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Concept of Small Power Reactor Installation without Refueling during Lifetime (SVBR-75/100)

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1. Introduction

Under SSC RF IPPE scientific supervision the certain Rosatom organizations in Russia have been carrying out works on the Project of modular small power fast reactor SVBR-75/100 cooled with lead-bismuth coolant. This coolant was mastered in Russia for the nuclear submarines' (NS) operating reactors. The purpose of the development is construction of a unified modular type reactor with a high level of the inherent self-protection and passive safety, long lifetime without refuelling, short schedule of construction and assembly works on the NPP site, which will be economically effective both in the developing countries (small and medium size modular NPPs for the different purposes) and in the developed countries (large size modular power units) for the immediate future when the uranium fuel is used (the low cost of natural uranium, opened fuel cycle), and for the far future (the high cost of natural uranium, closed fuel cycle, mixed uranium-plutonium fuel, plutonium breeding).

The proposed work plan for the first year is given below:

- Justification of the selection for the reactor capacity providing realization of the stated aims.
- The analysis of potentials of multipurpose usage of such reactors.
- Selection of the structure, content and dimensions of the reactor core which provide an opportunity for the reactor to operate using different kinds of fuel in compliance with the requirements to non-proliferation and receiving the breeding ratio to be over one.
- Identification of the initial events (equipment failures, personnel errors and their multiple superposition, outside ill-intentioned actions), on which the safety analysis should be based.
- Justification of the reactor installation's (RI) safety concept assuring a high level of inherent self-protection and passive safety.
- Identification of the technologies required for realization of the concept.

2. Justification of the Selection for the Reactor Capacity Providing Realization of the Stated Aims

Many developing countries do not possess cheap resources of fossil fuel and wish to use nuclear power supply sources.

At the same time the modern market of nuclear technologies offer, as a rule, the large power nuclear power plant (NPP) which cannot be used in the developing countries where there are no power grid electricity transmitting lines.

Moreover, such NPPs need the developed infrastructure and high-qualified operating personnel that are also absent in the developing countries.

When choosing the type of the NPP to be used in those countries, it is necessary to take into account that many of them are located in the regions which are politically unstable and where the possibility of sabotage and acts of terrorism is very high.

When choosing the electric power of the NPP power unit, one should also take into account that the needs for power of the potential consumers are very different (i.e. the different power NPPs must be designed). For that reason, an opportunity to construct on the basis of the unified reactor module the large, medium, and small power-units of various purposes, including nuclear thermal electric power plants (TEPP) and nuclear desalinating power complexes (NDPC) is very attractive.

When the reactor power was chosen, the following criteria were taken as the vital ones:

- the economical prospects;
- simplicity of construction and operation;
- prevention of accidents due to the passive systems and inherent self-protection properties of the reactor design (including heat decay removal via the walls of the reactor vessel) and due to minimization of safety systems;
- an opportunity to provide core breeding ratio (CBR) ≥1 when operating by using mixed uranium-plutonium oxide or nitride fuel. This ensures solution to the problem of fuel-providing for many centuries when closing the fuel cycle;
- an opportunity to standardize the RIs (multipurpose usage, serial production);
- nearness of the scale factors of RI SVBR-75/100 and NS's RIs;
- low specific consumption of lead-bismuth coolant (LBC).

The estimations have revealed that the optimal reactor power is a level of ~100 MWe.

The selected power level (100 MWe) provides the following:

- reactor's heat decay removal is entirely passive, heat is removed via the monoblock vessel to the passive heat removal system (PHRS) tank;
- complete plant fabrication of the reactor monoblock, RIs are produced in large quantities that improves the quality of works and reduce the cost;
- the reactor monoblock can be transported by railway, road or marine transport (with fuel in a nuclear and radiation-safe state due to LBC "freezing" in the monoblock vessel that also meets non-proliferation requirements);
- the schedule of constructing the NPP unit can be considerably reduced as modules are delivered in high plant readiness and the assembling scopes are sharply reduced. (This improves the conditions of receiving the credits for NPP construction and reduces the period of capital investments' payback);
- an opportunity to renovate the NPP unit with replacement of the RIs by the new ones in 50 … 60 years. This postpones to 50 years the necessity to construct the replacing power capacities;
- the cost of decommissioning the unit can be considerably reduced as no radioactive materials remain in the main reactor building after the monoblock has been removed;
- the NPP units with light water reactors (LWR) which RIs have expired their lifetime can be renovated by installing the necessary number of RI SVBR-75/100 in the empty steam generator (SG) and main circulation pump (MCP) rooms with use of some equipment, buildings and structures.

The relatively high cost of bismuth was a cause that the measures reducing the specific mass of coolant in the RI monoblock were developed.

The analysis of experience of developing the different capacity RIs has revealed the LBC specific mass decreases at reducing the RI nominal power.

Along with this, reduction of the LBC specific mass is limited. It is caused by the fact that at small dimensions of the core, it is impossible to provide CBR ≥ 1. Computations have revealed that in case of equivalent electric reactor power of ~ 100 MWe and corresponding dimensions of the core, CBR ≅ 1 is provided not only for the mixed nitride fuel but also for the less dense but well mastered MOX fuel.

Reduction of the LBC specific mass in the small-sized fast reactor (FR), for which the core power density is several times less than that of the sodium cooled FR, is also achieved by elimination of the in-reactor storage of spent nuclear fuel (SNF) and in-reactor refueling mechanisms (rotating plugs, etc.).

In this case, refueling is performed once during the core lifetime. For that purpose, a special refueling equipment is used, it is also used for reloading the fuel of all power-unit's reactors. The refueling technology is similar to that of LBC cooled NSs' RIs. (When using RI SVBR-75/100 in developing countries, the refueling equipment is not delivered).

Another way of reducing the LBC specific mass is increasing its average velocity in the RI and diminishing the length of the LBC circulation circuit. However, this way has its own constraints caused by the necessity to provide the safety requirements.

The first requirement is caused by the necessity to provide the power level of the reactor with naturally circulating LBC to be not less than 5 % of N_{nom}. This makes it possible to eliminate dangerous increase of temperature in case of shutting down the MCPs.

The second requirement is caused by the necessity to provide effective separation of steam bubbles from LBC with steam surfacing to the LBC free level in case of an accident with leaking SG tubes. This is necessary for elimination of steam ingress into the core and impermissible increase of pressure in the monoblock vessel.

The necessity to meet the highlighted requirements resulted into development of the LBC circulation scheme in which core hydraulic resistance in the primary circuit equals to 90 % of the total core hydraulic resistance and hydraulic resistance in the SGs, in which LBC velocity is much less, only equals to 10 %.

With due account of the highlighted requirements, the specific mass of bismuth in RI SVBR-75/100 is ~ 1100 t/GWe. At this, contribution of the coolant's cost to the specific capital costs of the NPP construction is very slight.

3. The analysis of potentials of multipurpose usage of reactors SVBR-75/100

The "standard" reactor modules with electric power unit being ~100 MW provides an opportunity of their multi-purpose usage for: construction of the modular type NPP power-units of small, medium or large power; construction of the nuclear heat electric power plant (NHEPP) of 200…600 MWe which are located not far from the cities; renovation of the "old" NPP units which reactors have expired their lifetime; use as parts of nuclear desalinating power complexes.

All these heighten the quantity production of the reactor modules, enable to change over to the innovative methods of standard designing of the different power-units and industrial methods of in-line production of construction and assembly works that result in improvement of quality, reduction of the schedule terms and costs of works.

The scheme opportunities of multi-purpose usage of RI SVBR-75/100 is shown in Fig. 1.

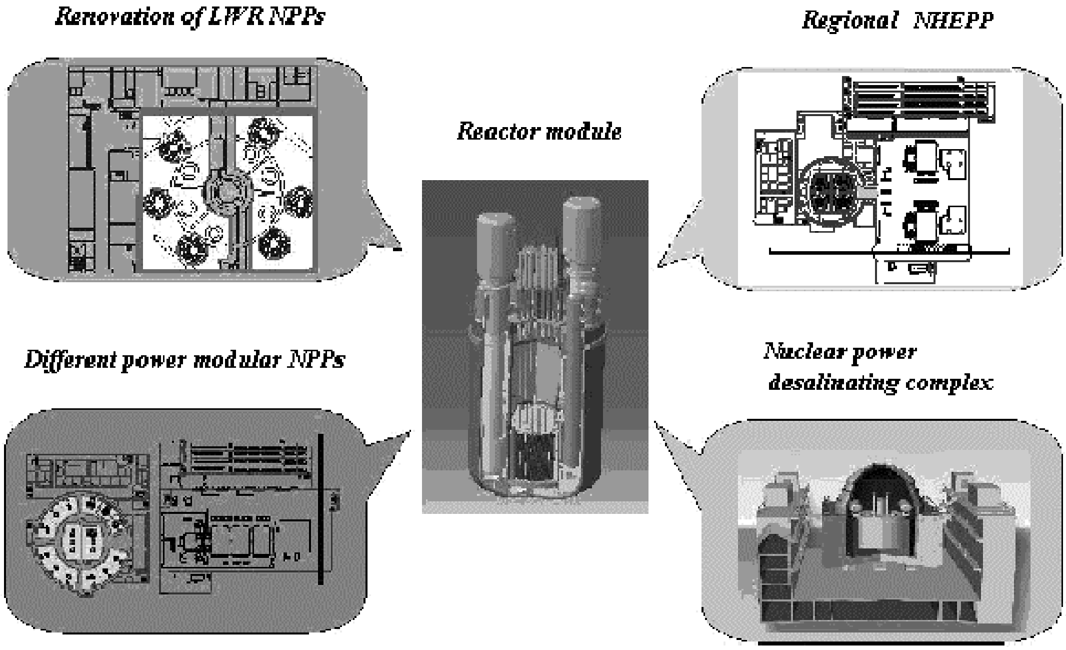

Fig.1 — The areas of multi-purpose usage of the SVBR-75/100 RI

From the standpoint of economical effectiveness, at the nearest phase of NP development in Russia the most attractive area of use of RI SVBR-75/100 is renovation of the “old” power-units of the NPP with VVER withdrawn from operating after the designed and extended service lifetime of the RIs has been expired instead of construction of the new replacing capacities. The real opportunity of implementing such renovation is verified by technical and economical investigations. The renovation means installation of the necessary number of SVBR-75/100 RIs in the original SG and MCP emptied compartments.

The estimations have revealed that the program of renovation of the existing VVER power-units will require ~300 RIs SVBR-75/100. This predetermines their production in quantities with corresponding reduction of the cost. As fabrication of the modules does not require the unique engineering equipment, the opportunity to create a competitive market of producers arises.

The results of technical and economic researches into the technical opportunity and economical expediency to renovate the 2^{nd}, 3^{rd} and 4^{th} units of the Novovoronezh NPP (NVNPP) on the basis of SVBR-75 RIs have revealed that such renovation would reduce the specific capital cost by a factor of two, as compared to the construction of the new replacement power capacities. Installation of the SVBR-75 RIs in the SG/MCP compartments of the 2^{nd} NVNPP unit is shown in Fig. 2, Fig. 3 (a plan view and a longitudinal section).

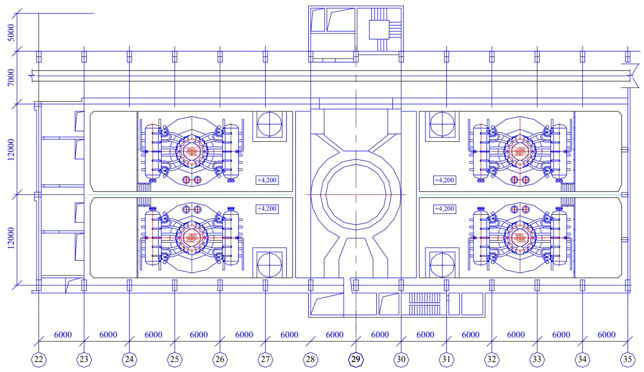

Fig. 2 — Arrangement of the 4 reactor modules in the building of the 2nd unit of the NVNPP (plan)

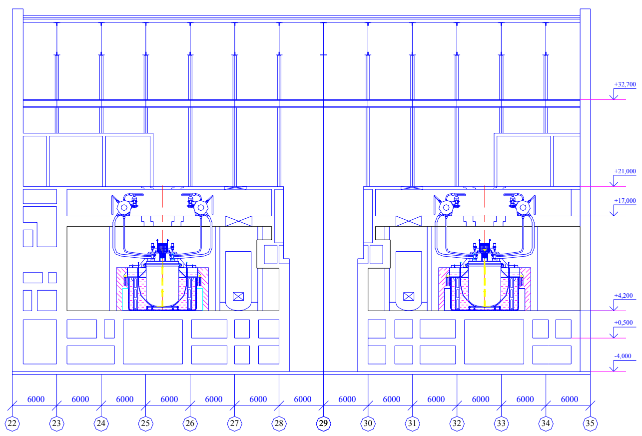

Fig. 3 — Arrangement of the 4 modules SVBR-75/100 in the building of the 2nd unit of the NVNPP (longitudinal section)

Experience gained by operating RI SVBR-75/100 in conditions of the NVNPP will make it possible with a minimal investment risk to launch sequential renovation of LWR units, which RIs have expired their lifetime and construction of modular NPPs with power-units of different power and purpose.

The improvement of the economical parameters of the NPP based on the SVBR-75/100 RIs is achieved due to lack of many safety systems necessary for the NPPs with LWRs which make the NPPs of this type considerably more expensive.

Another perspective area of use of the SVBR-75/100 RIs is to use them as parts of nuclear desalinating power complexes as in many developing countries fresh water resources are in deficit.

The NDPC consists of two parts: a transportable reactor unit (TRU) and a stationary complex that includes a turbine-generation installation and desalinating complex.

The on-shore located stationary complex is equipped with a dock structure that provides the TRU reception and protection against the external impacts and is functioning during the whole period of operating the NDPC. The stationary complex is a property of country-user. Construction, assemblage of the component equipment and operating is realized with maximal use of the domestic personnel, resources and industry.

The TRUs are delivered in compliance with the principle: “Build-Own-Lease”. This means that the Supplier leases the User the reactor unit for the time period determined by duration of the reactor core lifetime (~10 years). Such core lifetime will make it possible to keep the stable costs for the NDPC products (electricity, potable water and heat).

The RI is installed in the tight fail-safe floating (transport) unit with required stiffness. The TRU is equipped with all systems necessary for reliable operation of the reactor under the design basis scenarios and passive safety systems which make it possible to overcome the possible accidental situations without attracting the personnel and using any kinds of power. The TRU is entirely isolated from the environment that ensures lack of radioactive discharge and exhausts.

After being manufactured at the factory, the TRU with LBC “frozen” in the reactor is transported to the NDPC site and installed by sluicing in the “dry” dock that protects the TRU against falling objects and other external impacts.

On-site handling the radioactive materials including reactor refueling are not performed that makes it possible to reduce the requirements to the operating personnel and provide the nuclear materials nonproliferation mode.

In the events of all possible accidents, the radioactive products are retained in the TRU compartment, no radioactive contamination of the NDPC site happens that eliminates the necessity of implementing decontamination works.

On ending the core lifetime, the reactor unit is sluiced to the cooling compartment located on the NDPC site, which is protected against the external impacts. It remains there until solidifying of LBC (melting point is ~125°C). Another TRU is delivered to the User instead. After cooling, the Supplier will transport the spent unit with solidified coolant to the factory-manufacturer for refueling, necessary repair-reconditioning works providing the repeated use of the TRU for direct purpose.

On ending NDPC operating and Supplier’s transporting the last spent TRU, no radioactive materials will be left on the NDPC site.

Desalinating and electricity-generating installations do not belong to the systems important for safety, do not affect the RI safety and can be designed and manufactured according to the general industrial rules and standards.

Location of the reactor unit as well as the whole NDPC on the solid basis (in contrast to the variant of its location on the barge) ensures a high level of RI protection against the external impacts (including tsunamis), simplifies operating of the NDPC and considerably heightens reliability of transmitting the electricity, steam and potable water to the on-shore consumers.

The NDPC general view is shown in Fig. 4.

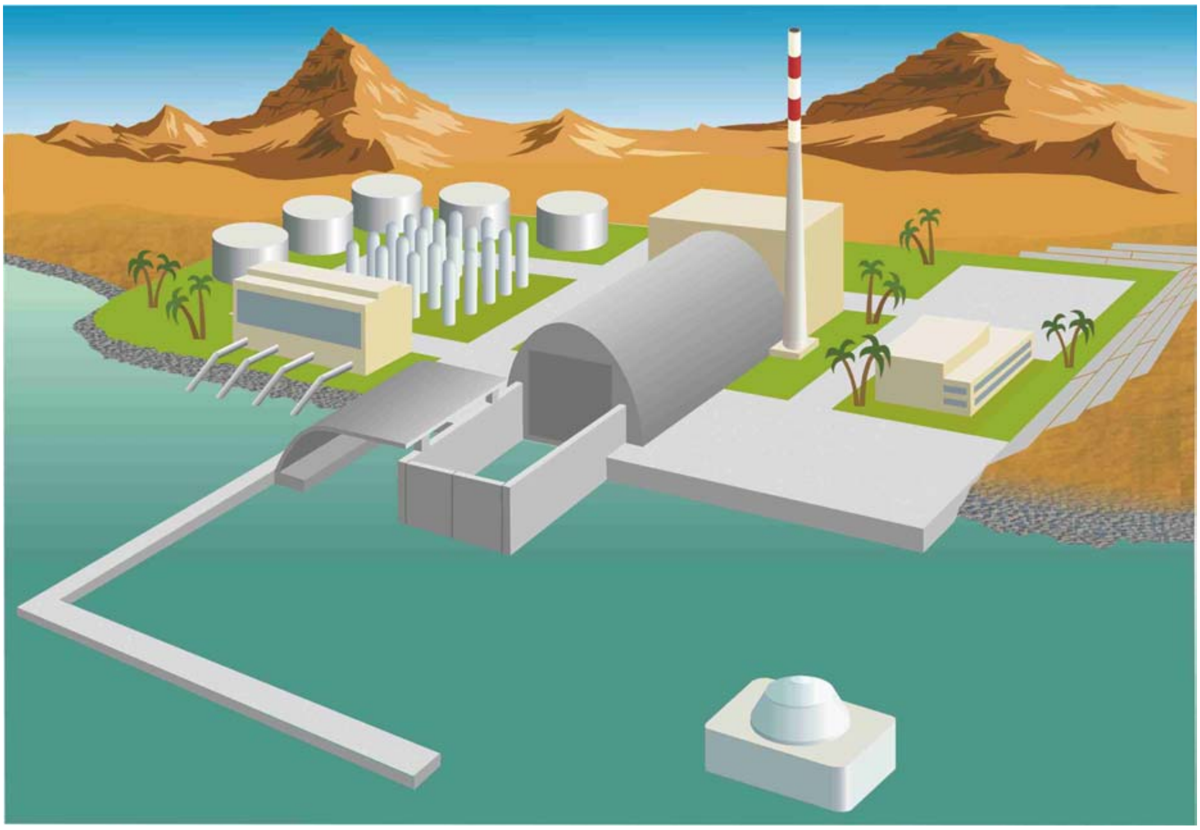

Fig. 4 — The NDPC general view

4. Selection of the structure, content and dimensions of the reactor core which provide an opportunity for the reactor to operate using different kinds of fuel in compliance with the requirements to non-proliferation and receiving the breeding ratio to be over one

- A wrappless design of the fuel sub-assemblies (FSA) is used. This ensures high cross heat-mass-exchange in the core and eliminates unallowable over-heating of fuel elements at large blockages of flow rate at the core inlet.
- On ending the lifetime, refueling can be performed at once, FSA-by-FSA. Being coupled with long lifetime (8-15 years), this eliminates the access to fuel and reduces the risk of unauthorized proliferation of nuclear fissile materials.
- The selected dimensions of the core (a diameter of 1600 ÷ 1700 mm and a height of 900 mm) at the volumetric fraction of fuel being 55 % (a diameter of fuel element is 12 mm, triangle lattice pitch is 13,6 mm) provide an opportunity to use different kinds of fuel (UO_2, MOX fuel with weapon or reactor Pu, TRUOX fuel, nitride fuel) without changing the reactor design and at meeting the safety requirements.

At this, for the variant of the core using oxide uranium fuel, uranium enrichment in U-235 does not exceed 20 %. This meets the IAEA requirements to non-proliferation.

It should be highlighted that the low values (25 … 30 %) of the LBC volumetric fraction in the core (“tight” lattice of fuel elements) and LBC specific mass do not deteriorate the safety parameters of RIs SVBR-75/100 in cases of shutting down the MCP and leaking SG tubes (as computations have revealed) but in the case of unauthorized insertion of positive reactivity as well. The latter is caused by a sufficiently high negative feedback being typical of small power reactors and a low time of delay of its temperature component at the coolant’s inlet in the core (extending of the lower lattice) coupled with sufficient heat-accumulation ability of the monoblock.

The following have been provided at the selected core dimensions and structure:

- The lifetime duration is ~ 53000 eff. hours in case the mastered oxide uranium fuel is used (CBR = 0.87);
- CBR ≥ 1 in case the MOX fuel is used, the reactor operates in the closed fuel cycle in the mode of fuel self-providing;
- CBR ≥ 1 in case the mixed nitride fuel is used, the reactor operates in the mode of fuel self-providing at a burn-up reactivity margin being less than βeff or in the mode of extended breeding with CBR = 1.13 at the plutonium doubling time being ~ 45 years;
- A burn-up reactivity margin is less than βeff, the lifetime duration is ~ 80000 eff. hours in case the uranium nitride fuel is used;

5. Identification of the initial events (equipment failures, personnel errors and their multiple superposition, outside ill-intentioned actions), on which the safety analysis should be based

It is proposed to consider the following initial events which will be the basis of the safety analysis:

- unauthorized extraction of the most effective absorbing rod;
- at the core inlet the coolant pass-through section is 50% plugged;
- all MCPs are shut down;
- steam intake to the turbine-installation and feed-water supply are terminated;
- guillotine rupture of several SG tubes;
- leak in the reactor monoblock vessel;
- blacking out the NPP.

6. Justification of the reactor installation’s safety concept assuring a high level of inherent self-protection and passive safety

The following basic approaches and technical solutions have been realized in the RI SVBR-75/100 design:

- A monoblock (integral) design of a pool type is used for the primary circuit equipment. Valves and LBC pipelines are completely eliminated, all the necessary devices of the coolant’s technology are installed inside the monoblock vessel. These practically eliminate the severe accidents of the LOCA type (loss of coolant accident);
- A two-circuit scheme of heat decay removal is used that simplifies the RI design, reduces the RI cost and cheapens RI maintenance and operating;
- The levels of coolants' natural circulation (NC) in the heat-removal circuits are sufficient enough to ensure reactor cooling without dangerous over-heating of the core. This provides passive heat removal in the accidental conditions;
- A reactor monoblock with a safeguard vessel is installed and fixed in the tank of the passive heat removal system. The tank is filled with water and also performs the neutron protection function. The supporting structures provide seismic resistance of the RI;

- A steam-generator operating in compliance with a multiple NC scheme and producing saturated steam is used. This ensures the best lifetime and operating parameters, e.g. reliable RI operation at any power levels (including the mode of heat decay removal via the SG), simplicity of maintaining LBC in a liquid state at low power levels;
- A slow-rotating gas-tight uncontrolled electric engine of the main circulation pump, that has a bottom hydro-static bearing and which electromotor power does not exceed 500 kW, is used. This eliminates the necessity to seal the rotating shafts, enables to use the ball-bearings with greasing and provides the necessary against-cavitation condition at the suction of the MCP impeller due to hydrostatic pressure of the coolant column;
- The RI equipment can be repaired or replaced.

The natural properties of lead and bismuth, physical features of the FR coupled with an integral (monoblock) design of the primary circuit equipment make it possible to eliminate deterministically an opportunity of the certain severe accidents.

High boiling point of coolant increase reliability of heat removal from the core and safety due to lack of the heat removal crisis phenomenon and being coupled with a safe-guard vessel eliminates the accidents of the LOCA type.

Low pressure in the primary circuit enables to diminish the thickness of the monoblock vessel’s walls and to reduce the constraints of the temperature change rate according to the thermo-cycling strength conditions.

LBC reacts with water and air very slightly. Development of the processes caused by primary circuit's tightness failure and SG intercircuit leaks occurs without hydrogen release and any exothermic reactions. There are no materials within the core and RI that 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.

In the case of failure of all active cool-down systems and total blacking out the unit, elimination of core melting caused by residual power release and keeping the monoblock vessel intact are ensured by an entirely passive way due to heat accumulation in the in-vessel structures and coolant and heat removal to the PHRS water tank via the monoblock vessel with further water evaporation. The “grace” period necessary for achieving the safety operation limit is about five days' time. A scheme of heat removal to the PHRS tank is shown in Fig. 5.

In case of unauthorized insertion of positive reactivity at postulated failure of all emergency protection (EP) system, elimination of prompt neutron reactor runaway is ensured by a special algorithm of controlling the compensating rods, which is the part of the automatic control system. In this case, when the reactor operates at nominal power, during a certain time (~ 4 months) a reactivity margin controlled by an operator is much less than β_{eff}.

Besides, an efficiency of each rod is much less than β_{eff}, a velocity of moving the absorbing rods extracted gradually is technically limited. For that reason, the inserted positive reactivity has time for being compensated by negative feedbacks without dangerous increase of the core temperature.

In the case of EP system failure caused by the events not specified in the regulatory documents (for example, damage of all servo-drivers), there are fusible locks connecting a rod with a driver bar. When the coolant temperature exceeds 700 °C, the EP rods that are installed in the “dry” channels are separated from the bars and drop into the core due to their gravity.

For the considered fuel loads, the total void reactivity effect of the reactor is negative and the local positive void reactivity effect is less than $\beta_{eff.}$ and cannot be realized due to the coolant's very high boiling point and lack of the opportunity for gas or steam bubbles to arise in great quantities.

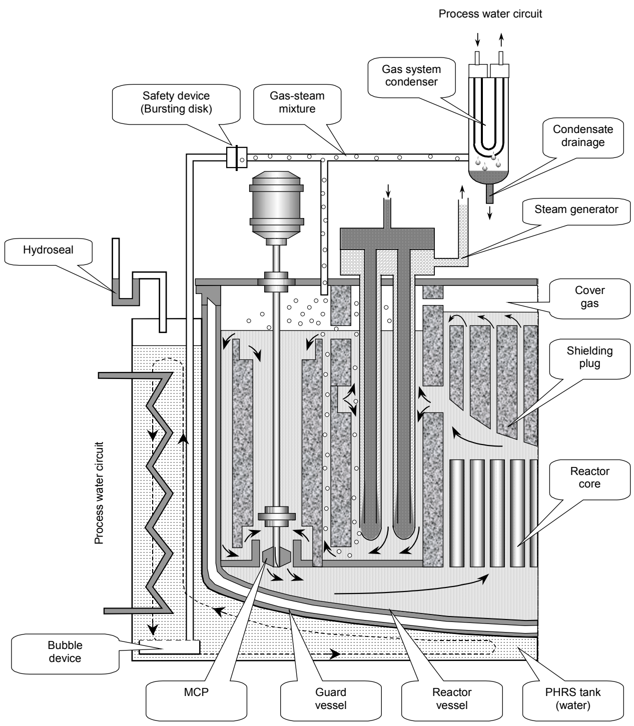

Fig. 5 A scheme of heat removal to the PHRS tank

Elimination of water or steam penetration into the core caused by a large SG leak and consequent over-pressurization of the monoblock vessel designed to be resistant against the maximum possible pressure under this condition are ensured by the LBC circulating scheme. This scheme provides that steam bubbles are thrown out on the free coolant level by moving up LBC flow. Then steam goes to the gas system condensers. In an event of their postulated failure, steam goes to the bubbler (PHRS tank) via the bursting membranes.

The carried out analysis has revealed that no equipment failures, personnel's errors or their combinations may cause core melting. The negative reactivity feedbacks ensure power decrease down to the level that does not cause core damage even in case of failure of all reactor shutdown systems and total blacking out.

The additional barriers of the safety providing system are the separate concrete cells of the RI (confinements) which restrict radioactivity release into the central reactor hall, and the protection shell of the central hall covering all RIs (containment) purposed to protect the reactor against the external impacts. In the event of accident tightness failure of the primary circuit, the high pressure radioactive exhausts (which can happen in LWRs) do not occur in LBC cooled RIs. For that reason, there is no need to design the containment of the unit and RI compartment to be resistant against high excess internal pressure. There is also no need to design the double containment with a water cooling system and corium catch.

The simple design scheme of RI SVBR-75/100, lack of the plant safety systems caused by the developed inherent self-protection properties of the RI, that made it possible to couple the functions of RI safety systems with those of normal operating systems, sharply reduce the probability of personnel's errors. The consequences of any personnel's errors and their combinations do not affect safety but only result in economical losses and the necessity to carry out unscheduled repair works.

RI safety does not depend on the equipment and systems of the turbine-installation. The key systems important for safety operate passively being independent on either right or wrong personnel's actions.

It should be highlighted that there are no valves or mechanical devices in the RI safety systems, which failure or shutting down being the result of someone's error or malicious actions or being the result of the over-standard external effects may cause blockage of the RI safety systems.

For that reason, actuation of the RI safety systems is assured by:

- Melting the locks of the EP rods (in case of exceeding the set LBC temperature) and their free falling down into the core due to the gravity;
- LBC natural circulation, heat transfer via the main and safe-guard vessels, air convection in the gap and heat irradiation, water boiling in the PHRS tank in the modes of emergency heat decay removal in the case of full NPP “blacking out”
- Rupture of the safety membrane that protects the monoblock from excess pressure at large SG leak and failure of the gas system's steam condensers.

As computations have revealed, the safety potential peculiar to the RI of the considered type is characterized by the following: even in an event of the postulated combination of such initial events as containment destruction, damage of the RI compartment overlapping and serious failure of the primary circuit gas system with direct contact of the LBC surface with atmospheric air in the monoblock vessel that is possible in the case of terror attacks, neither reactor runaway, nor explosion, nor fire occurs, and the radioactivity release is less than that requiring the population evacuation. These eliminate development of the corresponding anti-accidental measures.

The obtained results enable to conclude that the safety level of the SVBR-75/100 reactors is higher than that of LWRs and sodium cooled FRs. It is viable that the inherent self-protection and passive safety properties have been verified not only by computations but they can be really demonstrated at the stage of experimental operating RI SVBR-75/100 with realization of different postulated initial events and their combinations in controlled conditions without any economical and radiation damage.

Therefore, we can make a conclusion about SVBR-75/100 robustness and tolerance to the ill-intentioned actions such as sabotage, acts of terrorism, etc.)

7. Identification of the technologies required for realization of the concept

The list of back-up technologies is cited in Table 1.

Table 1. Back-up technologies

Problem	Back-up technologies	Status
1. Providing the reliability of primary circuit operating	Technology of maintaining the required quality of LBC during operating	Having tested by operating experience
2. Fabrication of fuel elements	Ribbed tubes for fuel elements are fabricated from steel EP-823	Having mastered in industrial production
	Pellets with oxide fuel mastered in FRs and VVER are used	
	Technology of fabricating the rod fuel elements of the container type	
3. Steels' corrosion resistance in LBC	Production of steel EP-823	Having mastered in industrial production
4. Steels' corrosion resistance both in LBC and in steam-water coolant	The industrial technology of fabricating (and welding) the bimetallic tubes for SGs which are corrosion resistant both in LBC and in steam-water coolant has been developed and mastered	Having mastered in industrial production
5. SG design	SG with bayonet tubes	Having tested by experience of operating RI BN-350
6. Instruments and devices for controlling and monitoring the parameters	Have been developed. Facility and full-scale tests have been realized	Having mastered in industrial production
7. Providing radiation resistance of the reactor vessel and in-reactor structures	Use of the in-vessel protection units with boron carbide	Having tested by experience of operating RIs with LBC
8. Keeping operability of the reactor's primary circuit elements when "unfreezing/freezing" LBC	Temperature-time schedule of RI "unfreezing"	Having verified by R&D results
9. Starting and cooling without connection with the turbine installation	Use of the autonomous cooling system with constant steam pressure in the SG maintained by the passive type devices	Having tested by experience of operating RIs with LBC