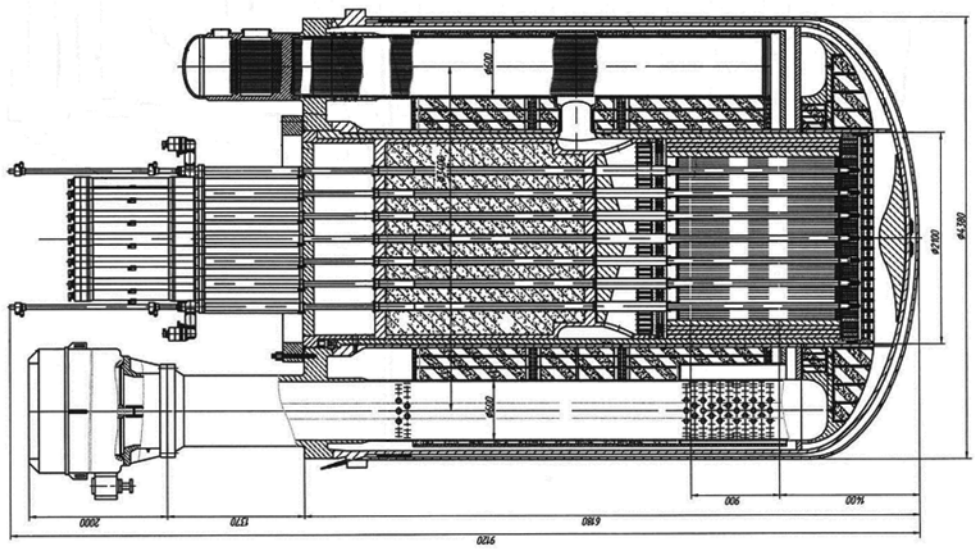


FIG. 5. Sketch of the RMB.

To ensure assembly, repair, maintenance and refueling, hatches designed to withhold extreme loads are installed over each reactor installation.

All the systems of the primary circuit are installed inside the strong vessel of the RMB. The removable unit (a basket with the core and control rods and a shielding plug) is installed in the central part of the RMB that is surrounded by in-vessel radiation shielding (boron carbide) with SG and MCP modules mounted on it (see Fig. 4). A sketch of the RMB is presented in Fig. 5.

8.4.4. Primary circuit circulation scheme

The in-vessel structures ensure the hydraulic connections between the equipment of both circuits (i.e. the main and auxiliary one) formed within the reactor monobloc, without the need of pipelines and valves. Within the

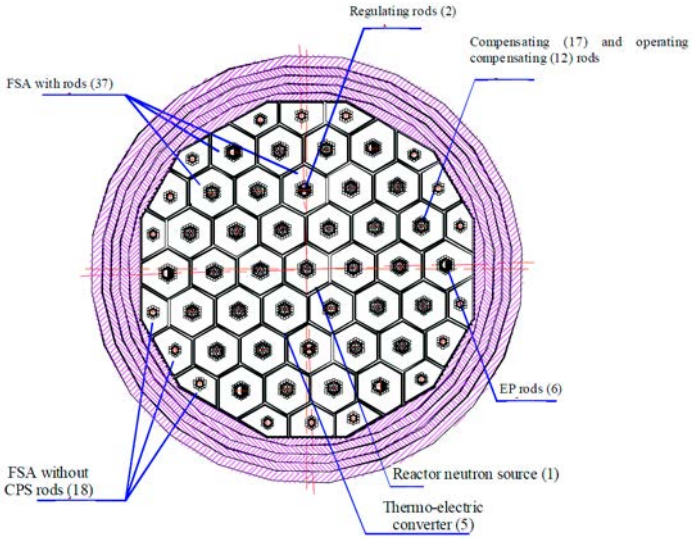


FIG. 6. Cross-section of the core of reactor SVBR-75/100.

main circulation circuit (MCC), the coolant flows according to the following scheme: it is heated in the core, and then the coolant flows to the inlet of the medium part of the intertube space of the twelve SG modules that is switched on in parallel. Then the coolant is divided into two flows. One flow moves bottomup in the intertube space and is transferred to the peripheral buffer chamber with a free level of the 'cold' coolant. The second flow moves top-down and is transferred to the outlet chamber from where it is transferred to the channels of the in-vessel radiation shielding. It is cooled when the flow moves up, and then it is transferred into the peripheral buffer chamber as well. Out of the peripheral buffer chamber, the main coolant flow is transferred over the downcomer circular channel along the RMB vessel via the inlet chamber to the MCP suction. Another part of the coolant is transferred to the MCPs suction over the circular channel formed by the MCP vessel and the shaft. Out of the MCP the coolant is transferred along the two channels of the block of the lower zone of the in-vessel radiation shielding into the distributing chamber, from where it is transferred to the reactor inlet chamber, thus closing the main coolant circuit.

The auxiliary circuit coolant circulation is formed in the channels designed for mounting the jackets of the absorbing rods of the reactor's control and protection system (CPS). The auxiliary circuit ensures cooling of the CPS absorbing rods, as well as the required temperature in the central buffer chamber and in the channel of the mass-exchangers that regulate the oxygen concentration in the coolant.

8.4.5. Reactor core

The shape of the core is close to that of a cylinder with dimensions $D_{eqv.} \times H_{core} = 1645 \times 900$ mm. There are ~12 500 fuel elements in the core, which are arranged in a triangle lattice with a pitch of 13.6 mm (the cross-section of the core is presented in Fig. 6).

No partial refueling is provided during the lifetime. For that reason, there is no need to use a casing for the fuel subassembly (FSA). A regular lattice of fuel elements is kept in the core. Therefore, the fraction of the coolant flowing through the cells with a non-standard geometry (at the core periphery or nearby the control rod channel) is considerably less than that for a design in which the FSA has a casing. This enables to reduce the maximal temperature of the fuel element's cladding. A shape of the outer border of the core is close to a cylindrical one, and this facilitates reduction of non-uniformity of power distribution. The design of the FSA is shown in Fig. 7.

The fuel element consists of a tube cladding made of EP-823 steel filled with UO_2 pellets, its outer diameter is 12 mm, wall thickness is 0.4 mm, and it has 4 screw ribs on the outer surface for spacing of the fuel elements. The volumetric fractions of the materials in the cell that corresponds to a single fuel element are given below:

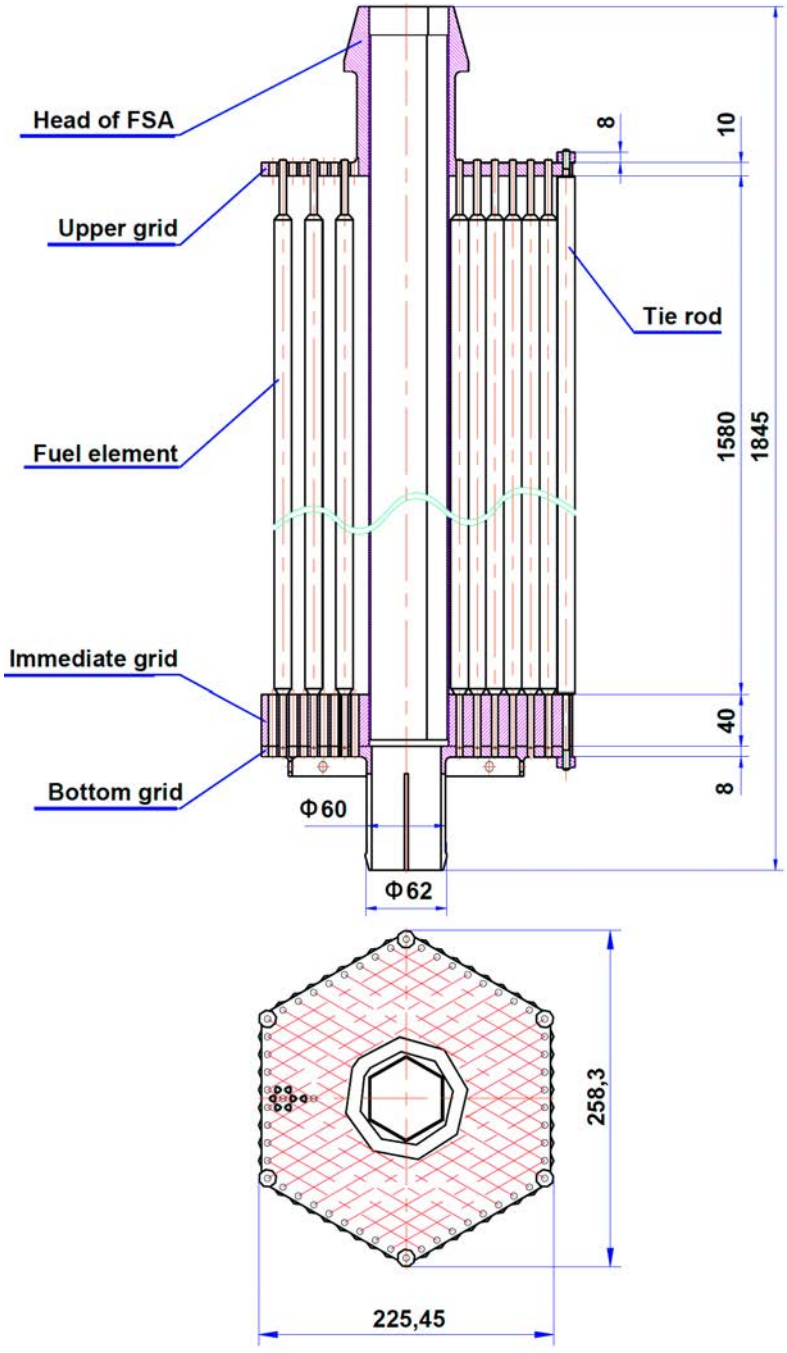


FIG. 7. Fuel subassembly of reactor SVBR-75/100.

— Volumetric fraction of fuel (homogenized over the total volume in the cladding) is ~0.615;
— Volumetric fraction of steel is ~0.105;
— Volumetric fraction of coolant is ~0.28.

In the lower part of the fuel element, right below the fissile zone, there is a steel bottom reflector. A compensating chamber for collecting the gaseous fission products is installed below this reflector zone. Above and below the core, there are the components of the reactor's removable unit; i.e. the lattices for mounting the fuel elements and FSAs, the coolant's inlet and outlet chambers, and the upper shielding elements.

Radially, a steel reflector, of ~ 240 mm width, surrounds the core. Beyond it are mounted the side screens of the in-vessel shielding.

A system of control rods is mounted along the FSA axis. The rods in the core are arranged in a triangle lattice with ~224 mm pitch. The number of rods and their design are set in compliance with the requirements for reactivity change compensation during the lifetime. The control rods system includes:

— Emergency protection (EP) rods;
— Additional emergency protection (AEP) rods (not shown in Fig. 6);
— Regulating rods (RR);
— Compensating rods (CR) and operating compensating rods (OCR), which ensure core subcriticality in the event of design basis (EP) and beyond design basis (AEP) accidents;
— Control of the RMB power having with RR;
— Compensation for the reactivity change during the lifetime having with CR and control of the operational reactivity margin having with OCR.

The absorbing elements (Aes) of CRs, OCRs and RRs are a cluster consisting of seven elements mounted in a triangle lattice and handled by a single control rod drive mechanism (CRDM). Each of the seven elements is made within the dimensions of a 4-ribbed tube of the fuel element cladding and uses boron carbide as an absorbing material. The Aes of EP and AEP are made of a single rod with boron carbide.

All Aes, except for the AEP ones, are fixed with the help of bars to the cogged bars of their CRDMs. The AEP rods have no CRDM and are kept in the cocked position with the help of fusible locks, which will fuse if the LBE temperature at the core outlet increases over the specified value (this can happen in the event of postulated accidents in which the standard EP does not actuate).

Each control rod channel displaces 19 fuel elements from the core. At the same time, they are used as the guide tubes for the FSA during fuel loading and unloading operation.

The casings of Eps, AEPs and the guide tubes without Aes are filled with argon. Cooling of Aes of RRs, CRs, and OCRs is provided by LBE flow inside their channels.

8.4.6. Steam generator module

The SG module is a recuperative submersible type heat exchanger consisting of a vessel and tube cluster with chambers of water and steam-water mixture. A tube system of the SG module that forms a heat transfer surface is made in the form of 301 Fil'd channels. Each Fil'd channel consists of an outer tube of 26 × 1.5 mm diameter containing co-axially mounted bottom and central tubes of 12 × 1.0 mm diameter. The outer and the bottom tubes are made of bimetal: the outer layer is made of austenitic silicon-doped steel that is high corrosion resistant in LBE at the given temperature; the inner layer is made of high nickel steel that is high corrosion resistant in water. The channels in the tube cluster are installed in a triangular lattice. The working length of the Fil'd channel is ~3.7 m. The circulation of the coolants in the module is realized as follows: LBE flows in the inter-tube space of the heat exchanger, water flows in the central tubes of the tube cluster, steam water mixture circulates in the circular gaps between the outer and central tubes of the tube cluster. The design features of the module are summarized as:

— Co-axially mounted tubes in the tube cluster, and spacer arrangement of the cluster in the vessel of the SG module;
— Lack of vibration loads;
— Lack of LBE leaks into the gap between the SG vessel and its in-vessel shell;
— Possibility to seal some leaked tubes of the tube cluster, and an possibility of entire replacement of the module, if necessary.

The design of the SG module is presented in Fig. 8.

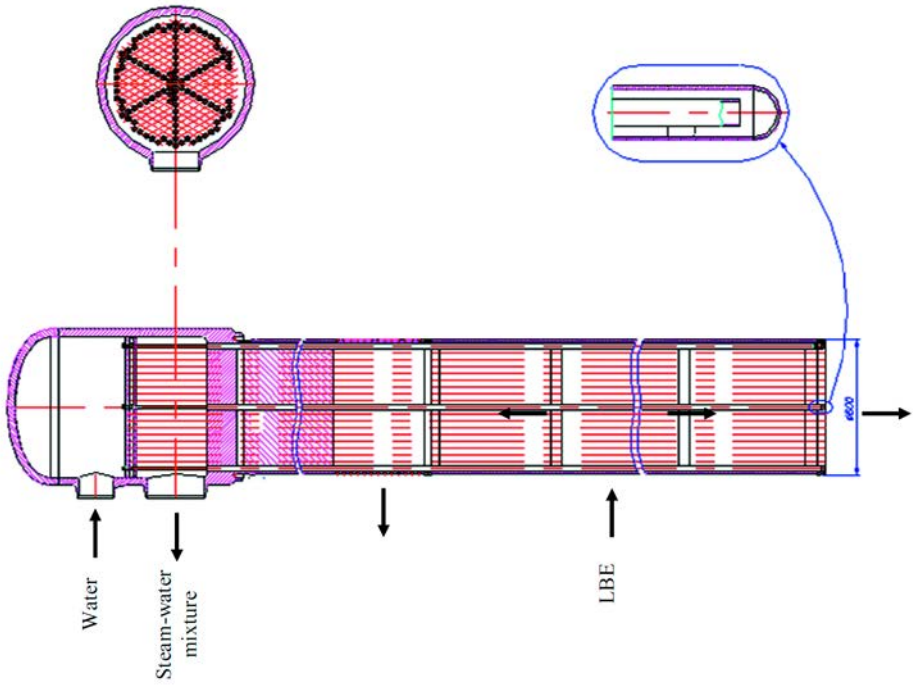


FIG. 8. The design of the SG module.

8.4.7. Main circulation pump (MCP)

The MCP is a pump unit consisting of a submersible axial pump and a gas-tight, non-regulated asynchronous electromotor.

Use of the gas-tight electromotor eliminates the necessity to equip the unit with a rotating shaft sealing, which separates the gas chamber of the pump from the surrounding atmosphere.

In addition to the electromotor, the basic parts of the MCP include a flexible clutch that connects the pump shaft and electromotor, an upper ball bearing and a bottom hydrostatic bearing of the pump, an impeller, a guide vane, and others.

The pump is installed in a cylindrical vessel with ~600 mm diameter. The upper flange of this vessel is fastened to the RMB vessel, while, the bottom part is fixed in the casing that is a part of the in-vessel RMB structures.

The design of the pump provides reduction of the axial thrusts acting on the impeller and upper bearing and increases the operability of the hydrostatic bearing. The level at which its hydraulic component is mounted ensures the operation without cavitation.

The pump head is ~0.55 Mpa, flow rate is ~2050 m^3/h, the number of rotations is ~750 rot./min.

Drawing of the MCP is presented in Fig. 9.

8.4.8. In-vessel structures

The in-vessel structures include the following:

— Shells for mounting the SG modules, MCP, core basket, shielding plug, and other necessary equipment;
— Bottom displacer that is part of the shielding;
— Structure elements which form a circulation circuit of the primary circuit.

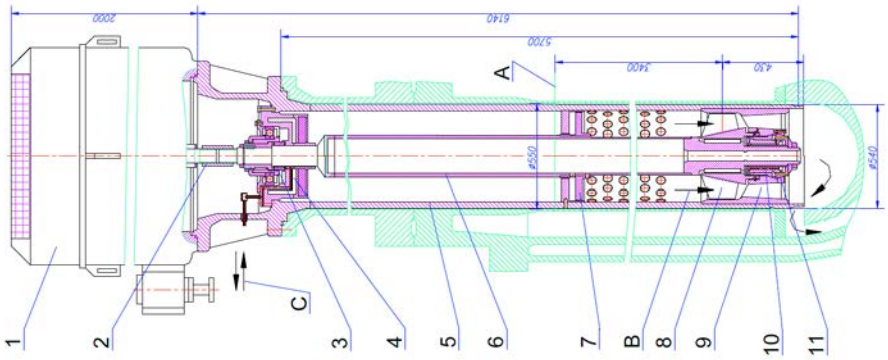


FIG. 9. Drawing of the MCP.

The in-vessel radiation shielding made of boron carbide is installed within the in-vessel structures. This ensures that steam-water radioactivity limits are met. It also ensures keeping the radiation exposure of the in-vessel structures and equipment at allowable level.

8.4.9. Shielding plug

The shielding plug of the reactor is part of the RMB removable unit that is replaced while refueling. The shielding plug is shaped as a step-type shell with an upper lid and bottom and has three chambers. The upper chamber forms a central buffer chamber of the RMB, the intermediate chamber filled with boron carbide fulfills shielding function along with a steel lattice installed on the bottom, and the bottom chamber that has distributing holes for the LBE flow forms the outlet chamber of the core.

8.4.10. Coolant technology

The devices, which maintain the necessary concentration of dissolved oxygen in order to eliminate structure materials corrosion, thus ensuring the required LBE quality and of the RMB primary circuits' surfaces in contact with LBE, are listed as:

— An electro-chemical sensor for monitoring the oxygen thermodynamic activity. This a capsule type sensor, which provides electric potential difference between a reference electrode installed inside the sensor and the flowing LBE. The electric potential difference change depends on the content of thermodynamic active (dissolved) oxygen in the LBE. The sensors are installed at the LBE entrance into the SG;
— Mass exchangers containing solid lead oxide, which provide for an increase of the dissolved oxygen content in the LBE, according to the indications of the oxygen sensor thermodynamic activity measurement;
— An injection device that allows the supply of the gaseous reagent mixtures based on hydrogen to the LBE at the MCP intake from the RMB buffer chamber. The purpose of this device is to purify the LBE and its circuit's surfaces from accumulated slag impurities of lead oxides after performing repair and refueling works, when the probability of air ingress to the primary circuit gas system cannot be excluded;
— The filters for purifying the LBE from impurities, which cannot be recovered by hydrogen (if necessary).

More detailed information concerning the coolant technology devices are reported in Refs [138–141].

To prepare lead-bismuth coolant of the eutectic content (44.5 ± 0.5% of lead weight and 55.5 ± 0.5% of bismuth weight), the high purity lead and bismuth with a total content of other impurities being not more than 0.01% is used.

The permissible content of impurities changes within a wide range up to 10^{-5}%. It depends on the microscopic neutron capture cross-section, the microscopic cross-section for activation with formation of long lived gamma active radionuclides, and the affinity of the respective chemical element to oxygen.

The particular values of the allowable content of impurities are assigned for each chemical impurity element based on the following criteria:

— Low influence on reactivity margin;
— No limitations due to gamma-irradiation on the duration of personnel's activities during primary circuit equipment repair;
— No considerable influence of impurities on oxygen control. To measure the impurities content, when their concentrations are very low, special methods based on high-sensitive chemical-spectral analysis have been developed.

It should be highlighted that the excessively high purification from impurities will cause considerable increase of the coolant's cost.

8.4.11. Refueling and storage of spent nuclear fuel

The refueling technology adopted for SVBR-75/100 provides at once loading of fresh fuel as a single cartridge (the new core) and cassette-by-cassette unloading of fuel at the end of the core lifetime.

Refueling is performed while LBE is in the RMB vessel. The LBE temperature is within 160–180°C, which is ensured by the system of external steam heating of the RMB vessel and by decay heat.

The refueling technology is based on special refueling equipment. This equipment includes an adapter box with a gate valve, refueling container for extracting the shielding plug (upper shielding of the reactor) and the core basket, the guiding device, a refueling shielding container for extracting the spent FSA, capsules for long storage of the spent FSA, and other equipment. This equipment eliminates radioactivity release into the environment, air ingress into the primary circuit, and provides irradiation protection of the personnel. When refueling is performed, the argon pressure in the primary circuit's gas system must be maintained slightly higher than the atmospheric pressure.

The extracted FSA are installed into the steel capsules filled with liquid lead. Then after the capsules have been sealed, they are installed in the 'dry' spent nuclear fuel (SNF) repository with natural circulation of the atmospheric air. The solidified lead forms an additional shielding barrier that provides long safe storage of SNF prior to its transportation for reprocessing. On ending the refueling, the shielding plug is mounted on the RMB vessel. At this point, startup and raising the power level of the reactor can be initiated.

8.5. BASIC CONCEPTS OF THE NUCLEAR POWER PLANT BASED ON SVBR-75/100

8.5.1. Approach

The increased safety characteristics of the SVBR-75/100 reactor installation (inherent safety, deterministic elimination of severe accidents) and its improved economical parameters are achieved virtually 'free of charge'. In addition to the use of fast neutrons, heavy liquid metal coolant, and an integral design of the primary circuit, a conservative approach was used in the design of the reactor installation.

This conservative approach is characterized by:

— Nearness of the scale factors between the SVBR-75/100 and the nuclear submarine reactor installations;
— Use of borrowed or narrowly scaled components, units, and certain equipment elements of the reactor installations, which have been verified by operating experience in nuclear submarine and other reactor installations; e.g. fuel pellets, fuel element claddings, fuel sub-assemblies, control rods, in-vessel devices, CRDM, LBE technology system devices, SG modules, drums-separators, autonomous cooling condensers, gas system condensers, seismic resistant support structure of the PHRS water tank in which the reactor monoblock is installed, refueling system's equipment;
— Use of operational parameters mastered in primary and secondary circuits;
— Orientation towards existing fuel infrastructure and technological capabilities of several machine–building enterprises.

This conservative approach makes it possible to reduce considerably the technical and financial risks, narrow down the possibility of mistakes and failures that are typical where implementing innovative nuclear technologies, as well as significantly reduce R&D and costs.

8.5.2. SVBR-75/100 inherent safety

Fast reactor physics and design features like the lack of poisoning effects, low value of negative temperature reactivity coefficient, and compensation for the fuel burnup processes by plutonium buildup allow having an operating reactivity margin less than the delayed neutrons fraction. Therefore, the risk for prompt neutron runaway is eliminated.

In the case of oxide uranium fuel, for which the reactivity margin is maximal, in the event of unauthorized insertion of positive reactivity and postulated failure of all EP CRDMs, a special algorithm of compensating rods control, which is part of the automatic control system, ensures the elimination of prompt neutron reactor runaway. When the reactor operates at nominal power during a certain period (~4 months), the reactivity margin controlled by an operator is much lower than 1 $. In this condition, the other compensating rods are disconnected from the control system.

Moreover, the efficiency of each rod is much lower than 1 $, the movement velocity of the absorbing rods extracted one by one is technically limited. For that reason, the inserted positive reactivity has time to be compensated by negative feedbacks without dangerous increase of the core temperature.

For the considered fuel loads, the total void reactivity effect of the reactor is negative and the local positive void reactivity effect is less than 1 $ and can be practically excluded due to the very high boiling point of the coolant and lack of opportunity for large gas or steam bubbles ingress.

Elimination of water or steam penetration into the core caused by large SG leak, and consequent possibility of over pressurization of the RMB vessel (which is, however, designed to withstand the maximum possible pressure under these conditions) is ensured by the LBE circulation scheme. This scheme ensures that steam bubbles are transported to the coolant free level by the upward moving LBE flow. Then the steam bubbles are removed to the gas system condensers. In the event of a postulated failure of this scheme, the steam bubbles are removed through the rupture membranes to the bubbler devices of the tank of the PHRS.

The loss of coolant from the RMB and the termination of LBE circulation through the core in the event of failure of the RMB vessel tightness (a beyond design basis accident) cannot occur because of the safeguard casing surrounding the RMB vessel with only a small free gap between the two vessels.

The principle of in-depth protection against radioactivity release into the environment is used in the SVBR-75/100 design. The following barriers exist against radioactivity release: the fuel matrix, the fuel element's cladding, primary circuit coolant, the SG tube cluster, the RMB vessel, the safeguard RMB casing, the tight box of the reactor installation, the shielding shell (containment) of the power-unit.

SVBR-75/100 meets the basic in-depth protection concepts. These are:

— Choice of type and power rating of the reactor;
— Conservative design approach;
— Maximal extent of factory fabrication and assembly of the equipment;
— Simplicity of design and operation;
— Inherent self-protection properties of the reactor installation;
— Passive heat removal systems;
— Slow pace of accident evolution;
— Use of design and technological solutions proven in practice;
— Long 'grace' periods for the personnel in the event of accidents.

The reactor installation is provided with two independent (without connection with a turbine installation) heat removal channels, each of them is able to remove ~3% of nominal power of the reactor. Each channel consists of the cooling condenser connected with a separator and cooled by water, and a condensate drainage pipeline with a direct acting regulator valve, which is opening if the pressure in the separator exceeds the nominal value.

In normal operation conditions, the autonomous heat removal system is used in the startup and cooling down modes.

In the stand-by mode, the condensers are flooded with water, and there are almost no heat losses. In the case of steam pressure increase up to a determined value, the valves will be opened, and the condensate will be drained to the separator. Thus, the heat-exchange surface will be cleared out, and the steam will begin to condense until its pressure decreases to a certain level determined in advance by adjustment of the regulator-valve.

The system employed to detect on accidental situation caused by SG leak is designed to prevent possible consequences of this initiating event; such as over-pressurization of the RMB vessel due to steam pressure and ingress of a steam-water mixture into the core.

In the SG leak event, the basic feature is the LBE circulation circuit with the up going streams the coolant's free levels. This provides reliable separation of steam-water mixture and prevents penetration of steam into the core with the descending primary circuit coolant's stream whose velocity is low.

In an event of 'small' SG leaks (up to full rupture of a single SG tube), two condensers of the gas system, which are cooled with water, are used. Their capacity enables to keep the reactor installation gas system pressure within 0.5 Mpa.

To withstand the large SG leaks (postulated rupture of several SG tubes; i.e. an initiating event that is beyond the design basis), the gas system is connected with the PHRS tank. This line is overlapped with a membrane-preventing device designed to rupture at gas system pressure of 1 Mpa that is not dangerous for the RMB vessel. In the event of membrane rupture, the steam will be condensed in the water tank. The volatile radionuclides of the cover gas will remain in the tank water and the radioactive non-condensed gases will be released into the atmosphere via the filter ventilation system. Radioactivity release will not exceed the permissible level.

Computations and experiments have shown [142] that if the conservative approach is used, safety operating limits on the fuel element cladding temperature are not achieved in the event of the following postulated accidental situations:

— Unauthorized extraction of the most effective absorbing rod;
— The coolant's through-pass section is plugged by 50% at the core inlet;
— All main circulation pumps are shut down;
— In-take of steam to the turbo-installation and provision of feeding water are terminated;
— Guillotine rupture of several SG tubes;
— Leak in the reactor vessel;
— 'Freezing' LBE in a single SG;
— Black out of the NPP.

The reactor installation safety does not depend on the state of the systems and equipment of the turbine-generator installation.

The reactor installation inherent safety properties conditioned by reactor feedbacks, natural properties of LBE and reactor installation design made it possible to combine the realization of safety functions (except for the emergency protection function) and normal operating function.

The safety systems do not contain elements where actuation can be blocked in an event of their failure or human action:

— In case of mechanical damage of the EP CRDMs with LBE temperature rising above dangerous values, AEP of the reactor operates passively due to the fusible locks holding the absorbing rods in the upper position;
— Residual heat removal when there is no heat removal via the SG is provided passively by transferring heat via the monobloc vessel to the water of the PHRS tank and the boiling of this water with steam removal to the atmosphere (the 'grace' period is about two days without damage to the core);
— In the event of rupture of several tubes, or failed operation of the gas system's condenser, SG leak containment is provided passively, due to the breaking membrane which discharges the steam to the bubbler (PHRS tank), at increasing steam pressure in the gas system above 1 Mpa.

Calculations of the safety potential of SVBR-75/100 have been performed. These calculations have shown that, in case of superposition of all postulated initiating events, (i.e. damage of shielding shell, reinforced concrete collapse over the reactor and tightness failure of the primary circuit's gas system with direct contact between LBE surface in the reactor mono-block and atmospheric air, and total black-out of the NPP), no reactor runaway, no

explosion and combustion occurs, and radioactivity release into the environment, does not reach values requiring population evacuation beyond the NPP fence. The probability of severe damage to the core is considerably lower than the value specified in the regulatory documentation.

This allows claiming not only reactor installation resistance to equipment failures, personnel errors and their multiple superposition, but also resistance to malevolent action when all special safety systems have been intentionally blocked. These robustness properties are typical to SVBR-75/100.

8.5.3. Modular structure of the nuclear power plant nuclear steam-supplying system

In Ref. [143] it is noted: "modular factory fabrication of nuclear power systems and their on-site assemblage will replace the existing expensive methods of construction". Economic advantages of a modular approach to NPP construction have been highlighted in Ref. [144]: "the measures on reducing the construction terms to the great extent influence on the total capital costs, especially at high rates because in the process of construction the payment of credits can achieve 25% of the total investment bulk and more. Modular production that makes it possible to manufacture and assemble the units in the factory but not on-site reduces the construction terms and hence the expenditures on payment of credits during the construction period".

Shortening of the investment cycle for NPP construction as a consequence of adopting modular structure of the NPP and delivery of factory-made modules allows reaching technical and economical parameters of the NPP that are comparable to those of steam-gas thermal power plants (TPPs) with short investment cycles. Hence, it makes it possible to reduce considerably the financial risks [145].

For developed countries with large power systems, it will be economically effective to use large modular power units. Maximal possible capacity of the modular power unit will not be restricted by the maximal possible power of the reactor. The modular structure of the power unit's nuclear steam supplying system (NSSS) allows putting the power unit in operation following a gradual power rising approach.

Therefore, capital investments payback term is shortened, and earlier electricity production allows for earlier credit repayment, as compared with a power unit based on a large capacity reactor.

In case the large modular power-unit is equipped with one turbine, and based on the existing technical level of turbine construction, the power-unit's capacity can be in the range of 1600-1800 MW(e). SSC RF (State Scientific Center Russian Federation)-IPPE, FSUE (Federal States University Enterprize) EDO 'Gidropress', and FSUE 'Atomenergoproekt' have developed a conceptual design of a two-unit NPP, which has a NSSS consisting of 16 SVBR-75/100 reactor modules and one 1600 MW(e) turbine [129]. This makes it possible to compare correctly the technical and economical parameters of that NPP with those of the NPP based on WWER-1500.

When the NPP unit capacity was selected, it was taken into account that specific capital costs of the NSSS's building ('nuclear island') would decrease with increasing the unit's capacity. This is because, with an increasing number of modules in the NSSS's building, the cost of the equipment and providing systems installed beyond the NSSS's building increases only slightly. For that reason, their contribution to the specific capital costs of the power unit will decrease. Such systems and equipment include the refueling equipment, coolant receiving equipment, equipment for coolant transport to the monoblocs at initial filling, etc.

The modular principle of the NPP design is the most economically effective in the case of reactors, in which the inherent safety properties against severe accidents have been realized to the maximal possible extent. In this case, the scale economical loss is abundantly compensated due to:

- Lack of various special safety systems operating in a waiting mode, which are necessary for traditional type reactors having the objective to diminish the probability of severe accidents and to reduce their consequences, but which do not eliminate the root causes of such accidents;
- Large scale production of the 'standard' reactor modules;
- Complete factory fabrication of the basic element of the reactor installation; i.e. the reactor monobloc in which the whole equipment of the primary circuit is installed;
- Reduction of the duration of the investment cycle.

Construction time and assembling works duration for the reactor compartment are considerably shortened. This reduces the capital costs correspondingly.

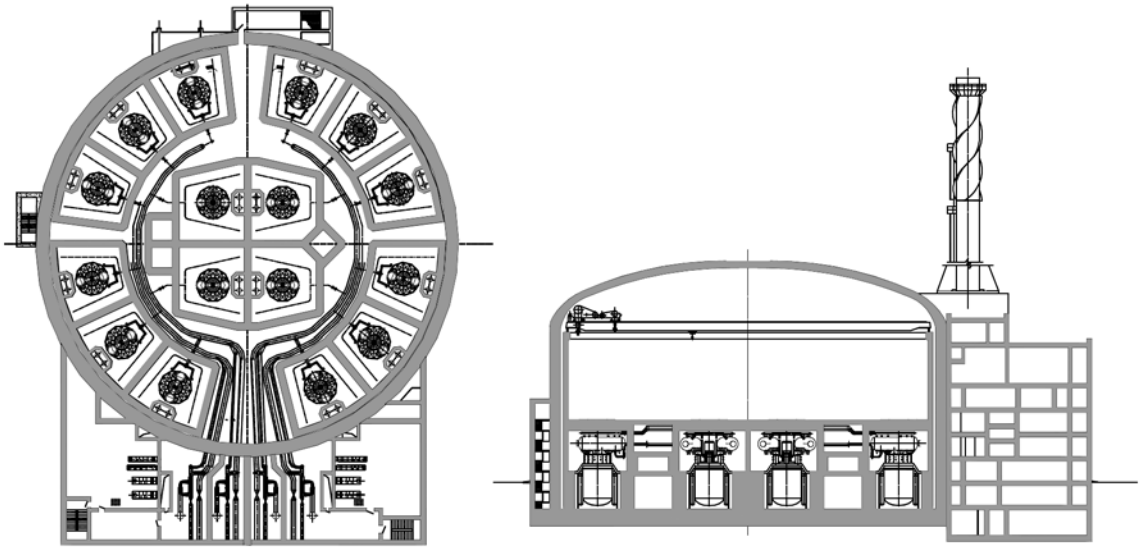


FIG. 10 (a) (b). Horizontal (a) and longitudinal section (b) of the main building of the NSSS of SVBR-75/100.

As the reactor module has only two states — actuated and shut down — control of the modular NSSS is carried out by an operator who uses the common power master unit. If there is any fault in a certain reactor installation, it is automatically removed from operation and can be cooled down autonomously with the turbine systems. Therefore, NSSS power is only slightly reduced.

Each module is mounted in a single compartment and is able to operate when the neighbor module is shut down for refueling or repair. The simple scheme of the reactor installations and their similarity allows to reduce the number of the operation and maintenance personnel at the modular NPP unit, as compared to a NPP unit with one large power reactor installation with lots of safety systems including protection systems, accidents detection systems, controlling, systems etc. For example, the safety systems of the AP-1000 reactor have 184 pumps, 1400 driver valves, 40 km of pipelines and cables [146].

The modular design of the power unit's NSSS allows attaining loading factors (LF) to be not less than 90%, with long reactor operating periods without refueling. When each reactor installation is shut down for refueling, the power-unit power is reduced only slightly.

Sequential refueling of each reactor installation (16 modules, ~8 years fuel residence time) of the NSSS is equivalent to batch refueling of the large-power reactor (1600 MW(e) annual refueling of ~1/8 of the core). Duration and periodicity of scheduled maintenance and repair works are determined by the requirements of the turbine equipment.

Licensing of a modular type large power power-unit will be much simplified by the construction of one reactor installation or of one small modular power-unit, after certification of the reactor installation. Once the reactor installation has passed tests, it will be possible for the serial standard installations to receive a certificate of conformance. Small power of the reactor installation ensures a comparatively low construction cost.

Horizontal and longitudinal section of the SVBR-1600 reactor compartment main building with the NSSS is shown in Fig. 10 (a) (b).

The basic technical and economical parameters of the two-unit NPP based on SVBR-75/100 in comparison with those of the two-unit NPPs with WWER-1500, WWER-1000 (V-392), BN-1800 and TPP with ten steam gas units PGU-325 are summarized in Refs [129, 147].

The results of technical and economical computations (Table 14) have revealed that, in compliance with the data obtained at the conceptual design stage, the technical and economical parameters of the NPP with two 1600 MW(e)

units, each based on the SVBR-75/100 type reactor installation, are better than those of the NPP based on the large power thermal and fast reactors and those of the TPP with ten units PGU-325 operating by using natural gas.

The schedule to construct this NPP can be ~3.5 years and with consideration of increasing capacities of the modular NSSS, the commercial production will begin no later than 3 years after the first pour of concrete. Accounting for the required additional financing to service the credit (if the construction is realized at the expense of borrowed capital), the advantage of the modular NPP over the other type NPPs based on the reactors with larger unit capacity will be even higher.

The costs were computed in 1991 Rubles for the Russian costs and terms when a Ruble was regarded to be equal to a US dollar for the opportunity of comparing the projects accomplished in different years.

However, it should be highlighted that evolutionary projects of the NPPs with light water reactors have been developing during several generations and the reserves of improving their technical and economical parameters are almost exhausted. At the same time, the innovative project of the NPPs with SVBR-75/100 that is actually the first generation design based on the conservative approach has considerable reserves for improvements. Despite this, due to lacking experience of practical realization, an additional margin of 17% over the standard one (60% of the reactor installation equipment cost) has been introduced into the costs of the capital investments in industrial construction of the modular NPP.

8.5.4. SVBR-75/100 improvement potential

Use of the conservative approach at the first implementation stage of SVBR-75/100 ensures a high potential for further project improvements, that will be realized as the corresponding R&D has been carried out and operating experience has been gained. The following two measures, in particular, have a considerable potential to reduce electricity production costs:

— Use of a SG producing super-heated steam that makes it possible to increase the thermodynamic cycle's efficiency and increase the electric power of the NPP by 10–15%;
— Increasing the LBE reactor outlet temperature up to the limit of 650°C fuel cladding temperature would allow an increase of the reactor thermal power by 10% without changes to the reactor design.

Realization of these measures for the NPP project with two large modular type power units (1600 MW(e) each being) will make it possible to reduce cost of electricity, as compared with the TPP with ten steam-gas units PGU-325. It will also allow reducing the specific capital costs of NPP industrial construction to the value that is typical of modern steam-gas TPPs.

TABLE 14. COMPARABLE PARAMETERS OF DIFFERENT POWER PLANTS [133]

Name and dimensions of the parameter	SVBR-75/100	WWER-1500 [119]	WWER-1000	BN-1800 [117]	HEPP with PGU-325
Set up power of the power-unit, MW(e)	1625	1550	1068	1780	325
The number of the units at the plant	2	2	2	2	10
Electric power necessary for plant's own needs, %	4.5	5.7	6.43	4.6	4.5
Efficiency of the net plant (power unit), %	34.6	34.4	33.3	43.6	44.4
Capital investments in the industrial construction of the plant, $/kW(e) (price of 1991)	661.5	680	819.3	860	600
Design cost of produced electricity, cent/kW·h (price of 1991)	1.46	1.62	2.02	1.6	1.75

8.5.5. Use of different kinds of fuel and fuel cycles

The design of SVBR-75/100 allows the operation using different types of fuel and in different fuel cycles without changing the reactor installation structure and deterioration of safety characteristics [148]. During the past decades, with current low uranium and enrichment cost, the most economical fuel proved to be oxide uranium fuel and proved to be the most cost effective fuel cycle the opened fuel cycle with postponed reprocessing.

The change to MOX fuel and closed nuclear fuel cycle (NFC) with core breeding ratios (CBR) exceeding 1 will be economically effective when the costs of natural uranium increase. Moreover, the expenditures for construction of the factories for SNF reprocessing, re-fabrication of the new fuel with plutonium and for their operation must be less than the corresponding costs of natural uranium, costs of its enrichment, costs of manufacturing the fresh uranium fuel and costs of long SNF storage. Under these conditions, it is possible to use both MOX fuel and mixed nitride fuel, if the latter offers economic advantages. In case nitride fuel is used, the lifetime duration can be considerably increased (according to recent calculations: from 7 years for the oxide fuel to 15 years for nitride fuel).

The transition to the closed NFC will prove economically more visible if the plutonium used to fabricate the first MOX fuel loads is coming from the own used fuel. The content of plutonium in SVBR used fuel is higher by an order of magnitude compared to that in thermal reactor (TR) used fuel.

As make up fuel, the TR used fuel (both from WWER and RBMK reactors) may be used in that closed NFC (similarly to the Direct Use of spent PWR fuel In CANDU (DUPIC) technology [149]). The principal scheme of the fuel cycle is shown in Fig. 11.

Fast reactors operating in an open NFC consume much more natural uranium than TRs. Therefore, the period of FR operation in an open NFC must be kept to a minimum.

Calculations [150] have shown that transition to the closed NFC is possible after the second reactor installation lifetime; i.e. in 16 years. In addition, during the first 16 years the consumption of natural uranium for 1 GW(e)-year energy produced will be ~5670 t (assuming oxide uranium fuel, CBR = 0.84). During the 60 years of the reactor installation service lifetime, the consumption of natural uranium calculated for 1 GW(e) installed capacity will be by 30–40% lower than the consumption of a PWR during the same time period. Moreover, this strategy would allow FR introduction in the closed NFC prior to reaching the equilibrium mode practically without consumption of natural uranium.

The flexibility of SVBR-75/100 relative to the fuel type and fuel cycle allows a timely and gradual transition to an economically justifiable closed NFC. Therefore, the SVBR-75/100 can contribute to the solution of the geological disposal of the long lived radioactive wastes (RAWs), taking into account that minor actinides are effectively utilized in the FR.

8.5.6. Handling the radioactive waste

During the process of operating the NPP, liquid RAWs are produced in very small quantities. This fact has been verified by experience of operating LBE cooled nuclear submarine reactor installations. The NPP design provides an installation for concentrating and solidifying the small quantities of liquid RAWs. At the end of the reactor installation lifetime, the radioactive LBE can be repeatedly recycled in the new reactor installations. In 1000 years of irradiation, the slight residual long lived radioactivity of LBE caused by ^{208}Bi and ^{210m}Bi radionuclides will be less than natural radioactivity of the uranium ore (in terms of U_3O_8). It will be only important at the final stage of NPP operation.

Lead-bismuth eutectic in the form of solid radioactive waste being disposed in the deep geological formations will not disturb the natural radioactivity equilibrium. The low chemical activity of lead and bismuth eliminates radioactivity release paths into the biosphere. This problem is similar to the problem of handling zirconium contained in the fuel element cladding of TRs, for which the long lived ^{93}Zr radionuclide is accumulated [151].

The quantity of tritium release into the environment due to unavoidable losses of water in the secondary circuit is comparable with the disposal of tritium from the world's operating NPPs (except for the heavy water NPPs, for which the disposal of tritium is higher by an order of magnitude).

Considering that the half-life of the majority of fission products does not exceed 30 years (except for ^{99}Tc, ^{129}I and ^{135}Cs), it is assumed that after vitrifying, they will be stored for about 500 years. After this intermediate storage

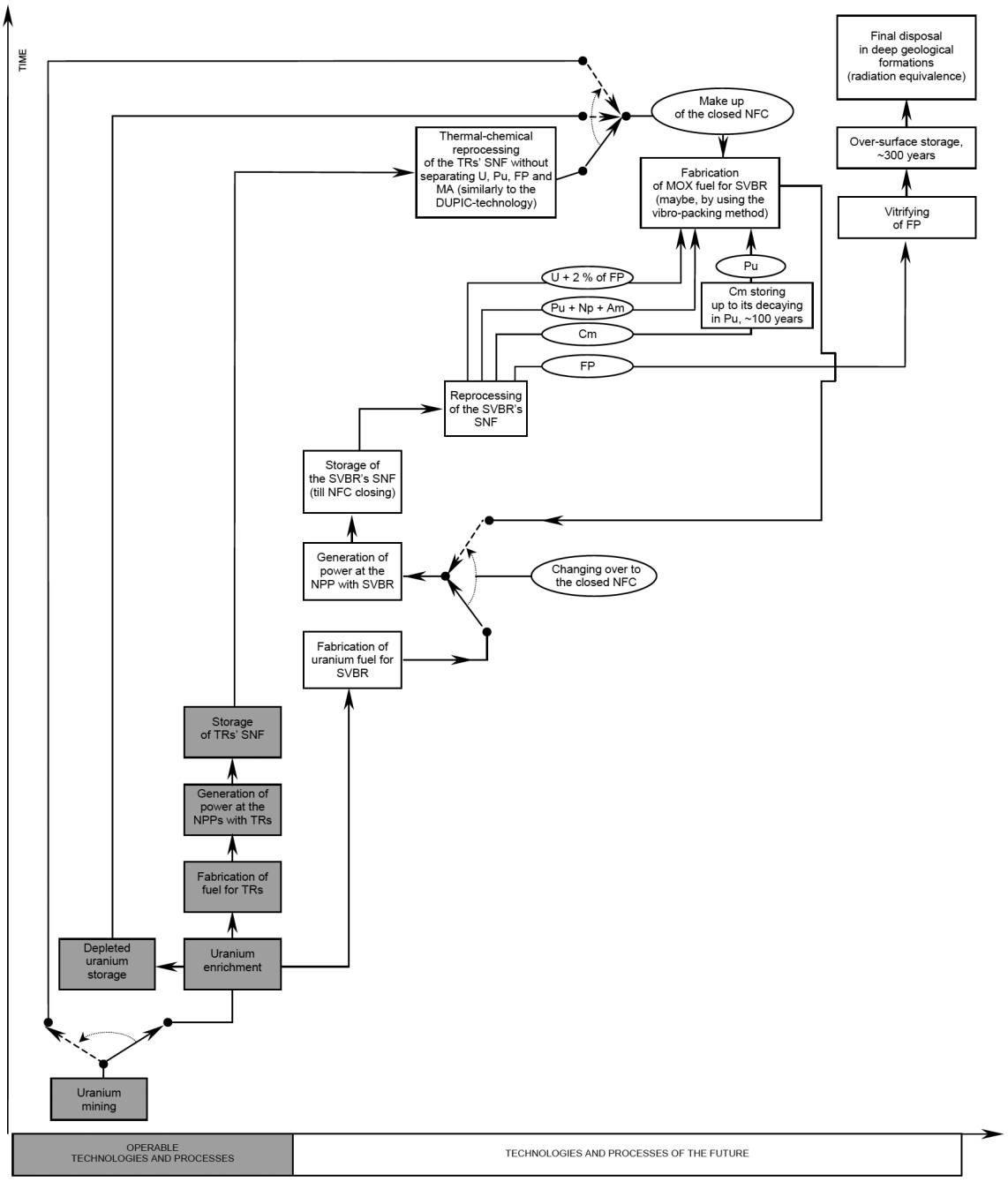


FIG. 11. The principal scheme of the fuel cycle of the NPT based on SVBR-75/100.

period, these vitrified fission products, whose radioactivity is then determined only by the long lived nuclides of technetium, iodine and caesium will be placed in deep geological formations.

Release of the transuranium (TRU) elements outside the fuel cycle is eliminated (except for very small losses at the chemical reprocessing stage of the used fuel).

To estimate the environmental impact of the SVBR-75/100, the specific radiotoxicity of the formed TRU elements (neptunium, plutonium, americium and curium) and long lived fission products (^{99}Tc, ^{129}I and ^{135}Cs) as a function of produced electricity was taken as a criterion. The radiotoxicity standard was adopted as the volume of water necessary for diluting radionuclides down to the concentrations for which the specific radioactivity of the obtained solution meets the sanitary requirements for drinking water.

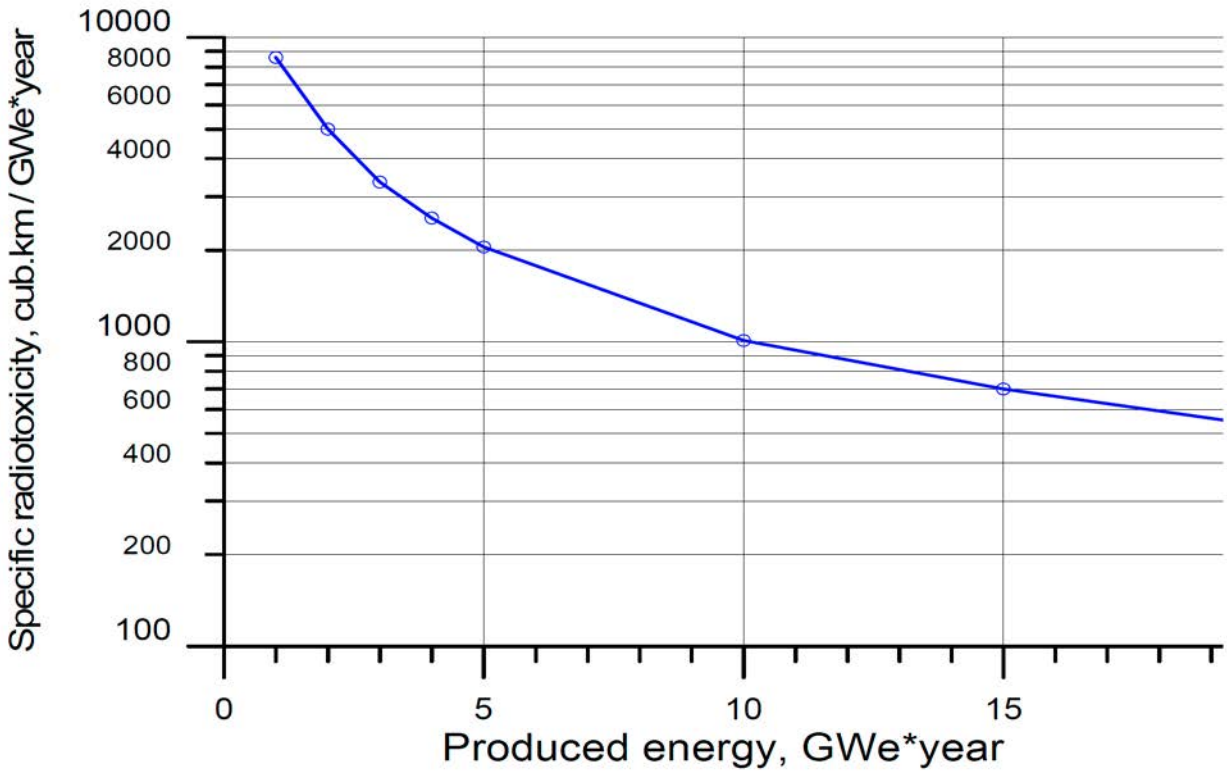


FIG. 12. Specific radiotoxicity versus cumulative energy production.

The specific radiotoxicity is determined as SNF radiotoxicity for the given installed power divided by the value of produced energy. The calculation (see Fig. 12) shows that the specific radiotoxicity of ^{99}Tc, ^{129}I and ^{135}Cs in the final disposal is 0.014 km^3/GW(e)*year without taking into account the losses at reprocessing. It is nearly equal to the specific radiotoxicity of natural uranium that is added to the fuel cycle each year.

The analysis of the obtained results reveals the environmental 'friendliness' of the SVBR-75/100 NFC, since the specific radiotoxicity decreases with increasing cumulative energy production. This is because the hard neutron spectrum in the reactor ensures efficient fissioning of both own Mas, and Mas built up in TRs.

8.5.7. Proliferation resistance

Non-proliferation of fissile materials means creating the conditions for which inappropriate use of fissile materials is made as difficult as possible.

It is evident that the problem of non-proliferation cannot be solved only by technical measures, since knowledge on the use of well-developed technologies of isotopic uranium separation and plutonium extraction out of spent fuel is available. For that reason, the complete solution to the problem of non-proliferation can only be achieved by coupling technological and political measures. The relationship of those measures will be different for nuclear and non-nuclear countries. During recent decades, all nuclear countries, legally possessing nuclear weapons, have solved this problem successfully using the measures of physical protection, accounting, control, and safeguard. For these countries, additional non-proliferation measures based on technology innovation are justified they do not reduce NP competitiveness.

In the case of NPP introduction in developing countries, additional technology innovation based measures should be taken along with the political measures and international control.

The small dimensions of the module and the properties of LBE offer a unique opportunity to realize the return of SNF without unloading it out in the user-country.

Fuel transportation in the reactor monobloc with solidified LBE creates an additional technical barrier to fuel theft. Solidified LBE in the monobloc, several years after shutdown, also eliminates the risk of nuclear and

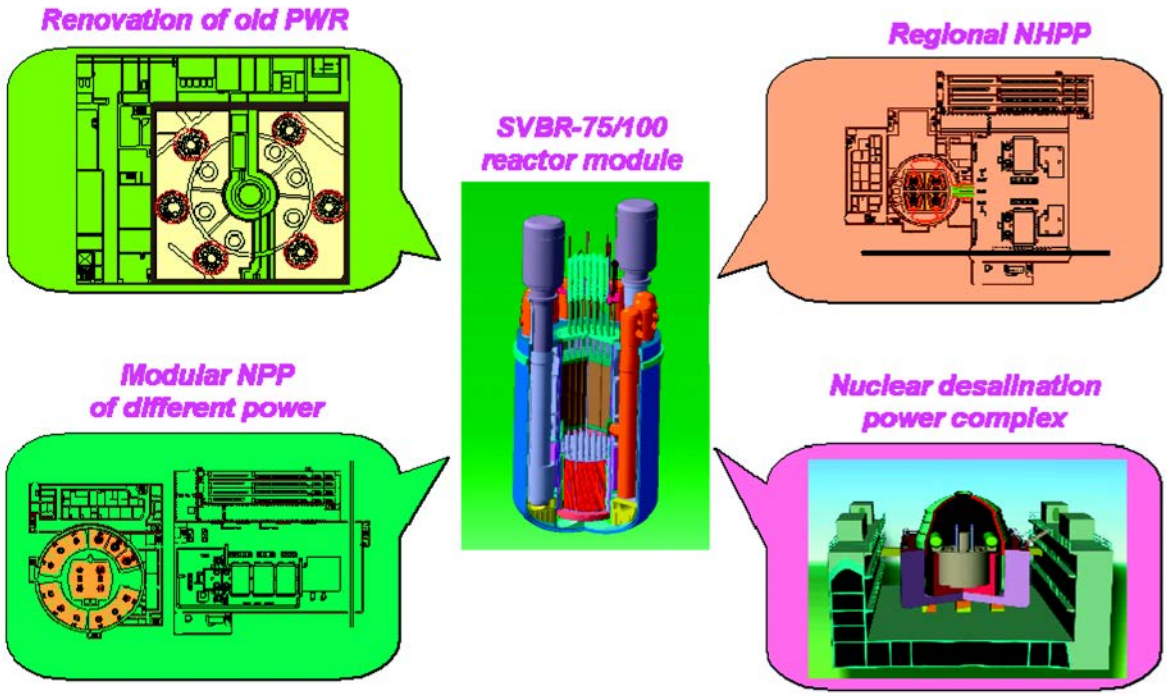


FIG. 13. Areas of SVBR-75/100 utilization.

radiation accidents, during transportation. It appears to be expedient to concentrate SNF reprocessing at international nuclear fuel cycle centers located in nuclear or developed countries.

8.5.8. Multi-purpose application of SVBR-75/100

The properties considered above offer the opportunity of a stage by stage realization of innovative NPP requirements. These requirements address fuel balance, safety, economics, reduction of risk of nuclear fissile material proliferation, and management of the long lived RAWs. The SVBR-75/100 possesses certain features, which makes it economically effective, and it is a good candidate for solving NP challenges for both the far and near future:

- The opportunity of complete factory fabrication of the reactor installation and its transportation to the NPP site by any kind of water or ground transport mean, including railway. This reduces considerably (by up to 3–3.5 years) the NPP construction time;
- Self-protection properties and simplicity due to the lack of special safety systems (the normal operating systems perform safety functions);
- Autonomous startup and shut down of the reactor with the help of ACS alone, without the use of the turbine;
- Sharply reduced probability for personnel's errors, whose consequences do not affect safety;
- Simplification and reduction of the maintenance cost of the reactor installation;
- Th multi-purpose use of 'standard' reactor modules with ~100 MW(e). The standard module can be used for constructing modular power units with medium or large power, or constructing regional nuclear heat power plant (NHPP) with 200–600 MW(e) which are located close to cities. Moreover, the standard module can be constructed as part of the renovation of NPP units, whose reactors have reached end of life, or used as part of nuclear desalinating plants. The multi-purpose applications allow increasing the production rate of the reactor modules, enable adopting modern standard design methods and industrial methods for production and construction works. All this results in quality improvements and cost reductions.

Multi-purpose utilizations of SVBR-75/100 are shown in Fig. 13.

Since extensive studies were performed in this area, it is worthwhile that the utilization of SVBR-75/100 modules for the renovation of NPPs based on TRs reaching their end of life is done by installing the appropriate number of SVBR-75/100 modules in the emptied SG and MCP compartments. The results of technical and economic feasibility studies of renovating the 2nd, 3rd, and 4th units of the Novovoronezh NPP (NVNPP) based on SVBR-75 have shown that the specific capital costs are reduced by a factor of two as compared with construction of new replacing power capacities [152]. The installation of SVBR-75 in the SG/MCP cells of the 2nd NVNPP unit is shown in Fig. 14 (a) (b) (longitudinal and horizontal cuts).

Reinforced concrete covering the reactor box will provide the function of the protection shell (containment).

A similar renovation approach can be used for almost all light water reactor (LWR) units. In this case, capital costs saving will be about $ 400M per GW(e) as compared with construction of the new replacing power units. The renovation scheme for a NPP WWER-1000 unit is presented in Fig. 15.

Experience gained by operating SVBR-75/100 under NVNPP conditions will allow, with a minimal investment risk, to launch sequential of LWR units renovation projects.

8.5.9. Opportunity to accelerate the pace of nuclear power development with use of the proposed nuclear power technology

The attractiveness of any nuclear power technology (NPT) can be characterized by its investment potential, which is determined by the maximal possible pace of NP development allowed by the economical parameters of the given NPT.

The highest requirements to the NPT investment potential occur at the NP steady development stage, when funding is required for the closure of the 'old' NPP units, the construction of replacing capacities, and the construction of new units ensuring the required pace of NP development.

The pace of NP development based on the own funds of the utility company is characterized by the doubling time of the total NPP power — $T_{2,}$ in compliance with the investment potentials of the used NPT, that is similar to the time of doubling plutonium in breeder-reactors. T_2 is determined by the following expression:

$$T_2 = \frac{100 \cdot C_{cap}^{d}}{LF_{av} \cdot T_y \cdot C_d}, \tag{12}$$

where:

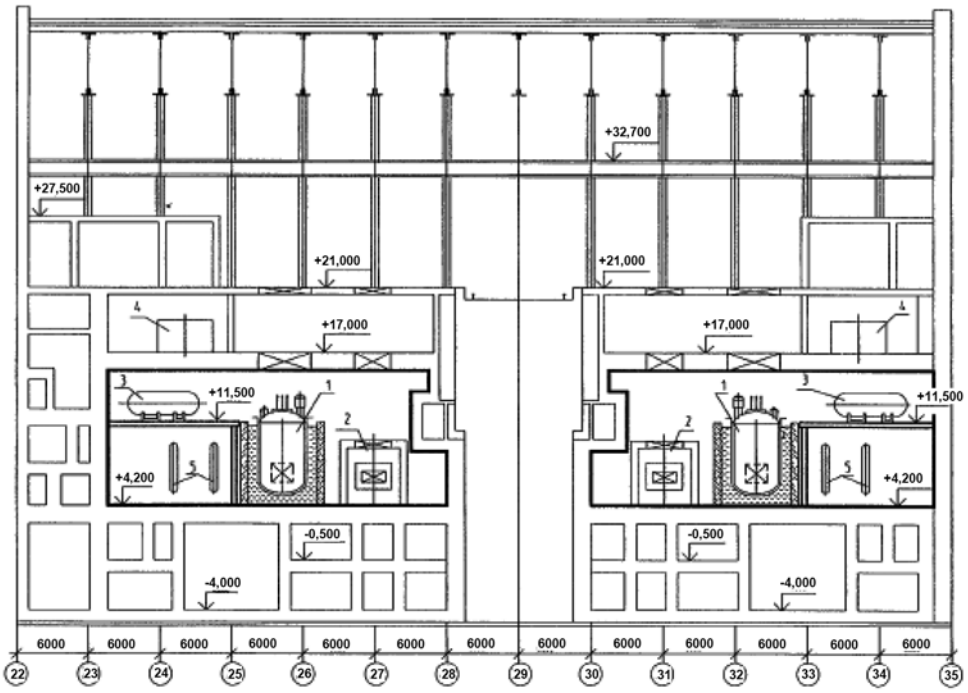
LF_{av} is the averaged LF over all NPPs of the utility company,
T_y is the number of hours in a year (h),
C_{cap}^{d} is the value of the specific capital costs of constructing the NPPs ($/kW(e)),
C_d is the investment component in the tariff for NP development (c/kW·h).

The investment component for NP development equals to

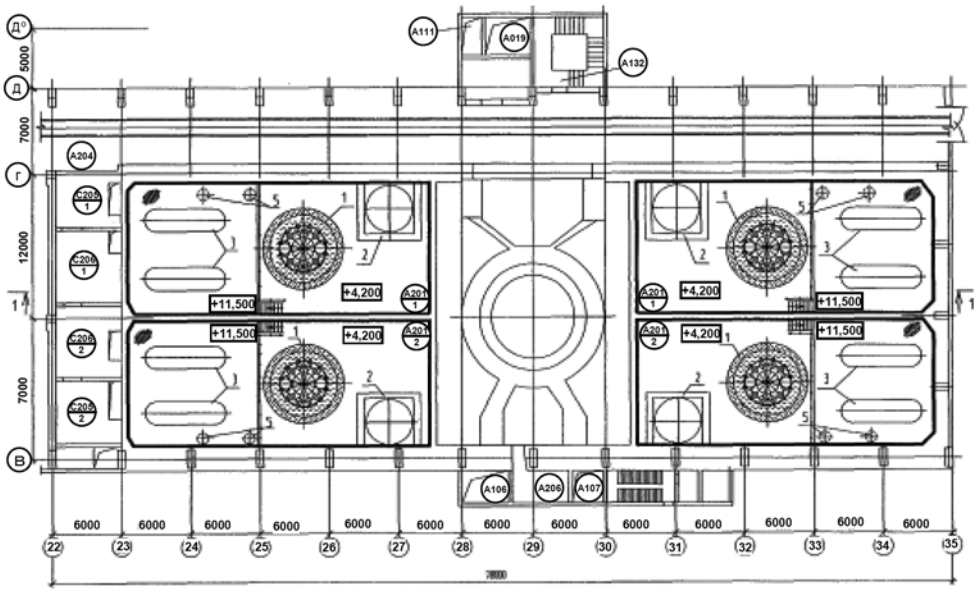
$$C_d = C_{inv} - C_r, \tag{13}$$

where:

C is the tariff, i. e. the cost of electricity at the wholesale market (c/kW·h),
C_{inv} is the total investment component in the tariff: $C_{inv} = C - C_e$, (c/kW·h),
C_e is the electricity cost that compensates for the expenditures for its production and includes all current expenditures except for the investment assignments for NPP construction (c/kW·h),
C_r is the investment component that compensates for the expenditures for construction of the replacing capacities i.e. simple reproduction of the NP replacing capacities (c/kW·h).



a)



b)

FIG. 14 (a) (b). Longitudinal (a) and horizontal (b) sections of the main building of the 2nd NVNPP unit after renovation.

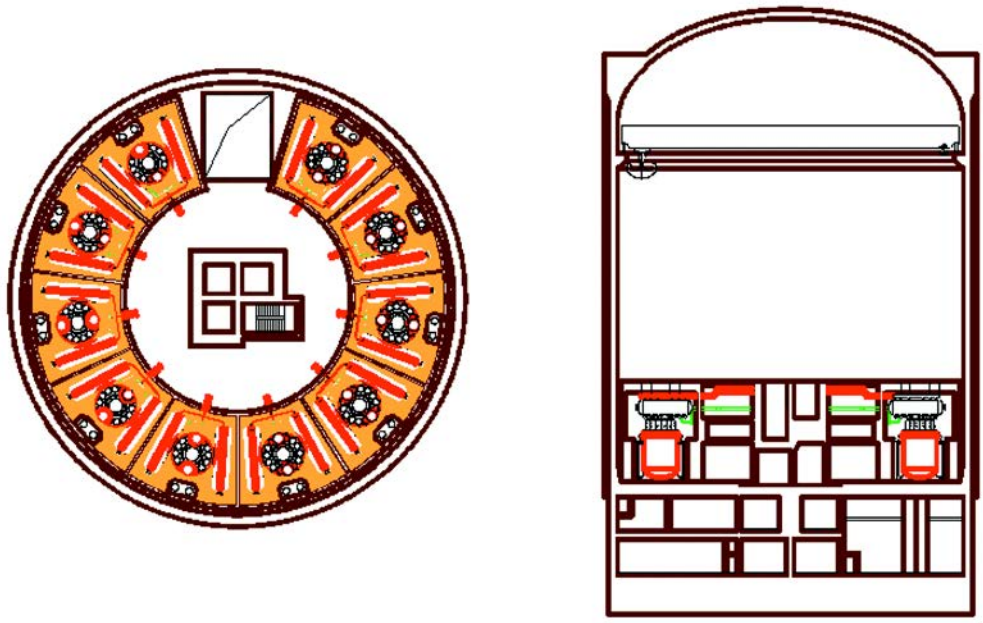


FIG. 15. Renovation scheme of a NPP WWER-1000 unit.

C_r is determined by the expression:

$$C_r = \frac{W \cdot C_{cap}^r}{N_{NP} \cdot LF_{av} \cdot 87.6} \tag{14}$$

where:

W is the pace of withdrawing NP capacities (GW(e)/a),
N_{NP} is the total power of the NPP (GW(e)),
C_{cap}^r is the specific capital cost of constructing the replacing capacities.

The NPT investment potential can be characterized by an investment breeding ratio (IBR) that determines the moving forces and trends of development (phase out) of the NP depending on the economical parameters used by the NPT and the cost conditions of the electricity market. The IBR can be calculated by using the following formula:

$$IBR = \frac{C_{inv} - C_r}{C_r} \cdot \frac{C_{cap}^r}{C_{cap}^d} + 1 \tag{15}$$

If for construction of the replacing capacities and construction of the new NPP units, the same NPT is used, $C_{cap}^r = C_{cap}^d$.

Then

$$IBR = \frac{C_{inv}}{C_r} \tag{16}$$

If $C_{inv} < C_r$ (in this case C_d is negative), then IBR < 1 and nuclear power will be phased out. If IBR = 1, then the total NPP power will be stable (simple reproduction). Only IBR > 1 provides NP increase — extended reproduction. It is evident that the relation T_2 connects the time of doubling and the IBR:

$$T_2 = \frac{N_{NP}}{W \cdot (IBR - 1)}. \quad (17)$$

If IBR < 1, then the notion of the doubling time has no sense. Instead, we should introduce the time of NP phasing out – T_{ph} that is determined by the expression:

$$T_{ph} = \frac{N_{NP}}{W - I_r} \quad (18)$$

where:

I_r is the pace of introducing the replacing capacities that in the considered case is lower than the pace of withdrawing the 'old' units (W).

The possible pace of introducing the replacing capacities is determined by the expression:

$$I_r = \frac{N_{NP} \cdot LF_{av} \cdot 87.6 \cdot C_{inv}}{C^r_{cap}} \quad (19)$$

However, the doubling time determined above does not reflect the absolute NP capacities increase pace. In practice, the more useful parameter is the value of the possible annual NP capacities increase pace (at IBR > 1).

Assuming linear increase of the total NPP power, the annual pace of NP capacities increase (Gw(e)/y) is determined by the expression:

$$I = \frac{N_{NP}}{T_2} \quad (20)$$

It can be seen that $I = W \cdot (IBR–1)$. If IBR < 1, I is negative, which means that the annual NP capacity is decreasing. Knowing the pace of NP increase I, the pace of withdrawing the 'old' units W and the pace of constructing the replacing capacities I_r (at $C_{inv} < C_r$), the investment-breeding ratio can be written as:

$$IBR = 1 + \frac{I}{W} \quad \text{with } C_{inv} > C_r \quad (21)$$

and

$$\text{with } C_{inv} < C_r \quad (22)$$

Calculations of the dynamics of the development (phase out) of Russian NP have been carried out for the following scenarios:

— Replacement of the phased out capacities and NP development using NPT based on WWER-1000;
— Replacement of the phased out capacities and NP development using NPT based on WWER-1500;

— Replacement of the phased out capacities using innovative technology based on SVBR-75/100, and development of NP using NPT based on WWER-1000;
— Replacement of the phased out capacities using innovative technology based on SVBR-75/100, and development of NP using NPT based on WWER-1500.

Table 15 summarizes the results of the parameters characterizing the dynamics of the NP development (C_r, C_d, IBR, T_2, I), for the four scenarios.

The obtained data is relevant to the stable stage of NP development, when other opportunities of 'low cost' capacity increases [153] (e.g. LF increase, unit lifetime extension, etc.) have been exhausted.

The calculations were carried out assuming the following initial data: C_{cap} = 400 $/kW(e) for renovating the units using SVBR-75/100 modules [147]; W = 1 GW(e)/a, which corresponds to the conditions of Russian NP; C_e = 0.72 c/kW·h; C_{cap} = 819 $/kW(e) for two-unit NPP WWER-1000 [149]; C_{cap} = 680 $/kW(e) for two-unit NPP WWER-1500 [154]; N_{NP}=22.5 GW(e); LF_{av} = 0.75; C = 1.4 c/kW·h [155].

The specific capital costs cited in Ref. [147] were calculated in rubles of 1991 in compliance with the normative documentation being in force. Thus, correct comparisons (with due account to inflation) of the economic parameters of NPP designs realized in different years was assured. When the ruble costs were converted to the costs in dollars, it was assumed that one ruble of 1991 equals to one dollar.

On the assumption that $C_r = C_{inv}$, the critical value of the specific capital costs can be obtained: C_{cap} = 1005 $/kW(e), at which IBR = 1, that is when NPP power will remain stable (simple reproduction). Thus, based on the obtained results, it can be concluded that renovating the 'old' NPP units based on substitution with reactor installations SVBR-75/100 will increase the expense of the utility's own funds approximately two times, with respect to the evolution of NPTs based on WWER-1000 and WWER-1500.

It should be noted, however, that the data in Table 15 cannot be considered as fully correct, because the values of the specific capital costs used in the calculations are in dollars of 1991, while the tariff values and the cost of electricity are in cents of 2003.

To provide comparability of prices, the tariff and cost values of electricity should be reduced to the prices of 1991 with due account of dollar inflation during this period. They are: C = 1 c/kW·h; C_e = 0.514 c/kW·h (the reduction coefficient is adopted to be 1.4 on the basis of extrapolating the data on the yearly indices of the consumer prices in the USA (USA Yearbook, 2000)). If this data is used, the results of the dynamics of NP development (phase out) will considerably change. They are summarized in Table 16.

These corrected results reveal that the most effective of the considered scenarios is combining the innovative NPT based on modular SVBR-75/100 for the renovation of the NPP 'old' units with the evolutionary NPT based on WWER-1500 for NP development.

In the considered case the critical value of the specific capital costs that provides simple reproduction of NP capacities (IBR = 1) equals to 718 $/kW(e).

Of course, the results obtained should be considered as tentative, since they are based on current parameters (tariff, electricity cost, specific capital costs, LF, total NPP power), used for calculating the NP development dynamics in 15–20 years.

TABLE 15. DYNAMICS OF NP DEVELOPMENT IN THE RUSSIAN FEDERATION ASSUMING VARIOUS NPT

Nuclear power technology	Parameter				
	C_r c/kW·h	C_d c/kW·h	IBR	T_2 years	I GW(e)/a
According to scenario No 1	0.554	0.126	1.23	99	0.23
According to scenario No 2	0.46	0.22	1.48	47	0.48
According to scenario No 3	0.27	0.41	1.48	47	0.48
According to scenario No 4	0.27	0.41	1.89	25	0.89

TABLE 16. DYNAMICS OF NP DEVELOPMENT IN THE RUSSIAN FEDERATION ASSUMING VARIOUS NPT (DOLLAR INFLATION HAS BEEN TAKEN INTO ACCOUNT)

Nuclear power technology	Parameter				
	C_r c/kW·h	C_d (C_{inv}) c/kW·h	IBR	T_2 (T_{ph}) years	I (I_r) GW(e)/a
According to scenario No 1	0.554	(0.486)	0.877	(183)	(0.877)
According to scenario No 2	0.46	0.026	1.057	398	0.057
According to scenario No 3	0.27	0.216	1.391	57.5	0.391
According to scenario No 4	0.27	0.216	1.471	47.8	0.471

Among the considered factors, only the increase of total NPP power results in an increase of the investment potentials. Increase of LF will result in increasing the electricity production and increasing the sales but it will require progressive increase of costs that will cause reduction of the economical effectiveness of this factor. The tariff increase will increase the NP investment potentials. At the same time, the increase of the cost of carbon based powercarriers that result in increasing electricity costs, as well as the increase of other costs like production and servicing, increase of salaries, will result in the increase of the capital and operating costs will and therefore lead to the reduction of the NP investment potentials. Therefore, it can be expected that the tendency revealed by the present studies will be retained.

8.5.10. Commercialization concept

Despite maximal possible use of experience of operating the LBE cooled reactors at the nuclear submarines, the conditions of operating the equipment of nuclear submarine reactor installations and NPP reactor installations are much different. The operating mode of nuclear submarine reactor installations is characterized by operation mainly at low power levels under lowered LBE temperatures, whereas the NPP reactor installations operate mainly at nominal power. Moreover, the requirements to the lifetime of the NPP reactor installations equipment are much higher with respect to those of the nuclear submarine reactor installations equipment. The technical and economical parameters also need direct verification.

These considerations require construction of an experimental industrial power unit with a SVBR-75/100 reactor. According to current assessments, the cost of its construction including the R&D cost will be ~$ 150–200M depending on the site. It should be highlighted that there will be only on one-time expenditures for the R&D and for constructing the experimental-industrial prototype.

The experimental-industrial prototype equipped with additional sensors and devices may be used for demonstration of the inherent self-protection and passive safety properties of the reactor installation under controlled conditions, while simulating all the possible super-positions of equipment failures, personnel's errors and malevolent actions.

After the certified tests of the experimental-industrial prototype have been performed and the design characteristics have been proved, the SVBR-75/100 will be ready for commercialization and wide use as a part of the NPP power-units of different capacity and for different applications.

8.6. CONCLUSIONS

— There is potential to considerably increase the investment attractiveness of nuclear power technology based on the use of fast reactors. Hence, their deployment appears likely, even under low natural uranium cost conditions;
— In particular, employing innovative NPT based on the 'standard' modular multi-purpose FRs with chemically inert lead-bismuth coolant (SVBR-75/100), possessing inherent self-protection and passive safety properties (deterministical elimination of severe accidents) has the potential to eliminate the conflict between safety and

economics requirements that has traditionally jeopardized reactor technologies, and hence achieve a high social acceptability level;

— The power level, safety, and non-proliferation properties of SVBR-75/100 make it ideal for deployment in developing countries;
— Among the different utilization areas, SVBR-75/100 is particularly suitable as replacement for NPP units that have reached the end of their lifetime deploying SVBR-75/100 as replacement allows higher NP development pace, as compared with the pace of NP increase if only large power LWRs are used;
— The modular structure of the power-unit's NSSS offers the opportunity of progressive switching over to 'standard' transportable reactor modules factory fabrication;
— At different stages of NP development, SVBR-75/100 can operate using different types of fuel in different fuel cycles without changing their design, Hence it is providing gradual and economical transition to the closed NFC;
— SVBR-75/100 is developed on the basis of a conservative approach with due account to experience of operating the LBE cooled reactors in nuclear submarines;
— The conservative approach adopted in designing the SVBR-75/100 allows reduction of R&D effort and cost, as well as of both technical and investment risks during construction of the demonstration plant. At the same time competitiveness of the modula concept can be achieved;
— Rosatom's Scientific and Technical Council No 1 on 15 June 2006 considered the prospects of the use of reactors SVBR-75/100 in nuclear power and recommended in 2007 to continue development of the technical design of the experimental-industrial power-unit with SVBR-75/100;
— The Federal Target Programme Development of the Nuclear Power Industrial Complex in the Russian Federation for the Period of 2007–2010 Years and up to the Year 2015 is foreseeing construction of the first experimental-industrial power-unit of the SVBR-75/100.

9. LIQUID LEAD COOLED FAST REACTORS

In the 1980s, a critical analysis on the experiences gained in the nuclear power sector after the events in the USA, (Three Mile Island (TMI) accident) and in the Russian Federation, (Chernobyl), lead to a revision of the nuclear power mission in scientific terms. This revision included the reactor concepts highlighting the principles of inherent (or natural) safety, while keeping in mind the need of NPP cost reduction [156, 157].

The words 'inherent safety' came into general use, but was interpreted by 'pragmatists' in the sense of just wider application of 'passive' safety and cooling features, feedback, etc.

In the late 80s, a NIKIET (Research and Development Institute of Power Engineering) team of physicians and designers, led by Director E.O. Adamov, continued computational and design studies initiated earlier at the Kurchatov Institute [158], and with FEI (IPPE), VNIINM (All-Russia Research Institute of Inorganic Materials), VNIITF (All-Russia Scientific Research Institute of Technical Physics), CNIIKM (Central Scientific Research Institute for Construction Materials) and other institutes joining the team, launched experimental investigations of unconventional NPP concepts [156, 158]. By the beginning of the 90s, a new statement of the nuclear power mission was developed and the BREST concept was defined, with the principle of inherent safety extended to the problems of waste and proliferation. With the notion of inherent safety having lost its originally clear-cut meaning, this extension came to be referred to as 'natural safety', a better sounding expression in Russian [159–164].

By now, the basic research has been completed, a pilot BREST-300 plant has been designed with the Beloyarsk NPP site in mind [165–168], and the supporting R&D effort [166] has been carried out, except for the final stage of in-pile fuel tests (in progress) and the rig testing of the basic equipment [167].

9.1. THE LEAD COOLED FAST REACTOR CONCEPT

Natural safety of fast reactors implies that severe accidents are eliminated despite any equipment failures, human errors or external impacts (except for nuclear and other extreme impacts leading to NPP destruction). Natural safety is ensured primarily by an equilibrium mode of fuel burning, with the reactivity margin reduced to $\Delta k_{tot} \sim \beta_{eff}$, when even its full input will never result in a prompt criticality excursion [169].

The initial period of reactor operation with the use of plutonium from thermal reactors, while its isotopic composition moves towards an equilibrium and fission products are accumulated, necessitates particular attention and special provisions.

The preferred fuel is mononitride, which has a high density and heat conductivity. It is also heat and radiation resistant. However, neutron losses in the reaction ^{14}N (n, p) ^{14}C and generation of environmentally hazardous ^{14}C is an issue to be addressed [170–177].

Given the relatively low temperatures and the clearance to the cladding, mononitride fuel with low C and O content may reach very high burnup levels. The burnup limit is therefore imposed by neutron-induced damage of the fuel claddings [171]. A large negative temperature reactivity effect dK/dT, at low temperatures, causes the reactor to 'shut down itself' in the event of failure of the pumps without letting the temperatures rise too high (residual heat will be removed by natural circulation of the coolant).

The fuel ensures BR≈1 even with a larger fuel rod pitch compared to oxide fuel. This ensures thermohydraulic conditions needed for heat removal from it by heavy liquid metal. The fuel fabrication process has been developed, and a production line has been set up at VNIINM to turn out fuel rods for tests in BOR-60, already in progress [173–179].

Heat removal from fuel occurs through lead. Lead has a high boiling temperature, a low chemical and radiation activity, and is not expensive. The circulation in a reinforced-concrete vault rules out any severe loss of coolant accident (LOCA) caused by circuit integrity loss, coolant boiling, burning or escape through cracks in the vault (high $T^{Pb}_{melt} = 327°C$). In the event of a pump trip, followed by passive reactor shutdown, residual heat will be removed by natural circulation, and by the contiguous air running in pipes immersed in the lead, for as long as may be required.

The 'wide' fuel rod lattice amply offsets the low velocity of the heavy metal and the high T^{Pb}_{melt}. This design solution is feasible due to the high-density fuel and low neutron moderation by the heavy metal. Moreover, it results that the Pb temperature gain, ΔT, fuel cladding temperatures, and the Pb pumping power requirements are smaller than in a Na cooled reactor.

The steam generator with water under high pressure, which may even be supercritical considering the high T^{Pb}_{melt}, is a problematic issue in view of the clear demonstration of natural safety.

Multiple rupture of SG tubes is fraught with severe consequences. Therefore, particular attention has been given to the elimination of reactor accidents with Pb temperature increase at the SG inlet to a hazardous, in this respect, level above ~700°C (from the nominal 540°C).

Based on the findings of FEI and CNIIKM experiments conducted over the last ~17 years, it has been shown that the corrosion control (oxide film maintenance) technology practiced effectively in lead-bismuth is also suitable for lead, provided it is appropriately modified. With oxygen concentration in lead kept at an optimal level of about 10^{-6} %, the experiments (~15 000 hours in duration) confirm high corrosion resistance of the steels developed for lead-bismuth. Moreover, it has been shown that deviations from the optimal oxygen concentration by more than an order of magnitude for hundreds or even thousands of hours can be acceptable. Facilities have been developed and are being tested for monitoring the oxygen level in lead and its automatic maintenance, for filtering of oxides and their subsequent removal from the circuit in case of an upset coolant chemistry [179–183]. Calculations show that flow mixing by pumps and turbulence will quickly restore the oxygen concentration conditions, which may be locally upset in the huge coolant volume for one reason or another. Naturally, precautions should be taken against pockets and stagnation zones, where prolonged departure from the design chemistry parameters may give rise to corrosion, as well as against mechanical damage to oxide films due to friction and vibrations. BOR-60 was used for the first test of a 'loop channel' with Pb circulation, equipped with a pump and an oxygen control system, and having 4 U-Pu-N fuel rods.

The silicon-alloyed ferritic-martensitic steels developed for Pb-Bi show the best corrosion resistance to Pb, as well as radiation resistance when exposed to fast neutron fluxes.

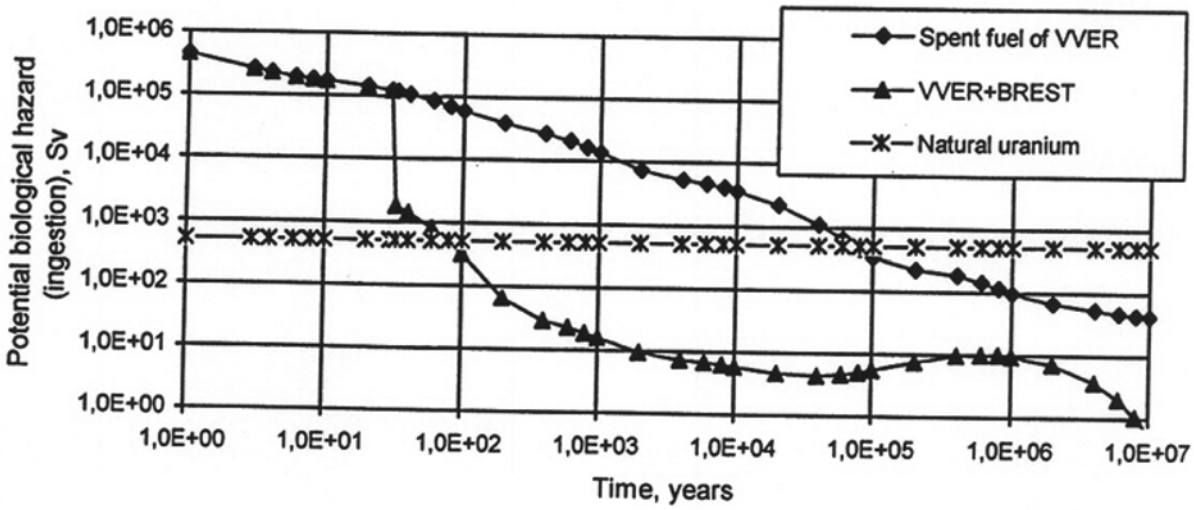


FIG. 16. Trends of radioactivity vs. time.

However, these steels have lower allowable temperatures with respect to austenitic steels (creep, etc.). In the fast reactor design with a wide-spaced fuel rod lattice and fuel having equilibrium composition, the Pb temperature gain ($\Delta T \sim 100°C$) is modest. Overheating factors are also limited, so that the Pb outlet temperature does not exceed ~540°C with the 'hot spot' of a fuel cladding temperature $T_{max} \approx 650°C$, which is tolerable for the steels in question. Low stresses in the cladding, free from pressure of swelling fuel and with fission gas pressure balanced by the external lead pressure (~10 atm), are prerequisite for high fuel burnup. At $T_{Pb} \geq 420°C$, these steels are beyond the region of low temperature radiation embrittlement.

Further progress of lead cooled fast reactor may be achieved by raising Pb temperature (perhaps, to 780°C, as in the STAR Concept (Argonne National Laboratory [ANL]) [176]), acceptable with high T^{Pb}_{boil}, for the purpose of increasing the efficiency factor and using high grade heat for industrial needs, such as hydrogen production. This would necessitate new heat resistant structural materials; e.g. metals (Nb alloys, etc.), ODS or ceramics (SiC) which would need extensive tests to be qualified at high fast neutron fluences and in lead environment.

9.2. NUCLEAR FUEL CYCLE CONCEPT

In terms of natural safety, the requirement placed on the management of RAW lies in keeping the natural radiation balance unaffected when, in the future, such waste is supposedly buried in the radioactive formations at former mining sites, where uranium was extracted together with its long lived decay products (Th, Ra), during their reclamation (see Fig. 16).

Neutron excess and fission of all actinides by fast neutrons allow reaching equilibrium with Pu. The fuel reprocessing technology is required to provide thorough removal of actinides from the waste to a remainder of $\sim 10^{-3}$, with the radiotoxicity of the latter comparable to that of the mined U together with Th and Ra, as well as separation of the long lived fission products, ^{129}I and ^{99}Tc, for subsequent transmutation in fast reactors [156].

For easier fuel fabrication, it is better to separate Cm and cool it for ~70 years until it decays into Pu. Moreover, Cs should be separated to facilitate waste disposal and for use elsewhere. For the same purpose, the radwaste should be kept in an interim storage for ~100 years.

The isotope ^{14}C arising from ^{14}N accounts for ~1% of waste radiotoxicity and is dangerous if emitted as gas into the air or if it is dissolved in groundwater.

Dry reprocessing techniques do not give rise to CO_2, but it is essential to have C fixed in stable compounds in the radioactive waste.

The limits to fuel concentration in water solutions make dry reprocessing techniques preferable for FR fuel. The preferred technological option is the electrochemical extraction in molten salts.

The radiation balance oriented technologies should be designed by the time of nuclear power deployment of fast reactors (2020 s), with certain modification of methods for reprocessing TRU fuel so as to provide the fuel for the new fast reactors [167, 168, 184, 185].

Political decisions together with control and physical protection arrangements act as deterrents to proliferation of nuclear weapons in the world, but they have proved incapable of stopping it entirely.

Circulation of ^{235}U and Pu in amounts in the order of thousands of tones in a large nuclear power industry of the 21st century opens up 'legitimate' paths of access to materials suitable for use in weapons, international control over which with an accuracy of tens to hundreds of kilograms is extremely difficult. There are no legal grounds for forbidding countries to build their own U enrichment and closed fuel cycle facilities for the sake of their energy self-sufficiency. The non-proliferation problem might be resolved without prejudicing the legitimate rights of nations as part of the general objective of developing economically efficient and safe nuclear power industry based on fast reactors which require neither U enrichment, nor separation of Pu from fuel of equilibrium composition, provided there is proper physical protection against fuel thefts for reprocessing elsewhere.

Fast reactor fuel requires moderate removal of fission products by reprocessing, which allows application of physical (plasma) methods of separation relying on the two-fold difference in atomic weight and incapable of separating Pu from U[2]. A thorium blanket in a fast reactor can generate ^{233}U to concentrations required for thermal reactors.

This is clearly a long term objective, and nuclear states (or international centers) will have a long time to resort to U enrichment and reprocessing of spent fuel from thermal reactors with separation of U for FR inventories, including, perhaps, such services for small countries. On the other hand, larger nations will hardly accept them, unless they see prospects for their subsequent independent development. It is today that we should start moving towards provision of proliferation-resistant fast reactors and closed fuel cycle.

Otherwise, the nations with an interest in nuclear power are likely to proceed to acquire the technologies of U enrichment and fast reactors with separation of Pu, and neither political nor technical palliatives will be able to stop some countries or malevolent groups from laying hands on nuclear weapons.

It appears that the development of fast reactors and their large scale deployment is a highly complex objective. Addressing any one of its aspects, such as coolant technology in isolation, may never be a complete job. The coolant technology problem e.g. requires recognition of its general aspects, and due consideration must be given to other design solutions with respect to fuel, structural material, etc... Some specific results of studies will be discussed below, with special emphasis on the problem of lead coolant.

9.3. BREST CONCEPT

Large NPPs with power units designed for hundreds or thousands of MW are best suited to meet economic and other requirements of the power industry, especially of nuclear power, and will remain the trunk line of its development in the 21st century (with small and medium sized power plants still being built wherever required and profitable).

BREST is a lead cooled fast reactor with nitride fuel of equilibrium composition and moderate power density (5–6 t Pu_{fis}/GW — twice as large as in Na cooled fast reactors). BREST power rating is 300 MW(e) and more.

The 300 MW(e) rating for BREST is the lower limit ensuring an appropriate neutron balance for fuel of equilibrium composition with the maximum pellet density (90% of theoretical value).

The limits for Pb velocity (~2 m/s), T_{clad}^{max} = 650°C, T_{Pb}^{melt} = 327°C (with a margin of ~100°C), and for the increase in Pu content dictate the temperature conditions and the configuration of the core: T_{Pb}^{in} = 420°C, T_{Pb}^{out} = 540°C. In the maximum neutron flux region, $d_{f/rod}$ = 9.4 mm, the cladding is 0.5 mm thick, the cladding-to-pellet gap is 0.25 mm, the fuel rod pitch is 13 mm, its height is 1.1 m (filled with Pb) and together with the gas plenum, 2 m, $\Delta P \approx 1.5$ atm. The Pb/fuel volume ratio is ~1.5, with the use of spacer grids.

[2] Studies on plasma techniques of nuclear and fuel reprocessing have started at the Nuclear Fusion Institute of the Kurchatov Centre.

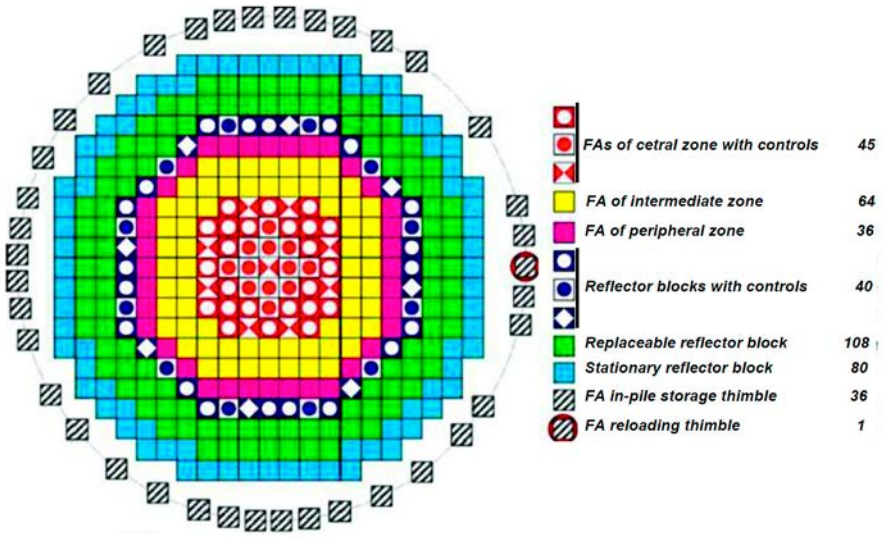


FIG. 17. Brest-OD-300 core layout.

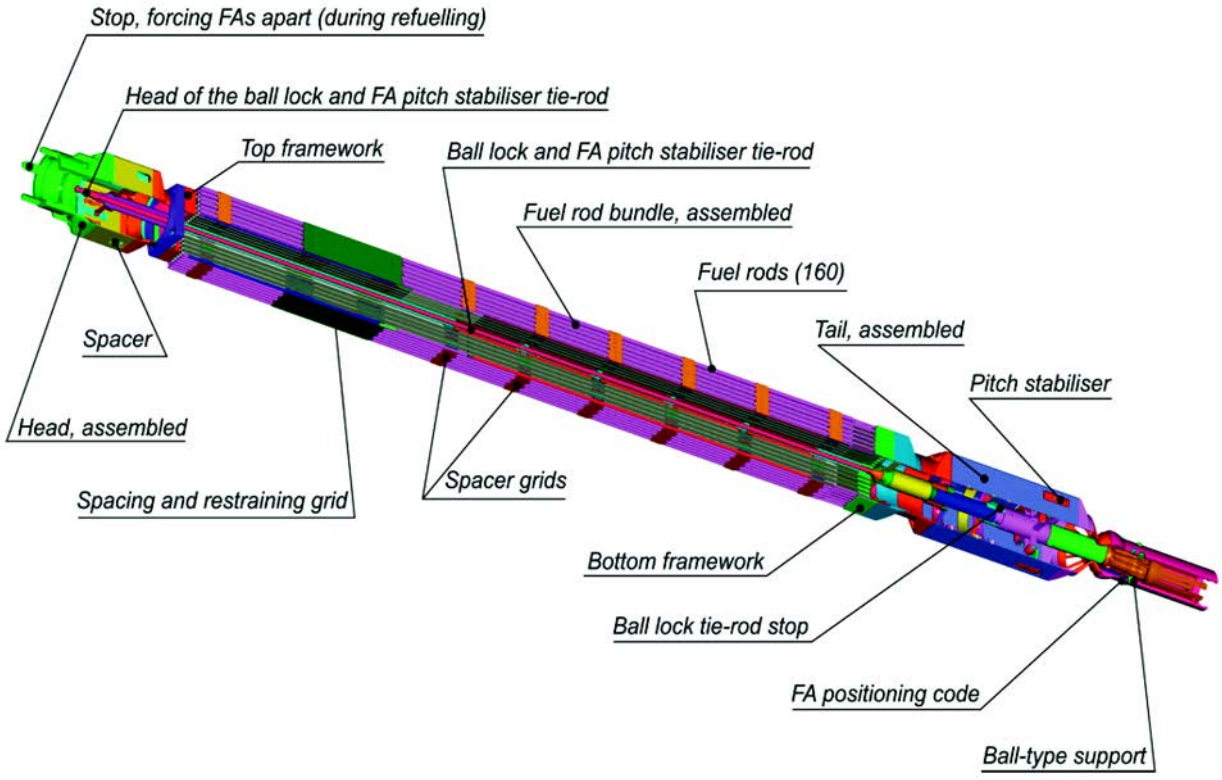


FIG. 18. General view of BREST-OD-300 fuel assembly.

The fuel assembly has no shroud ($D_{pb}^{hydr} \approx 14$ mm), which prevents flow blockage; it measures 17 × 17 cm (square cross-section) and has a clearance of up to 5 mm (neighboring assemblies are moved apart by the refueling mechanism).

There are three profiling zones using fuel rods with diameters 9.4, 9.8, and 10.2 mm, which allow matching power and Pb flow rates and stabilizing the distribution of power and temperatures during burnup (Figs 17–19).

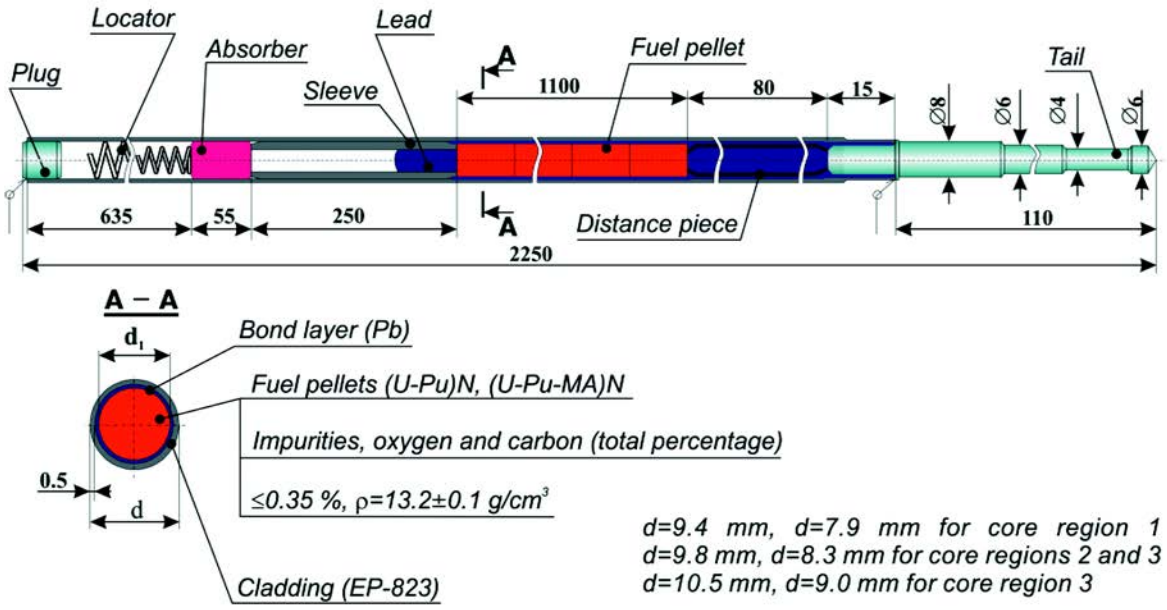


FIG. 19. General view of fuel rod.

Central zone, high-worth (ΔK ≈ 2%) passive shutdown (upon trip of pumps) and hold-down system.

Two-position absorbers in tubes are raised by Pb flow and sink, being heavier than lead, when the flow stops.

In the reflector, absorbers and Pb columns driven hydraulically, pneumatically or mechanically, with Pb, gas or cables fed from below, serve for compensation of the power and temperature effects of burnup and the Np effect, for regulation, emergency protection and passive coordination of flow rate and power.

$\Delta K_{tot} \approx 0.35\% \approx \beta_{eff}$

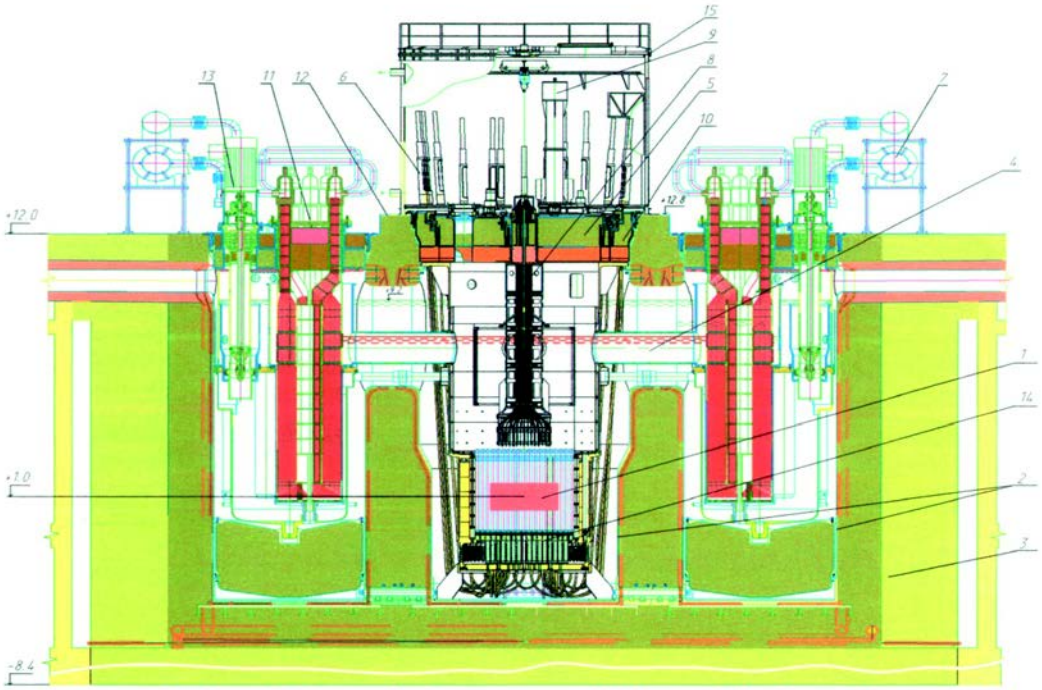
Reactivity is regulated from the bottom to leave the 'top' free for refueling.

9.3.1. Lead circulation arrangement

Lead is pumped from the SG outlet upward, to a pressure chamber with the Pb level of 1.5 m above the main one. The pressure of 1.5 atm drives Pb down the clearance between the vault and the separating steel shell at a velocity of <0.5 m/s, and then upward. Part of Pb is going through the reflector — via square steel tubes with throttle orifices at the inlet, whose row closest to the core accommodates CPS rods. Further, Pb goes through the interim storage for unloaded fuel (cooled for 1 year in case of partial refueling or for 3–4 months in case of 'continuous' refueling before being moved to a buffer chamber of the on-site closed fuel cycle). The bulk of the Pb flow, heated in the core, will rise to the boundary with the gas 'cushion' (Ar) and then on, via 'passages' to steam generators, which are located in a reinforced-concrete vault in individual 'wells' sharing the global Pb levels and gas volume [179]. The coolant circulation is shown in Fig. 20.

This arrangement needs a thorough investigation in terms of resistance to possible oscillations and to occurrence of turbulences in the main flow of 'hot' Pb and its slow-moving part above the reflector, but it has the following characteristics with considerable safety implications:

— A delay of ~20 s before the flow rate drops after a quick pumps trip (without flywheels), and its matching to the decreasing power by the temperature effect (in case of emergency protection failure) without excessive temperature growth;
— Reduction of accidental temperature excursion due to heat exchange with the Pb mass on the way to the SG inlet;
— Separation of steam during SG leaks or gas bubbles entrained by Pb into gas space.



1-reactor core; 2-reactor vessel; 3-concrete; 4-piping collector; 5-large rotating plug; 6-control rod drive mechanism; 7-decay heat removal system; 8- upper core structure; 9-fuel handling machine; 10-small rotating plug; 11-steam generator; 12-upper stationary shield slab; 13-main pump; 14-core support structure; 15-protective dome

FIG. 20. The BREST-OD-300 reactor.

Small bubbles left unseparated are compressed at ~10 atm and will not give rise to a hazardous void effect of reactivity (calculations and experiments at the BFS (fast critical assembly) test rig show that small hydrogen quantities have an insignificant effect on reactivity).

In the hypothetical case of large bubble quantities, variations in the Pb density will stop or reverse the circulation in the affected circuit section.

The low reactivity worth of the fuel assemblies and the small reactivity difference between fresh and spent assemblies, along with the moderate power densities, allow quasi-continuous refueling during dayor week long load reduction without shutting down the reactor.

Besides an increased capacity factor, with the facilities of the ‘equilibrium’ closed fuel cycle (CFC) co-located with large FR plants (~1 GW and up), there are additional significant benefits for economy safety, and non-proliferation:

— Considerable reduction in the cooling time, fuel quantities at a plant, size of its in-pile and external storage facilities, in long distance shipment of radioactive and fissile materials, as well as the number of organizations involved;
— Simpler, but reliable physical ‘protection in depth’, against theft or illicit removal of fuel for reprocessing elsewhere. With Pu separation (and U enrichment) excluded, proliferation-resistance is enhanced;
— Reduction of the Pu quantity needed to start fast reactors and their CFC. This off-sets the limitation due to the lower power density;
— During equilibrium operation, reduction of the impact of uncertainties in physical and operational parameters, as well as of the deviation in the fuel composition from beginning of operation.

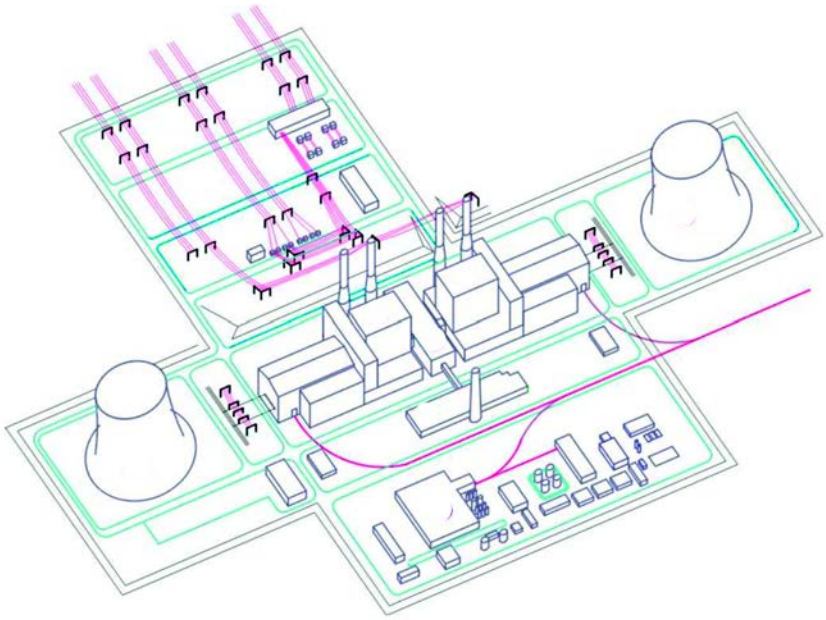


FIG. 21. General layout of NPP with two reactor units BREST-1200.

Preliminary design studies performed for a 2400 MW(e) BREST plant with two reactor-turbine sets, 1200 MW(e) each, suggest that the plant is likely to show much better economic performance in comparison both with traditional NPPs and with advance combined cycle facilities (Fig. 21).

However, the economical estimates made at the conceptual stage were very approximately, especially since they were using Russian methodology of the early 1990s, which have changed since then in a considerable way. Nevertheless, a qualitative comparison may provide a true picture of the basic aspects, and highlight the key characteristic of the BREST concept; i.e. the low nuclear fuel price and a lower cost of an NPP built assuming principles of natural safety.

On-load refueling places special requirements on refueling mechanisms as well as on the thermalhydraulic conditions for replacement of a spent fuel assembly with a fresh one. These issues need investigation, development and verification at the pilot prototype reactor BREST-300. Considering the still insufficient knowledge related to the radiation and corrosion resistance of the steel chosen for fuel claddings, conservative burnup levels are assumed in the design; i.e. 6% h.a. on the average and 10% as a maximum. These problems are among those covered by the R&D programme to be carried out at BREST-300 itself, where studies will be made both on fuel with high burnup (there seem to be no basic limits for reaching 15 or, perhaps, even 20% h.a.) and on refueling. Table 17 shows comparative economic efficiency of BREST-1200 plant, NV NPP-2 and TPP with CCP-450.

9.4. BREST-1200

The reduced neutron leakage due to a larger reactor size makes several percent of neutrons available for other uses, within the same conceptual approaches to fuel, core configuration, Pb circulation, control and refueling systems, as well as reprocessing procedures.

The first and foremost objective is to increase burnup by providing a larger fuel-to-cladding clearance, which together with a decrease of the fuel pellet density was one of the provisions made in the first concept of BREST-1200.

An increase in Pb volume proportion (or core height reduction) allows, at constant Δp, increasing the Pb flow rate, reducing ΔT and raising T^{out} to 550°C or slightly higher and thus higher steam temperature and efficiency.

TABLE 17. COMPARATIVE ECONOMIC EFFICIENCY OF BREST-1200, NVNPP-2 AND TPP WITH CCP-450

Indicator and its unit	BREST-1200 NPP	WWER-1000, NVNPP-2	CCP*-450 TPP
Power unit installed capacity at the generator terminals, MW(e)	1 296.5	1 068	450
Number of power units	2	2	2
NPP installed capacity (at the generator terminals), MW(e)	2 593	2 136	2 250
Fuel type	U-Pu nitride fuel	Enriched U dioxide	Natural gas
A set of the main reactor (boiler) equipment	2 × BREST-1200	2 × WWER-1000	5 × CCP-450
Power unit service life, years	30	30	30
Annual utilization of installed capacity, h/year	7 000	7 000	7 000
Annual generation of electricity, mln. kW·h/year	18 151	14 952	15 750
In-house electricity consumption, % of generation (service water system with cooling towers)	5.15	6.43	4.5
Annual delivery of electricity to the grid, mln. kW·h/year	17 216	13 991	15 041
NPP efficiency			
Gross, %	46.3	36.2	48.0
Net, %	43.9	33.3	45.8
Specific consumption of fossil fuel for electricity delivered, g.c.e./kW·h	—	—	277
Annual consumption of fossil fuel, thousands T.c.e/year	—	—	4 166
Investment in NPP construction, rel. units	1	1.19	1.006
Relative electricity production cost	1	1.65	1.82/2.2

* Captive power plant

But higher power, N, and larger core diameter, D, result in a smaller ratio of surface areas to volume, $S/N \sim 1/D \sim 1/\sqrt{N}$; i.e. detract from the worth of reactivity controls, and present difficulties for arrangement of steam generators, pumps, air pipes for decay heat removal and actuators of pneumatic and hydraulic drives. Higher reactor power achieved by increasing the core diameter, with larger dimensions of the top plate and rotating plug, leads to a design that is more complicated and has certain limits [180].

That is why the Bauman University and MIFI (Moscow Engineering Physics Institute) together with NIKIET are also looking into other options — reactors of different geometry with a larger surface area-to-volume ratio. One of them is an annular configuration which is free from 'end parts' and in which azimuthally power tilts are limited by even harmonics. However, it is difficult to make efficient use of the inner space of an annulus, while the gain in design is not very rewarding.

In terms of design, a promising option is a square bean-shaped reactor of length, L, with lengthwise constant neutron flux distribution, with a height, H, close to that of BREST-300, a width $M \approx D_{BREST\text{-}300}$ and having two 'mirrors' — BREST-300 halves — at both ends.

The reactor is quite similar to BREST-300 in thermal hydraulics, fuel rod geometry, and temperatures, except for the cross-sections of fuel assemblies, which may be as large as 20 × 20 cm or even 25 × 25 cm (N.B., the large neutron flux gradient in zone 3 should be taken into consideration).

The fuel-to-cladding clearance in the main part of the core should be made larger than in BREST-300, to achieve higher burnup, and as large as practicable in the 'mirrors'.

Such a reactor will make full use of the surfaces and space around it and is easily configured with the turbine hall and a long turbo generator (or two, on either side), as well as with two CFC facilities at its ends. Given standard fuel, physics, thermal hydraulics and equipment (SG, pumps, etc.), it may have any power, dictated by the grid conditions, starting with 300 MW. Its problems however, include:

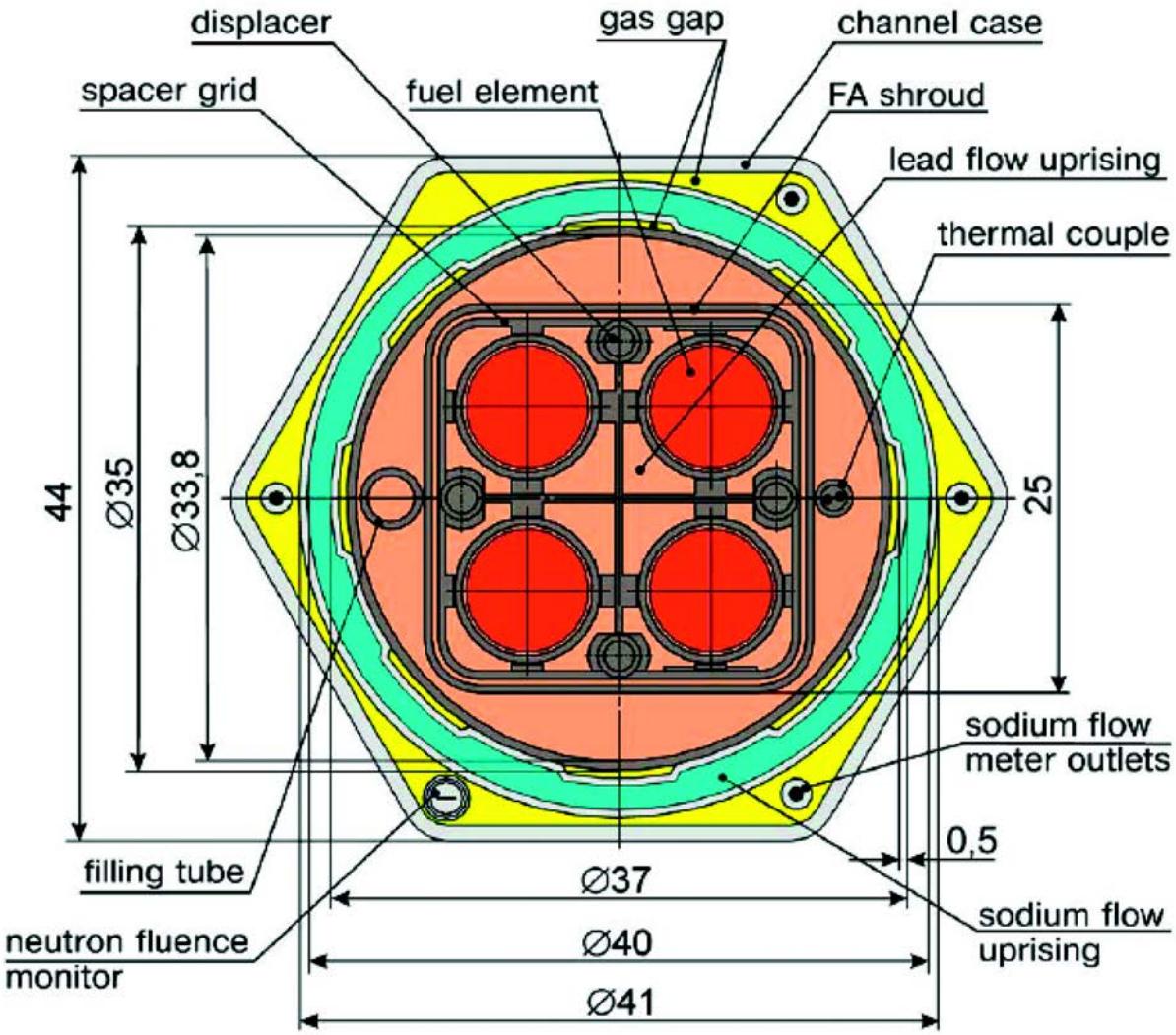


FIG. 22. Cross-section of SCLC in the heat exchange zone.

— Sensitivity of the power distribution to reactivity perturbations. These effects can be easily managed with the help of regulating devices distributed along the reactor axis and operating in response to local neutron or temperature sensors: an ideal option would be to have 'passive' regulators responding directly to T_{Pb}^{out};
— Resistance of flat vault walls to internal pressure from 1 atm (at the top) to ~10 atm (at the bottom), and of the separating steel shell to pressure differences (up to 10 atm) between the downward and upward Pb flows;
— Absence of rotating plugs, causing the difficult engineering problem of providing an in-pile reloading machine, which would have to travel in the gas cavity in longitudinal and transverse directions and would be exposed to radiation, temperature of 200–400°C and Ar atmosphere with Pb vapors.

A longer term option under investigation is the possibility of switching from a steam turbine to a gas turbine cycle with the primary objective of ruling out severe consequences of the hypothetical accident with multiple SG tube ruptures.

A self-contained lead cooled channel (SCLC) (see Fig. 22) with a built-in pump, 4 BREST-type UN-PuN fuel rods, an oxygen monitoring and maintenance system, and with heat removed by Na, had its first tests in the BOR-60 reactor (2 campaigns in 2003, 5 months in all).

The inlet and outlet Pb temperatures were 540 and 615°C, respectively, the fuel cladding temperature was found (calculated) to be as high as 660°C, and the burnup reached 0.44%.

No corrosion damage to the pump, fuel assemblies and fuel rods have been discovered, the cladding condition on the inside is being studied. The SCLC design was modified to allow for the test findings.

The pilot BREST-300 reactor is being developed to current regulations and rules with a view to bring its design as close as possible to a reactor of natural safety; with regard to a commercial BREST plant, its basic design features may be identified by extrapolating the operational data of the prototype, and then verified under working conditions of the reactor. The most important objective faced by BREST is to ensure an equilibrium mode of fuel burning and a total reactivity margin of $\Delta k \sim \beta_{eff}$, see Table 18.

For the most part (see Table 19), the power effect is compensated by passive features; i.e. coolant flow rate feedbacks to reactivity (power), designed as lead columns compressed by gas pressure from above and moved by Pb flow at the core inlet. The central zone of the core accommodates a passive shutdown system in the form of tubes with heavy absorber rods made of WB_2, which enter the core driven by the lead flow and sink in it when the flow stops, or when the outlet Pb temperature rises to a certain level T_m.

Errors in predicted k_{eff}, BR, reactivity effects, worth of reactivity controls, as well as the tolerances in dimensions, densities, fuel isotopes, etc., should be minimized, including inaccuracies in neutronic calculations.

To this end, some benchmark U-Pu-Pb compositions were studied at critical test facilities, BFS (FEI) and ROMB (VNIITF), with their subsequent computational analyses at FEI, Kurchatov Institute and NIKIET and updating of the neutronic characteristics in regard to Pb and MA in the BREST energy range [169, 180].

Main results are summarized in Tables 20–22.

TABLE 18. REACTIVITY EFFECTS

Effect	Value, % Δk/k
Isothermal (heating from 380 to 420°C), $\Delta\rho_{is}$	–0.08
Power effect with change in N:	
from N=0 to N=0.3 N_{nom}	–0.14
from N=0.3 N_{nom} to N=N_{nom}, $\Delta\rho_{power}$	–0.10
Reactivity increase over fuel life (micro), $\Delta\rho_{burnup}$	0.08
Fuel swelling over fuel life time (micro), $\Delta\rho_{swell}$	–0.13
Decay of ^{239}Np, $\Delta\rho_{Np}$	–0.08
Operating reactivity margin, $\Delta\rho_{op}$	0.04
Effective fraction of delayed neutrons, β_{eff}	0.364

TABLE 19. REACTIVITY WORTH OF CONTROL AND PASSIVE SYSTEMS

Control and passive systems	Worth Δρ, %
2 automatic control (AC) rods	0.08
18 reactivity compensation (RC) rods	0.72
8 emergency protection (EP), scram rods	0.82
1 emergency protection (EP), scram rod	0.11
Peripheral shutdown system 2AC + 18RC + 8EP rods	1.6
Central shutdown system — 33+12 absorber rods	3.6
All CPS rods	5.0
Flow rate – to – reactivity feedback, 12 passive devices	0.15

TABLE 20. CRITICALITY CALCULATIONS ON BFS ASSEMBLIES

BFS assembly	MCNP-4B	MMKKENO	
	ENDF/B-VI	ABBN-93	ABBN-93 + JENDL-3.2 for Pb
61-0	—	0.9995(5)	0.9959(5)
61-1	—	0.9971(5)	0.9935(5)
61-2	—	0.9977(5)	0.9926(5)
77-1	1.00025(13)	1.0016(6)	0.9987(5)
77-1a	1.00074(13)	—	—
87-1	—	0.9979	—
87-2	—	0.9976	—

Note: the statistical calculation error is given in brackets.

TABLE 21. VARIATION BETWEEN CALCULATED VALUES AND EXPERIMENTAL DATA FOR SPECTRAL INDICES

Index	BFS-61	BFS-77	BFS-77
	Diffusion calculation TRIGEX	Diffusion calculation TRIGEX	Calculation by MCNP-4B
C238/F235	1.001 ± 0.024	1.056 ± 0.050	1.013 ± 0.050
F238/F235	0.968 ± 0.030	1.011 ± 0.030	0.985 ± 0.030
F239/F235	1.002 ± 0.015	0.996 ± 0.014	0.995 ± 0.014
F240/F239	1.050 ± 0.020	1.043 ± 0.033	1.103 ± 0.033
FNp^{237}/F239	—	1.079 ± 0.042	1.039 ± 0.042
Fpu^{238}/F239	—	1.056 ± 0.022	1.063 ± 0.022
FAM^{241}/F239	—	0.985 ± 0.039	0.907 ± 0.039
Fam^{243}/F239	—	1.15 ± 0.08	1.016 ± 0.080

TABLE 22. ROD WORTH IN BFS-77, (β_{eff})

Rod type	Experiment	Diffusion calculation TRIGEX	Calculation by MCNP
Test core of BREST-300	0.11	0.082	0.091 ± 0.05
Test core of BREST-300	0.10	0.080	0.13 ± 0.05
Driver	0.31	0.307	0.36 ± 0.05
Driver	0.30	0.276	0.32 ± 0.05
Absorber B_4C	1.16	1.16	1.23 ± 0.05
3 rods with B_4C	3.22	3.18	3.26 ± 0.05
4 rods with B_4C	4.01	3.97	4.15 ± 0.05
Reflector	0.07	0.042	—
Reflector	0.08	0.038	—

The results were also used for updating the nuclear data library ABBN-93 involved in BREST calculations.

The core composition and neutron spectra of the pilot and commercial reactors being fairly similar, in addition to direct calculations, the empirical neutronic characteristics of the former can be 'borrowed' with sufficient accuracy using methods of the perturbation theory methods.

It is not difficult to build 'defense in depth' against accidents and to estimate their frequency at a level upward of 10^{-4} per reactor-year, if the properties and behavior patterns inherent to the reactor and its components rule out sever accidents with fuel failure and catastrophic radioactive releases. The common reactivity-suppressing feedbacks, such as Doppler and thermal expansion effects (plus the BREST-specific threshold effect for T_{Pb}^{in}), the high density, heat conducting, high temperature nitride fuel of equilibrium composition, the moderate power density (up to 200 kW/L as comparative up to 800 kW/L in Na cooled reactor) together with the manageable chemical activity and low activation of the heavy, high boiling Pb coolant, add up to make BREST resistant to the most severe accidents:

— Runaway in case of errors or failures in the control system (held back by small reactivity margins) or in the event of coolant boiling, since the void reactivity effect hides in at high temperatures (e.g. 1700^0C and 2000^0C at 1 atm and 10 atm, respectively);
— Loss of residual heat removal (natural circulation of lead and contiguous air circulation, low unit power, widely spaced fuel rod lattice allowing easy lead passage, no boiling);
— Loss of coolant (low pressure, no boiling, burning or escape through cracks in the vault);
— Interrupted coolant flow, with the initiating events including destruction of pump drives and flywheels by external impacts (a delay of 20 s in Pb flow drop, flow rate reactivity feedbacks);
— Seismic events (reduced eigen-frequencies of fuel rod vibrations, the vault anchored to ground, as for large buildings), however, still in need of further substantiation;
— Releases due to failure of the top plate and containment – low in comparison with Na (low chemical activity, radioactivity and Pb vapor pressure);
— Lead was showed in experiments not to burn when exposed to air (experiments were performed at 1200°C Pb temperature).

A severe accident may be initiated when certain critical temperatures are approached: $T_{Pb}^{melt} = 327$°C, $T_{clad}^{melt} = 1500$°C, $T_{fuel} > 1600$°C (nitride decomposition), $T_{Pb} \sim 700$°C (at SG inlet due to multiple tube ruptures in case of failure of emergency pressure relief).

Figures 23–25 show the results of some transient analyses.

In the case of insertion of the maximum reactivity reserve at zero power, it takes ~35 s for the max cladding and leads temperatures to reach the maximum.

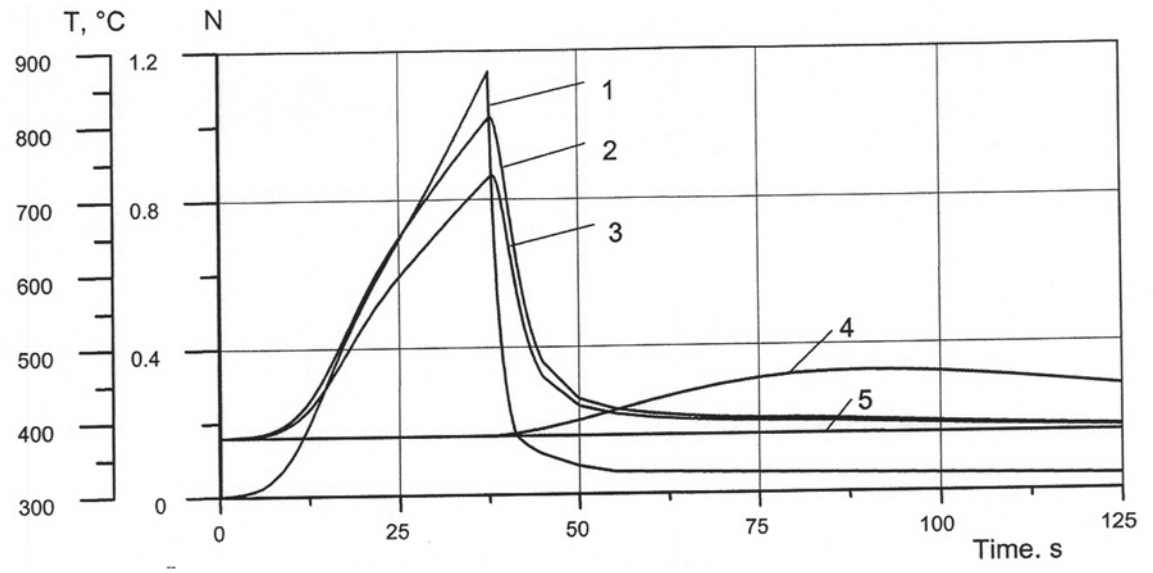
Some expert's opinion is acknowledged that the benefits of high-building pb temperatures apply to some slowly developing accidents.

In the case of main coolant pump trip at nominal power $\rho N \sim 2\% < \beta_{eff}$ leads to slow power decline (delayed neutrons) as compared to the flow rate decrease (in this case the heat-conducting fuel is a disadvantage). However, a more realistic 3-D calculation of 'hot' Pb mixing with the surrounding 'cold' lead shows T_{Pb} decrease at the SG inlet to ~650°C, or even below 600°C due to operation of only ¼ of the central shutdown rods, with T_{clad} reaching ~800°C. In case of prolonged reactor outage (without fission product accumulation) lead will begin to 'freeze' at the reactor periphery in a month's time. The large BREST concept is expected to have permanent heaters for utilization of the accumulated Sr and Cs.

In the event of steam header rupture accompanied by failures of emergency protection and of the MCPs, and with the water flow blocked, it will take Pb ~ 2 minutes to freeze in the SG tubes. Nevertheless, if Pb circulation in SGs is interrupted, it will still continue to circulate in the lines bypassing the SGs and pumps, with heat removed by natural air circulation without reaching temperature limits.

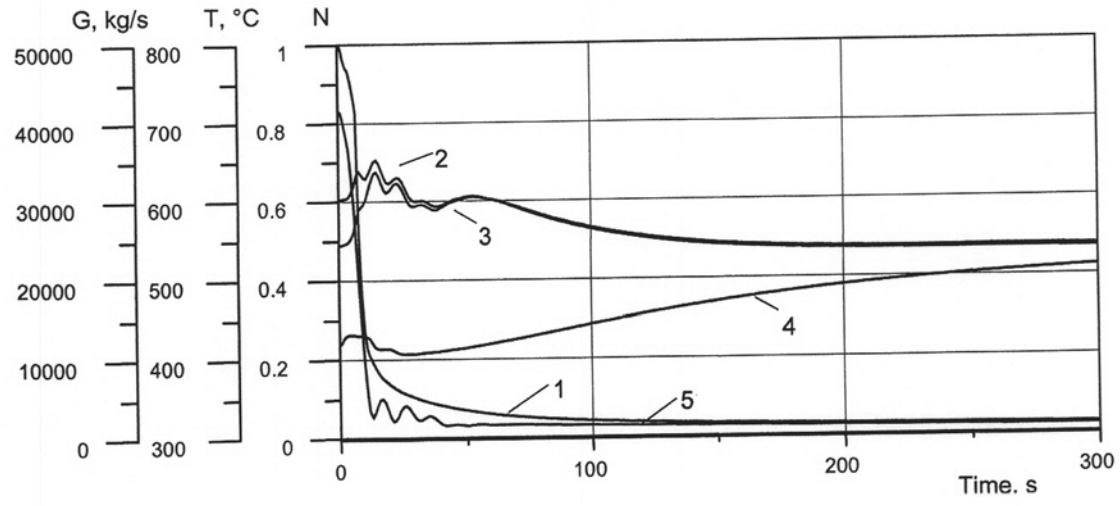
In this case, 3-D calculations also point to Pb temperature decrease at the SG inlet to ~600°C. Destruction of the top plate and containment, anticipated with T_{Pb} increase to 700°C without fuel failure, leads to the following releases within 4 days:

— Lead with concentration of ~0.02 mg/m^3 at the ground surface 10 km away from the NPP (twice the MPC in terms of Pb toxicity);



1-Reactor power; 2-Maximum fuel cladding temperature; 3-Maximum lead temperature at core outlet; 4-Lead temperature at steam generator inlet; 5-Lead temperature at core inlet

FIG. 23. Input of maximum reactivity margin at N=0.

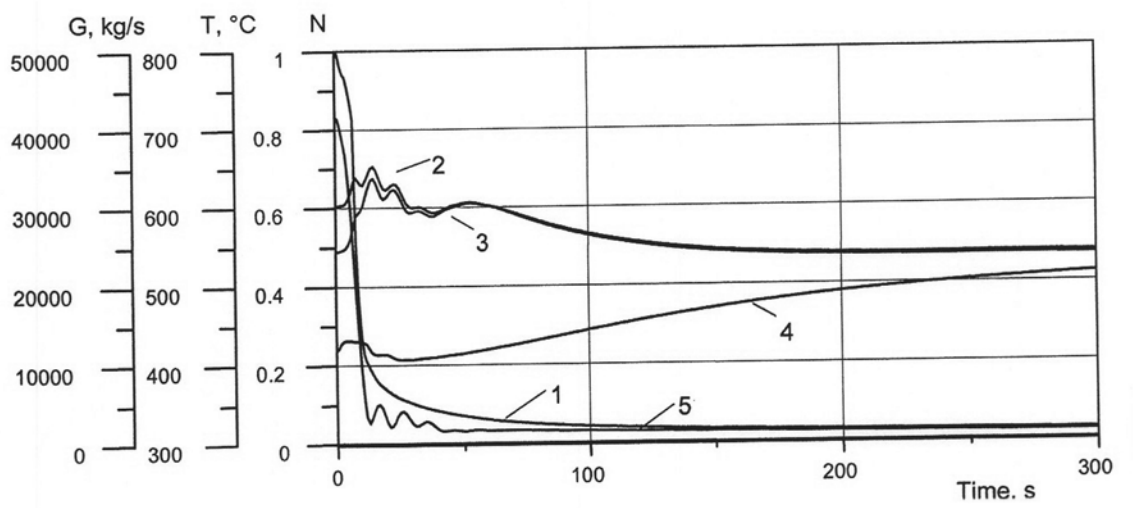


1-Reactor power; 2-Maximum fuel cladding temperature; 3-Maximum lead temperature at core outlet; 4-Lead temperature at steam generator inlet; 5-Coolant flow in the core

FIG. 24. Trip of all MCPs at nominal power.

— ~2 300 Ci (~3 Ci/day for ^{210}Po, which in terms of radiotoxicity is close to the total activity of all the other nuclides). This accident with a risk to environment is classified at Level 5 of INES, but would be downgraded to Level 4 of 3 if Bi, Po and other contaminants are removed, as provided for in the design.

Consideration has also been given to some hypothetical accidents, which appear to have no credible reasons for occurrence in BREST. A prompt neutron excursion with reactivity rise in excess of β_{eff} at a rate up to 50 β/s causes a surge of fuel temperature as large as hundreds of °C, but leaves the fuel intact. If the source of additional reactivity is not removed by the excursion, it will be followed by a series of excursions with the temperature growing as a function of the added reactivity ρ and power effect dρ/dT (but not of β_{eff}, τ or dρ/dt). In the event of



1-Reactor power; 2-Maximum fuel cladding temperature; 3-Maximum lead temperature at core outlet; 4-Lead temperature at steam generator inlet; 5-Coolant flow in the core

FIG. 25. Withdrawal of 2 AC rods + trip of all MCPs and feedwater pumps at N = N_{nom}.

cladding failure (with T_{Pb} ~ 1500°C) fuel whose density is close to that of Pb would be scattered by Pb flows all over the reactor. The possibility of its piling up on the reactor bottom and approaching criticality is prevented by Pb convection, which develops as the fuel heats up. The occurrence of $K_{eff} > 1$ and the excursion pitch depend on the fuel buildup rate, on the ratio of its density to that of Pb, as well as on its content of fission products and decay heat.

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ABBREVIATIONS

ACS	autonomous cooling system
ADS	accelerator driven system
AE	absorbing element
AEP	additional emergency protection
ALMR	advanced liquid metal reactor
ANPF	aircraft nuclear power facilities
ASTRID	advanced sodium technological reactor industrial demonstration
BN	bystrie neytrony (fast neutrons)
BFS	fast critical assembly (in Russian)
BOR	bystrij opytnyj reactor (fast experimental reactor)
BR (Reactor)	bystrij reactor (fast reactor)
BR	breeding ratio
BREST	bystriy reactor estestvennoy bezopasnosti (fast reactor natural safety)
CBR	core breeding ratio
CEFR	China experimental fast reactor
CFC	closed fuel cycle
CPP	captive power plant
CPS	control and protection system
CR	compensating rods
CRDM	control rod drive mechanism
DFR	Dounreay fast reactor
EBR	experimental breeder reactor
EFIT	European facility for industrial transmutation
EFR	European fast reactor
ELSY	European lead cooled system
ENHS	encapsulated nuclear heat source
EP	emergency protection
EUROTRANS	European research programme for the transmutation of high level nuclear waste in an accelerator driven system
FaCT	fast reactor cycle technology development
FBTR	fast breeder test reactor
FFTF	fast flux test facility
FR	fast reactor
FSA	fuel subassembly
GFR	gas cooled fast reactor
GIF	Generation IV International Forum
HEPP	heat electric power plant
HLMC	heavy liquid metal coolant
IBR	investment breeding ratio
IFR	integral fast reactor
IHX	intermediate heat exchanger
JSFR	JAEA sodium fast reactor
KNK	kompakte natrium gekühlte kernreaktoranlage
LBE	lead-bismuth eutectic
LF	loading factor
LFR	lead cooled fast reactor
LMC	liquid metal coolant
LMFR	liquid metal cooled fast reactor
LOCA	loss of coolant accident
LPFT	low-pressure feedwater tank

LWR	light water reactor
MA	minor actinides
MCC	main circulation circuit
MCP	main circulation pump
MEGAPIE	MEGAwatt pilot experiment
MOX	mixed oxide (mixed PuO_2+UO_2)
MSR	molten salt reactor
MUSE	multiplication of an external source experiments
NE	neutron excess
NFC	nuclear fuel cycle
NHPP	nuclear heat power plant
NP	nuclear power
NPP	nuclear power plant
NPT	nuclear power technology
NS	nuclear submarine
NSSS	nuclear steam supplying system
NVNPP	Novovoronezh nuclear power plant
OCR	operating compensating rods
P&T	partitioning and transmutation
PBWFR	Pb-Bi cooled direct boiling water fast reactor
PFBR	Prototype Fast Breeder Reactor
PFR	Prototype Fast Reactor
PGU	steam-gas installation
PHRS	passive heat removal system
PHWR	pressurized heavy water reactor
PWR	pressurized water reactor
R&D	research and development works
RAW	radioactive waste
RBMK	high-power channel-type reactor
RI	reactor installation
RMB	reactor monobloc
ROMB	rhombus
RR	regulating rods
RVC	reticulated vitreous carbon
SA	subassembly
SCLC	self-contained lead cooled channel
SCWR	super critical water reactor
SEFOR	South-West Experimental Fast Oxide Reactor
SFR	sodium cooled fast reactor
SG	steam generator
SNETP	Sustainable Nuclear Energy technology Platform
SNF	spent nuclear fuel
SNR	Schneller Natriumgekühlte Reaktor
SSTAR	small, sealed, transportable, autonomous reactor
STAR-LM	secure, transportable, autonomous reactor–liquid metal variant
SVBR	lead-bismuth cooled fast reactor
TMI	Three Mile Island
TPP	thermal power plant
TR	thermal reactor
TRU	transuranium
TWG-FR	technical working group on fast reactors
VHTR	very high temperature reactor
WWER	water cooled water moderated power reactor

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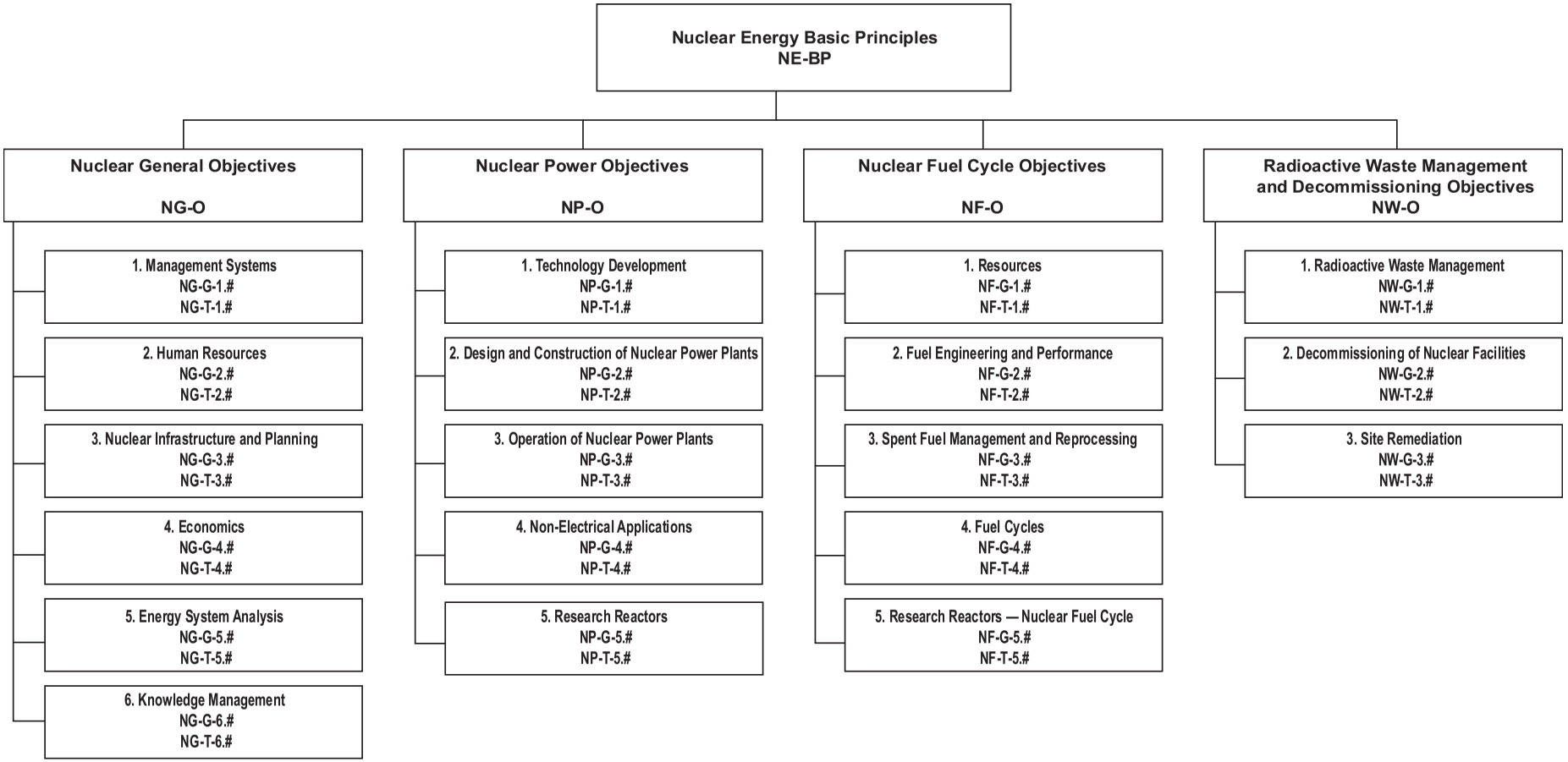

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