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PASSIVE COOLING SYSTEM FOR RESEARCH REACTORS

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Advantages of the technological scheme of the passive heat sink during the organization of the downward motion of the coolant in the core of the research reactor using natural circulation are shown. Examples of a numerical evaluation of the hydraulic characteristic of a cooling circuit based on the principle of natural circulation are given and the possibility of using such a scheme for medium-capacity research reactors is justified. The issues of the increasing of the power level while start-upp of such reactors, as well as the issues of increasing the safety of research reactors, are considered.

Key words: passive cooling system, research nuclear reactors, nuclear safety, pool-type reactor, tank-type reactor

INTRODUCTION

Research reactors include a wide range of civil and commercial installations, which, as a rule, are not used for power generation. Sometimes this category also includes experimental reactors, which can be more powerful than the research ones.

The main task of research reactors is to provide a source of neutrons for research and other purposes, and the process of their producing in the form of beams can be different depending on the purposes. The range of tasks that are set before these installations is wide, including analysis and testing of materials, production of radioisotopes. Among the applications are nuclear power, nuclear fusion researches, environmental studies, the development of new materials, the development of drugs and nuclear medicine [1].

As a rule, in such reactors circuits with forced cooling of the reactor core by the coolant pumped by pumps are used, but low-power reactors sometimes use a circuit with natural circulation of the heat carrier, for example, in the reactor IR-200 (Sevastopol, Crimea). Despite the advantages of the passive principle of reactor core cooling, it is practically not used because of the disadvantages of the circuits with lifting motion of the coolant in the core. Moreover, these schemes of heat removal from the reactor can not be called completely passive, since a secondary circuit with forced movement of the coolant is used.

To overcome these disadvantages, a passive cooling system with a downward coolant movement in the core of the reactor is proposed, the advantages of which will be discussed below.

Passive cooling system

Passive operation of systems is one of the key conditions for high reliability and operational safety. There is nothing that could disable or damage the mechanism of gravity, so the operation of systems on this principle is absolutely reliable. One of the applications of gravity is the system of natural circulation of the coolant, which develops due to the difference in the pressure of the hydrostatic column of the liquid in the circulation circuit, taking into account the difference in temperature and density of the coolant in the lifting and lowering pipelines

An important advantage of research reactors with natural circulation is that it is possible to reduce the volume of safety requirements with respect to them, but only taking into account their self-protection properties and on the basis of specific safety justification documents presented by their owners for independent examination. When using the adopted natural circulation schemes with the lifting motion of the coolant in the fuel assembly, it is very difficult to determine the thermal power of

the reactor by measuring the mass flowrate and raising the temperature in the core, which can be attributed to the disadvantages of such cooling circuits.

To eliminate this disadvantage and increase the intensity of heat removal, it is proposed to organize a natural circulation loop with the descending flow of the coolant through the fuel assembly for research reactors of the pool-type, as shown in Fig. 1, but for a research reactor of the vessel-type, as shown in Fig. 2. The use of such a technological scheme will allow the standard instruments to measure accurately both the coolant flowrate through the reactor core and the heating of the coolant in it, and, consequently, to control the thermal power.

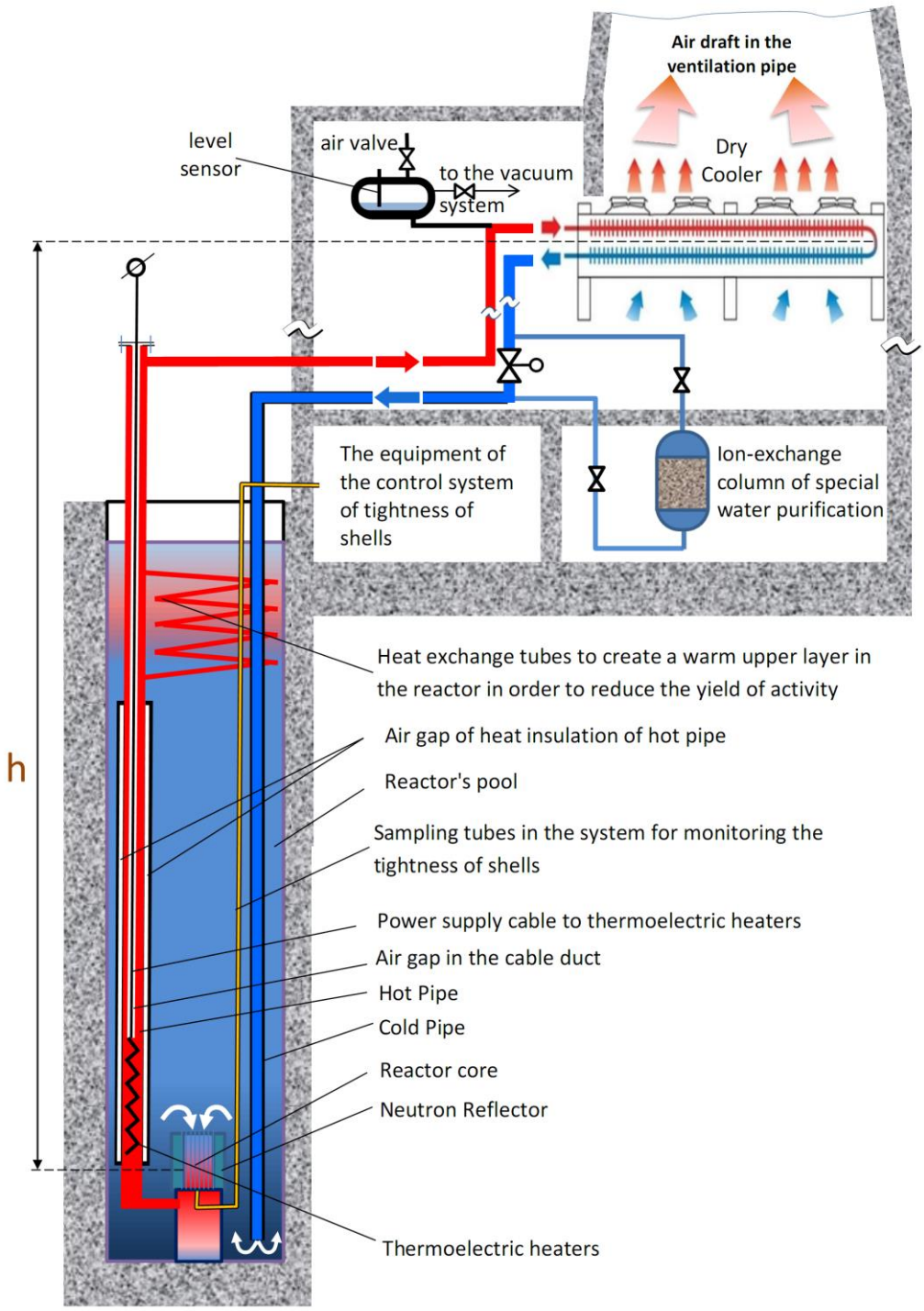


Figure 1 - Passive system for core cooling of a research pool-type reactor (red color indicates hot water, blue – cold water)

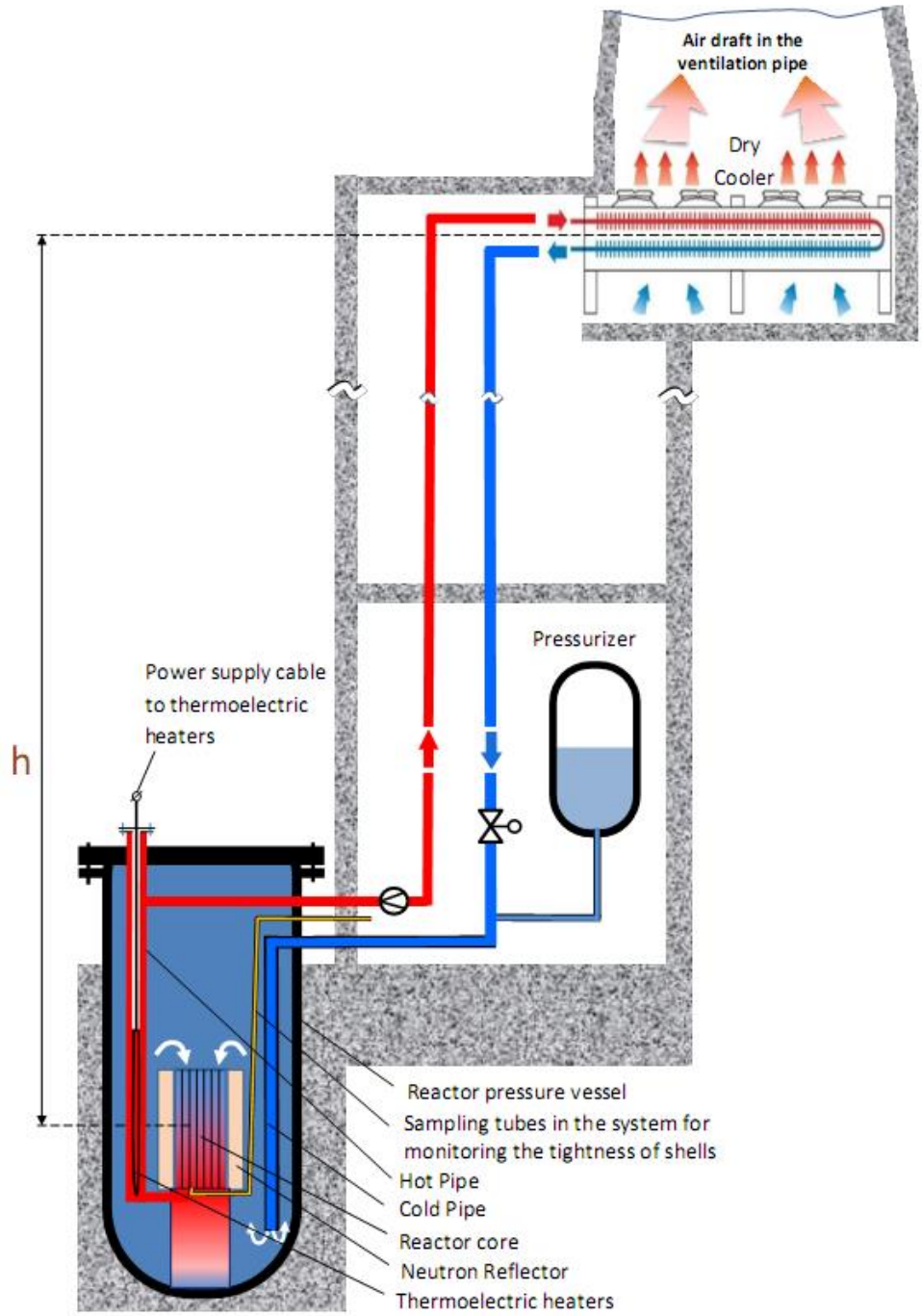


Figure 2 - Passive system for core cooling of a research vessel-type reactor (red color indicates hot water, blue – cold water)

Presented in Figures 1 and 2, simple technological schemes of the heat removal from the reactor core of research nuclear reactors provide key requirements for safety and reliability:

- The absence of electric pumps in the circulation circuit assures the absence of a situation with a dangerous decrease of heat removal from the reactor core in the event of electrical supply loss (blackout) or mechanical deterioration of moving parts of pumping equipment;

- Relatively large diameters of the circulation pipelines cause a low circulation speed in the circuit and, as a result, small hydraulic losses that allow a sufficiently high coolant flow in the reactor core even at a low natural circulation altitude;

- The use of dry cooling towers eliminates the drops and aerosol entrainment of the coolant into the atmosphere from the cooling circuit of the reactor installation, and the single-circuit scheme allows to obtain, the maximum average log temperature on heat exchangers, at heat exchange between water and air. The increase in the temperature of the coolant is particularly effective for research vessel-type reactor, which allows to significantly increase the power output at relatively small sizes of dry cooling towers;

- The downward movement of the coolant in the core of the reactor allows for each fuel assembly to install a sampling tube to control the leaktightness of the fuel gains, and the location of the sampling tubes of the system for checking the leaktightness in the reactor core does not interfere in any way with the operation of the reactor control and protection system, technological operations with fuel assemblies and experimental devices;

Especially it is necessary to emphasize the advantage of applying of the proposed technological scheme for research pool-type reactor as of the most simple and inexpensive ones:

- The low velocity of the coolant in the lifting pipeline leads to the fact that practically all radioactive nuclides of oxygen and nitrogen decay due to a short half-life at the time of outlet from the reactor vessel;

- The presence of the bulk of the water in the reactor at a relatively low temperature and in the upper part of the reactor at a slightly higher temperature makes it difficult to transfer the radioactive cooling liquid to the surface of the water. This temperature distribution is due to the fact that the pipeline with the descending coolant is not thermally isolated and cools the water in the reactor (Figure 1), whereas the main part of the surface of the pipeline for raising the hot coolant is almost completely isolated, with the exception of the pipeline section near the surface of the water. This upper section not only has no thermal insulation, but is also bypassed by heat exchange tubes that heat the upper part of the water in the reactor to create a stratification of the coolant by its density. All this creates conditions for the decay of radioactive parts directly in the coolant;

- A water purification system using ion-exchange columns in pool-type reactors can operate in a passive mode (without a pump), since the coolant is cooled to a sufficiently low temperature (below 50 °C), and the pressure difference on the coolant circulation column can be created by pressure drop on the control valve. However, if necessary, forced circulation of the heat transfer medium through the ion exchange columns can be made by a pump, which will allow to purify water in a stopped reactor.

The calculated temperatures for the heating of the coolant in the core of the reactor in the pool type range from 5 to 40 ° C, the average water temperature in the reactor is from 30°C to 40 °C with a heat output of 50 to 2000 kW at an effective circulating circuit height of 8 to 17 meters. It should be noted that for operation at high reactor power, it is required to provide a developed heat exchange surface for dry cooling towers and to ensure efficient natural convection of air in the heat exchangers because of the relatively low average log temperature difference between hot water and cooling air.

Limitations of the height of the circuit, and of the intensity of the natural circulation of the coolant inherent in the pool-type reactors, are eliminated when using the vessel-type reactor at elevated pressure (Figure 2).

The use of reactors of this type has great advantages over pool-type reactors:

- A large height of the circulation circuit, together with a large difference in the temperature of the coolant in the lifting and descending sections, provides practically any required thermal driving head of natural circulation and speed in the fuel assemblies;;
- High pressure in the circulation circuit causes an increase in the boiling point of the coolant, which makes it possible to provide high heating of the coolant in the reactor core while maintaining a large margin before the boiling crisis;
- The high temperature of the coolant in the circulation circuit ensures an increase in the efficiency of heat removal in dry cooling towers while maintaining the heat dissipation area in these heat exchangers. In addition, a large air heating creates a high traction force of natural convection in the ventilation pipe, which also increases the efficiency of dry cooling towers without electric fans.

Thus, such a heat sink system can ensure the operation of the research reactor at a sufficiently high power with a completely passive cooling system.

Some hydraulic characteristics of a possible natural circulation circuit

The calculated justification for the feasibility of realizing the technological scheme for cooling the reactor using natural circulation is facilitated by the fact that the temperature in the lifting and lowering areas remains practically unchanged and can be considered constant. Therefore, the thermal driving head of natural circulation is determined by a simple formula:

$$\Delta P = (\rho_c - \rho_h) \times g \times h$$

ΔP – thermal driving head of natural circulation in Pa;
ρ_c – density of water in the cold descending section of the circulation circuit in kg/m^3;
ρ_h – density of water in the hot ascending section of the circulation circuit in kg/m^3;
g = 9.8 – acceleration of gravity in m/s^2;
h – height of the circulation circuit in m (figure 1 or 2).

Calculated by this formula, the thermal driving head of natural circulation, depending on the difference between the temperatures of the coolant at the descending and ascending sections and the height of the circulation circuit, is shown in Figure 3.

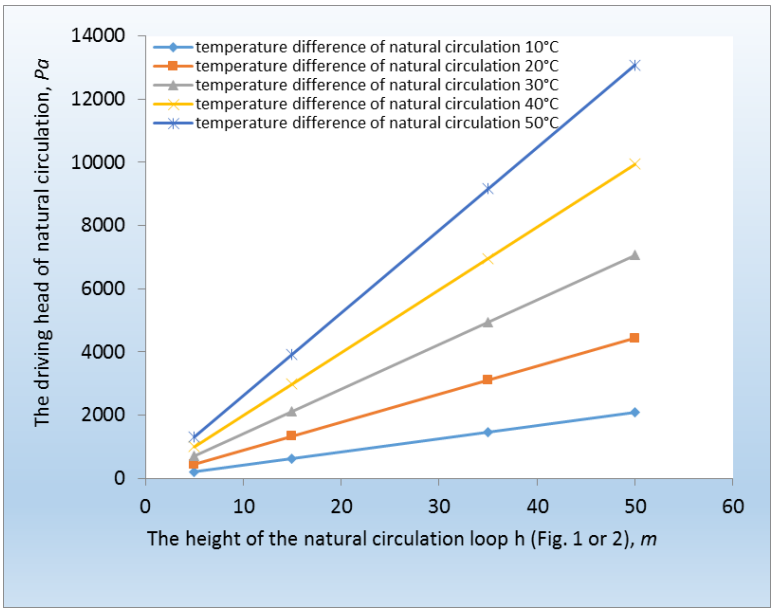


Figure 3 - The driving head of natural circulation, depending on the difference in temperature of the coolant at the descending and ascending sections and the height of the circulation circuit

The steady state circulation mode is set when the driving head ΔP of the natural circulation is equalized with the total hydraulic losses in the circuit including the head loss in the core of the reactor, the pipelines of the circulation loop, the control valves and dry cooling towers.

Hydraulic losses in the core of the reactor at low circulation rates were estimated by calculation for the types of fuel assemblies VVR-M2 and IRT-3M [2]. The results of the head loss for these fuel assemblies, depending on the average velocity of the coolant between the fuel elements, are shown in Fig. 4.

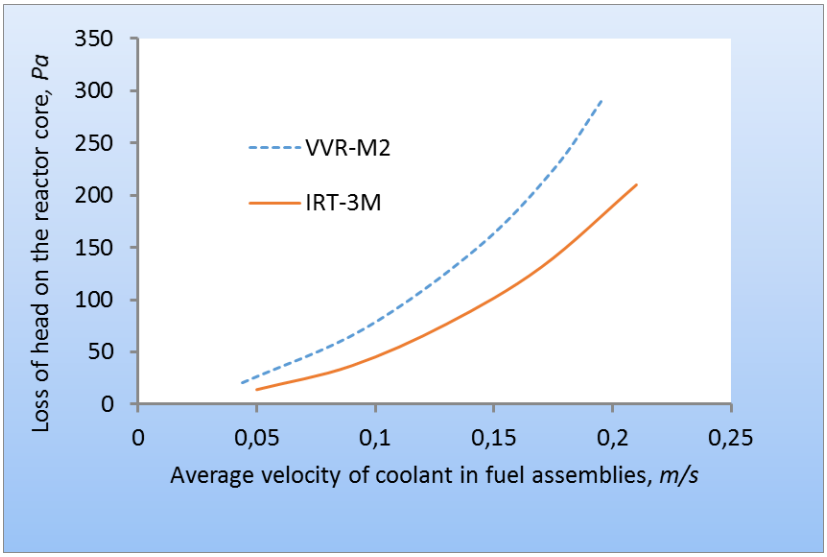


Figure 4 - Loss of head on the reactor core of research reactors with fuel assemblies VVR-M2 and IRT-3M, depending on the average velocity between the fuel elements

To evaluate the hydraulic losses in the part of the circulation circuit, excluding the reactor core, the following parameters were adopted:

- The length of the circulation pipelines of the circuit is 100 meters;
- Control valves with a coefficient of hydraulic resistance $\xi = 6.5$;
- 4 bends of pipelines;
- Total local hydraulic resistance $\xi = 8$.

For the analysis of hydraulic characteristics, three sizes of pipelines in the circulation circuit were considered - with a hydraulic diameter of 100, 200 and 300 mm. The calculated hydraulic characteristics of the circulation circuit for these diameters are shown in Fig. 5, depending on the volume flow, and in Fig. 6, depending on the speed in the pipelines.

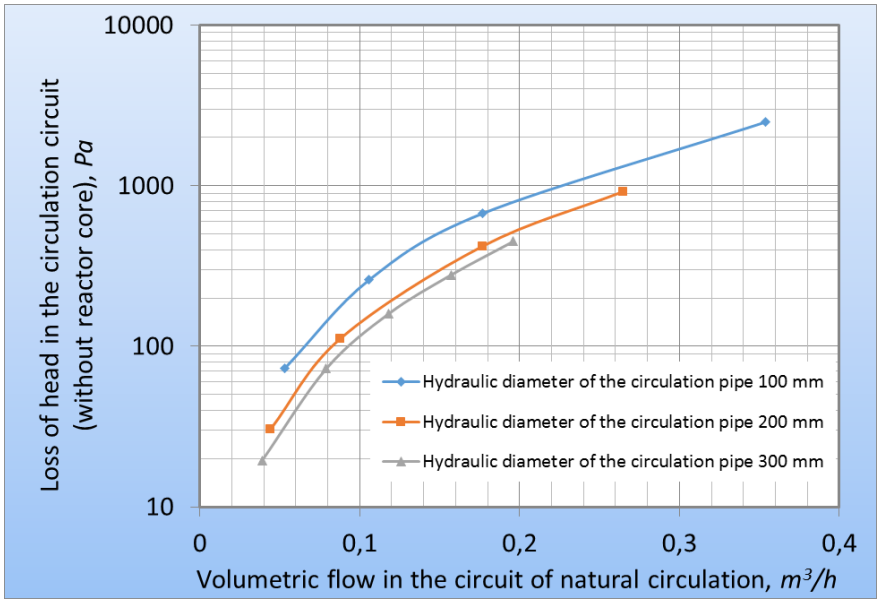


Figure 5 – Hydraulic losses in the circulation circuit (without reactor core) for sizes of pipelines with a hydraulic diameter of 100, 200 and 300 mm, depending on the volume flow

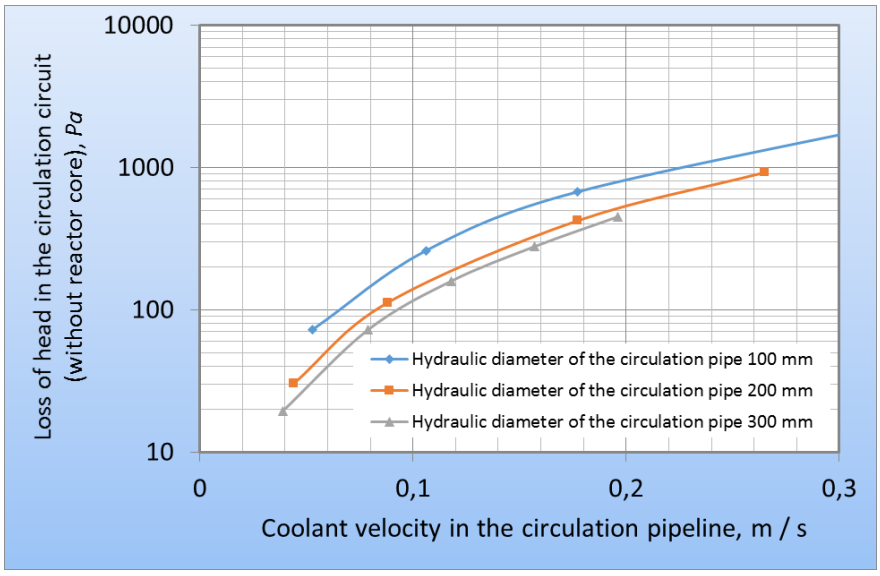


Figure 6 – Hydraulic losses in the circulation circuit (without reactor core) for sizes of pipelines with hydraulic diameters of 100, 200 and 300 mm, depending on the speed in the pipelines

From a comparison of the possible thermal driving head of natural circulation and the total hydraulic losses in the reactor core and the rest of the circulation circuit, it can be concluded that the proposed cooling scheme of the reactor is sufficiently effective. The ability to ensure high efficiency of heat removal from fuel elements makes it possible to create research reactor installaions with a sufficiently high power and to obtain a neutron flux density level of 10^{13}-10^{14} n / (cm^{2}s).

Raising the capacity of a research nuclear reactor with natural circulation to the required level

The descending movement of the coolant in the reactor core of the reactor according to the proposed technological scheme and the preferability of the maximum height of the natural circulation loop cause some disadvantages at the reactor start-up. In this mode, two additional systems shall be used, which are switched off after the reactor is brought to the specified power level:

- System for filling the circulation circuit with coolant;

- Preheating system for the ascending section of the natural circulation circuit.

For pool-type reactors, the main element of the first system is a vacuum tank with a level gauge connected to the upper point of the circulation loop. Vacuuming of this tank allows filling with coolant from the reactor tank all pipelines and dry cooling towers, while full filling is controlled by the appearance of water in this tank, from using a level gauge or a level sensor. For the tank-type reactors, the filling of the coolant circulation circuit is carried out from the system under pressure.

After the pipelines are fully filled with water, the heating section of the pipeline begins to heat up by tubular electric heaters located in the lower part of the riser, while the control valve on the cold line is closed. After warming up the lifting area, the control valves are opened and, in the presence of fans on dry cooling towers, they are put into operation. In the absence of fans, natural convection of cooling air develops due to draft in the ventilation pipe. This scheme of fully passive heat sink is preferable for vessel-type reactors at a high temperature of the coolant in the circulation circuit of the reactor. After the development of natural water circulation due to the initial heating of the coolant by tubular electric heaters, the operator raises the thermal power of the reactor core to the required level by the regulating bodies. When the reactor is brought to a predetermined power level, tubular electric heaters, whose energy release is approximately 1-10% of the reactor power level, are turned off. After this, the regulation of the circulation and heating of the coolant in the reactor core is controlled by the operator by opening or closing the control valve. The calculations of the dynamics of the reactor power output using RELAP5 confirmed the efficiency of this procedure with the disconnection of the tubular electric heating elements.

As an example, calculation of the dynamics of output to the nominal power level of 10 MW of the reactor with natural circulation is given, a simplified nodalization diagram of which is shown in Fig. 7. The core of the reactor is similar to the core of the research reactor SM-3, Fig. 8 [3]. The reactor specifications are given in Table 1.

Table 1. Basic data on the reactor with natural circulation with a capacity of 10 MW

Reactor characteristic	Value
Reactor type	Pressurized water-cooled and water-moderated reactor, Intermediate spectrum reactor with the neutron trap
Power output, [MW]	10
Max thermal neutron flux density, [$cm^{-2}s^{-1}$]	5×10^{14}
Fuel	Uranium dioxide, 90% enriched U-235
Core geometry	Square with the neutron trap in the center
External dimensions of the core, [mm]	420×420
The number of cells for fuel assemblies	32
Core height, [mm]	350
Hydraulic diameter of fuel assembly. [mm]	3.2
Coolant	Light water
Coolant flow rate, [m^3/h]	330
Core inlet temperature, [°C]	100
Heating of the coolant in the core of the reactor, [°C]	27
Core outlet pressure, [Pa]	3.0×10^{6}
Hydraulic diameter of circulation pipelines, [mm]	350
Height of circuit of natural circulation, [m]	35
Coolant velocity in fuel assemblies, [m/s]	1.15
Coolant velocity in circulating pipelines, [m/s]	0.97

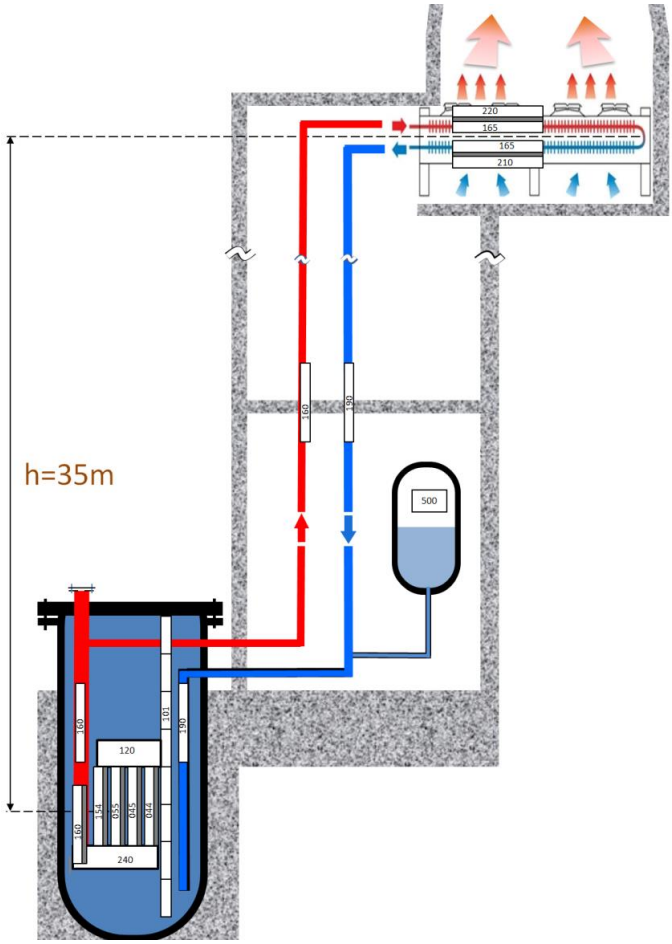
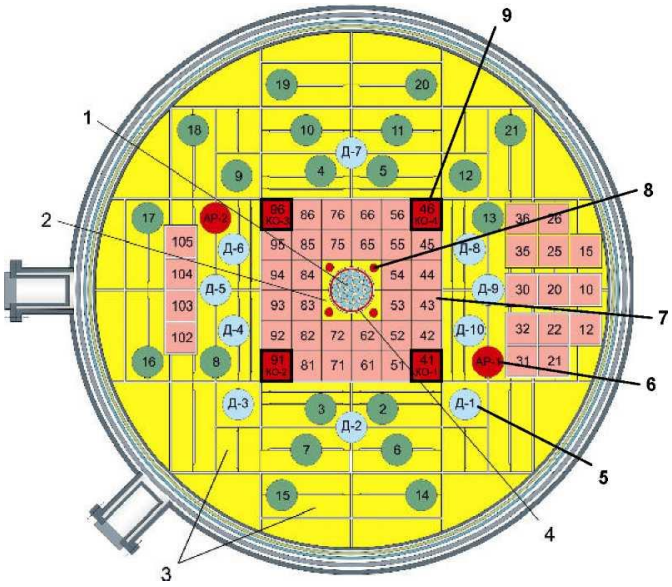


Figure 7 – Simplified nodalization diagram of the reactor with natural circulation with a capacity of 10 MW



1 - experimental channels in the neutron trap; 2 - beryllium insert; 3 - reflector beryllium block; 4 - central shim rod; 5 - experimental channel cell in the reflector; 6 -control rod; 7 - core cell with a FA; 8 - safety rod; 9 - shim rod

Figure 8 – The SM-3 core arrangement

Raising the capacity of a research nuclear reactor with natural circulation to the required level is performed with fully open valves, which always remain open. ***At the first stage, electric heaters with a capacity of 100 kW*** at the bottom of the pipeline lifting section are turned on, ***which operate for 2 hours, and then are switched off.*** After one hour of operation of the electric heaters, the reactor power is gradually increased from zero level to 10 MW for 1 hour.

Below is the dynamics of the change in the thermohydraulic parameters of the reactor in the initial stage of the reactor operation. Figure 9 shows the change in the average coolant temperature above the core (BR-120) and at the outlet from the core (BR-240).

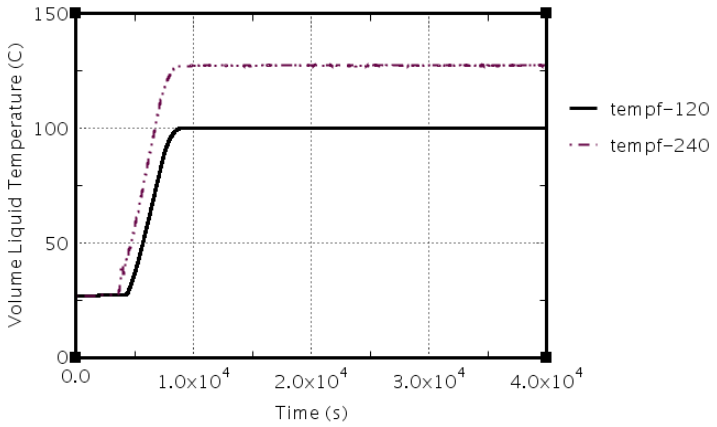


Figure 9 – The dynamics of the average temperature of the coolant above the core (BR-120) and at the core outlet (BR-240)

When the power of the fuel assemblies is increased and the natural circulation is not sufficiently developed, a short-term peak in the temperature of the maximum heat-stressed fuel element occurs (Fig. 10).

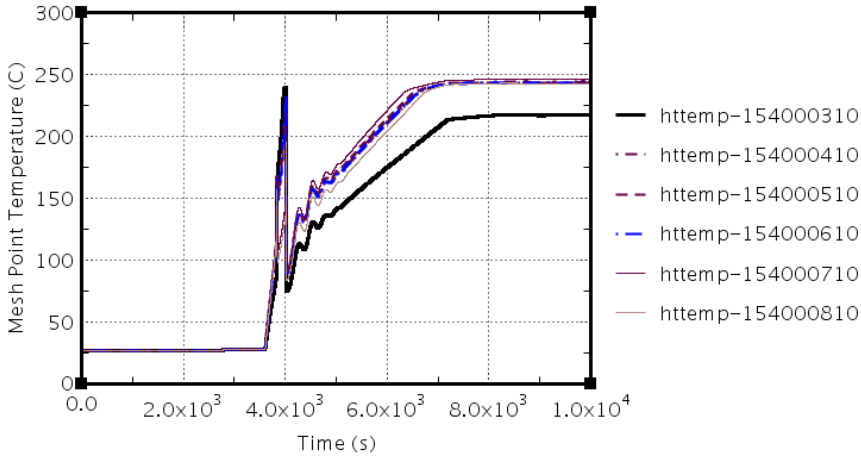


Figure 10 – Dynamics of temperature change of the maximum heat-stressed fuel element in high-altitude areas

The velocities of the coolant in the fuel assemblies of the reactor core are practically the same and do not depend on their power (Fig. 11).

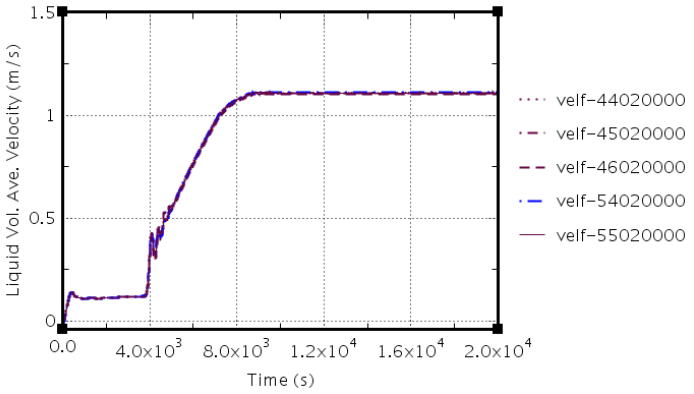


Figure 11 – The rates of the refrigerant in fuel assemblies of the reactor core

The change in refrigerant velocity in the lifting and lowering sections of D 350 mm pipelines is shown in Figure 12.

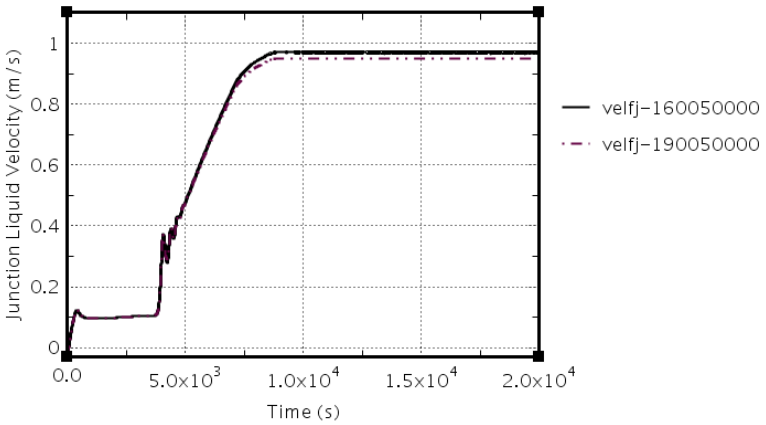


Figure 12 – Changing the refrigerant velocity in the lifting and lowering sections of the pipelines D 350 mm

The transition to the steady-state values of the coolant flow in the reactor occurs approximately 10,000 seconds after the start of the reactor output to the power. (Fig. 13)

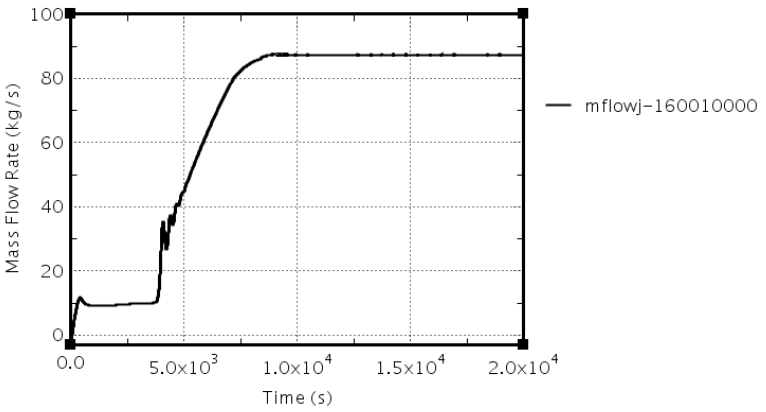


Figure 13 – Dynamics of the change in the coolant flow in the natural circulation circuit

Thus, the use of electric heaters in the lifting section of the pipeline of the natural circulation loop provides a descending movement of the coolant in the reactor core when it reaches a high level of thermal power.

The shutdown of the research reactor installation

The shutdown of the reactor installation is carried out either by the control devices or emergency protection system. After shutdown the reactor, if there are fans in dry cooling towers, they are also switched off. The low level of residual energy release in the reactor core does not imply any additional measures to remove residual heat from the fuel assembly, except of the natural circulation.

Drainage from the circulation loop of the pool-type reactor

Drainage of water (coolant) from the circulation loop of the pool-type reactor after its shutdown is effectuated by opening the air valve on a low-pressure vessel, which leads to the connection of this vessel to the atmosphere and drainage of water from the circulation pipelines into the reactor vessel under the gravity force.

Accidental situations

Due to the simplicity of the technological scheme, the number of accidental situations with a possible worsening of heat removal from the reactor core is severely restricted. For pool-type reactors, the circulation pipe may break or unintentionally close the valves on the pipeline of the circulation loop. For reactors of the vessel-type, an accidental situation may occur with the failure of the safety valve when the pressure rises or the situation with a sharp decrease in pressure due to the depressurization of the pipeline. When the reactor is operating at the nominal power level, a decrease in the flow rate of the coolant at the outlet from the reactor core below the set value leads to the operation of the emergency protection and the reactor stops. The same happens with increasing or decreasing pressure in the vessel-type reactor. In all these cases, the operation of the emergency protection system takes the reactor into a safe state.

Measures that need to be taken to reduce the negative radioactive impact during the depressurization of the circulation loop: pallets with leakage sensors and free draining of the coolant into the collecting tank for liquid radioactive waste should be installed in all the rooms where the circulation pipelines pass. Depressurization of the fuel assembly is identified by the system for checking the leaktightness of the gains, the reactor is shutdown manually by personnel or automatically and the defective fuel assembly is unloaded from the reactor.

Reactivity accidents with passive acting absorbing devices should be considered for specific cases of reactor core.

Conclusion

Analysis of the possibilities of using the principle of natural circulation for removing heat from research reactors of different purposes, showed the following:

1. For research reactors, it is possible to abandon forced circulation systems in favor of systems with natural circulation, which increases the stability of the reactor core cooling system and the safety of the reactor installation as a whole. Even a complete loss of electricity does not affect the intensity of the heat removal from the core of the reactor.
2. The parameters of the reactor installation with the capacity of ***10 MW*** and thermal neutron flux density $\boldsymbol{5\times10^{14}}$ ***neutrons/cm***$^{-2}\times$***s***$^{-1}$ with natural circulation of the heat carrier are given. After carrying out thermohydraulic calculations using RELAP5, it is shown that it is possible to ensure a downward movement of the coolant in the core of such a reactor using electric heaters in the lifting section of the circulation pipeline in the starting mode.

3. Absence of complicated electromechanical equipment with their duplication will allow to significantly reduce the cost of reactor installations and reduce the requirements for maintenance personnel.
4. The creation of simple and reliable nuclear reactors that do not require complex maintenance will make it possible to expand their scope, in particular, to use radiological centers for radiotherapy and the production of short-lived isotopes for the preparation of pharmacological preparations

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