LIQUID METAL COOLED REACTORS

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THE ENCAPSULATED NUCLEAR HEAT SOURCE FOR PROLIFERATION-RESISTANT LOW-WASTE NUCLEAR ENERGY

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Abstract

Encapsulated Nuclear Heat Source (ENHS) is a small innovative reactor suitable for use in developing countries. The reference design is a 50MWe lead-bismuth eutectic (Pb-Bi) cooled fast reactor. It is designed so that the fuel is installed and sealed into the reactor module at the factory. The nuclear controls, a major portion of the instrumentation and the Pb-Bi covering the core are also installed at the factory. At the site of operations the reactor module is inserted into a pool of Pb-Bi that contains the steam generators. Major components, such as the pool vessel and steam generators, are permanent and remain in place while the reactor module is replaced every 15 years. At the end of life the sealed reactor module is removed and returned to an internationally controlled recycling center. Thus, the ENHS provides a unique capability for ensuring the security of the nuclear fuel throughout its life. The design also can minimize the user country investment in nuclear technology and staff. Following operation and return of the module to the recycling facility, the useable components, including the fuel, are refurbished and available for reuse. A fuel cycle compatible with this approach has been identified that reduces the amount of nuclear waste.

1. INTRODUCTION

The encapsulated nuclear heat source (ENHS) is a small innovative reactor suitable for use in developing countries. The research on this concept is being conducted under the US Department of Energy (DOE) Nuclear Energy Research Initiative (NERI). The research was initiated in the summer of 1999[1]. The most innovative feature in the ENHS is the fact that the nuclear fuel is encapsulated in the primary coolant vessel throughout its life. The heat generated in the reactor core is transferred through a uniquely designed section of the primary coolant vessel wall to a pool of secondary coolant. The primary and secondary coolants are a lead bismuth eutectic. The secondary pool also contains the steam generators.

2. PERFORMANCE OBJECTIVES

The ENHS performance objectives are those established by DOE for the Generation IV(GEN IV) reactors [2]. These are summarized in the box below. The objectives address the entire fuel cycle as does the ENHS concept. The key features in the ENHS concept that support achievement of these objectives are the approach to the reactor module design and the use of fuel cycles that are compatible with its fast neutron spectrum.

Generation IV
Performance Objectives

- Competitive Busbar Cost of Electricity
- Acceptable Risk to Capital
- Limited Project and Construction Lead Times
- Low Liklihood of Core Damage
- Demonstration of No Severe Core Damage
- No Need for Offsite Response
- As Low as Resonable Achievable Radiation Exposure
- Tolerant to Human Error
- Solutions for all Waste Streams
- Public Acceptance of Waste Streams
- Minimal Waste
- Minimal Attractiveness to Potential Weapons Proliferation
- Intrinsic and Extrinsic Proliferation Resistance

The latter characteristic and recycling of the fuel are fundamental to the minimization of high level waste. The ability of ENHS to achieve the first two economic objectives has not yet been determined, but the approach is expected to result in the absolute minimum construction and lead times.

The safety objectives will be met with significant margins. It should be possible to demonstrate through full scale testing that there will be no core damage under severe accident conditions, and no need for offsite response. Since the nuclear operations onsite are minimum and the response to any balance of plant events only impacts the availability, the plant should be very tolerant to human error. Po210 is generated in the coolants and needs to be given greater attention than it has been . However, there are data that indicate that its presence should not be a major problem[3].

A key driver in the design was the desire to achieve an improvement in the proliferation resistance. We believe this has been done, and with the realization of international recycling facilities the approach will permit the minimization of radioactive waste while retaining a high level of proliferation resistance.

3. SYSTEM DESCRIPTION

3.1. Nuclear Island

The ENHS reference design produces 50MWe from 125MWt using natural circulation in both the primary and secondary Pb-Bi coolants. The reference coolant was selected as a lead bismuth eutectic (Pb-Bi) of 45% lead and 55% bismuth. In principle, the concept would also work with sodium as the primary coolant. The installed nuclear island layout is illustrated schematically in Figure 1.

The heated primary coolant flows up from the core into downcomer passages that form the primary vessel wall. The secondary coolant flows up the primary vessel wall cooling the primary coolant as it flows down to the core inlet. The heated secondary coolant is then cooled by the steam generators in the secondary pool and returns to the bottom of the pool vessel. These flows are achieved by natural circulation. The natural circulation achieves design simplicity and avoids the need for active components, but it requires a rather tall 19m primary vessel. An alternative design that uses a so called “lift pump” to introduce voids above the core outlet plenum by pumping cover gas into a sparger has been evaluated. This design is only 10m high and reduces the coolant mass significantly. This improvement is achieved at the expensive of adding an active component to pump the cover gas.

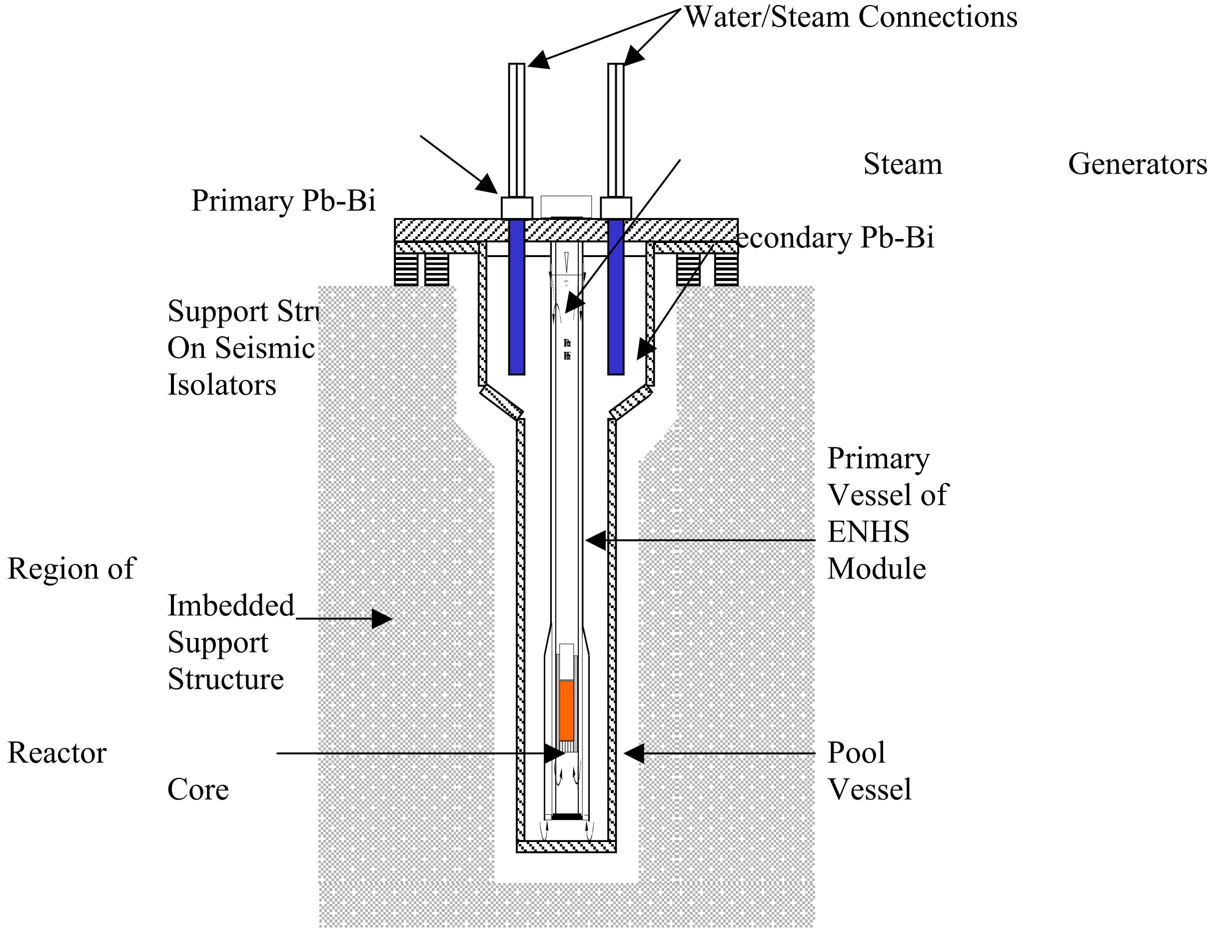

FIG. 1. A Schematic Vertical View of a Single ENHS (Not to scale.)

An innovative tube-in-tube steam generator module has been designed. Each module produces 15.625 MWt of steam at 135 $bars_g$ (2000psig) and 482C (900F). In the reference single module design, eight of these fit around the reactor module in a pool vessel that has an upper diameter of 4.6 meters.

3.2. Manufacturing and Shipping

The ENHS module is designed for shop assembly to the maximum extent possible. This means all structure and equipment internal to the primary coolant vessel, including the fuel is installed at the factory. Parts internal to the primary coolant vessel will be designed for the life of the module. This includes reflectors and the control drive mechanisms that are internal. Cabling for instrumentation and preparation of the structures that mate with the pool vessel are all completed in the shop to support rapid installation at the site. In addition, Pb-Bi coolant, sufficient to cover the core, will be poured into the vessel and frozen. This approach is intended to both physically protect the fuel against shipping damage and provide additional discouragement to theft.

It is anticipated that special equipment for shipping and installation would be fabricated and assembled along with the reactor module. This equipment would be used in the shop to erect the module to the vertical position so that the Pb-Bi can be poured into the core. It could later be used at the site to move the module from the horizontal shipping position into the vertical position for installation. This reusable equipment would be provided by the nuclear system supplier.

3.3. Installation at the Site

Upon arrival at the port of delivery the reactor module will be unloaded onto either rail transport or a special vehicle for road transport to the site. The transport fixtures and any special vehicles would be provided by the supplier and used repeatedly for delivery of multiple modules to one site, or for deliveries to various sites. The delivery would be made on a schedule consistent with completion of the secondary pool and installation of the steam generators. The module would be hoisted into the vertical position and lowered into the secondary pool vessel. When replacing a module however it would be desirable to make the exchange with the secondary Pb-Bi in place. It is anticipated this can be done while filling of the reactor module with primary Pb-Bi. The empty weight of the module will not be sufficient to cause it to sink into the pool to its operating depth. Therefore, as the reactor vessel is filled it will sink until fully immersed and mated with the secondary pool closure surface. The process of filling the module with hot Pb-Bi and sinking it into a pool of hot Pb-Bi will melt the Pb-Bi in the core. At this point the module will be structurally attached and seal welded to the pool vessel. Instrumentation and control cables will be connected to complete the installation.

The installation will be done with the Pb-Bi at an acceptably low temperature and using fixtures that maintain the inert cover gases over the Pb-Bi surfaces. These installation fixtures would be used repeatedly by the module supplier. In addition, special systems will be provided for Pb-Bi heating, circulating and purification. There is an open question as to whether or not it is possible to passively control the coolant chemistry quality throughout the operating life. This is a very desirable characteristic that might be possible with a sealed coolant systems. If this is not possible, it will be necessary to provide a permanent active system to maintain the coolant chemistry. Installation of the reactor module and purifying the coolants is the last major task. Following checkout of the nuclear instrumentation and controls and their integration with the plant supervisory system the plant can start acceptance testing.

3.4. Operation

The plant may have multiple modules, either in individual pools or in a single larger pool. The plant, whether single or multiple modules, would be operated from a single control building. The operations would be limited to monitoring and maintenance. Planned maintenance would be limited to the balance of plant and steam generators. The lack of pumps and valves in the nuclear system and the expectation of maintaining a sealed corrosively benign coolant environment support meeting this objective.

Plant startup will be initiated with a single switch and will proceed under the control of redundant supervisory computers. The system will be heated to 350C and the reflectors will be raised to bring the reactor critical. A preferred scheme for raising the temperature and initiating the natural circulation has not been selected. It may be feasible to do this using low reactor power. However, there may be other reasons for providing trace heating on the module and this equipment may play a role in the startup process, at least until sufficient decay heat is available to sustain the natural circulation.

Full power operation on natural circulation in both the primary and secondary coolant has been demonstrated analytically [4]. This feature is compatible with passive load following and autonomous control. The operational simplicity and improved safety margins should support a significant reduction in the size of the operating staff and cost of operations. The number the staff will be dominated by those needed to maintain the balance of plant. The number of staff trained in nuclear technology will be minimal. The fact that there is no servicing of the module, including no refueling, is further reason for a minimal size nuclear staff.

3.5. Recycling and Waste Disposal

At the end of 15 years of operation and a short cooling period, the module is removed from the pool to a storage location on site. This will be done by reversing the steps used for installation. The module will remain there, continuing to cool. Following solidification of the core in Pb-Bi the module is expected to be suitable once again for shipping to a recycle facility. Thus, the fuel is locked inside the ENHS from "cradle to grave". This feature restricts access to the fuel and simplifies safeguarding throughout the life of the module. The ENHS module also provides substantial barriers to accessing neutrons for use in production of fissionable material.

The module recycling would be completed in a few international monitored and controlled facilities located strategically through out the world. The total Pu and minor actinide inventory in the proposed ENHS-based energy system of a given capacity is fixed. Most of this transuranics (TRU) inventory is well secured inside ENHS modules. The only high level radioactive waste anticipated from this energy system consists of the following: (1) The fission products (FP) extracted in the dry fuel recycling process. (2) Trace losses of TRU and FP that can not be recovered from the spent fuel clad material during recycle. (3) Trace losses of TRU and FP waste from the fuel fabrication process. (4) Structural material activation products. It may be possible to reuse the vessel following annealing. The waste has not yet been characterized and quantified, but it is anticipated that it will be small and not a proliferation concern.

Recently Greenspan et al. [5] proposed a fuel cycle scheme that would further reduce the amount of high level radioactive waste. In this reference it is noted that the core life limit of ~105 GWd/tHM maximum burnup is due to the radiation damage to the cladding and fuel support structures. By partially reprocessing this spent fuel just to extract the volatile and some of the semi-volatile FP and mixing the product with makeup fuel the fuel can be recycled. The make-up fuel, approximately 7% of the fuel loading, can be spent fuel from light water reactors (LWRs) from which the volatile and semi-volatile FP have been removed. It may be possible to repeat this process many times. So far we have not carried out the burnup analysis beyond an average discharge burnup of 20%.

In addition to the reductions in high level waste, ENHS will create a minimum of low level radioactive waste because there is no need to provide normal servicing of the module. Some low level waste will result from the removal and replacement of the module every 15 years.

4. REACTOR MODULE DESCRIPTION

4.1. Design Parameters

As mentioned, two configurations for the ENHS reactor module are being evaluated. Table 1 gives some of the design parameters of both the natural circulation version of the module, ENHS1 and the lift-pump version, ENHS2.

TABLE I. SELECTED DESIGN PARAMETERS OF REPRESENTATIVE ENHS MODULES FOR $125MW_{TH}$

Design parameter	ENHS1	ENHS2
Primary Pb-Bi coolant circulation	100% natural	With lift-pump
Average linear heat-rate (W/cm)	60	60
Average discharge BU* (MWd/tHM)	52,000	52,000
Core life* (effective full power years)	20	20
BU reactivity swing	<1$	<1$
Maximum excess reactivity	<1$	<1$
Core height (m)	1.25	1.50
Core diameter (m)	1.98	1.87
Fuel rod diameter (cm)	1.0	1.0
Clad thickness (cm)	0.1	0.1
Lattice (hexagonal) pitch (cm)	1.45	1.50
Overall module height (m)	19.6	10.1
Outer module diameter (m)	3.24	3.35
Number of rectangular channels in IHX	135	245
Inner dimensions of channel (cm x cm)	40 x 2.5	50 x 1.0
IHX channel length (m)	13	6
Weight of fueled module for shipment (t)	360	300
Coolant core inlet/outlet temperature (^{o}C)	400/564	400/543
Primary-to-secondary mean ΔT (^{o}C)	49.1	47.3
Number of steam generators per ENHS	8	8
Steam generator module diameter (m)	1.0	1.0
Active length of SG tubes (m)	7.5	7.5

* Limited by radiation damage to clad @ $4x10^{23}$ n/cm^2 >0.1 MeV.

It can been seen that a major reduction in height is realized with the lift-pump design along with some reduction in shipping weight. In order to obtain sufficient heat transfer the primary coolant flow channels in the lift-pump design have been reduce to 1 cm and over 100 tubes are added. Appendix A provides a schematic of the ENHS1 module.

4.2. Intermediate Heat Exchanger

As mentioned, one of the key issue of feasibility was a concern about the unique characteristic of transferring heat through the primary vessel wall. This wall was conceived initially as a corrugated thin walled structure. The corrugations provided the necessary increase in heat transfer area between the primary coolant flowing downward along the inside of the wall and the secondary coolant flow upward on the outside of the wall. Both the fabrication and structural support of this configuration presented problems. Several more practical alternatives have been identified. The reference design uses rectangular tubes that are 2cm by 40cm with 2mm wall thickness. Section view A-A in Appendix A depicts the top view of rectangular coolant flow tubes that the primary coolant flows downward through. The secondary coolant flows upward through similar alternate channels formed by the tubes. Both the structural and thermal performance feasibility of such a design have been confirmed[6], but fabricability issues remain and other more innovative as well as conventional tubular designs are being considered.

4.3. Core Design

A key requirement of the ENHS is to have a long life core without access to the fuel. To accomplish this it means that the initially loaded core must have a very long life. From a safety standpoint it is desirable to do this without a large excess of reactivity throughout the life. The design domain for cores that meet these design objectives for at least 15 years of full power (EFPY) is defined in Greenspan et al.[1]. What limits the core life is the radiation damage to fuel structural materials, assumed to be $4x10^{+23}$ n/cm^2. The corresponding peak fuel burnup is approximately 105,000 MWD/tHM. The fuel considered for these cores is metallic Pu-UZr fuel with $10^{w}/_{o}$ Zr. Typical Pu concentration is 11-12 $^{w}/_{o}$ of HM. By adjusting the lattice p/d ratio and the Pu $^{w}/_{o}$, it is possible to change the slope of k_{eff} vs. burnup.

4.4. Safety

A series of transient simulations for the ENHS were performed using the DNSP computer code and have demonstrated large margins to damage of either the fuel or structures [7]. In addition various operational transients such as normal startup, the following postulated accidents were studied:

A postulated $1.5 reactivity insertion from:

- low power followed by a scram,
- Loss of heat sink without scram,
- Steam line break without scram.

Although more than one dollar of external reactivity was inserted in the postulated reactivity accident, the strong negative feedback mitigated the consequences. The core response is benign and slow. The fuel integrity is retained even in this event for which there is no credible mechanism.

The system heatup during the LOHS without scram is very slow and will reach a maximum in the range of 600-700^{o}C, in a few days, after which the temperature will decrease according to the decay heat curve, and finally the system is expected to freeze if not restarted. Under these conditions the small amount of remaining decay heat will be removed by conduction through the solid lead to the surface of the containment vessel.

The steam line break produced only a minor effect in the ENHS, and following the SG dry-out the behavior is similar to the LOHS accident.

The temperatures in the fuel and structures are sufficiently in low in these events that it is very likely that a series of severe accident tests can be implemented in a prototype to demonstrate the inherent safety of the ENHS.

5. STATUS OF RESEARCH

The ENHS project was proposed to the NERI as a three year effort and we are more than half way through the project. A number of the feasibility issues have been resolved but much design refinement and optimization remains to be completed. Progress to date has been encouraging and we hope to develop continued interest both domestic and international in continuing the research. Currently we have collaborators from Japan (Toshiba and CRIEPI) and from the Republic of Korea, and we would welcome the participation of other interested organizations. On completion of the currently planned three year project we would hope to identify resources that would permit further design optimization and some key feature testing.

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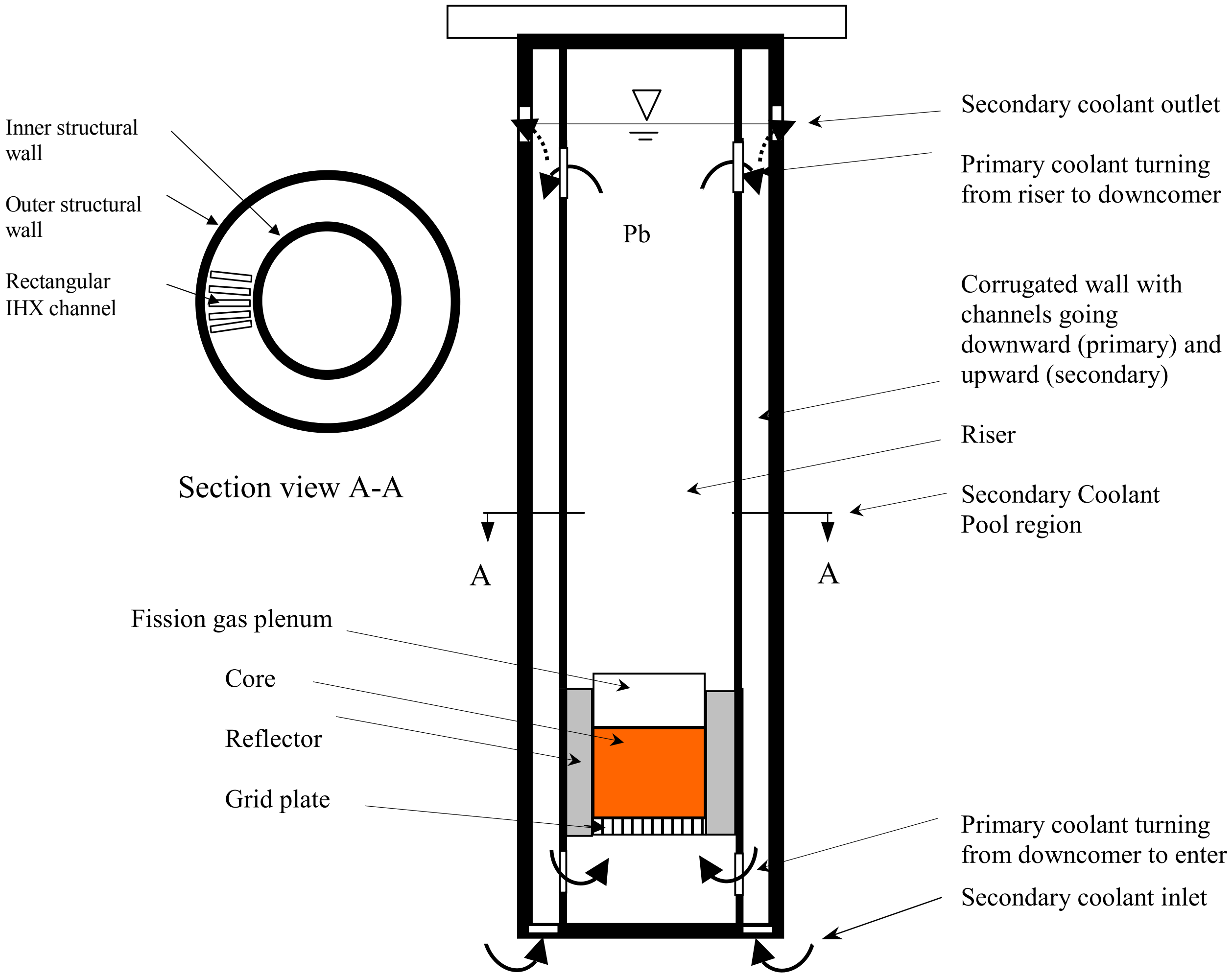
Appendix A
ENHS1 100% NATURAL CIRCULATION MODULE
(Schematic, not to scale)



LONG LIFE MULTIPURPOSE SMALL SIZE FAST REACTOR WITH LIQUID METALLIC-FUELED CORE

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Abstract

A concept of a long-life multipurpose nuclear reactor (MPFR) system, which can be installed closer to the beneficiary, or energy consumption area, is proposed to meet the requirements for energy production in the future. The use of liquid plutonium-uranium metallic alloys as a nuclear fuel potentially has the advantages such as excellent inner conversion rate, high burn-up values, and relatively small reactivity swings. The modular-type reactor of 300 MW thermal output does not need fuel reloading and decommissioning on the site and can be transported from the factory in a fabricated form. Considerations about the capability of in-vessel separation of the main fission products to prolong the core lifetime without fuel reloading or shuffling are made and discussed from the viewpoint of the influence on the core burn-up characteristics. The burn-up analysis shows the feasibility of the long life refueling-free MPFR core concept.

1. INTRODUCTION

There are many efforts to find clean alternative energy sources that permit independence from the limited fossil fuel reserves to realize a stable energy supply without polluting the environment by high toxic gas emissions. Recently, the usage of fuel cells is proposed by the main automobile makers and power companies, however problems associated with hydrogen production in quantities enough for the large-scale development of new transportation systems and electricity generators remain. An expansive utilization of nuclear energy for large-scale electricity production seems the most economical and realistic solution. The recent concepts are aimed at the construction of a nuclear reactor that can be operated remotely without access to the site by operators or any other personnel and reasserting these power sources as environmentally friendly, safe, and economical. Much more attention is also paid to the safety aspects of the reactor design and operation. Such ideas as to make the power reactor system small in size for factory fabrication and overland transportation with fuel irradiation time comparable to the reactor lifetime have been but forward. The maintenance-free modular-type reactor would also sufficiently increase the proliferation resistance of radioactive materials with reduction of spent fuel flows per unit of generated energy and consequently, the reduction of the radioactivity leaks into the atmosphere.

From the beginning of development of the Liquid Metal Fast Breeder Reactors (LMFBR), the minimum reactor doubling time has been considered as the main goal parameter for nuclear systems, and the sodium coolant had been selected to satisfy this purpose. LMFBRs have various weak points such as fuel swelling, fuel damage due to irradiation and outlet temperatures lower than 550 ^{o}C. A system with a higher operational temperature would allow us to extend the field where we can use the nuclear reactors for several purposes at the same time. It can be the energy source for alternative fuel production to generate hydrogen gas from water by thermo-chemical reaction and the energy source for electricity generation with high efficiency. To cover the region with core operational temperatures from 600 ^{o}C up to 1200 ^{o}C to satisfy the conditions for hydrogen production during long operational lifetime without refueling, the combination of liquid fuels and liquid metallic coolants seems to be the mutual solution. It was found that large a negative reactivity feedback based on large volumetric expansion of the fuels in the liquid-fueled systems would effectively improve the core characteristics. The Los Alamos molten plutonium reactor experiments (LAMPRE) showed the feasibility of the neutronic behavior and material compatibility of liquid metallic-fueled reactors with sodium coolant[1,2,3]. The safety analysis[4] described by H. Endo et al. showed high potential of liquid metallic-fueled fast reactors to meet the requirements for the future nuclear power plants and develop a compact system with improved safety features. To increase the operational temperatures, the usage of alternative coolants to chemically active sodium should be considered. The technologies developed in Russia showed that the eutectic lead-bismuth alloy has good neutronic properties and appears to be the most promising candidate to replace sodium[5].

To develop a multipurpose nuclear heat source module with a long-lived core, we propose here a new concept of the multipurpose fast nuclear reactor (MPFR) system with liquid metallic-fueled core. For the development of the MPFR system, we suggested two steps with two types of reactor cores. The system employs liquid metallic fuel and sodium coolant at the first step with the core outlet temperature of 650 ^{o}C. For the second step, the combination of liquid metallic fuel and lead coolant is considered with the core outlet temperature of 1150 ^{o}C. The paper describes the MPFR concept at the second step of the development. The neutronic characteristics of the proposed liquid metallic-fueled core are discussed from the viewpoint of elongation of core life by in-vessel separation of fission products (FPs).

2. MULTIPURPOSE FAST REACTOR CONCEPT

Noting that fossil fuel resources will be exhausted in several decades, we considered the concept of a power plant necessary to replace energy produced by fossil fuels utilization. The potential of a nuclear power plant for non-electric applications is quite high and it would be a candidate for the alternative energy source in the 21st century. It should cover the operational temperature region between 550 ^{o}C and 1200 ^{o}C, which had been a blank zone in the previous utilization of the nuclear reactors. To satisfy the conditions mentioned above, the MPFR system we propose is:

- Multipurpose. The power plant generates thermal energy, which can be utilized for the electricity generation, production of the alternative fuel such as hydrogen, and the nuclear fuel by breeding.
- Siting-independent, simple and economical. The MPFR is designed to be located close to the energy consumption area. It can be used for an underground urban siting or exploitation on small islands with relatively small grids. The reactor system without refueling has simple modular transportable structure. The maintenance is minimized, considering the core lifetime is over 15 years without the refueling or fuel shuffling at the reactor site. The core unit is made transportable and can be disassembled only at the factory using special equipment. The refueling system and many auxiliary systems are eliminated that results in reduction of capital costs and the area of the power plant site. The elongation of the core lifetime can be achieved due to the inherent separation of the FP materials from the fuel alloy and the elimination of such radiation damage of the fuel material as an expansion due to swelling.

- Safe. The balance between active, passive, and inherent safety features is the basis to obtain public acceptance for the future nuclear systems. The active safety feature: high reliability active control of the reactor system is introduced by employing the floating B_4C rod driven by coolant flow and external gas pressure. The passive safety feature: negative reactivity feedback mechanism, the self-controllability during anticipated transients without scram is initiated by employing the large thermal expansion capability of the liquid fuel. The inherent safety feature: reactivity insertion mechanism for the reactor termination, the ability to self-terminate a hypothetical core disruptive accident, is initiated by partial discharge of the liquid fuel to the out-of-core region.
- Environmentally friendly and has low waste emission per accumulated output. Long operational lifetime without refueling, which can be achieved by inherent FPs separation capability mainly due to the density difference between fuel and FPs, made it possible to decrease the spent fuel flows per unit of generated energy.

The MPFR systems are characterized as small fast reactors with the thermal power of 150 MW for the sodium-cooled system (1st design step) and 300 MW for the lead-cooled system (2nd design step). The configuration of the MPFR system is shown in Fig. 1. The MPFR consists of separated transportable modules, “Energy Unit” and “Core Unit”, connected with each other at the reactor site by short piping system equipped with isolation valves. Such a configuration extends the capability for reactor siting close to the energy consumption area and decreasing the total area of the reactor site.

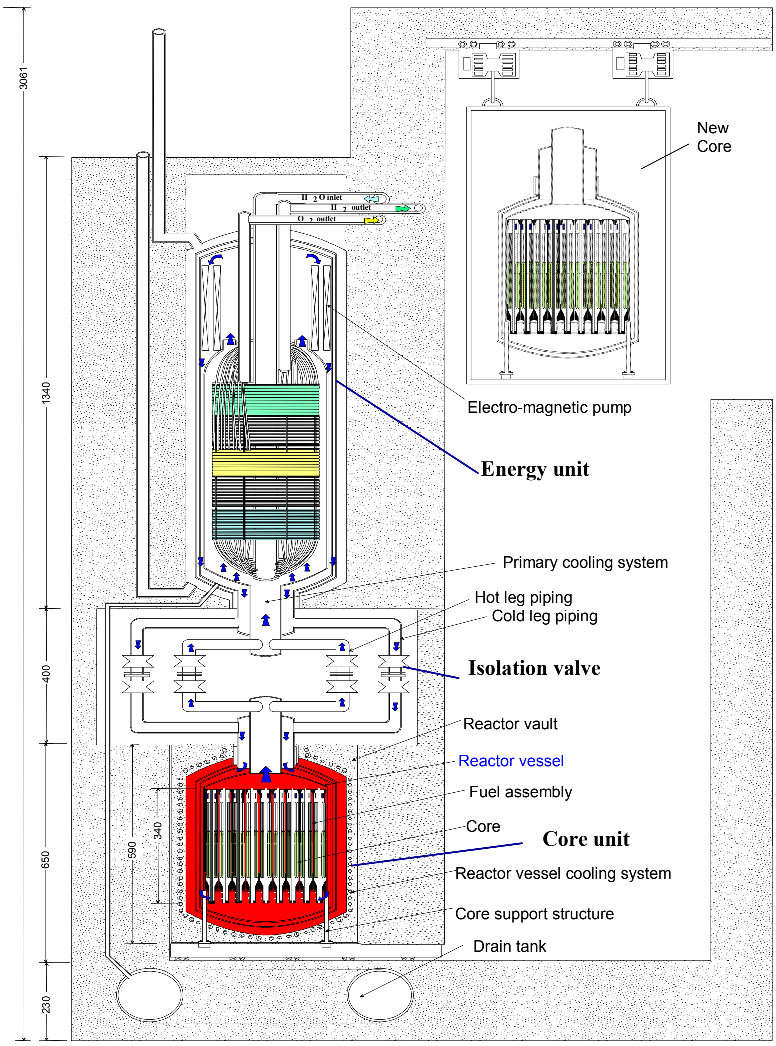

FIG. 1. Configuration of the MPFR System

Depending on the system implementation and energy conversion requirements at the siting point, potential users can select the supplied "Energy Unit". It can be the unit for electricity production system such as Intermediate Heat Exchanger with Steam Generator or the unit for heat conversion system with or without a hydrogen production device. The typical geometry of the "Energy Unit" is about 13.5m in height and 3.5m in diameter. In the case of thermo-chemical reactor using Mg-I cycle the volume of the single H_2 reactor in one "Energy Unit" is insufficient to convert the thermal energy generated by a single core unit. Hence, one "Core Unit" is combined with a number of "Energy Units". The production of electricity and hydrogen with high efficiency at the second design step will be performed by AMTEC-combined system which requires operational temperature around 1000 ^{o}C. The detailed design of the "Energy Unit" is out of the scope of this study.

The "Core Unit " with fresh-fueled core and reactor inner structures can be supplied from the factory to the site in a completely fabricated form. Its typical dimensions are about 6m in height and 3.5m in diameter. The design is done in the way to simplify the core structures and eliminate the equipment such as rotational plug, fuel handling machine, reactor deck, and control rod drive machine, which are massive but is essential for ordinary LMFBRs. The "Core Unit" consists of the reactor vessel with bottom-supported fuel tank, which are surrounded by the core former. The vessel is surrounded by a reactor vault made of concrete, which contained cooling systems. It should be emphasized that the used "Core Unit" can also be transported after the expiration of the core lifetime with the concrete reactor vault used for the radiation shielding during transportation. In this case, the total mass of the "Core Unit" and reactor vault is less than 200 ton.

The core mechanical structure configuration of the proposed MPFR concept is rather different from the structure of conventional reactors (shown in Fig. 2). The fuel remains liquid during reactor operation. The liquid Pu-U metallic fuel alloy is contained in a tantalum tank and cooled by lead supplied through a number of coolant channels assembled within the fuel tank. An additional portion of the fuel is located above the active core in the so-called fission product separation region, where the volatile and light weight FPs are partly sustained or separated to be processed. The core system is connected to the heat exchanging system though the number of valves and can be separately replaced by new one. Electro-magnetic pumps located within the unit for hydrogen production and heat exchanging pump liquid lead, which was selected for the coolant and reflectors due to its excellent neutronic characteristics[5]. The reactor power is controlled by a number of rods with neutron absorber inserted from the core bottom.

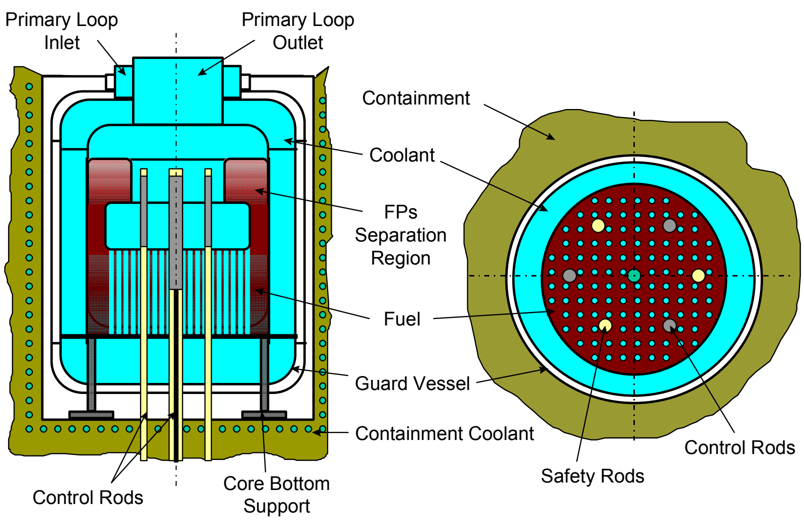

FIG. 2. The MPFR Core Concept ("Type B")

We suggested Pu-U alloys with relatively high melting temperature (about 870 °C) at the beginning of core life for the basic case of the core fuel to satisfy the condition of fissile material production in one-region breeder core concept. Additional elements, such as Mn added to this alloy do not dilute the fuel volumetrically, but slightly decrease the melting temperature of alloy (vary from 750 °C to 800 °C depending on Pu enrichment 10%-20%).

TABLE I. REACTOR SPECIFICATIONS

Item	*Parameter*
Thermal power, [MWt]	300
Average core power density, [W/cc]	186
Fuel material alloy	20% Pu - 80% U
Plutonium composition in fuel (^{238}Pu / ^{239}Pu / ^{240}Pu / ^{241}Pu/^{242}Pu) , [%]	2.8 / 49.9 / 25.6 / 14.8 / 6.9
Uranium composition in fuel	Natural
Coolant material	Lead
Structural material	Coated Tantalum
Control mechanism	B_4C control rods
Active core height, [mm]	700
FP separation region height, [mm]	700
Core diameter ID/OD, [mm]	1350 / 1390
Radial reflector thickness, [mm]	400
Coolant channel OD/ID, [mm]	10.0 / 7.6
Pin pitch-to-diameter ratio	1.55
Number of coolant channels	5983
Coolant temperature Outlet/Inlet, [°C]	1150 / 900
Coolant mass flow rate, [kg/sec]	8146.64
Core material volume fractions (Fuel/Coolant/Structure), [%]	60 / 25 / 15
Coolant velocity in the channel, [m/sec]	2.6
Fuel burnup, [MWd/tHM]	308400
Core operational lifetime, [years]	~ 30

The limitation on metallurgical knowledge about the materials able to sustain liquid Pu-U fuels under high temperatures, the excellent strength and heat-transfer properties of tantalum, its corrosion resistance against plutonium fuels and good fabricability led us to the conclusion that tantalum should be the material for the core structure. The parasitic capture cross section would be intolerable in the epithermal or thermal reactor, and, though relatively large in fast neutron spectrum, its effect on neutron economy in a fast reactor can be made small by careful design. In this configuration, the tantalum structure is considered with special coating made at the lead coolant side. The technology based on the oxide films developed for submarine nuclear reactors[5] cannot be applicable in this case. The experiments performed with lead and lead-bismuth with EP 823 steel cladding showed that these materials could be used as reactor coolants for temperatures up to 650 °C. The oxide films are very sensitive to the concentration of oxygen in the coolant, which should be kept in a narrow range. If the operational temperature exceeds 480 °C for lead and 650 °C for the lead-bismuth, the oxygen corrosion and liquid metal corrosion take place. However, for the study we assumed to use the lead coolant under the temperatures higher than experimentally evaluated, hoping that the material technologies will be developed in the future.

The proposed core configuration permitted to increase the fuel volume fraction up to 60 % with relatively small core size. The calculations of thermal-hydraulics, criticality search and burn-up showed that 135 cm diameter and 70 cm height core capable to sustain criticality over 30 years without refuelling. Obviously, such a long operational core lifetime cannot be realized with the conventional cladding material technologies. The core design specifications are shown in Table 1.

3. NEUTRONICS CHARACTERISTICS OF THE MPFR CORE

The neutronic characteristics of the MPFR cores were investigated. The core neutron kinetics parameters and reactivity feedback coefficients were calculated for the beginning and end of core life conditions (BOCL and EOCL, correspondingly) based on a two-dimensional homogeneous cylindrical reactor model using the neutron diffusion code CITATION[6] and the code for perturbation calculations PERKY[7] with the JENDL-3.2[8] neutron cross section library. Region dependent cross sections were generated using the code for cell homogenization calculations SLAROM[9]. The burnup calculation study was done by one point, one group burn-up calculation code ORIGEN 2.1[10] with updating the necessary cross section library data for structural materials and activation products of the metallic fuel. The active core was treated as one homogeneous region and burn-up calculations based on a constant flux approximation have been performed for each 365 days of core life. The effect of FP separation was not taken into account during the burn-up calculations. One burn-up calculation and two calculations of static neutronics were performed in each cycle. No cooling time was assumed between burn-up steps and before reprocessing. The coolant void reactivity worth was calculated using exact perturbation theory, and the coolant density, fuel density and Doppler coefficients were obtained using the first-order perturbation theory and 70-group cross section data. The core expansion coefficients were obtained from the differences of eigenvalues calculated for perturbed and unperturbed core conditions using 70-group cross sections. The results are summarized in Table 2.

TABLE II. MAIN CORE CHARACTERISTICS AND REACTIVITY FEEDBACK COEFFICIENTS

Parameter		*BOCL*	*EOCL*
Doppler Constant, [Tdk/dT]		-9.96×10^{-4}	-6.57×10^{-4}
Fuel temperature coefficient, [Δk/kk'/°C]		-2.45×10^{-5}	-2.49×10^{-5}
Coolant temperature coefficient, [Δk/kk'/°C]	Core	3.63×10^{-7}	3.33×10^{-7}
	Overall system	-1.77×10^{-7}	-1.72×10^{-7}
Structure temperature coefficient, [Δk/kk'/°C]	Core	8.57×10^{-7}	8.38×10^{-7}
	Core vessel	-2.46×10^{-2}	-3.74×10^{-2}
Core radial expansion, [Δk/kk'/ΔR/R]		1.02×10^{-3}	1.04×10^{-3}
Core axial expansion, [Δk/kk'/ΔH/H]		9.72×10^{-4}	1.01×10^{-3}
Delayed neutron fraction		3.91×10^{-3}	3.53×10^{-3}
Peak neutron flux, [$cm^{-2}s^{-1}$]		$1.9 \times 10^{+15}$	$2.7 \times 10^{+15}$
Prompt neutron lifetime, [sec]		2.04×10^{-7}	1.79×10^{-7}

TABLE III. COMPARISON OF REACTIVITY FEEDBACK COEFFICIENTS (SMALL METALLIC-FUELED CORES)

Parameter	*MPFR Core*	*Solid Metallic-Fueled Reference Core*
Power output, [MWt]	300	450
Core diameter and height, [cm]	135 x 70	153 x 120
Breeding ratio (max.)	1.10	1.19
Doppler constant, [Tdk/dT]	-7.76×10^{-4}	-4.01×10^{-3}
Fuel temperature coefficient, [¢ /°C]	- 0.742	- 0.073
Coolant temperature coefficient, [¢ /°C]	0.014	0.127

A comparison of the reactor characteristics with a conventional 150 MWe solid metallic-fuelled fast reactor[11] is shown in Table 3. The liquid metallic fuel has much larger volumetric expansion coefficient for the given temperature changes compared with solid fuels have, and it can be seen that fuel density coefficient is rather larger negative than that of the conventional designs with metallic-fuelled solid cores. In the current design, positive reactivity insertion caused by changes in the geometry of core structure and coolant heat-up will be compensated by large negative fuel density reactivity and Doppler feedback.

The influence of several FPs on core criticality is evaluated. In the fuel design of the conventional FBR, the lifetime of the fuel pin is mainly determined by the in-pin pressure due to rare gas migration and the cladding swelling due to the neutron irradiation. We suggested that the resident time of the fuel in the core could be extended without adding any special burnable absorber nuclide, like ^{237}Np or ^{231}Pa. In the MPFR core concepts two considerations are introduced to elongate the core lifetime by the removal of major FPs from the liquid fuel.

- A fuel tank with selective venting of the rare fission gases through porous media. It would decrease the inner pressure of the fuel tank and prevent the thermal creep failure of the fuel cladding.
- Effect of the density separation between the fuel alloy and FP elements produced in the fuel matrix. The density of several FPs is lower than the density of liquid fuel and the volatile and gaseous FPs will migrate upward the core to the FPs separation region.

Such a concept is schematically illustrated in Fig. 3. The burn-up calculations showed that an extremely high fuel burn-up of about 308.4 MWd/kg can be achieved by employing fuel with relatively high uranium content in the alloy composition. The goal of the survey was to reach high conversion of fertile material (^{238}U) with reduced fissile in the fuel, and the maximum breeding ratio of 1.1 has been obtained.

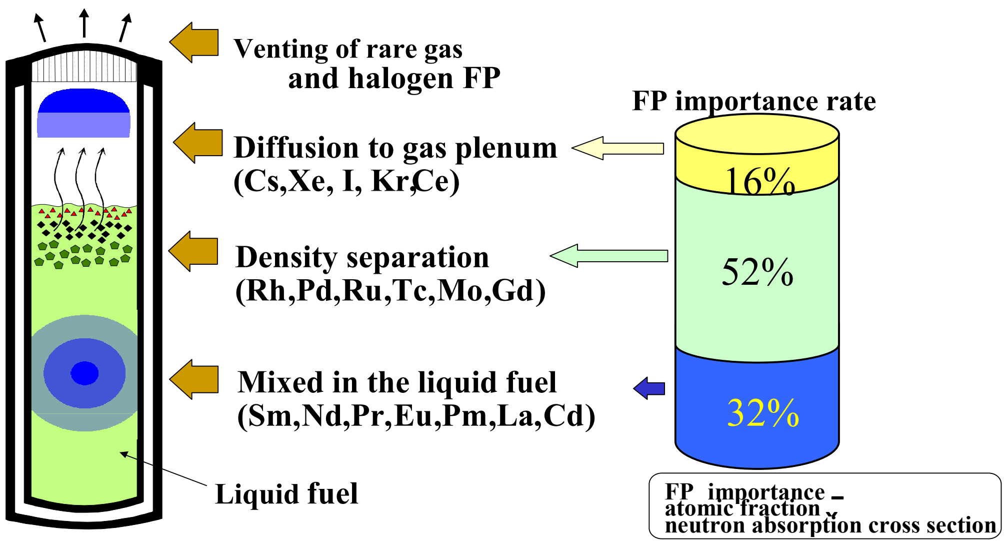

FIG. 3. Effect of the In-Pin Natural FP Separation

Introducing the “FP importance”, defined as the product of the atomic fraction multiplied by the neutron absorption cross section for each FP nuclide produced during the core operation, preliminary evaluations showed the following results on the efficiency of the density separation.

- The importance of the rare gases and volatile FPs such as Xe, Kr, Cs, Ce, and I corresponds to 16% of the total FP importance in the core. These elements will be separated from fuel and diffuse to the gas plenum from the fuel tank.
- The importance of the non-volatile and light FPs elements such as Rh, Pd, Ru, Tc, Mo, and Gd with melting temperatures higher than the fuel alloy corresponds to 52% of the total FP importance. These elements will be deposited near the surface area of the fuel tank.
- The importance of the non-volatile and light FP elements such as Sm, Nd, Pr, Eu, Pm, La, and Cd with melting temperatures lower or close to that of the fuel alloy corresponds to 32% of the total FP importance. This group of FPs will be resolved and mixed with fuel alloy in the fuel tank.

The calculated importance of several FPs on reactor poisoning is shown in Fig. 4. These preliminary evaluations indicated that about 70% of the total FP importance would be decreased due to the inherent separation mechanism. For the case of the initial composition in the fuel alloy 20% Pu – 80% U, the influence of the removal of the main FPs on the behaviour of the effective multiplication factor and fuel burn-up were evaluated and shown in Fig. 5.

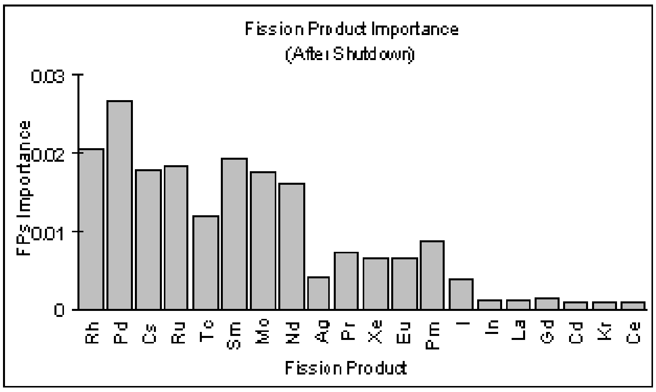

FIG. 4. The Importance of Fission Products on Reactor Poisoning

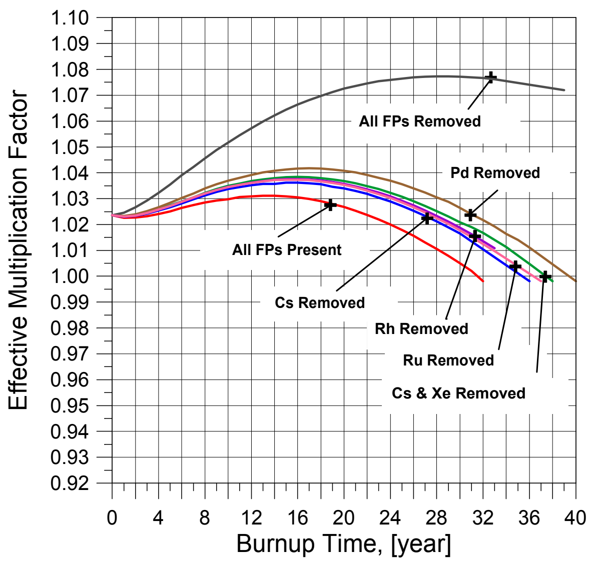

FIG. 5. The Effect of Fission Product Removal on the Effective Multiplication Factor

Two extreme cases can be seen: one corresponds to the case without FPs in the active core and the other is with all FPs being present in the core. In this computational model the initial concentration of the major FP under survey was supposed to be zero at the beginning of each next step of the burn-up calculation. Consequently, we considered that most of the FP elements would be separated from the active core region.

4. SAFETY ANALYSIS OF THE MPFR CORE

4.1. Method of Evaluation

The safety analysis described in several sources [(4), (12)] showed the high potential of liquid metallic-fuelled fast reactors to meet the requirements for the future nuclear power plants and develop a compact system with improved safety features. As it was already mentioned, the neutronic characteristics of the MPFR systems have been discussed by Netchaev et al. [(13)] The core neutron kinetics parameters and reactivity feedback coefficients have been calculated in a two-dimensional homogeneous cylindrical reactor geometry using the neutron diffusion code and the code for perturbation calculations. Using these data, we tried to simulate the anticipated transients without scram for the reactor core. For the worst main events among ATWS we selected ULOF and UTOP accidents.

The characteristics have been evaluated by using a transient analysis code ARGO [(4)] which had been well qualified for the safety analysis of LMFBRs. In this code, the primary and secondary coolant systems, the decay heat removal system, and the core are modelled by a flow network model that includes the models for flow mixing in the plenum, and heat transfer between the coolant and the in-vessel structure. The core is represented by four channels: three core channels with different power, and reflectors. The reactor power transient is calculated by the point kinetics model with six groups of delayed neutron precursors. The reactivity feedback effects are calculated taking into account the spatial distributions of the reactivity coefficients of the core materials including those for the coolant, Doppler and fuel volumetric expansion. Thermal expansion is considered for the reactor vessel and core support structures. It should be mentioned that to perform these calculations, the ARGO code was upgraded to evaluate the system, which uses the lead coolant instead of sodium.

4.2. Unprotected Loss-of-flow Transient

The ULOF was initiated by the loss of power supply to the main coolant pumps with failure of the reactor scram. There are several conditions, which can be assumed as critical conditions for the judgement of the self-controllability or self-shutdown capability of the MPFR core. They are the lead boiling, the loss of the material integrity of structure, and the melting of fuel cladding. The cladding has a quite high melting temperature (about 2977 ^{o}C) and, probably, will not occur before the coolant boiling would start. The most critical event in the system with liquid plutonium fuels is the diffusion of plutonium through the grain boundaries of the structural materials under high temperatures. It was found in the LAMPRE reactor and well described by Andelin et al. [(3)] . Probably, this effect will limit the core life under ATWS, however due to the lack of the experimental data, we considered that the limiting factor for the ULOF should be the boiling of the lead coolant.

According to the coolant heat-up due to mismatching of the coolant flow and reactor power shown in Fig. 6, negative reactivity is introduced for the reactor self-shutdown. Fig. 7 shows the typical transient behaviour of the reactivity components of the MPFR core for the coolant flow halving time of 10 seconds. It can be seen that the Doppler coefficient for the core is quite small compared with conventional metallic-fuelled cores. The hard neutron spectrum associated with high neuron leakage reduces the Doppler reactivity feedback by reducing the epithermal neutron population. Moreover, as the concentration of ^{238}U in the core is continuously decreasing and the plutonium produced during the core life makes the neutron spectrum harder, the result is a decreased Doppler coefficient at the end of core life. As already mentioned, the main contribution to the negative net reactivity is introduced by the thermal volumetric expansion of the liquid fuel materials. The positive components introduced by the core structure and core coolant heat-up are small and do not drastically affect the overall core performance.

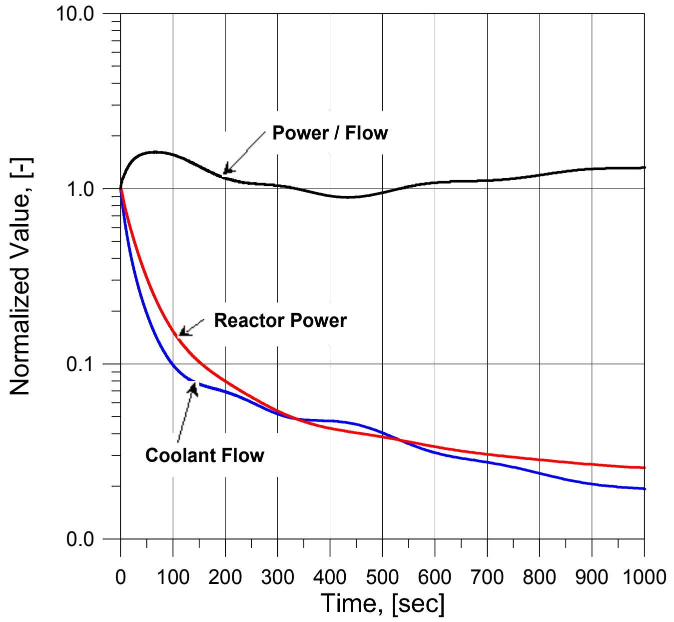

FIG. 6. Reactor Power and Coolant Flow Rate (ULOF.)

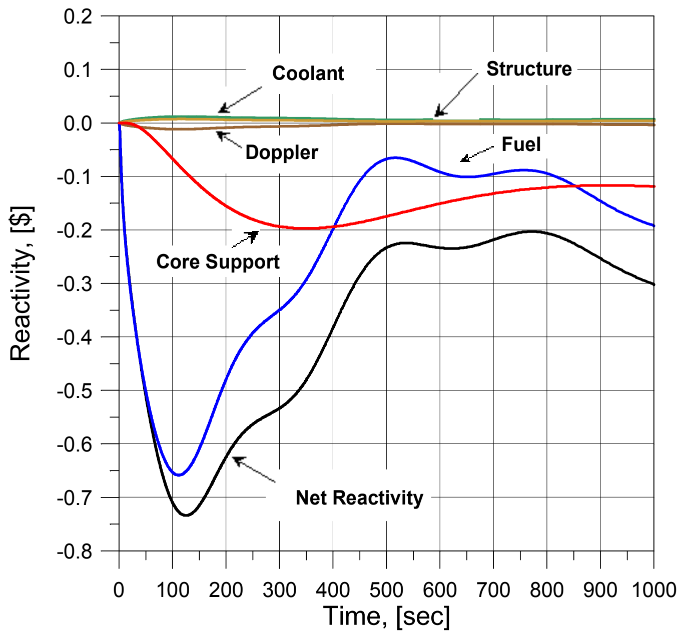

FIG. 7. Reactivity Balance (ULOF).

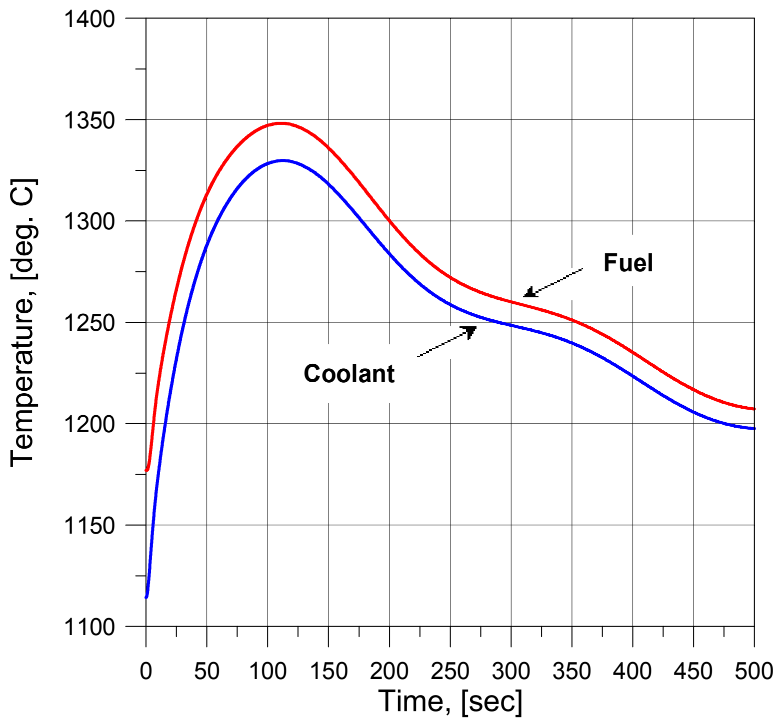

FIG. 8. Temperature of Core Materials (ULOF).

TABLE IV. RESULTS OF PARAMETRIC SURVEY FOR ULOF ACCIDENT (ARGO CODE)

Case	Coolant Flow Halving Time	Coolant Boiling
ULOF	0.2 sec	Yes
	1 sec	Yes
	3 sec	Yes
	3.5 sec	No
	4 sec	No
	5 sec	No
	10 sec	No

The maximum coolant temperature of 1329 ^{o}C was found at the core outlet after 124 seconds from the beginning of the transient (shown in Fig 8). Later, the temperature decreased and showed a tendency to stabilize at some level. The boiling temperature of lead coolant is about 1737 ^{o}C. It is found that the MPFR has large tolerance of temperature before coolant boiling can be started. We suggested using the lead coolant as neutron reflectors in the MPFR core design. In the condition of coolant boiling, passive shutdown should also be achieved by the negative reactivity insertion due to increased core leakage from the core upper reflector where voiding will be formed. The void reactivity worth for this region corresponded to –0.63$, that is larger than the coolant void reactivity in the upper part of the core (+0.54$). The boiling mechanism should play important role to maintain the core integrity during the coolant-boiling phase.

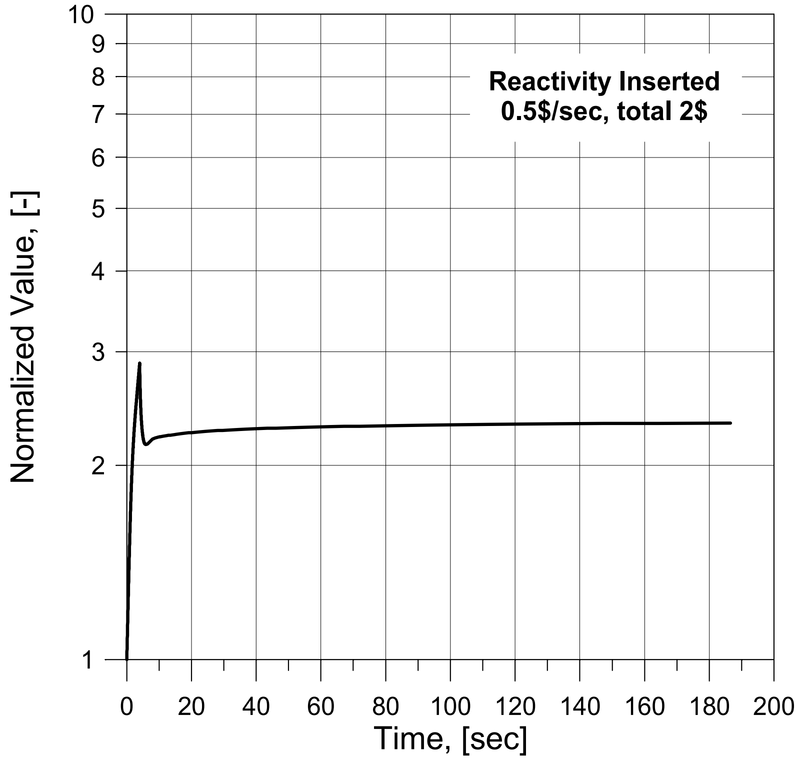

FIG. 9. Reactor Power and Coolant Flow Rate (UTOP).

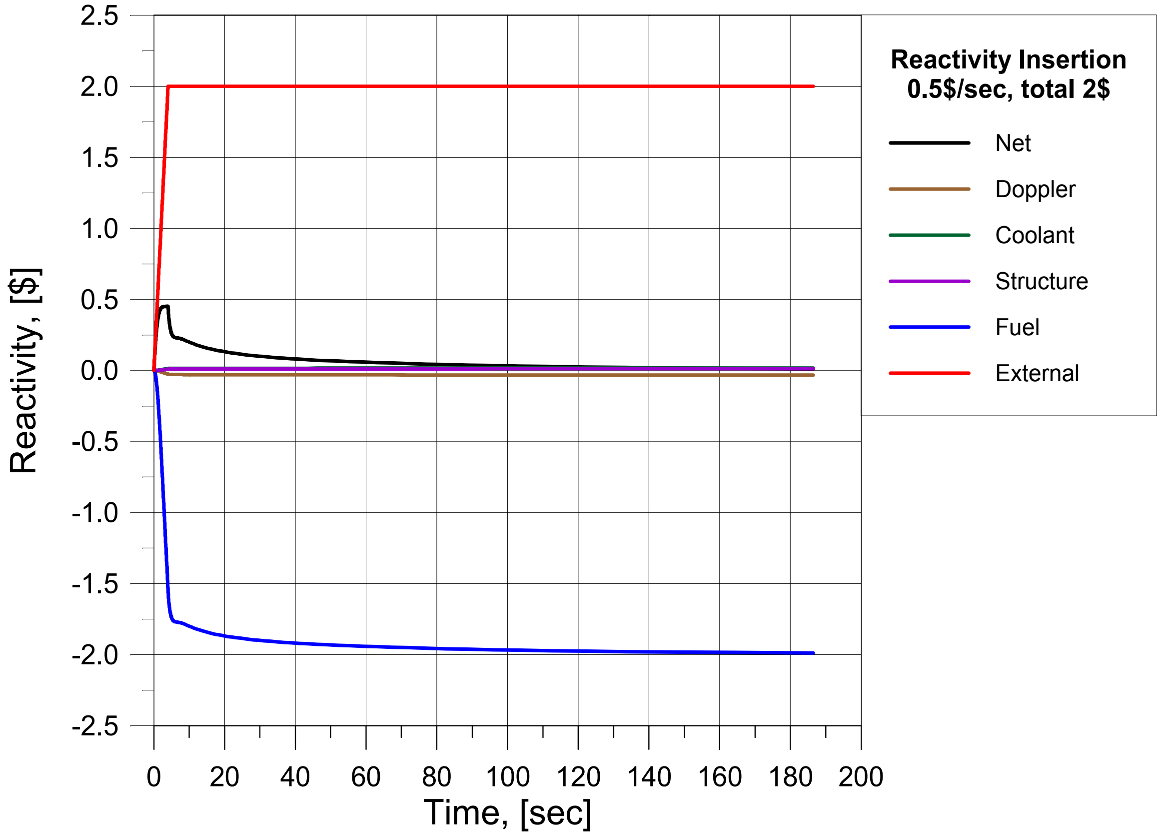

FIG. 10. Reactivity Balance (UTOP).

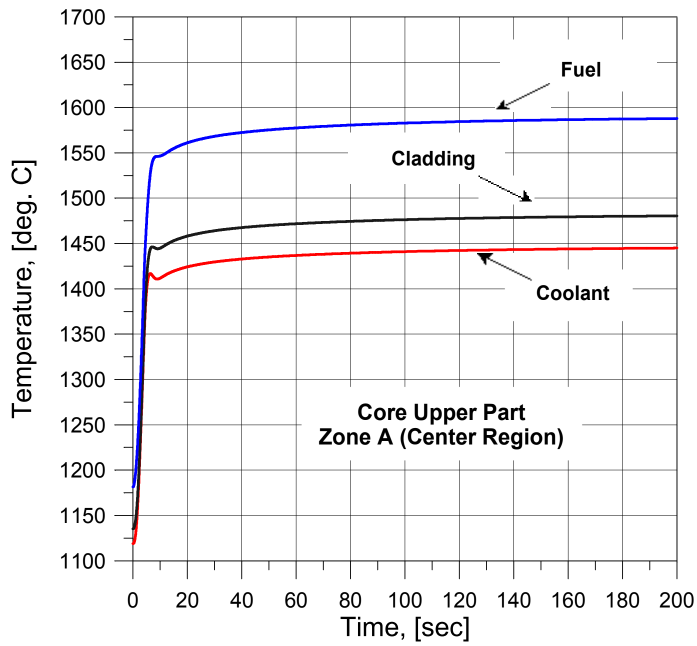

FIG. 11. Temperature of Core Materials (UTOP)

TABLE V. RESULTS OF PARAMETRIC SURVEY FOR UTOP ACCIDENT (ARGO CODE)

Case	Reactivity Insertion Rate	Coolant Boiling
UTOP	1$ /sec, Total 6$	Yes
	0.5/sec, Total 4$	Yes
	0.03$/sec, Total 4$	Yes
	0.03$/sec, Total 3$	No
	0.5/sec, Total 3$	No
	0.5$/sec, Total 2$	No
	0.03$/sec, Total 1$	No

Even in the event of cladding failure, the core damage and corresponding re-criticality would be prevented by the negative reactivity insertion due to the partial fuel discharge from the active core region by the lead coolant. Consequently, the core disruptive accident can be prevented for the ULOF case.

The coolant flow during ULOF is reducing in according with the pump and coolant inertial characteristics. It has many uncertainties to predict the proper halving time for the coolant flow. Conventional designs use 10 seconds halving time but for the electro-magnetic pumps (EMP), which are used in the MPFR concept, a smaller value should be considered. For example, for the PRISM reactor proposed by General Electric [14] which uses EMP and sodium coolant, the coolant flow halving time of 0.2 seconds is considered. Even with such a short time, the reactor can maintain self-shutdown against ULOF transient because of the large neutron leakage introduced by the removal of the sodium coolant from the Gas Expansion Modules located around the perimeter of the active core. For the MPFR core, we performed parametric studies for the ULOF case with a different coolant flow halving time.

The results of the transient calculations are presented in Table 4. The reactivity showed similar behaviour during the ULOF transients for all the cases with no boiling. The flow halving time of 3.5 seconds was obtained as a limit for MPFR. It should be mentioned that such conditions are extreme because the lead has larger inertia than the sodium coolant and, perhaps longer flow halving time should be considered.

It can be concluded that the investigated MPFR core demonstrated good self-shutdown characteristics against ULOF transient.

4.3. Unprotected Transient Overpower

For the reactor design, metallic fuels have such superior characteristics as high density and high thermal conductivity. At the same time, the melting temperature of metallic fuel is quite low. For conventional LMFBRs designs, the temperature of the metallic fuel is kept low and good heat removal should be provided. Particularly for that reason, reduction of the control rod worth should be considered to limit the reactivity insertion in the case of UTOP. Since the control rod required reactivity is limited by the fuel burn-up reactivity margin which is necessary to reach the long operational core lifetime, the fuel burn-up reactivity should be reduced by proper design. Here we have a dilemma for small reactors: to increase the fuel volume fraction, to change the fuel faster, or to have a bigger volume core with many control rods.

The solution can be found by using liquid metallic fuel. Having a rather large volume fraction of fuel in the active MPFR core (about 60%) and adjusting the fuel composition in the burn-up study, we succeeded in decreasing the burn-up reactivity margin [13]. The total burn-up reactivity swing for the MPFR core corresponded to about 8$, that is less than conventional reactors have. Thus, the reactivity insertion due to accidental withdrawal of even a number of control rods can be kept small.

Considering prevention and mitigation of CDA initiated by UTOP, the MPFR core has special characteristics that should be taken into account for judgement. The fuel is already molten at the normal operational conditions and it should stay molten during all transients. The integrity of the fuel tank and the coolant channels at high temperature is under question, and at the current step of the study is not evaluated experimentally or numerically (same problem as in the ULOF case). If local coolant boiling occurs, the consequence of events will depend on the location of the void. Of course, it is expected that negative reactivity will be inserted immediately due to the inherent thermal expansion of fuel. As the critical condition for the MPFR core under UTOP, we selected boiling of the lead coolant.

The UTOP was initiated by the insertion of external reactivity with normal operation of the coolant pumps and IHXs. The results are shown in Figs. 9, 10, 11 for the external reactivity of 2$ with a ramp rate of 50 cents per second. It is rather a large value compared to conventional reactors where a 1$ with ramp rate of 3 cents per second is usually assumed. The characteristic response is an increase in power to about 2.7 times full power, which is than countered by the fuel expansion and other feedback mechanisms, which returned the power into the 2.1 times full power range. The maximum temperatures did not reach the boiling temperature of coolant. In this case, the MPFR core could maintain the passive self-shutdown. The results of a parametric survey are presented in Table 5. We found that the MPFR core is not very sensitive to the reactivity insertion rate, but it has the limit for the total reactivity insertion worth that corresponds to 3$.

CONCLUSIONS

The basic concept of the multipurpose fast reactor (MPFR) with liquid metallic-fuelled cores is proposed and investigated. It was found that the MPFR satisfies such design characteristics to be the potential candidate to establish energy sources as alternatives to fossil fuels in the next century. The multiple energy conversion capability for electricity generation with high efficiency and production of hydrogen, which can be used as an alternative fuel in transportation systems, is achieved by application of liquid fuel technologies with high reactor operational temperatures. The extended capability for reactor siting close to the energy consumption area is based on the concept of a separated transportable long-life core and energy units with long operational lifetime without on-site refuelling. High conversion ratio and high fuel burn-up of the MPFR core were achieved by means of a hard neutron spectrum and selection of the proper fuel composition. The MPFR core is designed to realize the partial separation of fission products from the fuel, which was considered and studied. The results of the preliminary analysis showed that from the viewpoint of neutronics, the core lifetime could be extended by several years if the fission products separation occurs. Neutronics calculations indicated that the thermal expansion of liquid fuels would cause a decrease in reactivity that would be larger in magnitude than any other thermally induced reactivity changes. This creates the balance between the active and inherent safety features of the MPFR cores, such as self-controllability and ability to self-terminate which are out of the scope of this paper.

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SMALL POWER SODIUM COOLED FAST NUCLEAR REACTORS

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Abstract

1.5 MW(e), 12 MW(e) and 170 MW(e) small power sodium cooled fast reactors have been developed. The reactor plants were developed as universal power units for economically effective energy and industrial steam generation and heat supply. The main features increasing the power unit economic efficiency are:

- serial fabrication of standard RPs at the factory and delivery of reactor vessels in ready made form;
- realization of self-protection principles and use of passive systems in RP;
- use of standard machine room equipment, fabricated in accordance with the rules of conventional heat power engineering;
- use of turbine plant with thermodynamic coefficient, exceeding the corresponding value for the plants of PWR type.

For MBRU-1.5 and MBRU-12 RPs it is proposed to use a core without FA replacement during the whole service life (30 years) and for BMN-170 RP it is proposed to use a core with a 4 year operating period and 1 year between the refueling shutdowns. During the whole service life a minimal number of operating personnel will be needed for the plant servicing. The personnel functions will be periodically to observe the parameters of technological process.

Passive principles are used in the main RP safety systems:

- a passive type system of emergency residual heat removal system provides heat removal directly through the reactor vessel forced air cooling due to the natural air chimney effect;
- an emergency reactor shut-down system is provided by emergency protection rods with active-passive action.

1. POWER UNIT WITH MBRU-1.5 REACTOR PLANT

The power unit consists of fast sodium cooled reactor plant SFRP (small fast reactor plant) and turbine plant with the turbine of P-1.5-35/5 type.

Main technical data of SFRP power unit

Parameter	Value
Maximal electric power, MW	1.5
Maximal heat supply, Gcal/hr	7.0
Gross efficiency, %	27.2
Capacity factor, %	90
Lifetime, years	30

Nuclear power stations of 1.5; 3.0 MW etc. can be developed on the basis of power units with SFRD for remote, thinly populated regions. The features of the reactor plant with SFRP are:

- ultimate safety level;
- high efficiency;
- minimum maintenance of the reactor plant at operation;
- compactness.

The ultimate safety level of the power unit is assured owing to:

- high reliability of the reactor plant, provided by the design solutions, 100% factory fabrication and assembly of the reactor vessel and reactor plant equipment;
- use of design solutions on the main components of the reactor plants and sodium coolant technology, confirmed by the experience of long operation of BN-350 reactors in Shevchenko and BN-600 reactors at Beloyarskaya NPS;
- inherent self-protection features of fast reactors with sodium coolant (negative temperature and power reactivity coefficient, negative sodium void reactivity coefficient, low coolant pressure).

To increase the power unit stability to shock wave and aircrash the reactor compartment is located below ground under a protective concrete plate. If the power units are to be located in the zones with increased seismic activity it is planned to use special shock-absorbers to decrease seismic loads upon the reactor equipment and safety systems. Fire protection of sodium systems of primary and secondary circuits is provided by protective safety housings with constant checks of their tightness.

Main theses of the power unit economic efficiency increase concept

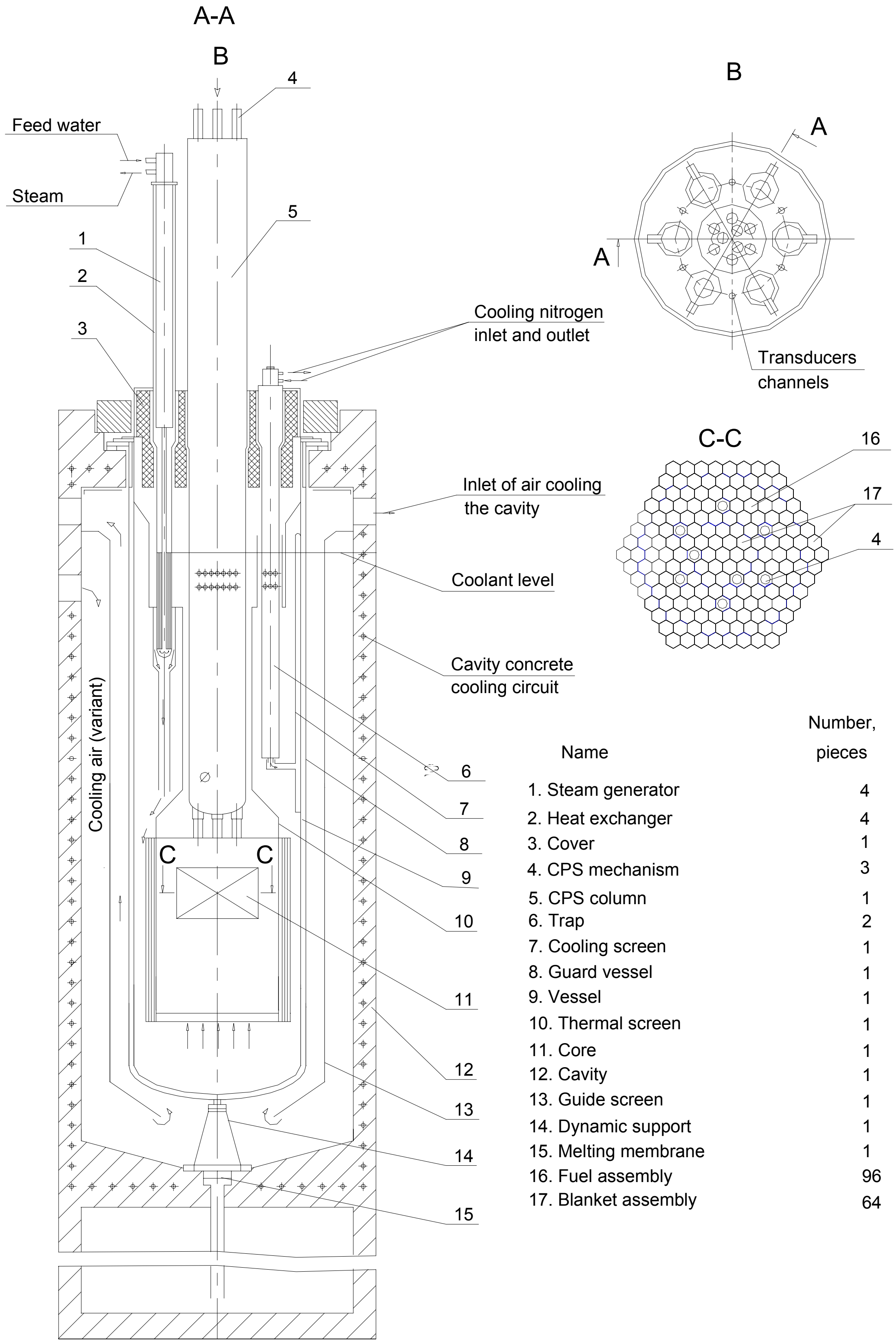
The main features increasing the power unit economic efficiency are:

- serial fabrication of standardized reactor plants SFRP at the factory and their delivery in a ready-made form to the site decreases the labor content at the power plant site and construction time;
- the use of self-protection principles and of passive systems in the reactor plants, allows a decrease in the number of auxiliary systems and safety systems;
- use of standardized equipment in the machine room (water preparation systems and turbogenerator), fabricated in accordance with the rules of conventional heat power engineering;
- use of turbine plant with thermal-dynamic efficiency, exceeding the corresponding value for the plant of PWR type.

SFRP reactor plant

The features of SFRP are:

- natural coolant circulation in the primary and secondary circuits;
- integral lay-out of primary and secondary circuit equipment.



Name	Number, pieces
1. Steam generator	4
2. Heat exchanger	4
3. Cover	1
4. CPS mechanism	3
5. CPS column	1
6. Trap	2
7. Cooling screen	1
8. Guard vessel	1
9. Vessel	1
10. Thermal screen	1
11. Core	1
12. Cavity	1
13. Guide screen	1
14. Dynamic support	1
15. Melting membrane	1
16. Fuel assembly	96
17. Blanket assembly	64

FIG. 1.1. The SFRP reactor design.

The reactor plant consists of the reactor, intermediate heat exchangers, located in the vessel, surrounded by the guard housing, and steam generators.

The reactor plant consists of three-circuits of integral type; primary and secondary circuit equipment are located in one and the same vessel of 0.45 m external diameter and 10 m height. In the lower part of the vessel (Fig. 1.1) there is a core, surrounded by radiation protection. The reactor cavity has a cover on which there are the following elements: column CPS with CRDM of reactor control and protection systems, two filter-traps with nitrogen coolant cooling and electromagnetic pump for coolant circulation through the trap, four heat-exchange modules in the lower part of which there are heat exchangers of primary-secondary circuits and in the upper part there are steam generators.

The reactor vessel is surrounded by the guard vessel from outside. The presence of the guard vessel around the main one guarantees the preservation of coolant circulation through the core at the main vessel depressurization. The reactor lay-out provides heat removal from the core both in nominal and in emergency operation modes due to natural circulation in the primary and secondary circuits owing to the distribution of corresponding heat-exchange surfaces along the height. At the third circuit failure heat removal from the reactor vessel is provided owing to natural air convection through the reactor cavity.

SFRP dimensions allow its transportation by road, railway and water.

During the whole service life a minimum number of personnel is required for plant maintenance. The function of personnel will be mainly to observe the parameters of technological process periodically. It is proposed to use a core without FA replacement during the whole service life of the reactor. So practical non-maintainability of the reactor plant eliminates such problems as erroneous non-professional or even ill-intentioned personnel actions and safety provision when performing the potentially hazardous operations connected with the core refueling.

Technical and economical data

Parameter	Value
Thermal power, MW	5.5
Electric power, MW	1.5
Sodium temperature at the core inlet, ^{0}C	330
Sodium temperature at the inlet to intermediate heat exchangers, ^{0}C	480
Sodium temperature at steam generator inlet, ^{0}C	460
Sodium temperature at steam generator outlet, ^{0}C	280
Fresh steam temperature, ^{0}C	435
Steam pressure at SG outlet, MPa	3.44

The following passive principles of operation are realized in the main safety systems of the reactor:

- emergency residual heat removal system of passive type provides heat removal directly by SFRP vessel forced air cooling due to natural chimney effect;
- the emergency reactor shut-down system is equipped with emergency protection rods of active-passive action.

The low fuel rating of the core at 9 kW/l does not require the emergency cooling system: even in case of the rupture of coolant circulation circuit through the core the removal of residual heat with its dispersal through the reactor vessel to the cooling circuit of the reactor cavity without dangerous overheating of FEs is provided. Equipment and systems of the reactor plant are designed for the preservation of serviceability at magnitude 8 design earthquake and provision of safety at maximum design earthquake of magnitude 9 by MSK scale.

The safety level of the power unit with SFRP exceeds safety criteria of the regulations in force now at least by several orders. The probability of severe damage or melting of the core in beyond-design accidents does not exceed 10^{-8}.

Core characteristics

Parameter	Value
Thermal power, MW	5.5
Specific power, kW/l	9
Linear FA power, W/cm	10
Height, mm	750
Diameter, mm	1200
Number of FAs with enriched uranium	96
Number of FAs with depleted uranium	19
CPS rods number	9
Operating period, years	up to 25
Fuel type	UO245
Enrichment in U 235, %	45
Maximum burnout, % h.a.	5.0
Reactivity losses for burnout, % Ak/k	2

Turbine plant

High parameters, (435°C temperature and 3.44 MPa pressure) of steam generated by the reactor plant allow the use of turbine plant of the P-1.5-35/5 type, used in fossil plants, in the power unit.

Main P-1.5-35/5 turbine plant data

Parameter	Value
Maximum electric power, MW	1.5
Fresh steam pressure, MPa	3.44
Fresh steam temperature, ^{0}C	435
Feed water temperature, ^{0}C	160

2. MBRU-12 REACTOR POWER UNIT

The main part of the power unit MBRU-12 is a sodium cooled fast reactor:

Main technical data of the power unit

Thermal power, MW	48
Electric power, MW	12
Gross efficiency, %	25
Lifetime, years	30
Number of the reactors	1
Number of turbine plants	1

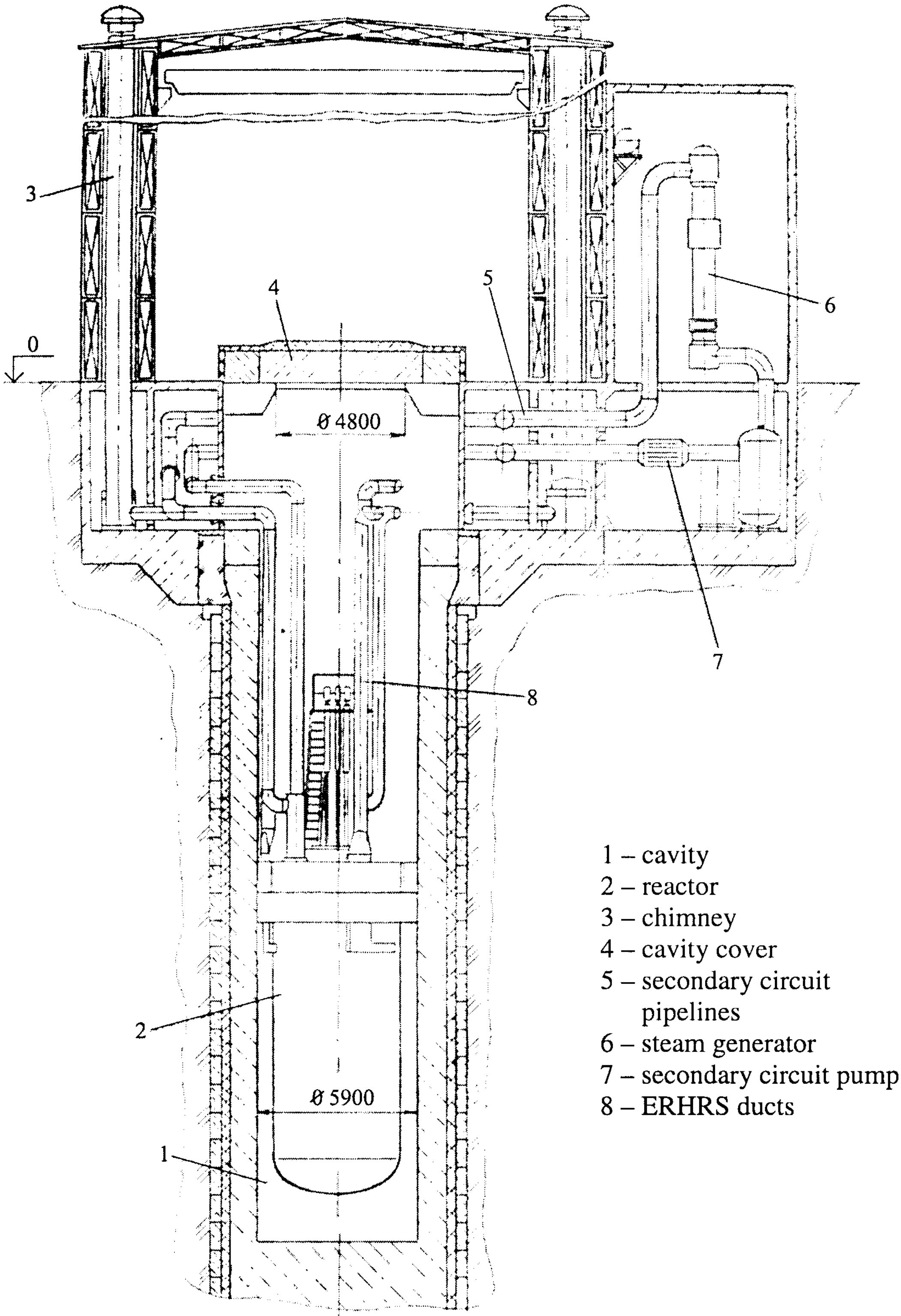
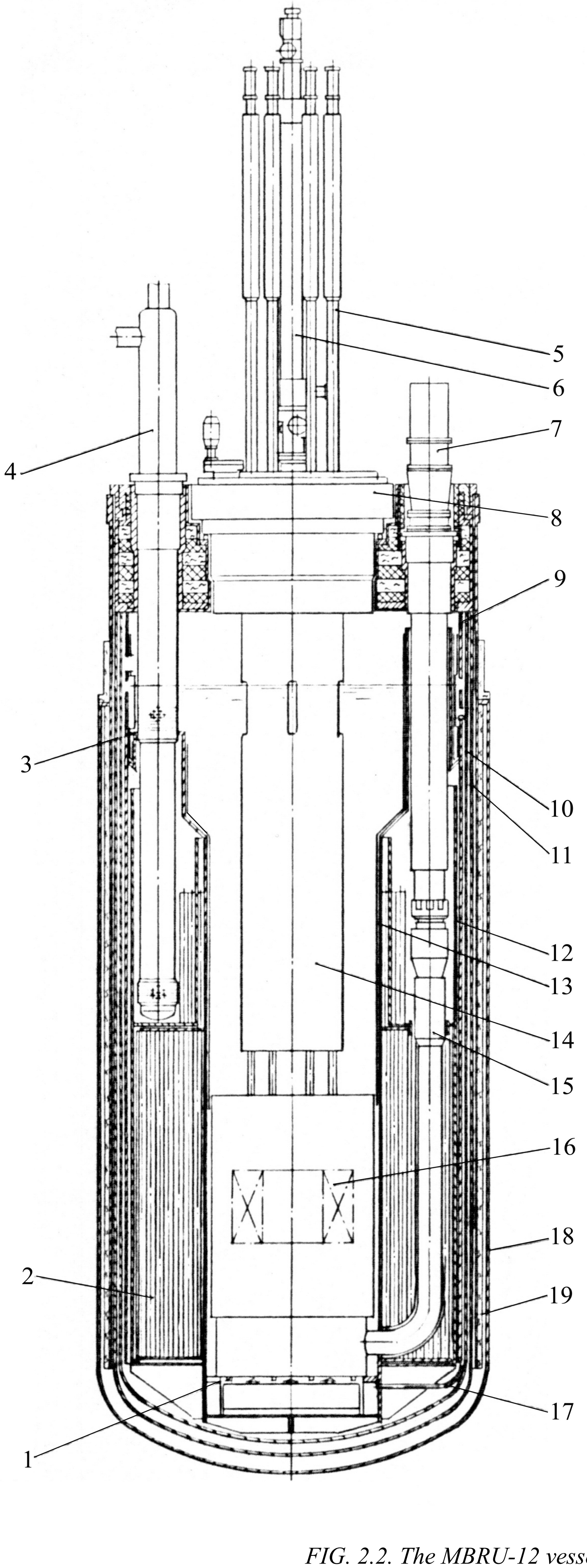


FIG. 2.1. The MBRU-12 reactor design.



1 - Pressure chamber
2 - In-vessel protection
3 - Thermal sealing
4 - Intermediate heat exchanger
5 - CPS mechanism
6 - FAs rearrangement mechanism
7 - Primary pump
8 - Rotating protection
9 - Pressure shell
10 - Reactor vessel
11 - Guard vessel
12 - Thermal screens
13 - Hot box
14 - CPS column
15 - Pressure piping
16 - Core
17 - Pipeline for sodium supply for thermal stabilization of the vessel
18 - ERHRS vessel
19 - Separating shell

FIG. 2.2. The MBRU-12 vessel.

Main design solutions

The MBRU-12 reactor plant (Fig. 2.1) is a three-circuit plant with sodium coolant in primary and secondary circuits and water coolant in the third circuit. Primary sodium radioactive circuit equipment (core, four IHXs, two primary main circulating pumps, two impurities traps) are placed in a common vessel (Fig. 2.2). The vessel has a cover on which there is a rotating plug with CPS drives and FA rearrangement mechanism installed. The main reactor vessel is arranged in a guard vessel (GV), which ensures coolant circulation through the core in the event of main vessel depressurization.

The transfer of heat from the primary circuit to steam-turbine plant is provided by an intermediate sodium circuit consisting of 2 (1) loops each containing SG, secondary circuit pump and communication pipelines. Electricity in the power unit is generated by 12 MW steam-turbine plant.

Main technical characteristics of MBRU-12 reactor

Primary sodium flowrate, kg/s	290
Primary sodium temperature, ^{o}C	
- at core inlet	330
- at core outlet	480
Secondary sodium temperature, ^{o}C	
- at steam generator inlet	460
- at steam generator outlet	280
Steam parameters:	
- pressure, MPa	9.0
- temperature, ^{o}C	435

Reactor core

The reactor core (Fig. 2.3) is composed of BN-350 and BN-600-type FAs which have been proven industrially. For the fuel, U-Pu dioxide using weapons-grade Pu is proposed. The use of uranium enriched oxide fuel is also possible.

A feature of the core is the elimination of FA refueling outside the core during reactor operation requiring primary circuit depressurization. Compensation for loss of reactivity for fuel burnout (not more than 0.8%, Δ k/k per year) is provided due to the in-core rearrangement of FAs once a year. Thereby the total lifetime corresponds to reactor service life.

Main core characteristics

Fuel	UO_2+PuO_2
Enrichment in Pu, %	23
Core fuel inventory, t	4.1
FA number	102
Blanket FAs number	120
CPS assemblies number	6
FAs wrapper size across flats, mm	96
Fuel element linear power, W/cm	200
Burnout:	
maximal, %, h.a.	15
average, MW day/kg	100

Safety features

The adopted integral design of the reactor with the placing of all equipment and systems with radioactive sodium into a common vessel, in combination with safety systems simplification provides for maximum compactness and safety both of the reactor and the reactor station in general. The three-circuit scheme of heat transfer guarantees the generation of pure steam and eliminates the ingress of radioactivity into steam-water circuit and to external steam users.

The enhanced stability of the power unit to external impacts (aircrash, shock wave) is provided due to placing the reactor and safety systems into a silo.The RP is designed to remain operable at earthquakes of magnitude 7 (to MSK-64 scale) and safety is ensured at magnitude 8 earthquake.

High reliability, stability in operation and radiation safety are the features which are intrinsic for fast nuclear reactors with sodium coolant. These features are based on:

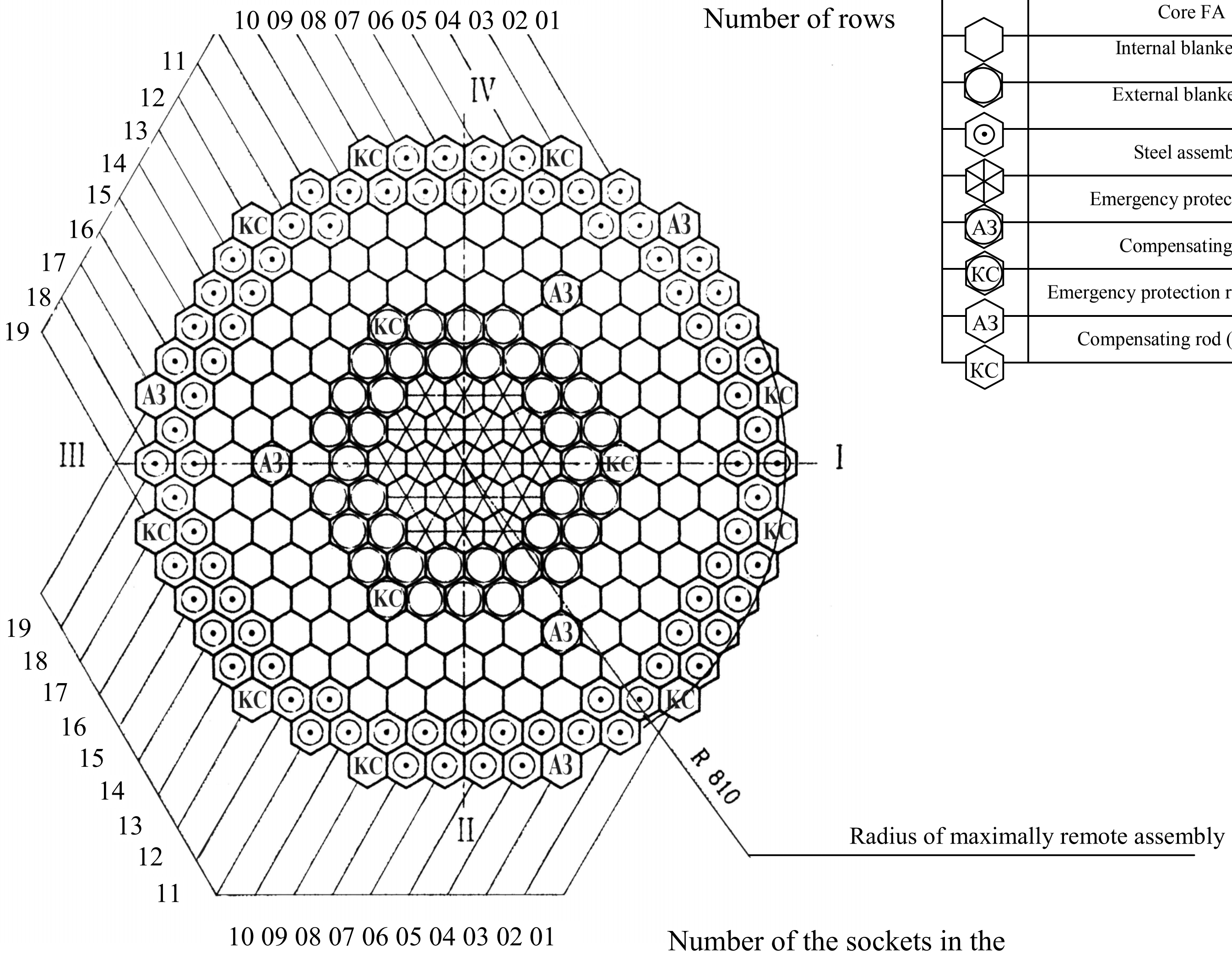
- reactor power self-regulation owing to stable negative feedbacks between physical processes and the core parameters;
- small reactivity margins in the core, absence of “poisoning” effects characteristic of other reactors;
- high stability of neutron fields in the core;
- low pressure of coolant in the reactor;
- excellent thermal-physical properties of sodium coolant;
- absence of corrosion problems for structural materials operating in sodium coolant.

Accidents with large loss of coolant, which are the most dangerous for other reactors, operating under excessive pressure, are practically excluded in the fast nuclear reactors. The probability of such an accident is assessed as 10^{-10} per reactor year. Inherent safety features of fast reactors have been maximally developed and supplemented in the MBRU-12 reactor (Figs 2.4 and 2.5). The possibility of uncontrolled reactor power increases accompanied by coolant boiling and loss is completely excluded (negative void reactivity coefficient). The reactivity margin for fuel burnup is reduced (by order of $2_{\beta\ eff.}$). Circulation of sodium in primary and secondary circuits in emergency modes is natural.

Passive residual heat removal system provides heat removal from the core to ambient air in accidents with complete loss of feed water supply to steam generators. In this case the reactor cooling down is provided by its vessel forced cooling by atmospheric air due to chimney effect. This system is constantly in operation and does not require connection. In an ultimate accident with power unit complete de-energization and non-actuation of all reactor shutdown means the reactor cooling is provided without coolant boiling and fuel melting (Fig. 2.6).

The principle of multi-barrier localization of radioactivity is realized in full measure for the MBRU-12 reactor:

Core assemblies cartogram (beginning of the fuel life)



Denotation	Name	Number	Additional notes
	Core FA	102	
	Internal blanket FA	36	
	External blanket FA	78	
	Steel assembly	19	
A3	Emergency protection rod	3	
KC	Compensating rod	3	
A3	Emergency protection rod (stand-by)	3	
KC	Compensating rod (stand-by)	9	

FIG. 2.3. The MBRU-12 reactor core.

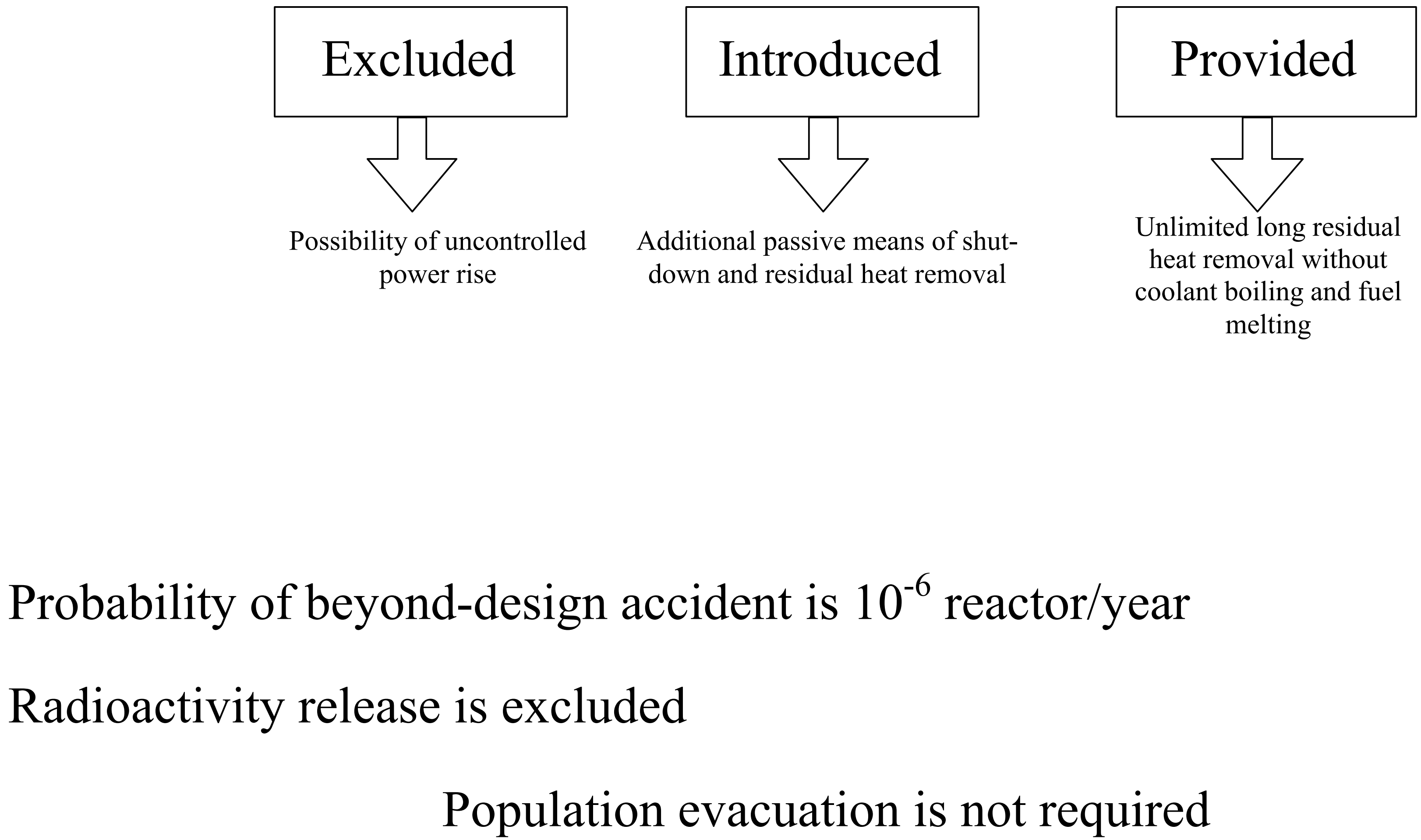


Fig. 2.4. Safety characteristics of MBRU-12

1. All the equipment items and sodium systems of primary circuit are located in the reactor vessel; external pipelines with radioactive sodium are excluded completely;
2. Reactor is located in a leak-tight GV, which is able to localize radioactivity release in the case of sodium leakage from the reactor;
3. The design use of the elements of path for vessel forced cooling by the air of ECCS represents a second GV capable of localizing the radioactivity escape in a beyond design accident associated with main and auxiliary vessels depressurization;
4. The RP is placed in a concrete metal-lined silo. The well lining together with the silo cover form an auxiliary barrier to the propagation of radioactive substances;
5. A three-circuit flow scheme for heat transfer from the reactor to working medium of steam-turbine cycle with pressure barriers between circuits has been used; it excludes radioactivity ingress into steam even in the event of the reactor plant's heat exchange equipment loss of tightness.

The probability of severe beyond-design accidents, the development of which leads to the reactor core damage, is considerably lower than 10^{-6} per reactor year.

Radiation safety

The development of the facility project is oriented on meeting the following radiation safety criteria:

1. Radioactive releases during normal operation and corresponding equivalent individual radiation doses for the population shall be not less then an order of magnitude lower than those established by the current norms in Russia (not more than 0.02 mSv/year for the whole body and 0.06 mSv/year for individual organs);
2. The values of equivalent individual radiation doses for the population on the boundary of sanitary restricted zone and beyond its boundaries in a design basis accident shall not exceed the doses established for normal operation (0.2 mSv for whole body and 0.6 mSv for individual organs per accident);
3. The values of equivalent individual radiation doses for the population on the boundary of the sanitary restricted zone and beyond its boundaries in beyond design accidents shall not exceed 5 mSv for whole body and 50 mSv for individual organs for the first year after the accident.
4. The preliminary safety analysis of the facility confirms meeting the criteria for design basis and beyond design accidents.

There is no accident in the MBRU-12 reactor where population protection measures will be required concerning radiological releases.

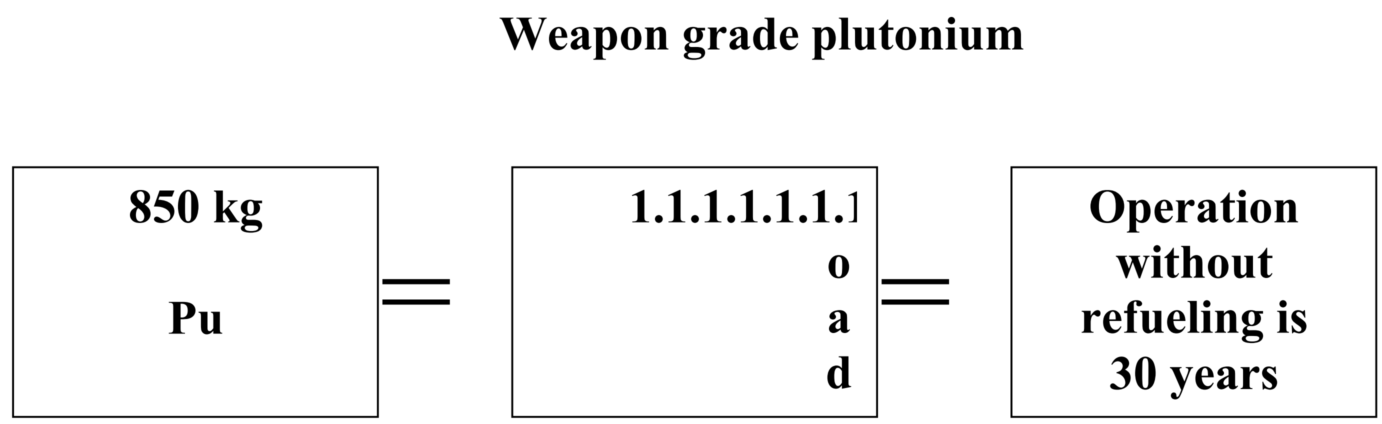
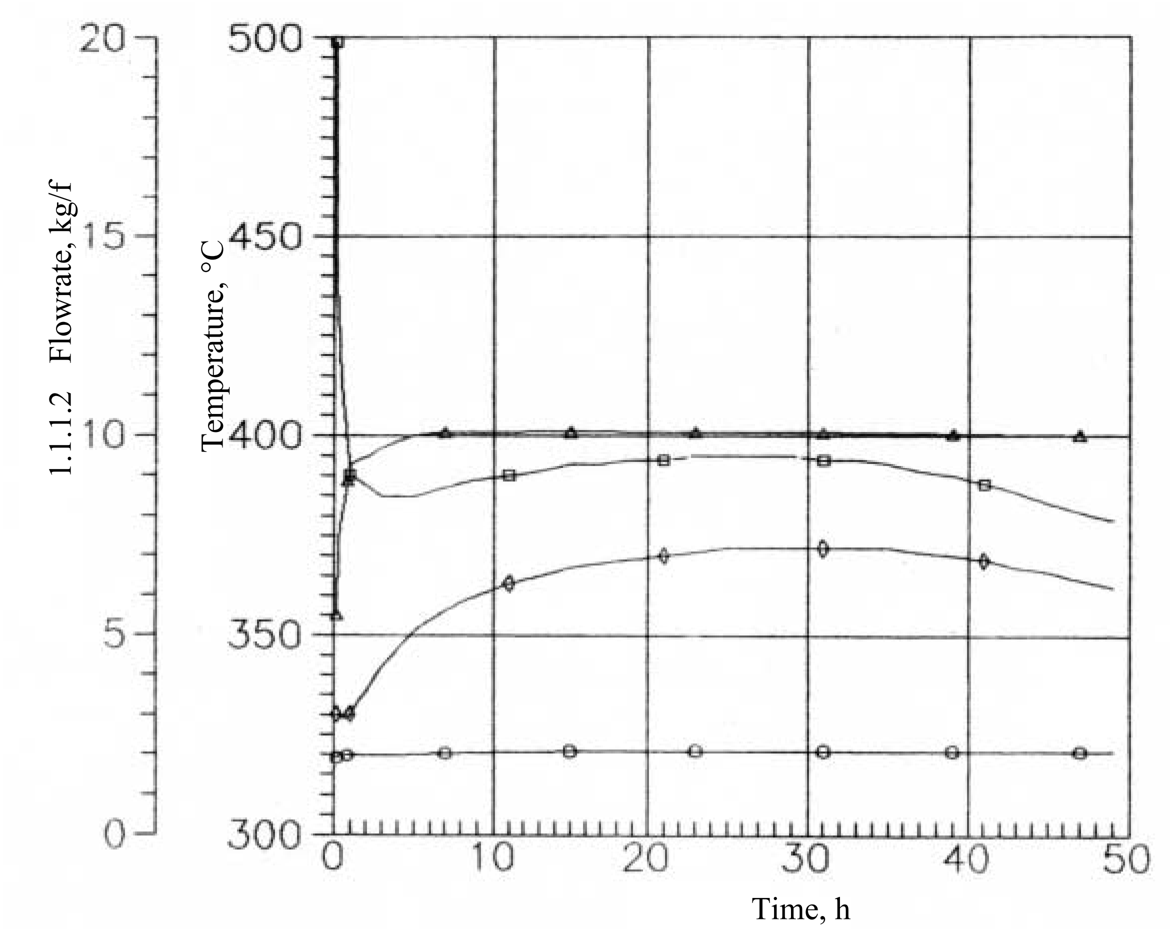


Fig. 2.5. A long core life.

Dependence of the reactor plant parameters on time
in the mode of residual heat removal using ERHRS
at the failure of two channels by air path



□ — sodium temperature at the core outlet;
◊ — sodium temperature at the core inlet;
△ — sodium flowrate by primary circuit;
○ — air flowrate by ERHRS circuit.

Fig. 2.6

BMN-170 REACTOR POWER UNIT

The nuclear power unit is based on 400 MW(th) fast nuclear reactor MBN-170 with sodium coolant.

Power units with BMN-170 reactors are universal power sources of the new generation for economically effective production of electricity, industrial steam of any parameters, fresh water and district heating as well (Fig. 3.1).

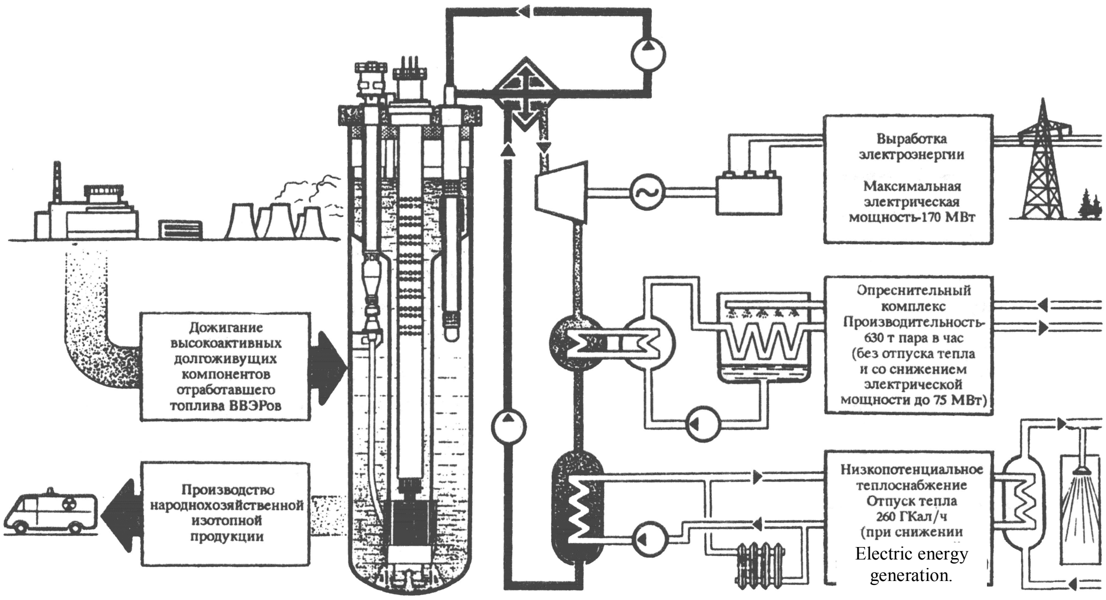


Fig. 3.1. BMN-170 reactor system.

Maximum electric power of 170 MW

Desalinating complex. Output – 630 tons of steam (without heat supply and with electric power decrease to 75 MW)

After-burning of highly active long living components of PER spent fuel

Low potential heat supply. Heat supply 260 Gcal/kg (at electric power decrease to 100 MW)

Fabrication of national economic isotopic products

The possibility of safe and reliable provision of the consumers with electricity and heat from power units with fast reactors has been confirmed by successful operation of NPPs with BN-350 and BN-600 reactors.For large NPPs of the new generation a 500 MW power unit is proposed which comprises three BMN-170 reactors and steam turbines of T-135/165 type.

Main technical data of the power unit

Electric power, MW	500
Gross efficiency, %	36
Lifetime, years	60
Construction time, years	5

A unit with one BMN-170 reactor can provide an industrial town of about 50000 population with electricity heat and potable water (using the desalinating plants of DOY-100 type).

Technical characteristics of T-135/165 turbine plant

Power of turbine plant, MW	
- nominal	135
- maximal	165
Initial steam parameters	
- pressure, MPa	14.0
- temperature, ^{o}C	500

The turbine permit operation with 10% overload.

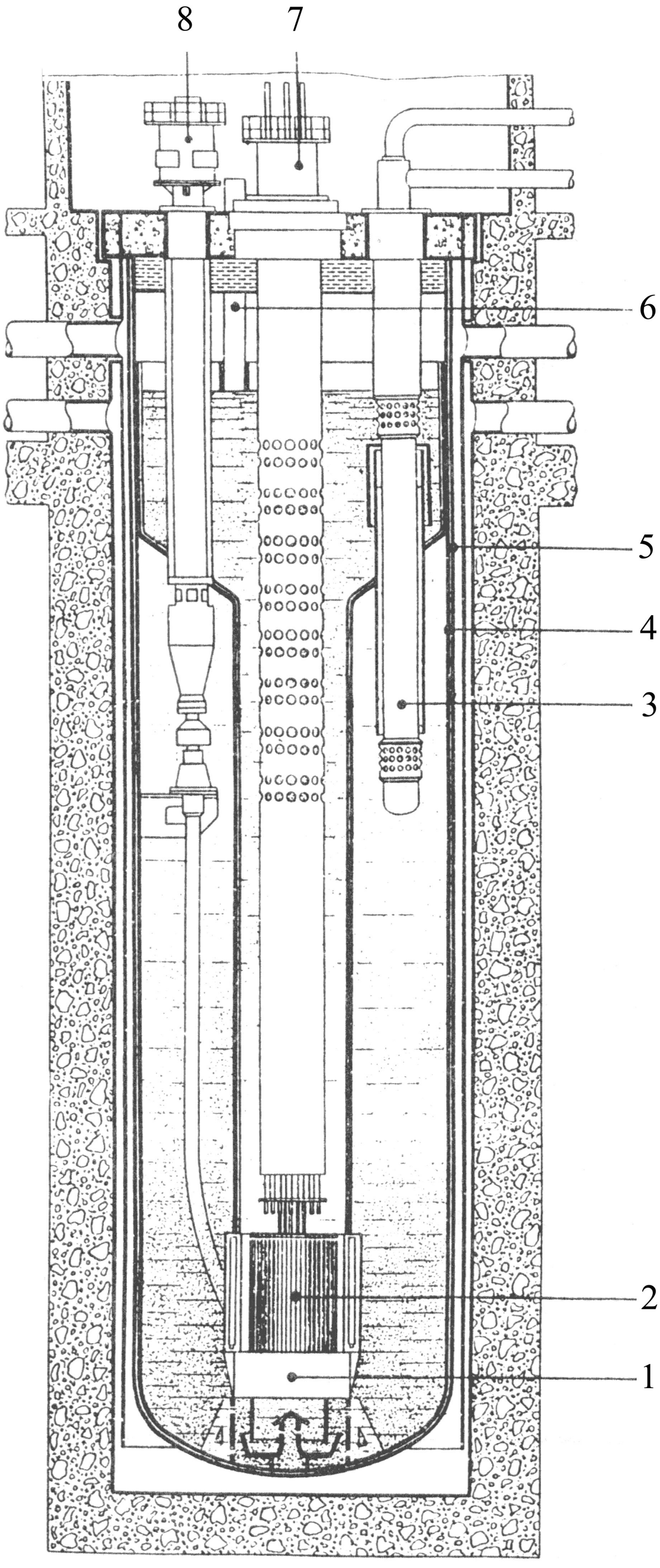
Engineering principles and decisions of 500 MW(e) power unit

The power unit consists of 3 modular unified reactors, which can be prefabricated and delivered to a NPP site in the assembled condition. The power unit includes the serially produced highly effective steam turbine plant and related machine room equipment unified for conventional heat power engineering. BMN-170 reactor plants have three-circuit flow scheme for heat transfer. An integral lay-out of the reactor (Fig. 3.2) with arrangement of all equipment and systems with radioactive sodium in a single vessel was adopted. This feature together with safety systems simplification, provides maximum compactness and safety of the reactor itself and the plant as a whole.

Enhanced stability of the power unit to extreme external impacts (aircraft crash, blast wave) is provided by location of the reactor and its safety systems in protected rooms of the reactor compartment below the grade. Enhanced seismic stability of the unit is provided owing to utilization of shock-absorbing supports for the reactor equipment and safety systems. The reactor plant remains operable at earthquake of magnitude 7 and safety is ensured at magnitude 8 earthquake (to MSK-64 scale).

Main design characteristics of the reactor plant

Thermal power, MW	400
Primary circuit sodium flowrate, tons/hr	7300
Primary circuit sodium temperature, ^{o}C	
- at core inlet	395
- at core outlet	550
Pressure in the reactors gas plenum, MPa	0.04
Secondary circuit sodium temperature, ^{o}C	
- at steam generator inlet	530
- at steam generator outlet	350
Sodium radioactivity, Bg/l	
- in primary circuit	$<3.7 \times 10^{11}$
- in secondary circuit	$\sim 3.7 \times 10^{4}$
Fuel type	PuO_2+UO_2
Fuel loading into the core, tons	4.65
Core lifetime, years	4
Time between refuelings, years	1
Nuclear fuel breeding ratio	1.1…1.3
Sodium void reactivity coefficient, % Δ k/k	-0.15
Specific metal mass of the reactor, t/MW(th)	1.68



1 – pressure chamber
2 – core
3 – intermediate heat exchanger
4 – vessel
5 – guard vessel
6 – filter-trap
7 – CPS column
8 – primary circuit pump

FIG. 3.2. Layout of BMN-170 components in a single vessel.

Safety characteristics

High reliability, stability in operation and radiation safety are the features which are intrinsic for fast nuclear reactors with sodium coolant (Fig. 3.3). There features are based on:

- reactor power self-regulation owing to stable negative feedbacks between physical processes and the core parameters;
- small reactivity margins in the core, absence of "poisoning" effects characteristic of the other reactors;
- high stability of neutron fields in the core, impossibility of forming local critical masses;
- low pressure of primary coolant in the reactor;
- excellent thermal-physical properties of sodium coolant (high thermal conductivity, large margin before boiling temperature under atmospheric pressure, large heat-accumulating capacity);
- absence of corrosion problems for structural materials operating in sodium coolant;
- ability of sodium to fix iodine radionuclides.

Accidents with large loss of coolant, which are the most dangerous for other reactors, operating under pressure, are practically excluded in the fast nuclear reactors. The probability of LOCA accident is assessed as 10^{10} per reactor year. The inherent safety features of fast reactors have been maximally developed and supplemented in BMN-170 reactor. The possibility of uncontrolled reactor power rising accompanied by coolant temperature increase (zero or negative void reactivity coefficient) is completely excluded. The reactivity margin for fuel burnout is reduced.

Features for reactor shut down and residual heat removal, based on passive principles of operation have been additionally introduced. They help to keep the reactor in a safe state, independently of support systems (power and compressed gas supply etc.) operability and personnel actions. For guaranteed shut down of the reactor additional rods are provided, which are inserted into the core under gravity in the case of sudden discontinuation of sodium coolant circulation or emergency increase of its temperature. This system provides the reactor shut down independently of the emergency protection automatic system actuation. Passive residual heat removal system provides heat removal from the core to ambient air at accidents with complete loss of feed water supply to steam generators. In this case residual heat is removed by forced air cooling of the reactor vessel due to chimney effect. This system is permanently operated and does not need to be actuated.

The principle of multi-barrier localization of radioactivity is realized in a full measure for BMN-170 reactor. All the equipment and sodium systems of primary circuit are located in the reactor vessel; external pipelines with radioactive sodium are excluded completely. The reactor is located in a leak-tight guard vessel, which is able to localize radioactive releases in case of sodium leak out from the reactor. The reactor plant is located in a concrete silo the walls of which are covered with leak-tight metal lining. Above the reactor roof the lining changes over into a shielding dome, which serves as an additional protection barrier on the path of radioactivity propagation.

A three circuit flow scheme of heat transfer from the reactor to working medium of steam-turbine cycle with pressure barriers between circuits has been used; it excludes radioactivity ingress into steam even in case of the reactor plant heat exchange equipment loss of tightness. Due to the BMN-170 reactor's physical properties and design decisions in emergency situations it has the ability for power self-limitation and self-shut down, for self-cooling during unlimited time, for localization and mitigation of radioactivity release within the passive protective barriers. Fast processes of spontaneous power rise, large coolant losses, overheating and melting of fuel are physically excluded in BMN-170. (Fig. 3.3). Probability of severe beyond-design accidents, development of which leads to the reactor core damage, is assessed in BMN-170 reactor as less than 10^{-6} per reactor/year, it meets the modern international requirements to safety level of the new generation reactors.

Radiation safety

The development of the facility project is oriented on meeting the following radiation safety criteria:

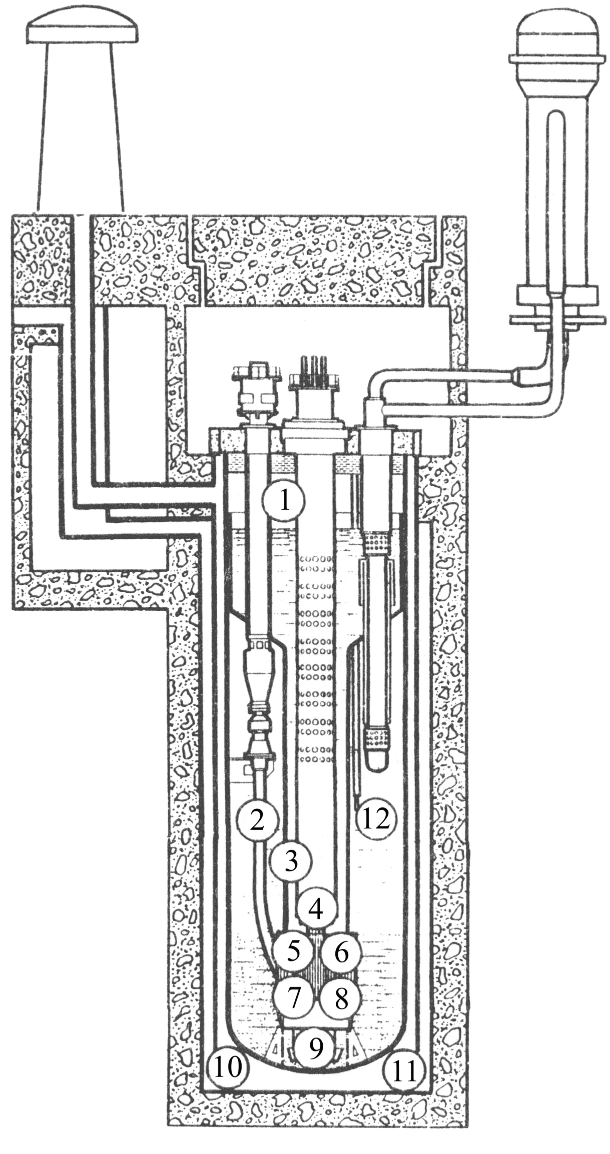
- radioactive releases during normal operation and corresponding equivalent individual radiation doses for population shall be not less than an order of magnitude lower than those established by the current norms in Russia (not more than 0.02 mSv/year for the whole body and 0.06 mSv/year for individual organs);
- values of equivalent individual radiation doses for population on the boundary of sanitary restricted zone and beyond its boundaries in design basis accidents shall not exceed the doses established for normal operation (0.2 mSv for whole body and 0.6 mSv for individual organs per accident;
- values of equivalent individual radiation doses for population on the boundary of sanitary restricted zone and beyond its boundaries in beyond design accidents shall not exceed 5 mSv for whole body and 50 mSv for individual organs for the first year after the accident.

The preliminary safety analysis of the facility confirms the criteria for design basis and beyond design basis accidents are met. It no accident in the BMN-170 reactor will population protection measures be required concerning radiological situation.

Economic indices of small power units

Power units with MBRU-1.5, MBRU-12, BMN-170 reactors have some positive properties, allowing to decrease their capital cost and to increase economical efficiency of power generation. They are:

- possibility of series fabrication of standardized modular reactors in a factory and their delivery in ready-made form to the site, decreases the labor content of installation works at the NPP site, construction time and expenses, including payment of rates for invested capital;
- no external refueling operations during the entire service life and considerable increase of the period between refuelings;
- utilization of self-protection principles and use of passive systems for residual heat removal from the reactor allow a decrease in the number of both auxiliary and safety systems and reduce the power unit equipment nomenclature;



1 – pressure – close to atmospheric
2 – high accumulating ability, high inertia of processes
3 – large margin before boiling (more 300^o C)
4 – reactor self-shutdown at temperature increase and flow-rate decrease
5 – high coefficient of fuel conversion, it provides minimum reactivity margin for burnout
6 – stability of neutron fields
7 – absence of the reactor poisoning
8 – negative values of thermal and power reactivity coefficients
9 – low corrosion activity of sodium relative to structural metals
10 – presence of guard vessel
11 – passive system of emergency residual heat removal by air
12 – radioactive debris confining by sodium and trapping by built-in traps.

Fig. 3.3. Inherent self-protection features of the reactor

- use of standardized machine room equipment, steam-turbine plant, condensate-feed water system, fabricated in accordance with the rules of conventional heat power engineering;
- high thermo-hydraulic efficiency;
- minimum thermal discharges to the environment and respectively minimum need in cooling water for condensers;
- possibility of different heat consumers connection to the power unit additionally increases its thermal and total economic efficiency;
- minimal number of operating personnel.

The possibility of reactor unit standardization simplifies the licensing procedure. For this type of the reactor the R&D program for its equipment development may be limited by the first reactor plant and for the following ones a standard license is to be supposed. Relatively small unit power of the plants allows to construct and put them into operation in sequence by line construction scheme for NPP of any power. Small power units are characterized by decreased investment risk and shorter times from a start of capital investment to finished product operation.

A LONG-LIFE SMALL REACTOR FOR DEVELOPING COUNTRIES, "LSPR"

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Abstract

There are many demands for small energy sources especially in developing countries, where the infrastructure is not sufficient and where it may be difficult to find sufficient technicians. In this case the preferred reactor should provide long-life, safe, simple, small, portable and proliferation-resistant characteristics. However, long-life and small-size are the two basic characteristics and the other characteristics can be derived from them. Both long-life and small-size require excellent neutron economy. The lead-bismuth-eutectic (LBE) cooled fast reactor shows excellent neutron economy for small reactors because of the large scattering cross section. The LBE-Cooled Long-Life Safe Simple Small Portable Proliferation-Resistant Reactor (LSPR) is proposed in this paper and the outline of its design and safety features are discussed. LSPR shows a stable response even at UTOP+ULOF+ULOHS accident, and can be restarted once the causes of the accidents have been removed, since there will be no damage anticipated in the fuel.

1. INTRODUCTION

Nuclear reactors will provide a considerable share of the electricity production of the world. These reactors benefit from large size to maximise economic competitiveness. However, large reactors can be constructed in only limited areas in developed countries. There are many demands and areas in the world, where large reactors do not fit. A typical example is the developing countries. There are many small energy demands in such areas, where small reactors are more suitable. Furthermore in this area, the infrastructure is often insufficient and it can be difficult to find the required number of technicians. In this case, not only small reactors but maintenance-free, long-life reactors are preferable. It is also possible that prospective users can not establish sufficient measures for proliferation resistance and physical protection. In this case proliferation resistance becomes a more important issue. In the present paper, a reactor is described, which shows long-life, safe, simple, small, portable and proliferation-resistant characteristics.

To realize this kind of reactor, a lead-bismuth-eutectic (LBE) cooled fast reactor is considered as the best candidate [1, 2]. It will be discussed in Chap. 2. In this chapter, the problems of LBE are also discussed. An actual design of the reactor is presented in Chap. 3. This reactor is called LSPR (**L**BE-Cooled **L**ong-**L**ife **S**afe **S**imple **S**mall **P**ortable **P**roliferation-**R**esistant **R**eactor). In the present paper the outline of the design and safety features of LSPR are discussed.

GUIDING PRINCIPLE FOR DESIGN

1.1. Long-life safe simple small portable proliferation-resistant reactor

We should consider all of long-life, safety, simplicity, small-size, portableness and proliferation-resistance. However, some of these characteristics are tightly related each other. For example, portability requires small-size. Portability is therefore considered as a similar characteristic to small-size. By investigating these characteristics from this point, it appears that only long-life and small-size are basic characteristics and the other characteristics can be derived from these two. Namely, a long-life small reactor can be transported to a site as a completed reactor and will be installed there. After operation for a certain period it can be replaced with a new one. A reactor on a barge is an alternative, which can be shipped to the site and operated as a power plant at a proper port. In these scenarios, highly skilful procedures such as fuel replacement are not required. The maintenance of the reactor becomes simple. Safety features are also improved. Small reactors contain only small amount of fuels and radioactive materials. Heat produced in an accident escapes easily from the core surface. The power shape is more stabilized. Especially for fast reactors, it is important that their void coefficients are shifted towards being more negative. From these discussions it appears that long-life and small-size are tightly related to safety and simplicity. In the above scenarios the reactor should always be closed during transportation and operation, and any fuels inside the reactor cannot be taken out. Therefore it can be said that the reactor is proliferation-resistant.

In the above discussion it appears that long-life and small-size are basic for realizing long-life safe simple small portable proliferation-resistant reactors. However, the latter items long-life and small-size are in conflict, because a small-sized reactor usually shows poor neutron economy, hence higher burn up can not be expected. The neutron economy, namely reactor criticality, limits both size and life of the reactor. Our discussion leads to the conclusion that the long-life safe simple small portable proliferation-resistant reactor requires excellent neutron economy. It is well known that fast reactors show much better neutron economy compared to thermal or epithermal reactors.

1.2. LBE-cooled small fast reactor

Following the above conclusion, we investigate small fast reactors. At present sodium is considered as the best coolant for fast reactors. The main reason is its superior cooling ability. It can make power density higher and doubling time shorter. The short doubling time was an indispensable requirement at early development and construction stages of fast breeder reactors in the 1960's and 1970's. It is reported that from a safety viewpoint the LBE was considered originally [3].

As previously mentioned the neutron economy is very important for realizing long-life small reactor. For the small fast reactor, it is expected that LBE coolant shows much better performance in terms of neutron economy than sodium coolant because of its large scattering cross section. It is reported that the LBE-cooled long-life small fast reactor shows better performance for neutron economy, burnup reactivity swing and void coefficient [1].

However, in the western world it has been considered for long time that LBE can not be used as a reactor coolant due to negative experimental results concerning the issue of corrosion. But in Russia this problem was solved by oxygen concentration control, and LBE was employed as submarine reactor coolant. It is reported that 8 nuclear submarines with LBE coolant were constructed and operated for about 80 reactor-years [3]. After the Russian research results became available openly, much research work especially for corrosion experiment were started worldwide. The corrosion problem is considered to be solved by choosing material, temperature, fluid velocity and oxygen concentration properly.

The most important merit of LBE compared to sodium is chemical inertness. LBE does not react violently with water or air.

The boiling temperature of sodium is 883℃, and it is not easy to protect against boiling in some severe accidents. If the void coefficient is positive, the accident may lead to a core destructive accident. The boiling temperature of LBE is 1670℃, and the possibility of boiling seems negligible. Furthermore, the void coefficient is more negative than sodium as mentioned before.

The density of LBE is about 12 times the density of sodium. The viscosity of LBE is large and the pressure drop is expected to be large. The Prandtl number is about 3 times that of sodium. For corrosion protection, the flow speed has to be set lower. These characteristics lead to a poor cooling ability of LBE. Then the power density of LBE-cooled reactor should be lower. However, for small power reactor power density usually restricted from the size of core determined by criticality conditions [4]. Therefore, the poor cooling ability of LBE is not so important for long-life small reactors.

For natural circulation capability, LBE-cooled reactors can offer better potential provided by larger core equivalent hydraulic diameter [5]. It improves the response of the reactor in accidents.

As mentioned before the LBE-cooled long-life small fast reactor shows better performance with respect to neutron economy, burnup reactivity swing and void coefficient, due to its large scattering cross section. The LBE shows also large shielding effects for neutrons and gamma-rays. Then the size of the reactor can be reduced.

The radioactive materials produced in the coolant during operation are also important. For sodium ^{24}Na should be considered. Its half-life is 15 hrs and emits high-energy gamma-rays (2.8MeV and 1.4MeV). Therefore the primary loop of sodium cooled reactor shows very high dose rate. On the other hand, LBE does not produce so much gamma-ray emitters, though Polonium is produced that is an alpha-ray emitter. Then the dose-rate around the primary loop of LBE is expected much lower than the sodium case.

2. LSPR

2.1. Plant concept

The LSPR plant conceptual plan and its main parameters are shown in Fig. 1 and Table 1 respectively. An integral type reactor design where steam generators are installed within a reactor vessel is employed, since severe reaction between the LBE reactor coolant and the steam generator water coolant is not anticipated. Nitride fuel, which has a high thermal conductivity, is chosen as the principal candidate because of its good adaptability with LBE coolant. Metal fuel, which has a potential to give improved performance is left for a future study concerned with the material issues

FIG. 1. Artist View of LSPR

TABLE I. MAIN DESIGN PARAMETERS

Item	Unit	Specification
Reactor Thermal Output	MWt	150
Reactor Electric Output	MWe	53
Reactor Outlet Temperature	℃	510
Reactor Inlet Temperature	℃	360
Reactor Vessel Diameter	m	5.2
Reactor Vessel Height	m	15.2
Core Barrel Diameter	m	3.4
Type of Steam Generator		Serpentine Tube
No. of Steam Generators	unit	2
Type of Pump		Centrifuge Pump
No. of Circulating Pump	unit	2
Total Pressure Drop	kg/cm2	0.7
Pb-Bi Coolant Flow Rate	ton/hr	12300
Pb-Bi Coolant Core Velocity	m/s	0.9
Feed Water Temperature	℃	210
Feed Water Flow Rate	ton/hr	2940
SG Outlet Steam Temperature	℃	280
SG Outlet Steam Pressure	MPa	6.47
Turbine Efficiency	%	35

Natural uranium or depleted uranium fuel assemblies are placed at the center of the core as an inner blanket, whereas plutonium fuel assemblies are situated outside of the inner blanket. In this core composition, the burn up of fuels will progress from the outer core into the inner blanket region, which is beneficial for sustaining reactivity through long-term burn up with a small reactivity swing. The reactivity change due to burn up is shown in Fig.2, where the curves are shown for the cases of lead coolant and also metal fuel respectively for comparison. For the reactor lifetime of 12 years, the expected reactivity swing is more or less than 0.1%, hence in any circumstance there will be no chance of exceeding prompt critical. In the long life core it is not easy to have high heat density but in this design 60 MW/m^3 can be achieved, which is reasonably acceptable compared with about $100MW/m^3$ averaged over the core and the blanket in typical fast reactors. Three control rods are placed within the core region. It is proposed and now under study not to use these control rods for power regulation but to use them only for the start up and for shut down of the reactor taking advantage of the very small excess reactivity. A newly proposed innovative burn up scheme named CANDLE (Constant Axial shape of Neutron flux, nuclide densities and power shape During Life of Energy producing reactor), where the burn-up region shifts along the central axis of core is under investigation for application in our laboratory. In this new burn-up scheme it is possible to have not only a higher power density but also a longer core lifetime, due to the fact that the excess reactivity can be set nearly zero.

The changes of coolant void coefficients are shown in Fig. 3, where the coefficients for the cases of lead coolant and metal fuel are shown for comparison respectively. In all cases the coolant void coefficients stay negative, whereas the coefficients appear to become positive if the coolant is changed to sodium. These facts clearly show that LBE cooled reactors have much safer characteristics than sodium cooled reactors for such an event with the coolant void reactivity insertion.

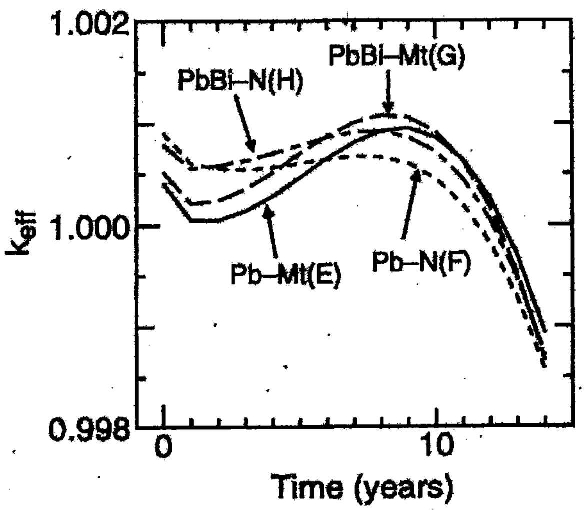


FIG. 2. Change of effective neutron multiplication factor with burn up

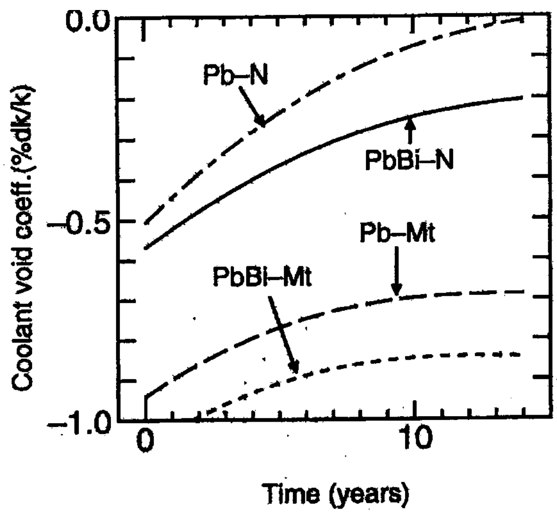


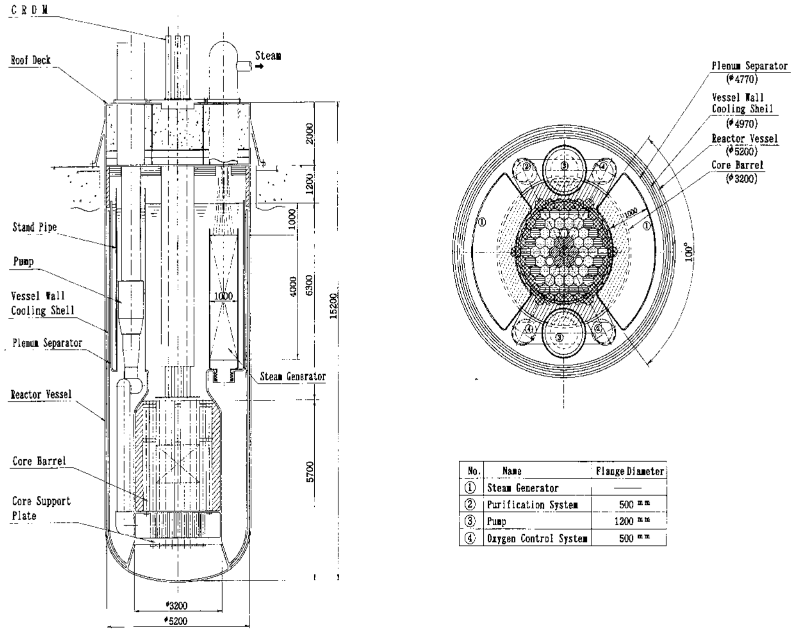
FIG. 3. Change of coolant void coefficient with burn up

Polonium produced by the neutron irradiation of bismuth will not be hazardous since the reactor vessel is closed and is not opened during regular fuel handling as mentioned above. If the reactor vessel is opened soon after the shut down, however, the high radioactivity is no doubt very hazardous, which is beneficial from the view of the proliferation resistances.

The concept of the reactor structure is shown in Fig. 4, where 2 units of steam generators, mechanical pumps, coolant purifying units and oxygen-concentration modulation units are respectively installed within a reactor vessel. Serpentine tube-type steam generators are employed because of their good space factors. The selection of the driving devices for the heavy metal coolant is one of the key issues in the reactor design. In this design mechanical centrifuge pumps are chosen from considerations of flexibility to the further study of the long life cores, possibly with higher pressure drop and of the versatility to be able to have adequate pump coast-down times. The natural circulation potential of the primary circuit is arranged to be from 30 to 40 % of the nominal primary flow at the nominal heat balance level.

A Steam Generator Auxiliary Heat Removable System (SGAHRS) is adopted, where the decay heat is removed by natural circulation through steam generators to air coolers, without having DRACS or PRACS in the reactor vessel. Instead, the RVACS is installed as a backup system to SGAHRS. In order to control the thermal conductivity and to enhance the function of RVACS, the reactor wall is designed to stay at the cold leg temperature at the power operation, but to rise to the hot leg temperature by the coolant overflow at the accident condition as shown in Fig. 4.

The system flow diagram is shown in Fig. 5, where the re-circulation in the secondary system with the free surface in the water drum at the power operation is proposed from the view of passive heat removal by natural circulation through steam generators. The heat balance shown in Table 2 indicates the turbine efficiencies of 35%, which seems to be an outcome of a design with enhanced safety and high operability.



No.	Name	Flange Diameter
①	Steam Generator	——
②	Purification System	500 mm
③	Pump	1200 mm
④	Oxygen Control System	500 mm

(a) vertical cross section (b) horizontal cross section

FIG. 4. LSPR Reactor Vessel

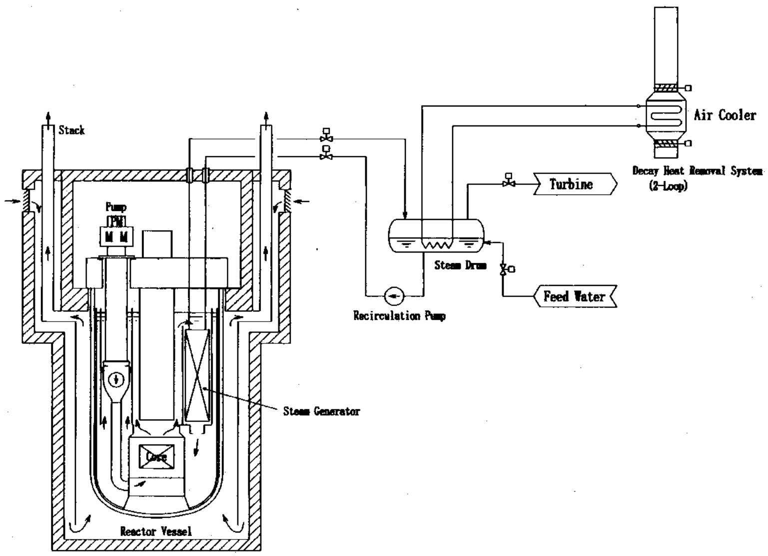


FIG.5. LSPR System Flow diagram

The plant layout of LSPR is shown in Fig.6. The conventional fuel handling system is not prepared from the view of long life core, however, the maintenance handling machines and maintenance spaces are accommodated, taking the pullout space of mechanical pump impellers and purifying units into consideration. In case the reactor vessel lifetime expires, it is planned to

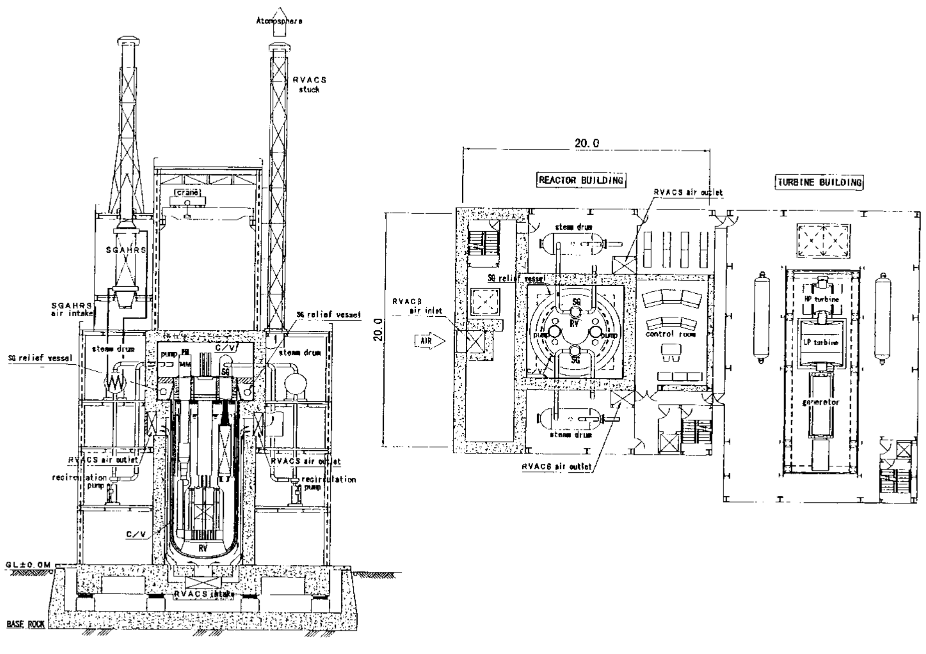


FIG. 6. LSPR Plant Layout

One of the main advantages of LBE cooled reactors to be able to simplify drastically the reactor engineering safety features accommodation compared with the other small reactors, by mitigating the effect of reactor coolant leakage accident effectively with the help of a simple guard vessel.

2.2. Passive safety

As a representative incident of an anticipated transient, loss of external electricity is commonly postulated, in which a diesel generator (DG) is expected to start up and to supply electricity to safety class demands in the conventional design. On the other hand, however, in the present study it is aimed to comply with passive safety demand of the decay heat removal system without the help of DGs. Ambient decay heat can be removed in the present design by SGAHRS through the steam generators to the air coolers by virtue of natural circulation for the anticipated transient of loss of external electricity.

The transient of power (TOP) due to a control rod withdrawal, the loss of primary flow (LOF) and the loss of heat sink (LOHS) due to the loss of heat removal capability of secondary system are widely postulated as accident events. Even though the loss of external electricity is commonly superposed on these events, this does not lead to any serious problems, as the reactor is safely tripped.

Severe accidents, where the failure of scram is superposed on the above-mentioned accidents, are surveyed hereafter. In UTOP (Uncontrolled TOP) the power excursion comes down to a stable state without scram by virtue of a negative feedback coefficient, since the maximum insertion reactivity in this study is only 0.25$ due to very low burn up reactivity swing. ULOF (Uncontrolled LOF), where all primary pumps are postulated to stall without scram, is shown to terminate with ambient margin. By virtue of the natural circulation, the decay heat removal system has been shown to have enough capability, on the condition that the coast-down half time of the primary pump is set to about 12 sec and the natural circulation of the primary flow is kept around 30% of the nominal flow level. ULOHS (Uncontrolled LOHS) terminates also without problem as far as SGAHRS holds the safety passive function. RVACS, which removes the excess heat through the reactor wall to chimney, serves as an inherent passive system to backup the function of SGAHRS in the accident condition.

If the above mentioned three severe accidents conditions are hypothetically superposed, which is the case not necessarily to be considered, this reactor shows a stable response as shown in Fig. 7. It seems to be possible to restart the reactor, even in these circumstances if the causes of the accidents can be removed, since there will be no damage anticipated on the fuel.

The tube rupture accident of steam generators, which brings water in steam generator tubes into the LBE coolant, is one of the most serious events. The high-pressure steam ejected in the reactor coolant is relieved to a steam relief tank situated above the reactor vessel through relief valves, which keeps the pressure in the reactor vessel below a certain level. The impact on the hydrodynamic behavior of the primary coolant and the behavior of steam bubbles due to a steam generator tube rupture as well as the possibility of production of the oxides of lead and bismuth have yet to be fully investigated.

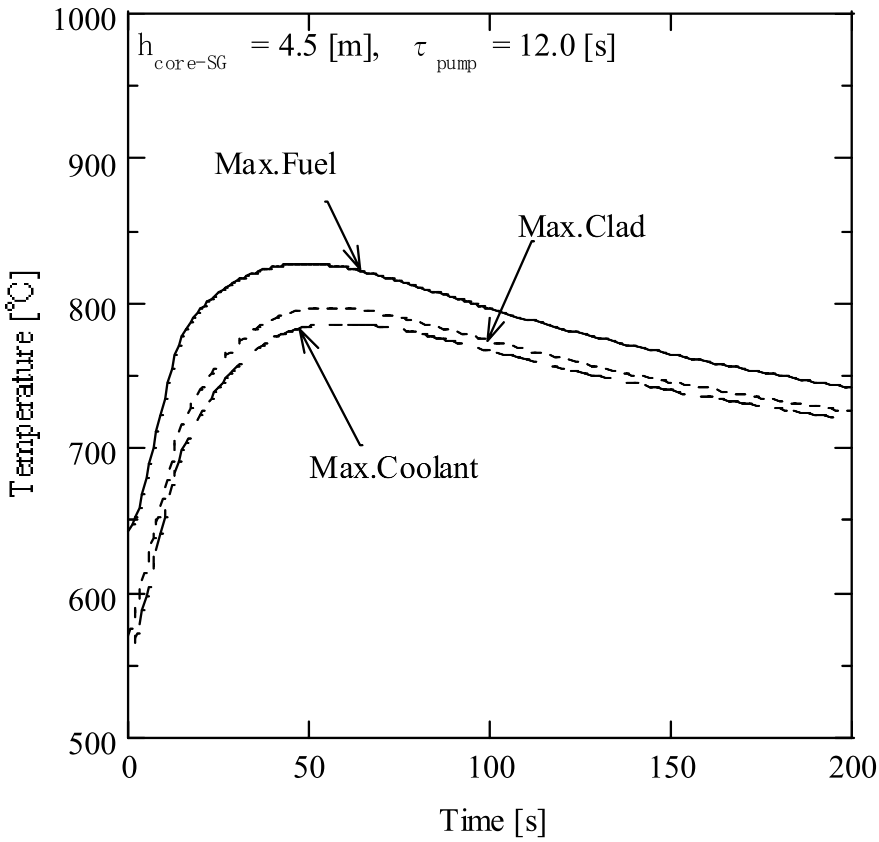


FIG. 7. Change of maximum temperatures of fuel, cladding and coolant during TOP+ULOF+ULOHS accident

2.3. Future issues

The present design study has been carried out conservatively, where the primary coolant velocity is set low compared with the other designs, considering the ambiguity of the material corrosion data against LBE coolant. The height of the reactor vessel is chosen to be high enough to give a sufficient natural circulation head with ambient margins.
It seems to be an option to mount a reactor compound on a barge, which might make the installation and dismantling easier at the site.

Alternatives to the present centrifuge mechanical pumps should be further investigated in future, such as the natural circulation without pumps, as well as the application of lift up pumps, that brings in gas bubbles in the coolant to increase the buoyancy force. In this design, however, it is emphasized to show a simple and feasible reactor accommodated with the requested functions with only the use of conventional and reliable devices.

In a future study the application of CANDLE burn up in the core design, the simplification of a passive decay heat removal system, the application of suitable counter measures against a steam generator tube rupture, as well as simplified maintenance techniques for in-vessel devices are the primary concerns in relation to the further improvements of safety and economy.

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NUCLEAR POWER COMPLEX BASED ON SVBR-75/100 SMALL REACTORS COOLED BY LEAD-BISMUTH LIQUID METAL COOLANT, COMPETITIVENESS, SIMPLIFIED LIFE CYCLE, SAFETY, NON-PROLIFERATION

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Abstract

An opportunity and expediency to use a small power fast reactor cooled by lead-bismuth coolant for a nuclear power complex producing electric energy, heat and fresh water have been considered in the report. Option for this type reactor has been substantiated. The results of experience of operating lead-bismuth cooled reactor installations in nuclear submarines have been considered. The technical solutions and characteristics of the reactor installation SVBR-75/100 have been investigated. The concept and parameters of the nuclear power complex for developing countries are presented.

1. INTRODUCTION

The real prospects of multi-purpose nuclear power sources of small and medium power and their introduction into the power infrastructure of different countries require substantiated solutions to two global problems: economic effectiveness and safety of these power sources. A small power nuclear power complex (NPC) can be a universal source of power supply all over the world. Here power supply is understood as the production of electricity and heat and seawater desalination using heat or electricity.

It is considered that optimal power of the NPC unit is 50…150 MWe. These units to the most extent are characterised by passive safety properties that reduce the risk of serious accidents. Moreover, operation and maintenance of these units is simpler.

The use of new nuclear technology can eliminate the contradiction between economics and safety requirements that is peculiar to traditional reactors. This technology is based on using lead-bismuth alloy - heavy liquid metal coolant (HLMC) - for reactor cooling. The technology of using lead-bismuth coolant (LBC) has been developed in Russia for nuclear submarine (NS) reactors under the scientific leadership of the Institute for Physics and Power Engineering [1]. Today there are real prospects for introducing this technology into civilian nuclear power (NP).

Interest in this technology is conditioned by the fact that the natural properties of the coolant – high boiling point and chemical inertness – allow the design of a safe reactor installation (RI) operating under low coolant pressure and with high safety level. It excludes development of any events (not only single personnel mistakes and equipment faults, but multiple superposition of these causes, acts of terrorism and malevolent activity) into severe accidents, which can be accompanied by explosions, fires, inadmissible exhausts of radioactivity and require population evacuation beyond the NPC site.

2. SUBSTANTIATION OF OPTION FOR LBC COOLED FAST REACTOR

Fast reactors (FR) cooled by liquid metal coolants (LMC) are classified as RIs in which safety is ensured mainly due to their inherent safety properties. This is associated with a number of their internal features:

The lack of poisoning effects in the FR, small negative values of the temperature reactivity coefficient, compensation of fuel burn-up processes by plutonium generation ensure the operating reactivity margin to be less than the delayed neutron share and eliminate the prompt neutron runaways in the operating reactor.

The high boiling point and latent evaporation heat, which are LBC natural properties, practically eliminate the possibility of primary circuit overpressurization or reactor thermal explosion during any conceivable accident as the pressure does not increase. The impossibility of coolant boiling enhances the reliability of heat removal from the core and safety due to lack of the heat removal crisis phenomenon.

In an event of accident overheating and simultaneous postulated failure of emergency protection systems (EPS), the reactor power decreases down to a level that does not cause dangerous overheating of the core is ensured by reactivity negative feedbacks. Additionally, the release of gas fissile products in great amounts is eliminated.

The coolant itself reacts with water and air only very slightly. The development of accident processes caused by leak-tightness failure of the primary circuit and steam generator (SG) inter-circuit leaks occur without hydrogen release and any exothermic reactions. Within the core and RI there are no materials which release hydrogen as a result 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 coolant's chemical inertness, the impossibility of its boiling in an event of the primary circuit leak, its property to retain iodine (the radionuclide that, as a rule, represent the major factor of radiation risk just after an accident), as well as the other fission products (inert gases are an exception) and actinides, sharply reduce the scale of radiation consequences of such an accident.

As computations reveal, extremely high safety potential peculiar to this type of RIs is caused by the fact that even if such initial events as containment destruction and primary circuit leak coincide (that is possible in an event of diversion or military attack), neither reactor runaway, nor explosion or fire occur; and the radioactivity release is lower than that, which requires population evacuation.

With due account that energy stored in the coolant (heating, chemical and compression potential energy) is minimal compared with other coolants, and taking into account previously mentioned physical features of fast reactors and RI integral design, one could look forward to designing the RI of an extremely high safety level. On the base of those reactors, NP would become not only socially acceptable for population but also socially attractive if it gained economical competitiveness with heat electric power plants using organic fuel (we have all the backgrounds for it).

A significant improvement of the economic parameters is caused by the inherent safety of RI SVBR-75 that allows the elimination of a great number of safety systems, localizing, protective, control and maintenance systems peculiar to the traditional types of RIs.

3. RESULTS OF EXPERIENCE OF OPERATING LBC COOLED REACTOR INSTALLATIONS

In the early 1950s, at almost the same time, the USA and the USSR launched their development programs on the NSs' RIs. Both countries developed two types of RIs: pressurized water reactors and LMC cooled reactors. In the USSR work was begun in 1952; lead-bismuth eutectic alloy was chosen as LMC.

In the USA sodium was chosen as LMC because of it possessed better thermo-physical characteristics. The ground-based RI prototype test facility and experimental NS "Sea Wolf" were constructed. However, operational experience revealed that this option for the coolant, which is a fire and explosion hazard in contact with air and water, had not proven itself. After several RI accidents occurred at this NS, it was decommissioned together with the compartment and replaced by a pressurized water RI.

R&D work on mastering LBC were also carried out in the USA. However, the chosen approach to solve the problem of structural material corrosion resistance, control and maintenance of the coolant quality (coolant technology) did not give any positive results, and the work was stopped.

In our country failures could not be prevented either. The history of engineering has revealed that, unfortunately, this is inevitable for the process of mastering any new technology. In 1968 at the initial period of operating the experimental NS of Project 645, an accident with partial melting the core occurred [2]. This accident was caused by use of LBC. As a result of poor knowledge of physical and chemical coolant processes in the operating reactor, there were no justified specifications on the coolant's impurity composition, instrumentation of controlling the coolant quality, and equipment for maintaining the required parameters of the coolant quality during operation. Those days information on LBC could be compared with that on water coolant when it was considered permissible to supply the steam boilers with the water taken from a running water system.

After this accident work on the coolant technology problem was launched. For many years, certain organizations were carrying out this work under the scientific supervision of IPPE. As a result, the problems were solved successfully, and it was confirmed further by many years of experience of RIs operating at the NSs. When operating the second generation RIs, there were no problems caused by structural material corrosion in the primary circuit and by violating the circuit purity standards.

Among the positive properties of LBC cooled RIs, which have been revealed during reactor operation, the following should be highlighted: simplicity of control, high manoeuvreability and short time required for reactor transition from a subcritical state to a power mode, capability of quick changing the coolant circulation mode under considerable change of its flow rate, almost total expire of the designed power lifetime by the operating core under normal and maximum permissible leak-tightness states of the fuel element cladding. When repair work and reactor refueling are performed, there is no need to carry out decontamination of the primary circuit which is associated with collecting, storing, transportation and reprocessing of masses of liquid radioactive waste (LRW).

The operating experience of LBC cooled RIs at the NS has demonstrated the possibility of safe RI operation for a certain time period, under conditions of small SG leak that do not cause any significant deviations from design technical parameters. Due to this fact, necessary repair work would not be urgently required, but can be carried out at a convenient time.

The measures developed for radiation safety make it possible to eliminate personnel irradiation by polonium beyond permissible limits even in the event of an accident in which LBC leaks into the NS compartment and further maintenance is required.

Substantiation of the possibility of multiple "freezing-unfreezing" of LBC cooled RIs was an important practical problem, since this procedure might be necessary in case of long NS outage. During LBC transformation from the solid state to the liquid one, RI damage can be prevented by a slight increase of the LBC volume during its melting and its rather high plasticity under low strength in a solid state. Special requirements to the temperature-time heating mode were developed for carrying out safe "unfreezing" the RI.

Experience of development and operation of LBC cooled RIs as parts of NSs and ground-based facilities-prototypes has made it possible to draw practically important conclusions on the arrangement and design of the primary circuit components for LBC cooled reactors of the nuclear power plants (NPP) [2].

The best technical and economical parameters should be expected for the integral (monoblock) arrangement of the primary circuit equipment. The most convenient design of the SG is one in which LBC circulates in the inter-pipe gap and water or steam circulates in the pipes. This design ensures that the SG can be repaired by plugging the leaking pipe without SG dismantling or opening the primary circuit.

4. MULTIPURPOSE REACTOR MODULE SVBR-75/100

RI SVBR-75/100 is designed for generating steam which parameters enables its use as a working medium in a thermo-dynamical cycle of turbo-generator installations. It is possible to vary the steam parameters in compliance with the specific needs. The base variant of RI SVBR-75 [3] was developed for generating saturated steam, i.e. the pressure, which is produced by the SG of the 2-nd unit of the Novovoronezh NPP. When turbo-generator with intermediate steam superheating is used, this allows the generation of electric power of ~ 75 MWe under condensation regime operation.

The design of RI SVBR-75 module has a two-circuit scheme for LBC heat removal in the primary circuit and steam-water in the secondary circuit. The integral design of the pool type is used for the RI primary circuit (see Fig. 1). The primary circuit equipment is installed inside the one vessel enclosed into the safe-guard vessel. RI SVBR-75 includes a removable unit with a core and drivers of control and protection rods (the reactor itself), 12 SG modules with multiple natural circulation over the secondary circuit, 2 main circulation pumps (MCP) for circulating LBC over the primary circuit, devices for controlling the LBC quality, an in-vessel radiation shield and a buffer chamber, which are parts of the main circulation circuit (MCC).

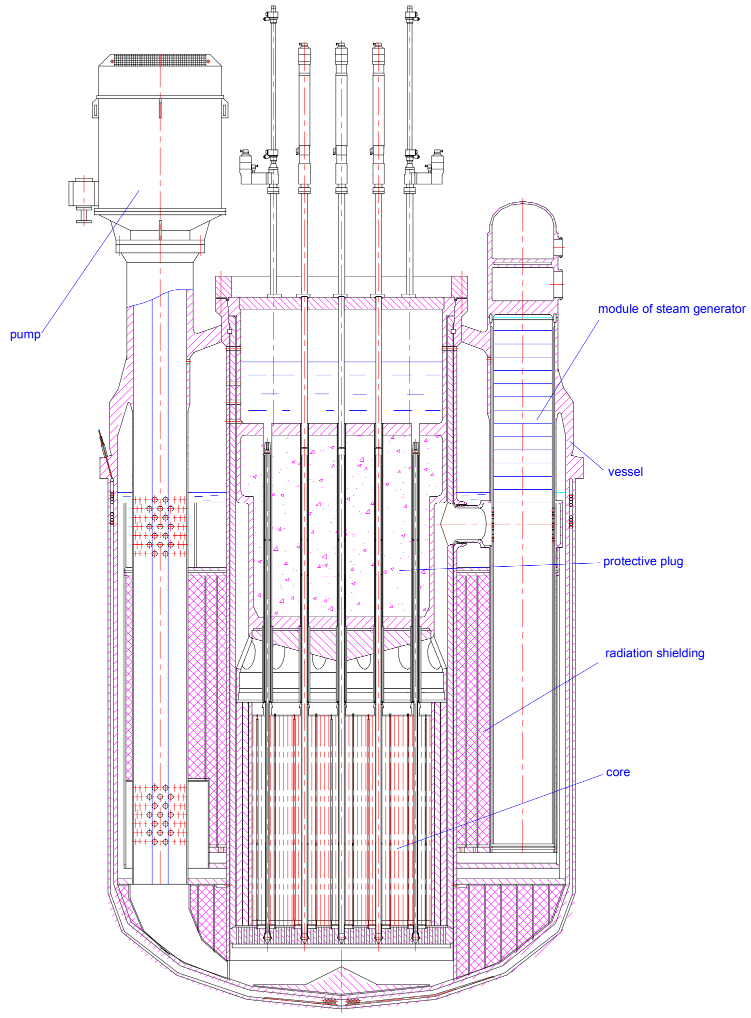

FIG. 1. Monoblock General View

The scheme of coolant circulation within the MCC is as follows: through the windows of the reactor outlet chamber the coolant heated in the core flows to the inlet of the SG twelve modules which have parallel connection. It flows from top to bottom in the intertube space of the SG modules and is cooled there. Then the coolant penetrates into the intermediate chamber, from which it moves in the channels of the in-vessel radiation shielding, cooling it, to the reactor vessel upper part and there it forms the free level of «cold» coolant (peripheral buffer chamber), further from the reactor vessel upper part the coolant flow moves to the MCP suction inlet.

The adopted circulation scheme with free levels of LBC that exist in the upper part of the monoblock and SG module channels, which are in contact with the cover gas, ensure reliable separation of steam-water mixture from the coolant in an event of accident leak of SG tube system, and existence of cover gas ensures the possibility of coolant's temperature changes.

A monoblock is placed in the pool and is mounted there. The pool is filled with water and is designed for cooling the RI in an event of beyond design accidents. The gap between the monoblock's main vessel and safe-guard vessel is designed to ensure the continuity of the circulation circuit in an event of accidents associated with leak of the monoblock's main vessel. The gap also provides the necessary heat insulation.

The design scheme of the secondary system provides operation of the SG that generates saturated steam with multiple natural circulation through the evaporator-separator, as well as scheduled and emergency cooling the RI by using the SG.

The design provides three systems of heat removal to the heat sink both in the case of scheduled and emergency cooling the reactor core. The first cooling system includes the equipment and systems of normal operating of the RI and the steam-turbine installation (STI). Cooling is realized by removing heat from the primary circuit via SG exchange surfaces, steam is dumped to the STI systems.

The second system of heat removal is an autonomous cooling system (ACS), which, besides a part of primary and secondary circuit equipment, includes a separator-cooling condenser (CC) circuit with natural circulation. Via this circuit heat is removed to the intermediate circuit water. This system ensures autonomous (independent on the STI systems) reactor cooling and autonomous operation of the reactor at a power level being ~ 6 % of N_{nom} under maintaining the required steam pressure. Connection/disconnection of ACS is realized without any operator action and without using any external power supply systems.

The third system of removing heat from the reactor core is a passive heat removal system (PHRS). Heat is removed from the monoblock's vessel to the water pool in which the monoblock is installed. This system provides cooling for the reactor core in an event of a postulated maximal accident associated with failure of all secondary circuit equipment, failure of the reactor protection system and total blacking out of the NPP. A non-interference period is determined by the time of boiling off water in the pool. This time equals to several hours.

The EPS consists of the two subsystems for bringing the reactor to the subcritical state:

The emergency protection rods subsystem. Their operating mechanisms drop the rods in case of de-energizing the electromagnetic clutches in response to EP signals. EP rods are designed with fusible seals that allow for dropping EP rods into the core when LBC temperature exceeds safe operation level in case of mechanical damage of the operation mechanisms;

The subsystem of the operating group of rods for reactivity compensation. Their operating mechanisms have springs that provide for drop of the rods in case of de-energizing the electromagnetic clutches in response to the EP signal.

The principal technical parameters of RI SVBR-75 are summarized in Table 1.

RI SVBR-75 operates for eight years without core refueling. During this period there is no need to carry out any fuel work. At the initial stage mastered oxide uranium fuel should be used. Further it will be possible to use MOX fuel. In this case, the core breeding ratio (CBR) will slightly exceed 1 and the reactor will operate in the plutonium closed fuel cycle by using only depleted waste pile uranium.

The eight-year fuel lifetime has been adopted for the first RIs with due account of the available experimental data validating operation ability of the materials in these conditions. Further, when operation experience is gained, the fuel lifetime will be increased to 10 - 12 years.

TABLE I. PRINCIPAL PARAMETERS OF RI SVBR-75

Parameter	Value
Number of reactors	1
Rated heat power, MW	268
Electric power, MWe	75*
Steam production rate t/h	About 487
Steam parameters:	
-Pressure, MPa	3,24
-Temperature, ºC	238
Feed water temperature, ºC	192
Primary coolant flow rate, kg/s	11180
Primary coolant temperature, ºC	
-core outlet	439
-core inlet	275
Coolant's velocity in the core	1,85
Core dimensions, DxH, m	1,65x0,9
Average value of specific volume core power, kW/dm^3	135
Average value of specific linear core power, kW/m	~ 22
Fuel:-type	UO_2
-U-235 mass loading, kg	1476
-Average U-235 enrichment, %	15,6
Lifetime reactivity change, %	-3,6
Power reactivity effect from T=200ºC to $N_{nom.}$, % of $\Delta k_{eff.}$	-0,85
Void reactivity effect at total drying the reactor, % of $\Delta k_{eff.}$	-2,1
Delayed neutrons effective share - $\beta_{eff.}$, % (BOL/EOL)	0,72 / 0,60
The number of compensating rods	37
The number of safety rods	6
Burn-up depth average value, % of h.a.	~6,0
SG numbers	2
Evaporator numbers in SG	6
Evaporator dimensions DxH, m	~ 0,6x4
Numbers of MCPs	2
MCP electric driver power, kW	400
MCP head, MPa	~ 0,5
Primary circuit coolant volume, m^3	18
Major reactor vessel dimensions, DxH, m	4,53 × 6,92
Designed earthquake	Of magnitude 9 (MSK)
Designed construction terms (months)	36

* Low thermodynamical efficiency (28 %) is caused by using the existing turbines of the NVNPP second unit.

Design and operation experience of NS's RIs was used to the maximum extent possible when RI SVBR-75 was designed. The total operation time for these reactors (along with the ground reactors-prototypes) is 80 reactor-years. When designing the RI, all the prior accidents that had occurred were taken into account and design faults of the RI were eliminated [2].

The closeness of the scale factor of RI SVBR-75 to that of NS's RIs, makes it possible to use practically developed technical solutions from that system and reduce the scope of R&D.

RI SVBR-75/100 [4] is designed on the RI SVBR-75 construction base and only distinguishes from it by the SG generating saturated steam of 9.5 MPa or superheated steam of 400 ˚C. This provides the electric power of ~ 100 MWe.

5. BASED ON RI SVBR-75/100 DUAL-PURPOSE NUCLEAR DESALINATING POWER COMPLEX FOR DEVELOPING COUNTRIES

Lots of developing countries in Africa and Asia suffer from a deficiency of fresh water and electric energy. The majority of these countries do not have sufficient of their own resources of fossil fuel to meet their demands. In some countries fuel transportation is difficult and there are no powerful electric power transmission lines. Market research conducted recently by IAEA has revealed that for these purposes small sized nuclear power sources of 100 MWe can be used economically effectively.

However, developing countries' particularities associated with the rather low level of education, technical culture, social and economical development, as well as local military conflicts that might arise, put forward special requirements to nuclear power technology, which are stricter than the requirements for nuclear power technology in the developed countries.

First of all, these requirements include inherent safety of the RI against severe accidents, that is based on RI properties ensuring safety not only in cases of personnel's errors, multiple failures of technical systems, and their coincidences, but in cases of sabotage, terrorist actions, etc. Besides, they must meet strict non-proliferation requirements.

Refueling in the User-Country must be eliminated and, therefore, the lifetime duration must be 10 years or more. Nuclear and radiation safe transportation of the reactor unit for refueling to the Manufacturer-Country must be provided. Radiation safe transportation of the reactor unit for refueling back to the User-Country must be provided. Thefts of fuel must be technically eliminated. Besides, competitiveness with the alternative sources of fresh water and electric energy must be ensured.

RI SVBR-75/100 meets these requirements the most completely. It has an extremely high safety potential, required duration of the lifetime, it ensures non-proliferation due to:

Use of uranium enriched less than 20 %;

Lack of refueling in the User-Country;

Opportunity of nuclear and radiation safe transportation of the reactor module after ending its lifetime to the Manufacturer-Country with LBC «frozen» in the reactor.

It is proposed to use the most effective desalinating technologies for seawater desalination. These technologies are based on different principles and use different kinds of energy. They are multi-stage evaporation (MED) and reverse osmosis (RO). It is proposed that at the NPC power capacity of each type of installation is 50 %. Also there are reserve power capacities for keeping the production rate of the NPC in the case of scheduled maintenance.

TABLE II. BASIC PARAMETERS OF THE NPS.

Reactor type	LBC cooled fast reactor (SVBR-75)
Core lifetime, thousand effective hours	70 (for the first lifetime) 100 (for the next lifetimes)
Operation time between refuelings, years	10...15
Steam capacity, t/h	460
Fresh steam pressure, MPa	4.6
Fresh steam temperature, °C	saturated
Parameters of turbine steam bleeding after high-pressure cylinder pressure, MPa temperature °C	 0.27 130
Turbo-generator power, MWe	75
The output of desalinated water, m^3/d	80 000
Method of seawater desalination	MED – 50% RO – 50%
Floating module displacement, t	1500...2000
The floating module overall dimensions, diameter × length, m	13 × 20
The protective dry dock overall dimensions, Length × Width × Height, m	40 × 25 × 25
Designed earthquake	of magnitude 9 (MSK)
Designed terms of construction, months	36
Approximate rent for the reactor module, $M/year	6.5
RI approximate specific operation and service cost, $/MWt*hour	8.1
Approximate cost of electricity (including RI rent, turbine-generator set amortization cost, cost of RI and turbine-generator set operation and service), $/MWt*hour	25
Approximate contribution of RI to the cost of fresh water (RI rent, electric power according prime cost for RO - ~6 кWt/m^3 – turbine-generator set amortization, RI and turbine-generator set operation and service), $/$m^3$	0.24

As desalination installations are not viable for RI safety, domestic industry and manpower should be used to the maximum extent possible for their construction. The existence of auxiliary boiler-house and power grid make it possible to operate these installations just after their construction prior to RI readiness.

This will enhance reliability of the complex in case of shutting down the reactor unit. Depending on the available resources, the RO installation will be able to operate due to electric power supply from the external power grid or MED will be able to operate due to incinerating fossil fuel in the reserve boiler-house.

A shore-mounted NPC design (in comparison with a barge-mounted one) simplifies the reactor module protection against external impacts (like gales or tsunamis) and possible acts of terrorism (fighting scuba divers). Also it simplifies NPC operation (excluding the expenses for providing barge floatage).

Reactor units are delivered as “Build-Own-Lease”. This means that the supplier leases the reactor unit for the time period determined by the reactor core lifetime duration (~10 years). Such a core lifetime will make it possible to keep stable costs for the NPC products (potable water and electricity).

Reactor installation is installed in the gas-tight and durable compartment of the floating unit with 1500 – 2000 tons of displacement. This unit includes all the systems necessary for safe operation of the reactor under design scenarios and passive safety systems for overcoming possible accidental situations. The compartment is separated from the environment; there is no discharge of contaminants from the reactor system.

After being manufactured at the plant, the floating unit with the freshly refueled reactor is towed to the nuclear desalination power complex and installed by sluicing in the closed dry dock (see Fig. 2), which protects the reactor unit against objects falling on it, as well as other design external events.

On-shore operation with radioactive materials including reactor refueling is not performed. For that reason, requirements for maintenance personnel are reduced and the risk of plutonium proliferation is reduced. In the event of an accident, radioactive products are kept in the RI compartment and on-site radioactive contamination does not occur; no decontamination operations need to be performed. At the end of its lifetime, the reactor unit is sluiced to the cooling compartment protected against external effects. It stays there for about a year until the level of residual heat release decreases and LBC solidifies (melting point ~ 125 °C) and forms a solid protective cladding. It is replaced by another reactor. After cooling, the supplier transports the unit with solidified coolant to the plant-manufacturer for refueling, necessary repair works, and renewal of expired equipment.

When withdrawing from operation, the reactor unit will be towed to the Manufacturer-Country after the necessary cooling time. After this removal no radioactive waste will be left on the NPC site.

The proposed NPC concept makes it possible to decrease the investment risk to a level typical of non-nuclear projects.

It is presumed that all control operations from the moment of initial startup to the final shutdown of the reactor will be automated to the maximum extent possible. When designing the NPC control system, experience of designing high-automated NSs of “Alpha” class, current achievements on control systems, reactor diagnosing systems, and RI inherent safety properties will be used to the maximum extent possible. The number of the NPC personnel will be ~ 70.

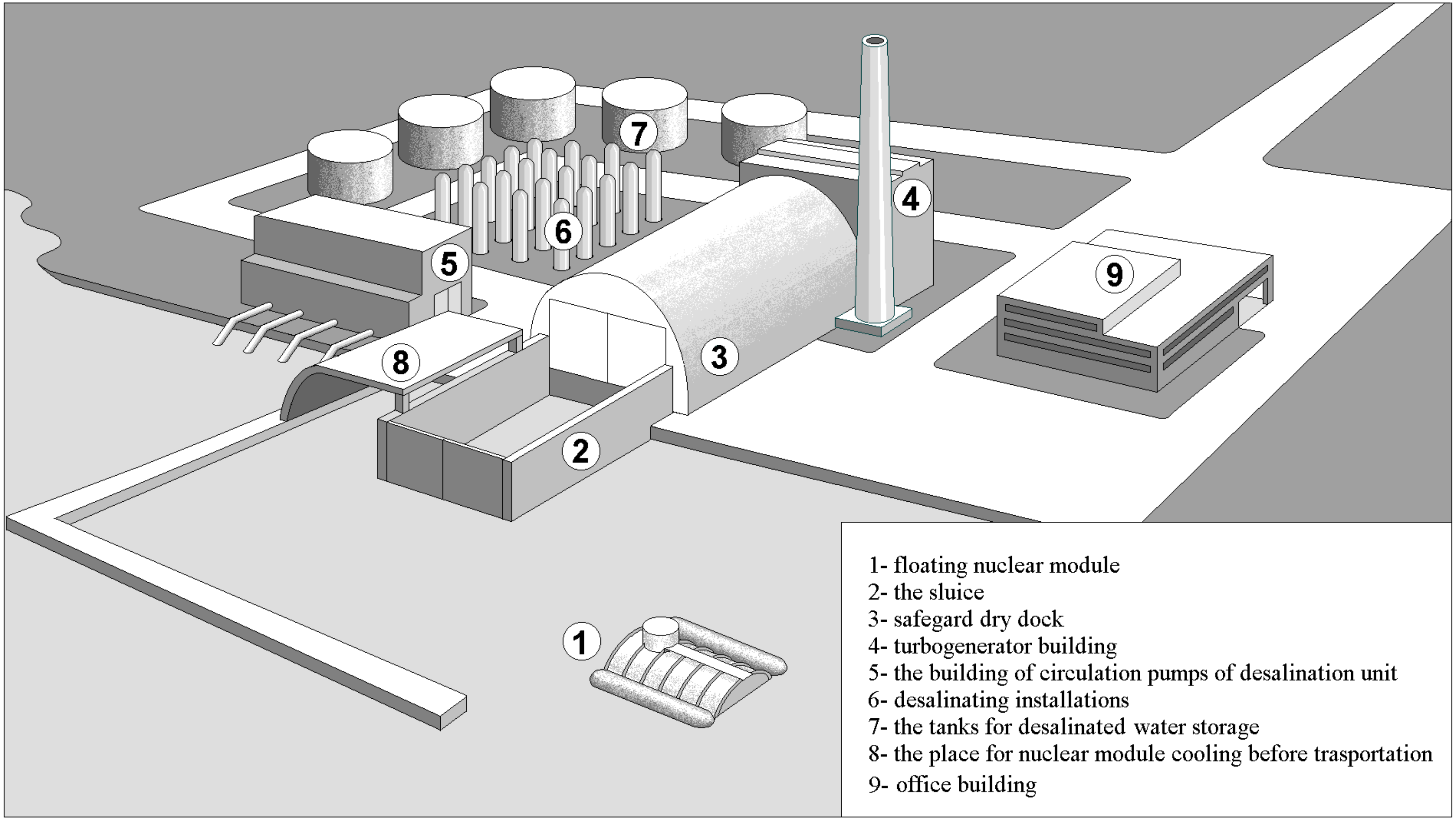

FIG. 2. Nuclear Power Desalination Complex. General View.

CONCLUSION

1. At present NPC development is retarded due to lack of real experience in developing countries, which is necessary to validate economical efficiency, safety, technology of managing radioactive waste, non-proliferation, resistance to malevolent actions. To overcome this barrier, it will be necessary to design full-scale prototype of that reactor and demonstrate the declared characteristics during 5-10 years of its operation.
2. The main feature of RI SVBR-75 is an opportunity to demonstrate its real safety including demonstration of its behavior in an event of accidents and to validate practically its design characteristics: economical parameters, safety, technology of managing radioactive waste, non-proliferation, resistance to malevolent actions.
3. A demonstrational prototype of the RI is expedient to be constructed in Russia. Lack of financial resources in Russia requires widespread cooperation of User-Countries, countries that produce a desalination equipment, and Russia (Russian enterprises), as a supplier of transportable RI modules.
4. If there is a real financing, the prototype can be designed by 2008 - 2010. Experimental operation of RI SVBR-75 and its international licensing will require other 5 - 7 years. At the same time, in order to begin their construction in quantities in 2015 - 2020, NPS design for particular sites can be launched.

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LIQUID METAL COOLED SMALL REACTORS (MDP AND 4S) IN CRIEPI

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Abstract

This paper describes two kinds of liquid metal cooled small fast reactors. A metallic fuel, which has inherent safety features, high breeding capability and so on, is applied to each reactor. One of them is a MDP (Modular Double Pool, 325 MW(e)), which has been designed to reduce the construction cost and to improve reliability by factory manufacturing of the most components. The other is 4S (Super Safe, Small and Simple, 50–100 MW(e)), which has been designed to obtain a long life core, while maintaining a negative sodium void coefficient during operation.

1. INTRODUCTION

The double pool type concept reactor[1)], in which most piping systems in the primary and secondary loops required in the loop type and/or pool type reactor concept are excluded to improve the economy, has been designed. The size of the reactor is also made small. The conceptual design of MDP (Modular Double Pool) [2)] has been carried out, in order to reduce the construction cost and to improve the reliability by factory manufacturing of most components. From the results of the safety evaluation, it was confirmed that a sodium water reaction in the steam generator does not affect the integrity of the reactor vessel and other systems and also the construction cost is similar to the LWR in the case that four units with 325 MW(e)/unit are installed as one power plant (1300 MW(e), 325 MW(e) $\times$ 4 units).

The innovative reactor concept of 4S (Super Safe, Small and Simple) [3),4)], which is different from the previous concept, is as follows: small diameter and long length fuel is selected to obtain a negative sodium void reactivity coefficient during the life time and the burn up of the core is controlled by the annular reflector surrounding the core. In this reactor, a long life core is obtained due to the long length core and the movement of the reflector upward, so the maintenance such as refueling is reduced and a high proliferation resistance is also obtained because the core is sealed for a long time. The unit power of the reference design is 50 MW(e) to keep the negative sodium void reactivity coefficient during the operation time (ten years), however, more power up to 100 MW(e) and longer life will be feasible in the advanced design of 4S[5),6)]. In the NERI program in US, ENHS (a lead-bismuth coolant reactor)[7)] is studied based on the design concept of 4S.

2. MDP (MODULAR DOUBLE POOL REACTOR)

For practical deployment of FBRs, it is necessary to reduce the construction cost. The construction cost of an FBR is expensive in comparison with an LWR due to the use of sodium as the coolant and the adoption of the intermediate heat transport systems. Therefore, the reduction and limitation of the sodium handling area and the reduction of the intermediate heat transport system are effective countermeasures for the cost reduction. The objective of this design study is to establish a modular reactor concept that can compete with a LWR on construction cost.

The double pool design concept is to reduce the distance in the intermediate heat transport system by installing steam generators and secondary pumps into the sodium filled annular space formed between the primary vessel and the secondary vessel. The reduction of secondary piping system allows reduction of piping support structures, sodium leak monitoring systems, pre-heating systems, etc. The reduction and limitation of the sodium handling area allows a reduction in the size and volume of the nuclear building, reduction of liner against sodium leak and limitation of sodium fire area. On the other hand, because of this compact arrangement, that is steam generators located adjacent to the primary reactor vessel, the structural integrity of the primary reactor vessel against any sodium-water reaction accident should be assured.

In addition to the simplification of the reactor module by the double pool system, standardization of design by seismic isolation, common utilization of equipment, reduction of plant site work and shorter construction schedule are considered to compensate the scale-disadvantage of small reactors.

2.1. Plant Design Description

Figure 1 shows the schematic diagram of the reactor module. Table 1 shows the main plant specifications. An electric power output of 325 MW(e) has been set, considering the size of the primary vessel, the secondary vessel and cavity wall combination that can be manufactured at a factory and transported to a site. The principal components of the reactor system consist of (1) the primary vessel which includes the core and its support structures, 4 intermediate heat exchangers, 4 primary circulating electro magnetic pumps integrated with cold traps, (2) the secondary vessel which includes 4 steam generators, 4 secondary circulating electro magnetic pumps, 2 cold traps, (3) upper internal structure, (4) fuel handling mechanism, (5) roof slab, and (6) short secondary sodium piping.

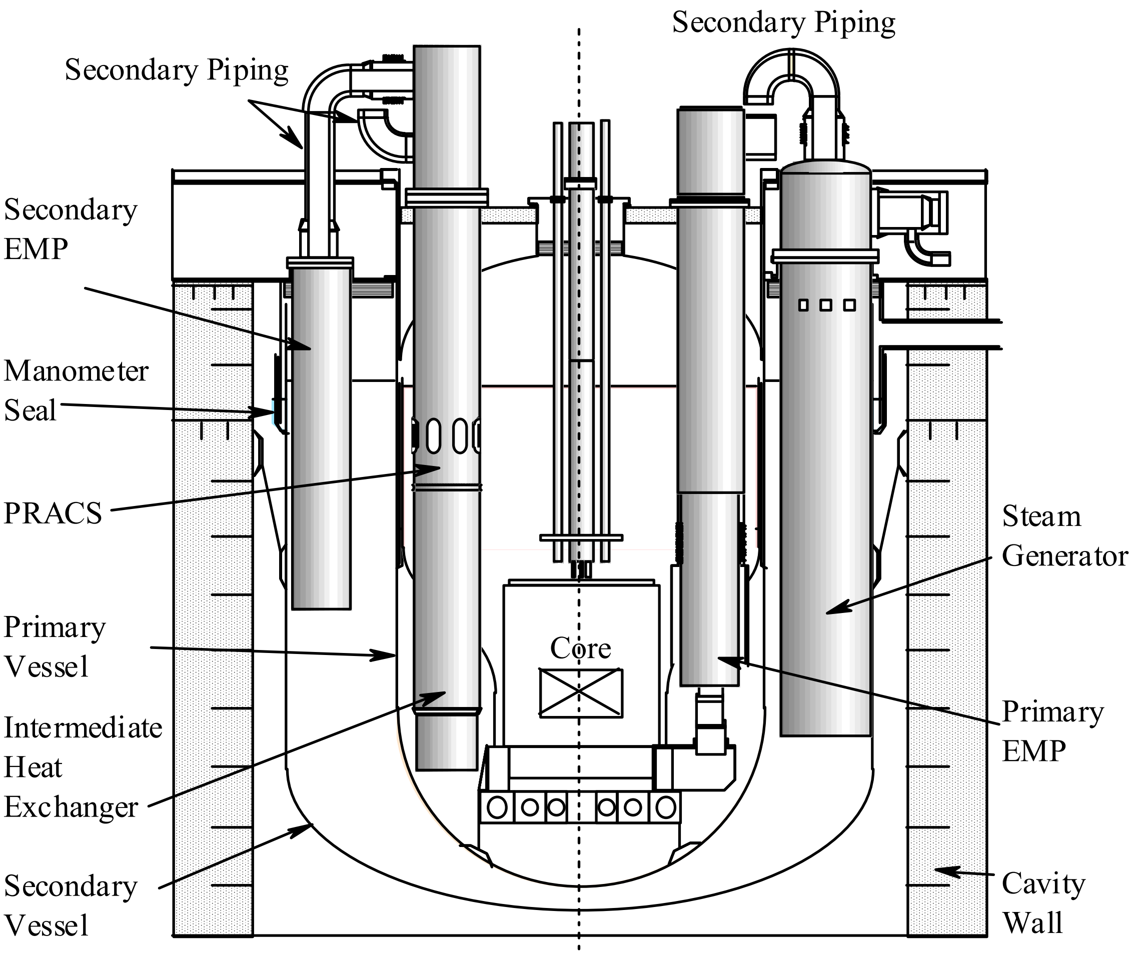


Fig. 1 Reactor Assembly of MDP

TABLE I. MAJOR PARAMETERS OF REFERENCE DESIGN

Item	Specification
2.1.1 Core Power	
Thermal Output	125 MW(th)
Electric Output	50 MW(e)
Core Inlet/Outlet Temp.	355/510ºC
2.1.2 Reactor Assembly	
Diameter/Length	2.5 m/23 m
Thickness	20 mm
Material	
Reactor Vessel	304 SS
Core Barrel	Mod.9Cr-1Mo
Reflector Guide	Mod.9Cr-1Mo
Others	304 SS
2.1.3 Fuel	
Composition	U-Pu-Zr
Pu Enrichment (Ave.)	19.5%
2.1.4 Core	
Diameter	83 cm
Length	4 m
Breeding Ratio	0.7
No. of Sub. Assembly	18
2.1.5 Reflector	
Material	Mod.9Cr-1Mo/Graphite
Thickness	15 cm
Length	1.5/2.0 m

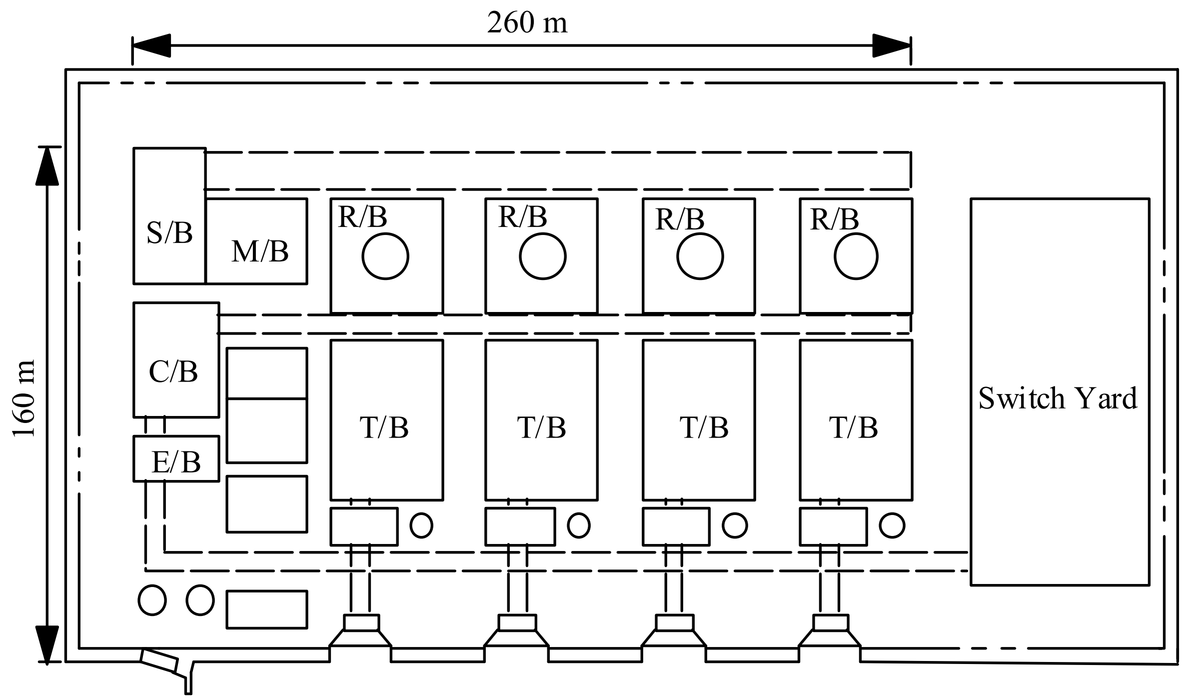
The sodium in the primary vessel flows upward from the core and flows downward in the tube side of the intermediate heat exchanger to reach the cold plenum. The sodium in the cold plenum is driven into the core by the primary electro magnetic pump. On the other hand, the secondary sodium pressurized by the secondary electro magnetic pump enters the intermediate heat exchanger through the secondary piping. The sodium flows upward in the shell side of intermediate heat exchanger and enters into the steam generator through the secondary piping. After heat transfer to water/steam, the sodium flows to the secondary electro magnetic pump.

The core design aims at passive shutdown capability based on the features of the metallic fuel and small-size core. A homogeneous core is used to achieve the compact radial core size. Reactivity control for normal operations is accomplished by six control rods, which constitute two individual shutdown systems (3+3), each of which can shut down the reactor independently. To prevent an ATWS, this separated shutdown system, low burn-up swing core and the primary flow coast down control system are adopted. Furthermore, features of the metallic fuel, thermal expansion of core and large heat capacity are expected to mitigate an ATWS.

The diameter of the primary vessel is 9 m and that of secondary vessel is 14.4 m. The primary vessel is supported from the roof-slab. On the other hand, the secondary vessel is supported from the cavity wall, and retains the secondary coolant. This secondary vessel also serves as a safety vessel, if a leakage occurs in the primary vessel. It functions to prevent the sodium level from lowering in the primary vessel. To cope with the thermal expansion of the secondary vessel and to form the cover gas boundary between the secondary cover gas space and cavity space, a manometer seal of low melting point alloy is used. This manometer also works to release the pressure in the case of a beyond design base sodium-water reaction accident. As for the pressure rise after a design base sodium-water reaction accident, 4 piping systems with rupture disks are used in common with 4 steam generators to release the pressure.

A seismic isolation system for the reactor building is adopted to achieve the standardization of design. The base size of the building is 40 m × 40 m, and the height is 55 m.

Figure 2 shows the plant layout for a 4-module constellation producing 1300MW(e). Each reactor has its own turbine generators, because considerations are given to the first operation of each module in order and to assure of the independence of each module. The control building, the fuel transport equipment, etc., are commonly used.



R/B ; Reactor Building
T/B ; Turbine Generator Building
S/B ; Service Building
M/B ; Maintenance Building
C/B ; Control Center Building
E/B ; Emergency Electricity Building

Fig. 2 Plant Layout (1300 MWe)

2.2. Safety in case of sodium-water reaction event

If water leaks from the heat transfer tubes in a SG, both isolation and blow-down of the water/steam system starts upon receiving the leak signal from the leak detection system. Considering the reliability of the leak detection system, the isolation system and the blow-down system, an event tree was made. A probabilistic estimation with the event tree has been performed to classify the sodium water reaction events sequences into two groups, DBE and BDBE. The analyses show that all sequences including multiple failures have lower probability than 10^{-7}/ry, which was determined as the BDBE criterion in this design study. On the other hand, the deterministic criterion of a single failure assumption is adopted to select DBEs though the probabilities of some sequences including a single failure are below BDBE criterion. This approach is taken to verify whether the mitigating systems and the responses of the

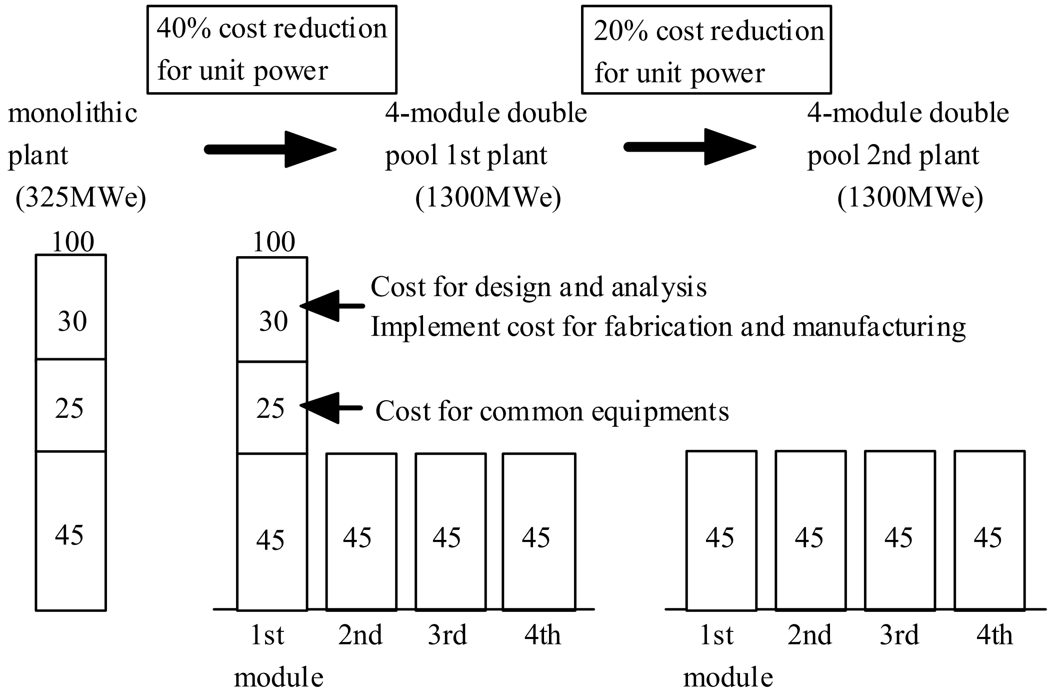


Fig.3 Effect of Standardization and Common Utilization of Equipments

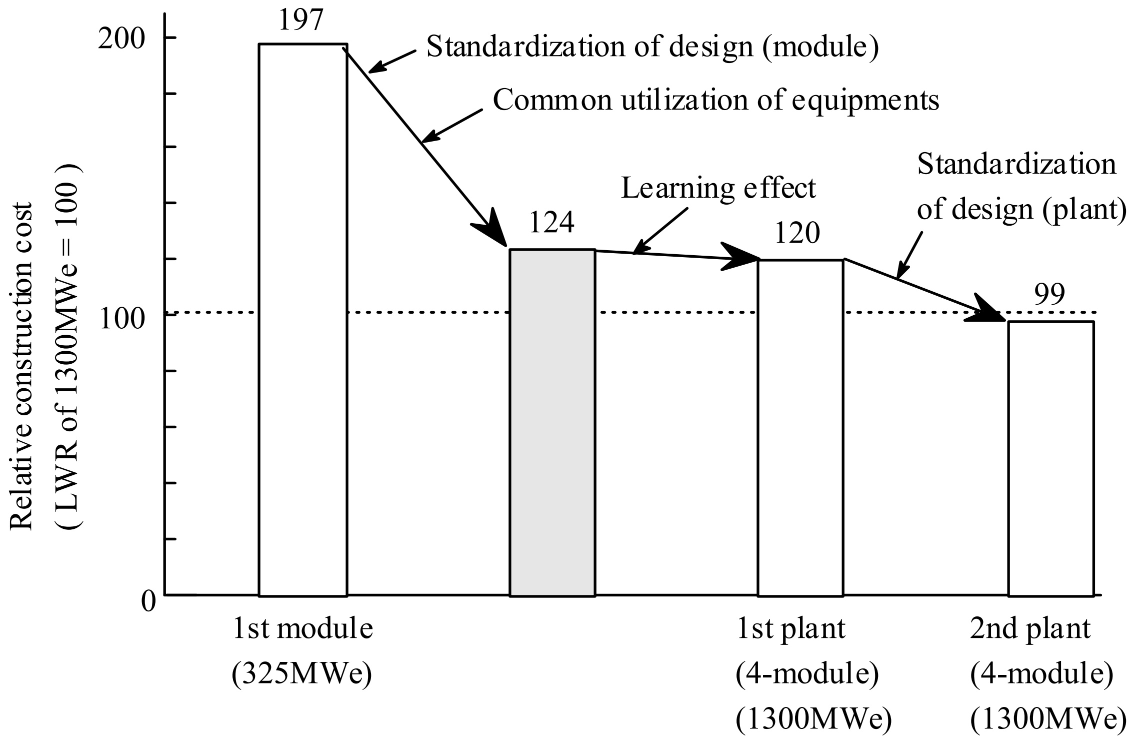


Fig. 4 Construction cost of MDP

components satisfy the safety targets. Moreover, the failure of all the leak detection systems, all the isolation systems and all the blow-down systems is analyzed to estimate the safety margin of this plant.

The analytical evaluation shows that the pressure rise for a DBE and a BDBE is much smaller than the limit of the allowable value. Furthermore, even for a sodium-water reaction event worse than BDBE, the primary vessel can withstand the external pressure with sufficient safety margin and the structural integrity of the containment boundary is confirmed.

2.3. Economy

The double pool reactor has a high potential to shorten the construction schedule because it reduces the sodium piping and sodium handling area, and provides for compactness of NSSS which is assembled and tested at a factory. The construction period of thirty-one months is confirmed. This short period contributes to the reduction of interest in construction.

In the construction cost for one module, 25% of the cost is for common equipment with the other modules such as fuel handling machine, and 30% of the cost is for design and analyses. Then, in the case of 4-module power plant, about 40% of the cost reduction will be achieved as shown in Figure 3. Figure 4 shows the cost reduction approach of MDP. The construction cost of 2nd 4-module plant will be competitive with the construction cost of a large scale LWR.

3. 4S (SUPER SAFE, SMALL AND SIMPLE-LMR)

Major parameters of the reference design are shown in Table 2. The reactor building is an embedded structure with seismic class A. It contains the reactor, secondary system, steam generator, coast down control system, power switch board and refueling pits. The dimension of building area is 26 m $\times$ 16 m, requiring only a small ground base.

3.1. Reactor system

The reactor system is shown in Fig. 5. Primary coolant flows out of the core, rises in the hot pool and descends in the intermediate heat exchanger through which the heat is transferred to the secondary sodium. It is pressurized by the primary electromagnetic pump at the bottom of the intermediated heat exchanger and flows down along the inner hole of in-vessel shielding structure. Then it turns at the bottom of the reactor vessel and enters the core.

The intermediate heat exchanger and the electromagnetic pump have an annular shape to form the annular path of flow. A space is provided within the outside of the core barrel, in which the reflector moves vertically. A sodium path is formed in this space to cool the reflector.

The primary pump has two annular single stator coils joined in series in order to keep safety at the loss of one coil.

A study is posed on the feasibility of 10 years stay in the core. The fuel of 4S falls within experience of the preceding reactor in aspects of fuel burn-up and neutron fluence. On the point of long stay time in the reactor, design needs to consider the margin for cladding corrosion in the design.

TABLE II. PASSIVE DESIGN FEATURES OF 4S

Item	Specification
Core Control System	Annular Reflector Movement (Nearly passive)
Primary Pump	Electromagnetic Pump
Primary Flow after Shutdown	Natural Circulation
Cavity Cooling	Natural Circulation
Containment Cooling	Natural Circulation
Secondary Pump	Electromagnetic Pump
Emergency Room Cooling	Heat Storage System
Safety Features Reactor Shutdown Shutdown Heat Removal	 Inherent Core Safety with Metal Fuel Natural Circulation

4S employs a burn-up control system with annular reflector in place of the control rod and control rod driving mechanism which requires frequent maintenance service. Replacement of the reflector is not required for the entire plant life. The core geometry with reflector control system should have the core equivalent diameter less than 90 cm and length of the reflector at least 1.5 m to meet requirements for negative void reactivity and no refueling for ten years.

The reflector drive mechanism consists of hydraulic system that operates at start up and shutdown and a ball screw that is connected to a motor which is operating during normal operation. The mechanism has six driving systems corresponding to the number of reflectors. The motor and the hydraulic system consist of six systems and one system, respectively

The reflector is moved upward at a rate of 1 mm/sec by the hydraulic pump for the start up. For shutdown of reflector, the scram valve is open and at 1 sec later, it causes the reflector to move downward at a rate of 10 cm/sec. As the reflector goes down 1 m, the reactor enters the sub-critical cold shutdown state. Length of 1 m downward movement of the reflector is limited by the capacity of the hydraulic cylinder. It can not move otherwise.

Upon completion of the start up by hydraulic actuation, the reflector is fastened by the hydraulic and moves up for burn-up control at a constant speed of 1 mm/day by a motor, which is designed so that the reflector is positioned by integration of generated power frequency. A reduction mechanism composed of paradox planetary gears is installed to attain the fine speed of 1 mm/day.

3.2. Advanced core concept

3.2.1 Reactor power up to 100 MW(e) [5]

The 4S core concept can be also applied to larger cores up to about 100 MW(e) (250 MWt). As the negative sodium void reactivity is primarily achieved by employing a small core diameter, one of the largest design restrictions is the equivalent core diameter, which is about 100 cm.

Other important restrictions are the coolant pressure drop and the linear heat rate of the fuel pin. The core height is primarily responsible for core lifetime and has some little effect on the pressure drop and the linear heat rate. The pressure drop linearly increases with the total length of fuel pin. As for the maximum

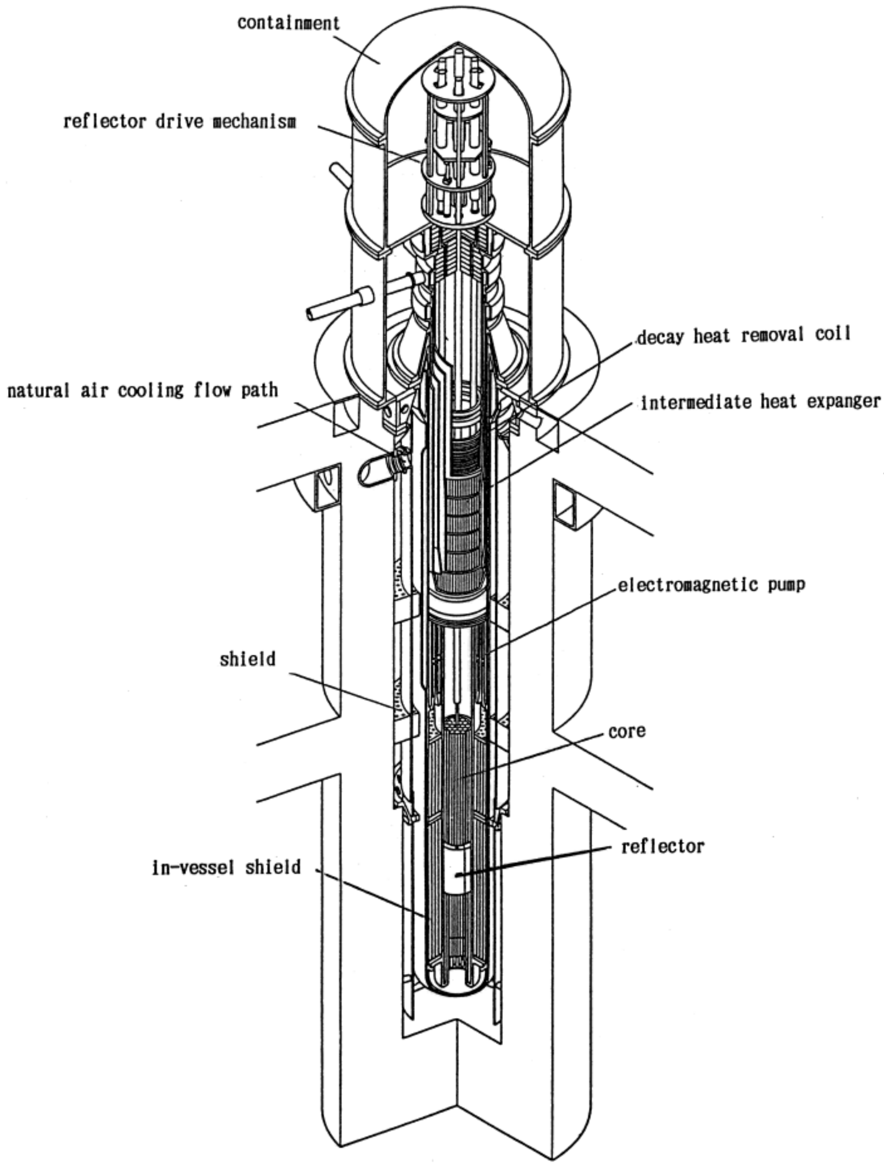


Fig. 5 Reactor Assembly

linear heat rate, increasing the core height has only a small effect because the axial power peaking factor becomes large. In the following design studies, the core height is fixed at 400 cm and it is assumed that the maximum permissible Pu enrichment is 20%. The design parameters of fuel assembly and core diameter for 80 and 100 MW(e) are 36 and 104.

Based on the above design parameters, 80 and 100 MW(e) cores can be obtained, however, the core life time is reduced to 5.8 years for 80 MW(e) and 4.7 years for 100 MW(e).

3.2.2 Long life core of 30 years[6]

The 4S core has a fast neutron spectrum, so conventional burnable poison materials are difficult to burn. Then we introduced moderator in the core to make the spectrum soft. The burnable poison assembly that is located at the center of the core. The assembly contains six burnable poison pins in which zirconium-dihydride (ZrH) and Gd is packed in a mixture. The ratio of ZrH to Gd is assumed to be 5:3 where the worth of Gd is maximized.

The reactivity loss due to fuel burn-up is monotonically decreased during the life time, otherwise the reactivity increase rate of the burnable poison, which is actually a decrease of negative reactivity, is large at the beginning of the operation in particular. In order to compensate the reactivity loss due to the burn-up and to sustain criticality, the upward speed of the reflector should be slower at the beginning than at other times.

The core height is reduced to 2 m compared to 4 m of the reference core. The length of reflector is increased to 2 m, but the upper part of it still consists of a cavity region to increase the reactivity. The effect of the cavity is enhanced by the B_4C shield, where leaking neutrons are absorbed. A 30 year long life core is obtained with four cases of core design: (a) metal fuel with Pu enrichment of 25%, (b) MOX fuel, (c) metal fuel with enriched uranium core and (d) metal fuel with lead coolant.

3.3. Basic Safety Features

The reactor concept of 4S is designed to enhance passive features shown Table 3, which include passive safety systems. In addition to these features, special attention is focused on excluding reactivity insertion during plant start up.

3.4. Economy

The material weight per output of 4S reactor structure is smaller than that of a large reactor such as SPX, owing to the simplified design concept. The economics of mass production would make the construction cost is near equal to that of a large LWR, if fabricating 10 units per one year for 10 years by the fully standardized design. Another cost merit of small reactors is expected to be that the total R&D cost for commercialization is dramatically reduced compared with a large reactor.

3.5. Proliferation resistance

4S is one wherein a long life core with small diameter is surrounded by annular reflector to control the burning and enhance the safety of the core. Its life time is set at ten or thirty years to eliminate the need of complicated refueling work. The fuel being a nuclear material is also sealed for ten or thirty years and subjected to a rigorous control by IAEA.

CONCLUSION

Nuclear energy can solve the dilemma of the energy crisis, economic development and environmental problems of the future. A study of both energy demand in the year of 2050 and all the energy technologies that could appear by the time reveals that the demand can not be met using the natural energy resource because vast quantities of energy will be required to account for rapid population growth and to preserve the world's environment. It also shows that fossil fuels create environmental problems and will be required for purposes other than the provision of energy. This strongly suggests that we must rely on nuclear energy, fast reactor, which can use natural uranium about 100 times as efficiently as conventional methods (LWR), as the best and the most logical solution. Two kinds of small fast reactor presented in this paper are expected to be utilized for different demands, such as large energy demand in a big city by modular MDPs or small energy demand in a remote area by 4S.

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