Power-independent cooling system for a high-power research reactor

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Abstract

This paper presents the results of the analysis of a universal system for cooling of the research reactors core. The cooling system based on the passive principle of natural convection of the coolant. This cooling system develops the concept of a reactor plant presented in [1]. This Chapter presents à three-dimensional model, technological and design diagrams of a reactor unit. The examples of the numerical assessment of transients during operation in normal and accident modes are shown to substantiate the possibility of using such system in research reactors of medium and high power, providing a neutron flux of more than 1×10^{15} cm^{-2}/s.

A fundamental feature of the presented passive heat removal system is the absence of active elements in the cooling circuits, such as circulation pumps, shut-off and control valves, as well as the presence of an intermediate circuit with a non-radioactive coolant, made according to the principle of operation of a heat pipe (thermic syphon). Such design excludes the release of the radioactive coolant into the surrounding environment for any depressurization of the circuits. The cooling circuits include only vessels, piping and heat exchangers. The absence of elements with mechanical moving parts can significantly reduce the equipment failures probability and increase the reliability of the cooling system while reducing its cost.

The versatility of the proposed system allows using it for different reactor plants of a wide power range, designed for various nuclear research areas

Keywords:

Research reactors, passive system, reactor cooling system, heat pipe, thermic siphon, nuclear reactor safety, natural convection

1. Introduction

Thermal hydraulics is recognized as a key scientific subject in the development of innovative reactor systems. New nuclear reactor concepts are in development around the world with an emphasis on demonstrating their technical feasibility, economic competitiveness and improved safety performance.

One of the important directions in the development of reactor technology is the creation of research reactors with a high neutron flux density, which have high thermal engineering reliability and safety. Perhaps the most optimal approach to the development of modern research reactor facilities is demonstrated by the Argentine company INVAP, recognized today as one of the leaders in the research reactor industry. Its main principles are:

a) Safety comes first.

b) Maximum simplification of the design.

c) Minimizing technical risks by relying on proven technology.

d) Compliance with IAEA safety requirements and best international practices.

The proposed innovative concept of a research reactor facility is based on improving safety by **extremely simplifying the design of heat removal systems** from the core and using only passive systems in them to organize the coolant circulation in all cooling circuits.

2. Compliance of the proposed concept with the design principles of advanced research reactors

In accordance with international standards, in developing the concept, special attention is paid to the safety requirements of research reactors [3], as well as to the conceptual provisions and design principles for advanced research reactors for research centers described in [1] and listed below.

2.1 Reliability

2.1.1 Use of technical solutions and equipment applied on the existing reactors

The natural circulation of the coolant is used in reactor installations in normal operation systems, for example, in the VK-50 power reactor or in the IR-100 research reactor, but it is usually used to cool the core after the reactor shut down.

The physical basis of the mechanism of natural circulation of water and air is simple and reliable. Modern calculation codes make it possible to estimate with sufficient accuracy the dynamic processes of development of the natural circulation over time, even in complex circuits. In addition, a great experience has been gained in the creation of systems using natural circulation both with boiling and without boiling of the coolant.

The use in the cooling circuit of only passive equipment (pipelines, vessels, heat exchanger) without mechanical moving parts ensures high reliability of the entire system [3].

2.1.2 Selection the flow rate and coolant pressure drop of the in the core

The rate of increase in reactor thermal power, organized by the regulations, and taking into account the dynamics of the increase in flow rate in the circulation circuit, makes it possible to reliably ensure the temperatures of the fuel elements lower than the boiling point on their cladding surface. The stability of the circulation process during the lifting movement of the coolant in the core provides high thermal reliability even with accelerated output to power, resulting in boiling on the cladding surface of fuel rods. When this occurs, the circulation of the coolant in the circuit is intensified due to a decrease in the average coolant density in the lifting section, an increase in coolant flow through the reactor core, which ensures a return to the normal thermal-hydraulic parameters of the reactor if the power does not exceed the limit of normal operation. It should be noted that when fulfilling the requirements for the reactor core to ensure the self-protection by natural processes (the negative "void" reactivity effect), boiling of the coolant in the core leads to negative reactivity and a decrease in thermal power.

Thus, the natural processes on which the principles of operation of a research reactor facility are built, allow us to ensure high levels of heat engineering reliability and operational safety.

2.2 Safety

2.2.1 Placing the core under a high water level

The high level of water above the core in the reactor vessel provides radiation protection for personnel during transport and handling operations.

2.2.2 Ensuring the preservation of the water level above the core with water in the Loss Of Coolant Accident

The presence of passive natural circulation valves on the inlet and the outlet pipelines makes it possible to maintain the circulation through the core in the event of a pipeline rupture. This ensures safe cooling of fuel assemblies in any accident situation with depressurization of the primary circuit.

2.2.3 No surface boiling at cladding of fuel rods and on reactor core elements

The absence of wall boiling at cladding of fuel rods and core elements during normal operation is achieved by ensuring high efficiency of natural circulation due to the low hydraulic resistance of the circulation circuit as a whole and the difference in hydrostatic pressure of the descending and ascending flow sections, creating a driving pressure of natural circulation. This difference is proportional to the height of the contour of the natural circulation and the difference in the average densities of the coolant in the descending and ascending flow sections. An increase in the heating of the coolant in the reactor core leads to an increase in this difference in average densities and to the circulation intensification.

Compliance with the regulations on the speed of the reactor output to the nominal power level and power limitation allows fulfillment of the requirement for the absence of boiling on the surface of cladding of fuel elements and core elements, if this requirement is present.

2.2.4 *Passive safety systems*

An important factor in the safety of a nuclear reactor with natural circulation is that all systems that provide heat removal in both normal operation and accident conditions are completely passive and do not depend on power supply. This is the main distinguishing feature of the proposed concept of a research nuclear reactor. Along with the presence of negative feedbacks on reactivity and sufficient efficiency of the elements of the control and protection system, this ensures the maximum level of safety and reliability of the reactors under consideration

2.3 Flexibility

The universality of the cooling system provides the possibility of implementing various layouts of the reactor core for a selected size of the reactor vessel. The access to the channels of the reflector located in the pool around the reactor vessel provides convenient irradiation and replacement of irradiation devices even during reactor operation

2.4 Efficiency

In [1], the following criteria are proposed as indicators of the efficiency of a nuclear research reactor:

- high level of neutron flux density in experimental devices of the reactor;
- deep fuel burnout in unloading assemblies;
- high "reactor quality" is defined as the ratio of its maximum neutron flux to its heat output;
- a variety of experimental spaces.

All these listed qualities are fully inherent in the proposed concept of a research reactor unit.

2.5 Simplicity

Extreme simplicity of the circulation loop in the passive cooling system provides ease of maintenance of the reactor and the absence of the need for scheduled preventive maintenance of complex equipment of the reactor forced cooling system (pumps, shut-off and control valves, check valves) due to their absence. In [1], the advantages of the technological scheme of a passive heat sink in the organization of the downward coolant movement in the core of the research reactor using natural circulation are shown. **It is very important that the simplification of the circulation loop makes it possible to carry out a high-quality and reliable analysis of accident situations in order to substantiate the safety of the reactor unit, and the number of possible accidents is reduced drastically.**

3. Implementation of the concept of passive cooling for a research reactor with a pressure vessel

The defining parameters for organizing the natural circulation of the coolant are the **hydraulic resistance** of the circulation circuit and the difference in the **average densities** of the coolant in the ascending and descending sections. The difference in average densities is directly related to heating the coolant at a certain heating power and mass flow rate. With regard to a reactor facility, the main share of hydraulic losses in the circulation loop should be in the core to ensure the maximum velocity in the fuel assembly. All other hydraulic losses in the primary circuit can be minimized by increasing the flow area and reducing the speed of the coolant in sections of the circuit, as well as due to the absence of shut-off and control valves.

Based on the balance between the driving head of natural circulation, determined by the difference between the average densities of the coolant in the ascendant and descendant sections, and the total hydraulic losses in the primary circuit, it is possible to estimate the required height difference *h* between the heat exchanger and the core is determined from a simple relation:

$$h = \frac{\Delta P_{\text{р}}\ (G) +\ \Delta P_{\text{гт}}\ (G) +\ \Delta P_{\text{хт}}\ (G) +\ \Delta P_{\text{то}}\ (G)}{\Delta\overline{\rho}\left(G, N_p^{max}\right) \cdot g}$$

where

N_p^{max} –maximum power of the reactor unit, W;

G – required flow rate of the primary coolant at the power N_p^{max} , kg/s;

$\Delta P_{\text{р}}\ (G)$ – head loss in the reactor, including the core, at the coolant flow rate in the primary loop G , Pa;

$\Delta P_{\text{гт}}\ (G)$ – head loss in a pipeline with a “hot” coolant at a coolant flow rate in the primary circuit G , Pa

$\Delta P_{\text{р}}\ (G)$ – head loss in a pipeline with a “cold” coolant at a coolant flow rate in the primary circuit G , Pa;

$\Delta P_{\text{р}}\ (G)$ – head loss on the heat exchanger at the coolant flow rate in the primary circuit G, Pa;

$\Delta\overline{\rho}\left(G, N_p^{max}\right)$ – difference between the average densities of the coolant in the lifting and lowering sections of the primary circuit;

g – gravitational acceleration, m/s^2.

To demonstrate the potential of a research reactor with a passive heat removal system, a pressurized reactor unit is considered. A 3D model of such reactor is shown in Figure 1.

As a non-limiting example, a reactor is considered, the vessel design of which is similar to that of the PIK research reactor vessel (Gatchina, Russia) with an external heavy-water reflector in a zircaloy reservoir located in a tank (pool) with light water

The final heat absorber (coolant) of the reactor plant can be either heated (evaporated) water in an open reservoir or circulating atmospheric air. In both cases, the system can operate using only natural convection of water in the primary and secondary circuits, as well as natural convection through the air heat exchanger (if a chimney is used) or water (if a reservoir is used) in the final absorber circuit. Such conception largely simplifies the research the reactor plant and increases its reliability.

The use of an open reservoir as the final absorber reduces the installation cost. However, the problems with deposits on the surface of the heat exchange tubes may arise. Therefore, a design with a ventilation tube is considered as a basic option for heat transfer to the environment (air).

3.1 Possibility of creating a universal cooling loop for nuclear research units

The using of cooling system of a reactor plant to remove heat of a given power allows it to be used as a universal system for reactor plants for various purposes with identical thermal power. The reactor core is formed on the basis of the assigned tasks and can be easily transformed in case of changing the directions of scientific and technical activities of the reactor plant.

When using the design of a reactor with a core in the vessel and an external reflector in the pool (Figure 1), it is possible to easily adjust the configuration of the irradiation volumes in the reflector to the assigned tasks and, if necessary, change the number, the location and the size of the irradiation channels.

3.2 Three-dimensional model of the reactor cooling loop

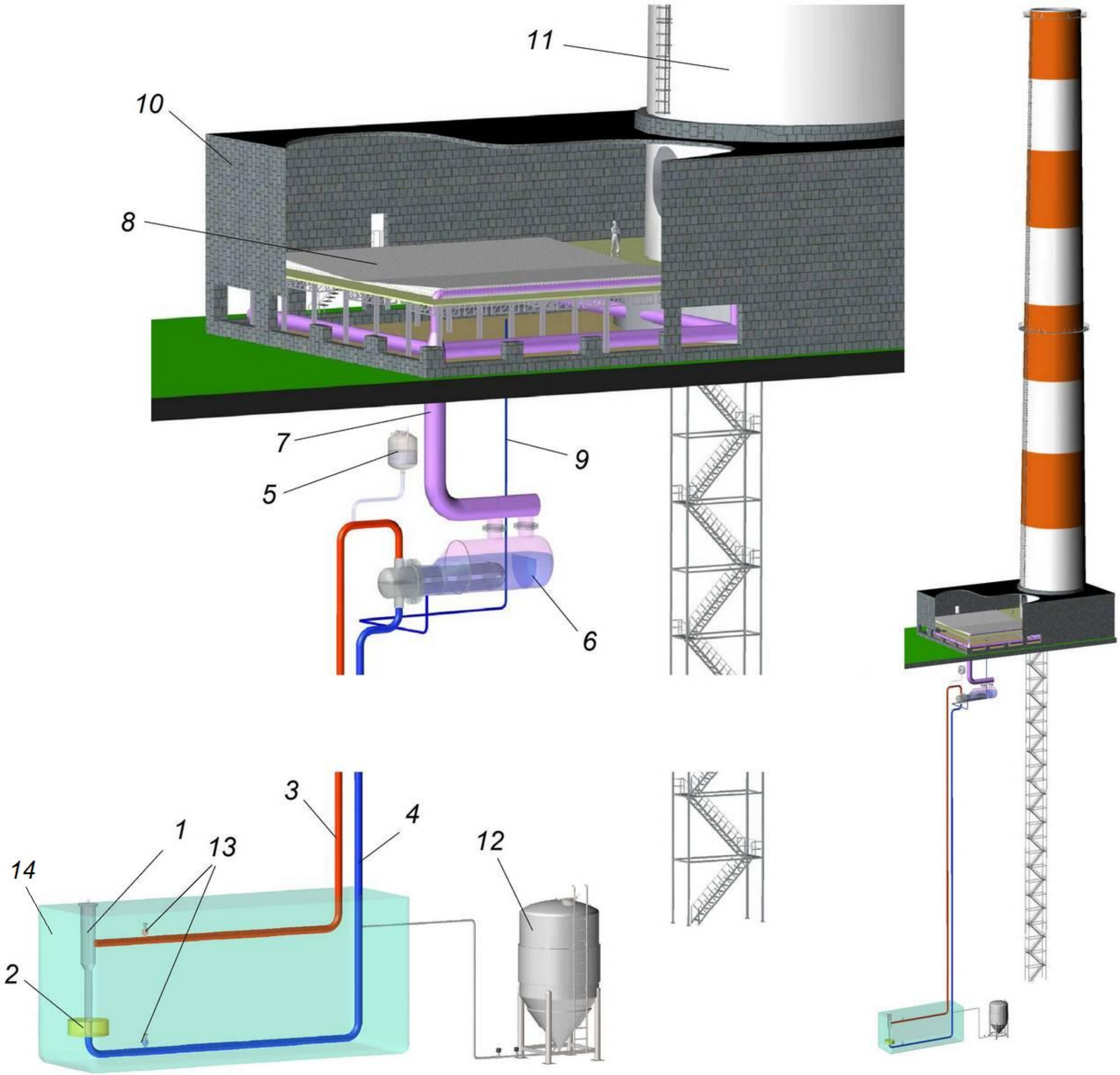
To visualize the proposed concept, Figure 1 shows a 3D model of the cooling system for a 25 MW reactor unit.

3.3 Basic parameters of the reactor plant

The main thermal and neutron-physical parameters of the considered research reactor plant with a capacity of 25 MW are presented in the Table 1.

Table 1. Main parameters of a reactor with natural circulation of the coolant

Reactor characteristics	Value
Reactor type	Pressurized water-cooled reactor with intermediate neutron
Power, MW	25
Maximum neutron flux density, $cm^{-2}s^{-1}$	$1{,}17\times10^{15}$
Fuel	UO_2, 20% of U-235 enrichment
Core geometry	Cylindrical shape with a neutron absorber in the center
Number of cells for fuel assemblies, pcs	30
Fuel assembly type, pcs :	VVR-KN
5-pipes, pcs	6
8-pipes, pcs	24
Number of cooling circuits	3
Primary coolant	Light water
Core diameter, mm	Ø480
Core height, mm	600
Primary circuit coolant flow, [$t\ h^{-1}$]	463,3
Core inlet temperature [°C]	118
Core outlet temperature, [° C]	164
Coolant heating in core, [° C]	46
Maximum fuel rod temperature Максимальная температура твэлов, [° C]	242
Pressure loss in core, [Pa]	14100
Core outlet pressure, [Pa]	5.5×10^{6}
Hydraulic diameter of primary circuit lines [mm]	400
Height of the primary circuit with natural circulation [m]	80



1 - reactor vessel; 2 - neutron reflector; 3 - outlet (hot) pipeline; 4 - inlet (cold) pipeline; 5 - pressurizer; 6 - heat pipe (steam generator); 7 - steam pipe; 8 - air heat exchanger; 9 - return of the heat pipe condensate; 10 - air heat exchanger body; 11 - exhaust pipe; 12 – monte-jus; 13 - valves of the shortened natural circulation circuit; 14 - reactor pool

Figure 1 - Three-dimensional model of cooling circuits of a 25 MW reactor with heat removal by natural circulation: on the right - general view in compliance with the scale, on the bottom left - enlarged fragments of the reactor pool and on the top left - a heat pipe (thermosyphon) circuit with a chimney

Table 2 shows the characteristics of fuel assemblies adopted for the computational analysis:

Table 2. Characteristics of VVR-KN fuel assemblies [3]

Parameter	Value
235U enrichment, %	19,7
Uranium density, g×cm^{-3}	2,8
235U weight in fuel rod, g	
8-pipes	250
5-pipes	199
Number of fuel rods	
8-pipes	8
5-pipes	5
Fuel rod thickness , mm	1,6
Core thickness , mm	0,7
Fuel rod cladding thickness , mm	0,45
Heat transfer area surface , m^2	1,34

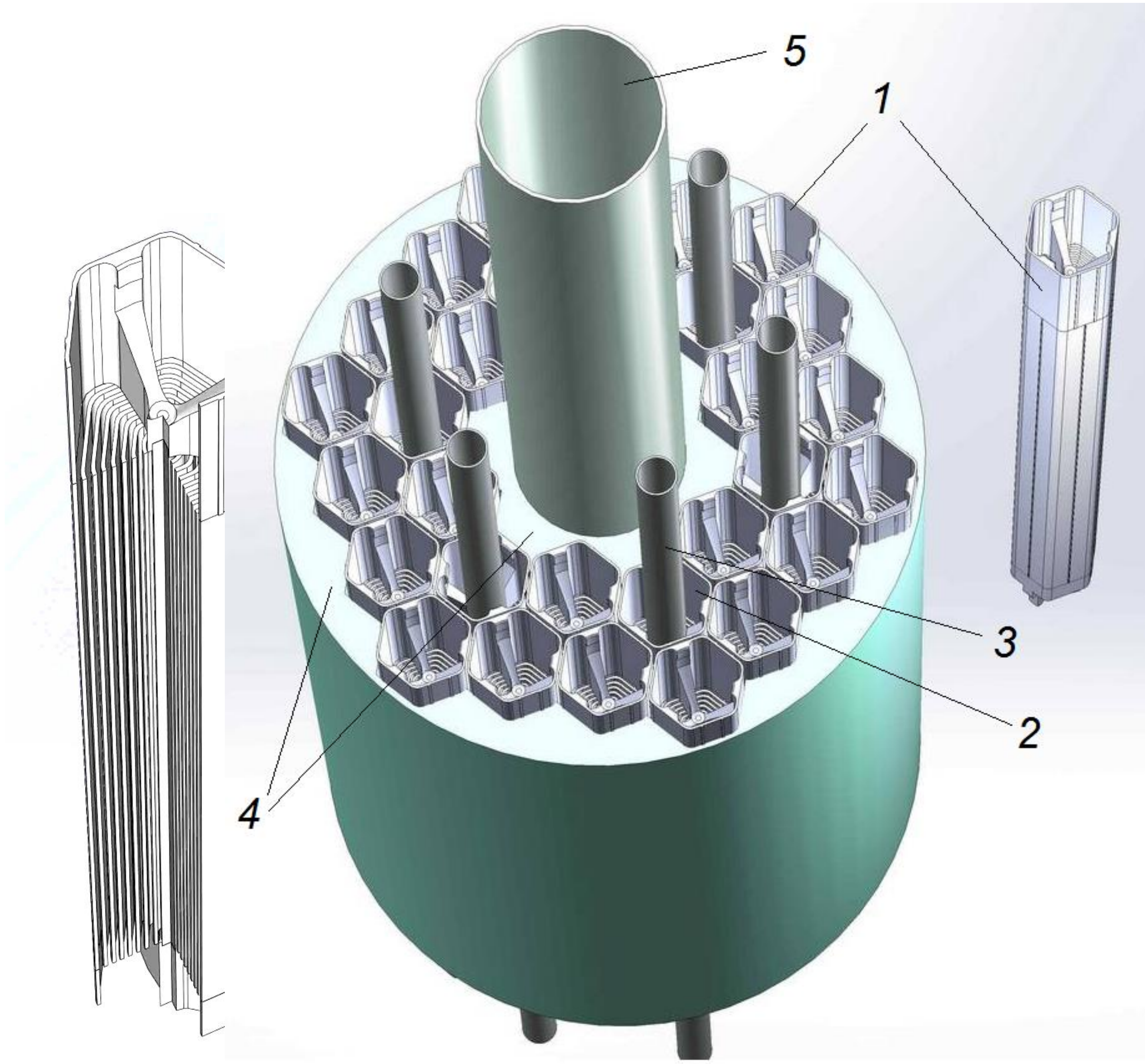
3.4 Reactor core

As a reactor core with a central moderation cavity (Figure 2), a design of 30 VVR-KN fuel assemblies, consisting of 24 fuel assemblies containing 8 rods each one (1), of six fuel assemblies containing 5 rods each one (2) and of control rods located in the center (3), is considered. The material of the displacer (4), the reactor vessel and the central channel (5) can be zirconium alloy.

3.5 Cooling system of the reactor plant

The proposed for use three-circuit scheme of heat removal from the core to an external heat sink provides not only the efficiency, but also the radiation safety. The coolant circulation mechanism in all circuits is based on a passive principle using gravity. In addition, it is assumed that there are no pumps and shut-off and control valves, which drastically reduces the number of possible accidents and the need for preventive maintenance. Heat dissipation is carried out using the following circuits:

- heat removal loop from the core with natural circulation;
- intermediate circuit (heat pipe or thermosyphon);
- heat transfer loop to the final absorber.



1 –fuel assemblies with 8 rods of the VVR-KN type; 2 - fuel assemblies with 5 rods of the VVR-KN type; 3 – control rods channel; 4 - displacer; 5 - central moderation cavity

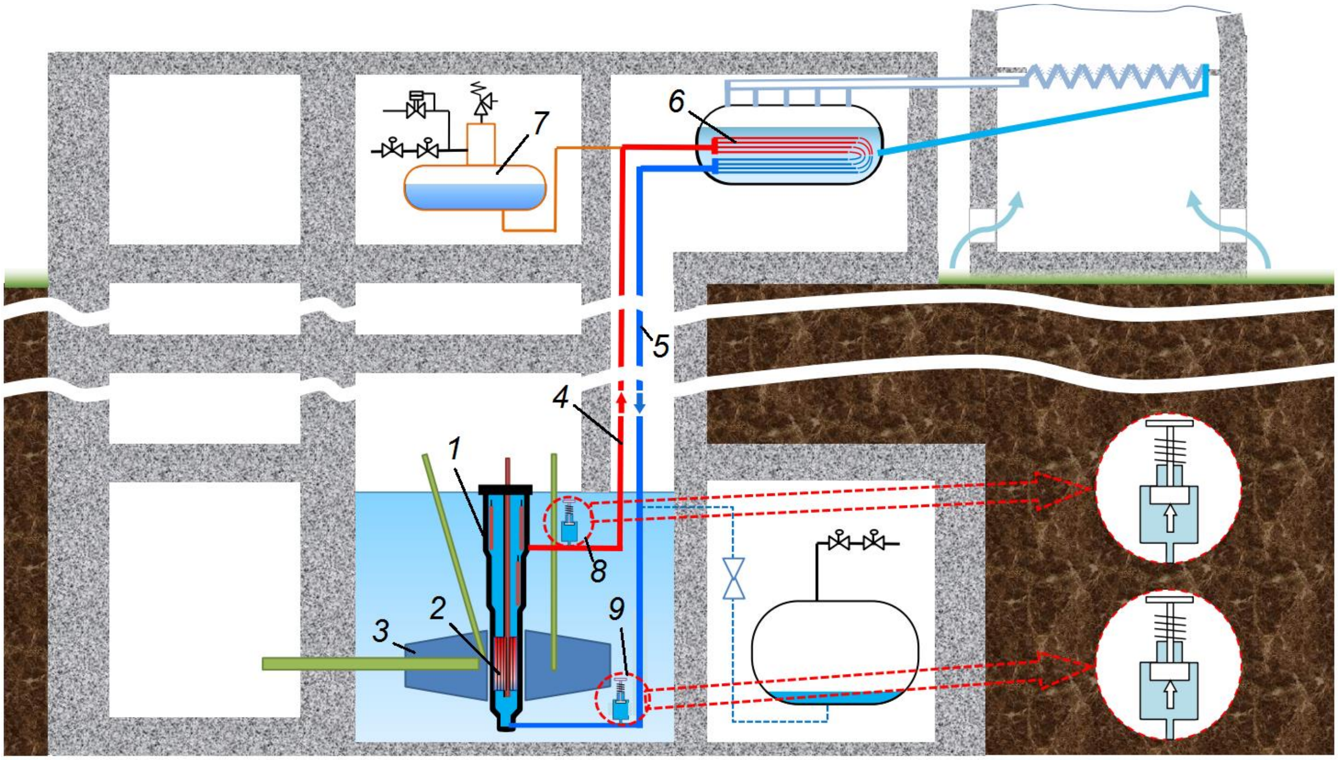
Figure 2 - Three-dimensional model of the reactor core

3.5.1 Natural circulation circuit of core heat removal

To provide the heat removal from the core to the heat pipe, the primary circuit with natural circulation is used. It includes the reactor vessel 1, the outlet (hot) pipeline 4, the inlet (cold) pipeline 5, the heat pipe heat exchanger 6 and the pressurizer 7 (Figure 3).

The required heat removal capacity during natural circulation is achieved by minimizing the hydraulic resistance in the primary circuit and a large height of the circulation circuit. The minimization of the hydraulic resistance is ensured by the large diameter of the circulation pipelines (4 and 5) and the complete absence of shut-off and control valves.

Most of the hydraulic losses in the primary circuit occur in the section of the core 2. On the inlet 5 (lower) and outlet 4 (upper) pipelines in the pool, short natural circulation valves 8 and 9 are installed, which are kept closed by the high pressure in the circuit (Figure 3)



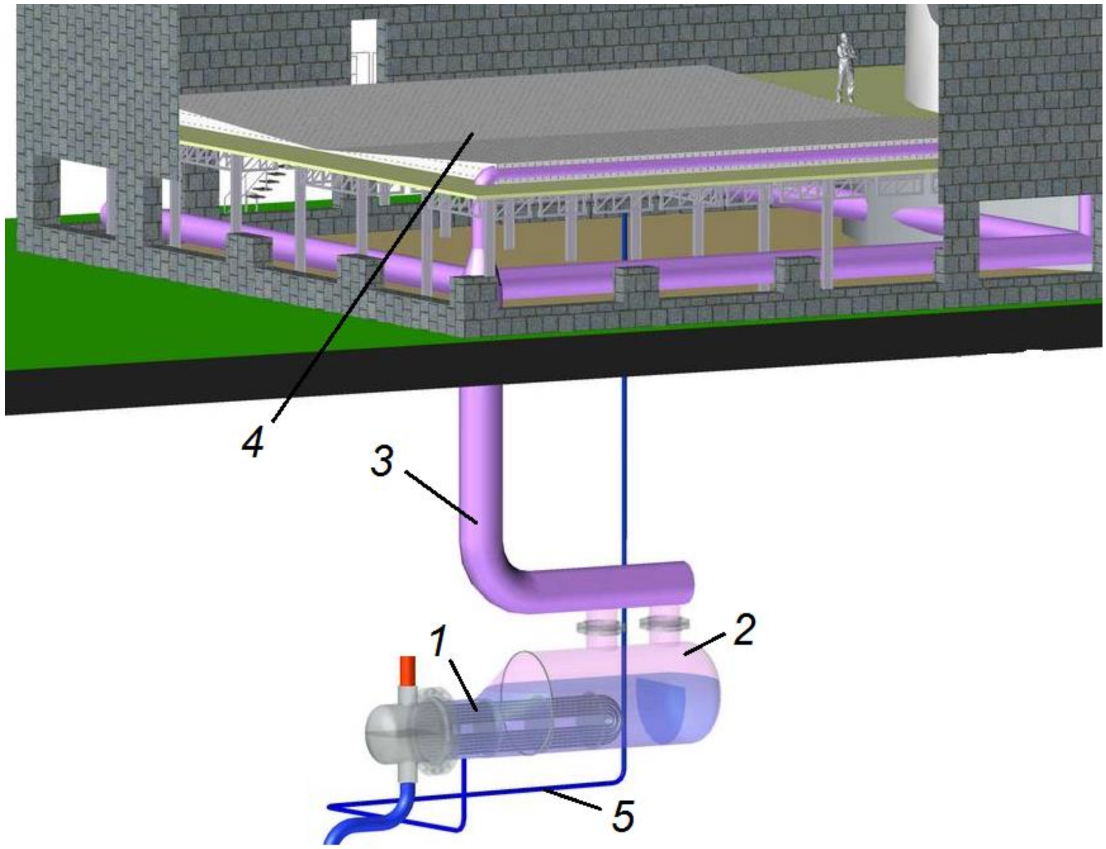
1 - reactor vessel; 2 – reactor core; 3 - neutron reflector; 4 - outlet (hot) pipeline; 5 - inlet (cold) pipeline; 6 - heat pipe steam generator; 7 - pressurizer; 8 - upper valve of the shortened natural circulation circuit; 9 - lower valve of the shortened natural circulation circuit

Figure 3 - Scheme of circulation of the primary coolant during reactor operation at power

After the reactor shut down, the circulation is maintained due to the residual heat release, but the circulation rate is significantly reduced.

3.5.2 Intermediate circuit (heat pipe)

An important role in the implementation of the concept of a safe reactor is played by the intermediate circuit between the primary circuit and the heat transfer circuit to the final absorber, made in the form of a heat pipe (Figure 4). The presence of this circuit prevents the risk of the release of radioactive coolant into the environment in any accident with a rupture of the pipelines of the cooling circuits.



1 - steam generator tubular; 2 - steam generator vessel ; 3 - steam line; 4 - air heat exchanger-condenser; 5 - condensate return pipeline

Figure 4 - Intermediate circuit diagram (heat pipe)

A heat pipe is a heat transfer device capable of transmitting large heat powers at low temperature gradients. It is a sealed structure, partially filled with a coolant. In the heated part 1 (in the heating zone, or the evaporation zone), the liquid heat carrier evaporates with the absorption of heat 2, and in the cooled part (the cooling zone, or the condensation zone 4), the produced steam flowing through the steam line 3 from the evaporation zone condenses with the release of heat. The movement of steam occurs due to the difference in saturated steam pressure, determined by the temperature difference in the zones. The return of the liquid to the evaporation zone is carried out through the pipeline 5 due to the force of gravity.

3.5.3 Third circuit (heat transfer to the final absorber)

Контур передачи тепла конечному поглотителю представляет собой обычную вытяжную трубу большого размера, в которой создается циркуляция охлаждающего воздуха через воздушный теплообменник-конденсатор за счет разности гидростатического давления подогретого воздуха в трубе и атмосферного воздуха (Рисунок 5). Следует отметить, что в корпусе воздушного теплообменника циркулируют только нерадиоактивные среды (пар и конденсат тепловой трубы, атмосферный воздух).

The third circuit transferring the heat to the final absorber is a conventional large chimney, in which the cooling air circulates through the air heat exchanger-condenser due to the difference in the hydrostatic pressure of the heated air in the tube and the ambient air (Figure 5). It should be noted that only non-radioactive media (steam and condensate of the heat pipe, atmospheric air) circulate in the air heat exchanger body.

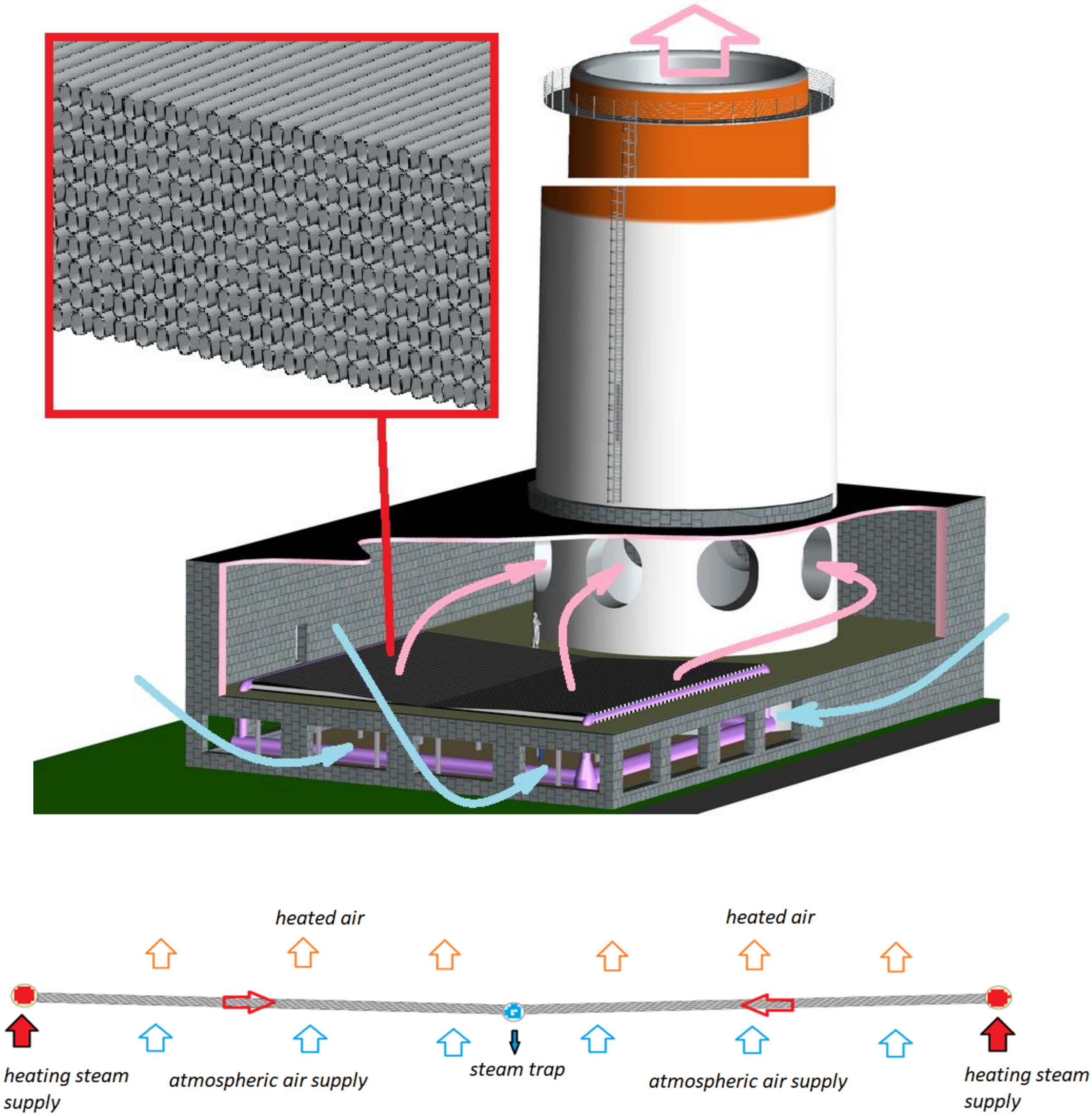


Figure 5 - Scheme of heat transfer to the final absorber, a fragment of the section of the tubular air heat exchanger (above) and the scheme of circulation of the coolant in the air heat exchanger (below)

3.6 Intra-pool natural circulation circuit and refueling of fuel assemblies

An exception to the principle of rejection of shut-off and control valves in the cooling circuits of the reactor plant is the use of in-pool natural circulation valves operating on a passive principle and opening when the coolant pressure in the primary circuit decreases (Figure 6). The opening of the valves ensures the automatic creation of a shortened natural circulation circuit through the core inside the reactor pool in the event of a rupture of the circulation pipelines or a pressure decrease in the primary circuit in a shutdown reactor.

The reactor operates under pressure, therefore, to increase the primary circuit pressure before reaching the power, the rod of the sealing element, for example, with a sealing ball surface, must be mechanically lifted using a rope, after which the circuit is sealed from water in the pool. This will avoid leakage of the coolant from the primary circuit when the pressure increases up to the nominal level and the reactor operates at power. After that, the cable must be released, ensuring spontaneous opening of the valve when the pressure in the primary circuit decreases, for example, during the pipelines rupture.

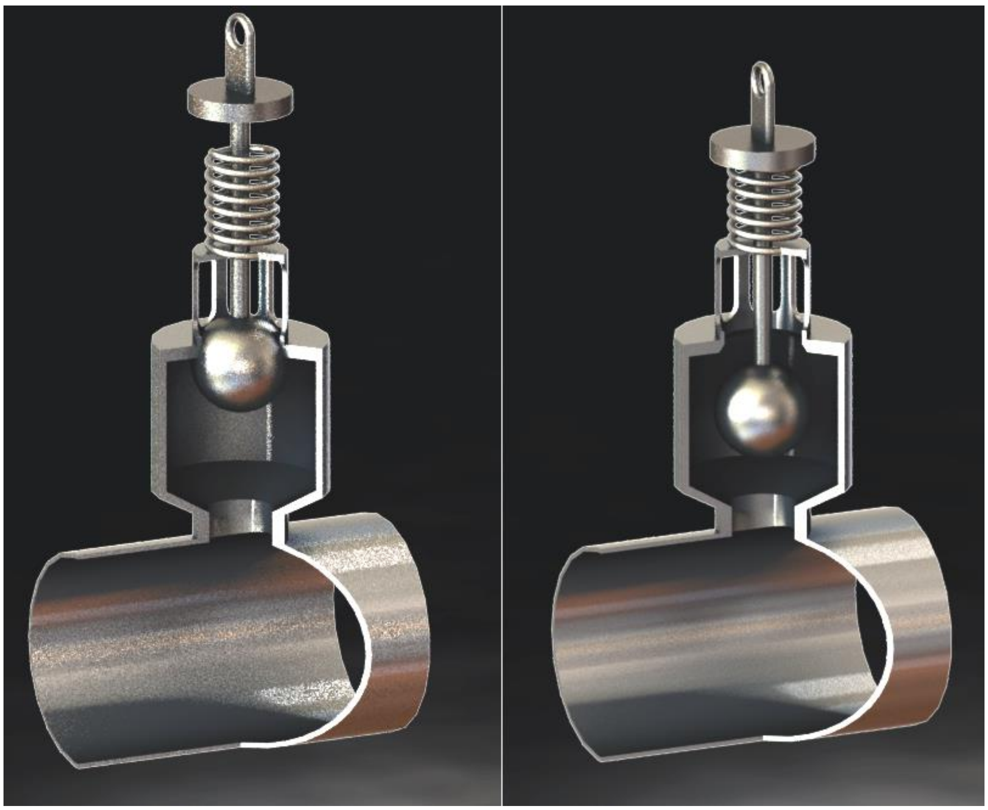


Figure 6 –Sectional view of the valve of the shortened natural circulation circuit: on the left - in a closed state, on the right - in an open state

The valves also open during normal operation, if there is a need to reduce the coolant level in the primary circuit to overload the reactor with an open lid. For this, the pressure in the pressurizer is released, which leads to the downward movement of the sealing element under its own weight, thereby opening the circulation circuit for cooling the fuel assemblies. During this operation, the coolant from the upper part of the primary circuit and the pressurizer is drained directly into the reactor pool (Figure 7, top). The organization of in-pool natural circulation makes it possible to remove the lid of the reactor vessel to carry out refueling of the fuel assemblies. After that, the surplus coolant is drained into the monte-jus after opening the corresponding valve

(Figure 7, below), then the reactor vessel cover is installed in place, and the sealing elements of the valves of the shortened circuit are transferred to a closed state, after which the compressed gas immediately starts squeezing the coolant out of the monte-jus into the primary circuit pipelines and the pressurizer. During this operation, the purge valve in the upper part of the primary circuit must be open, allowing the circuit to be completely filled with water before the pressure increase.

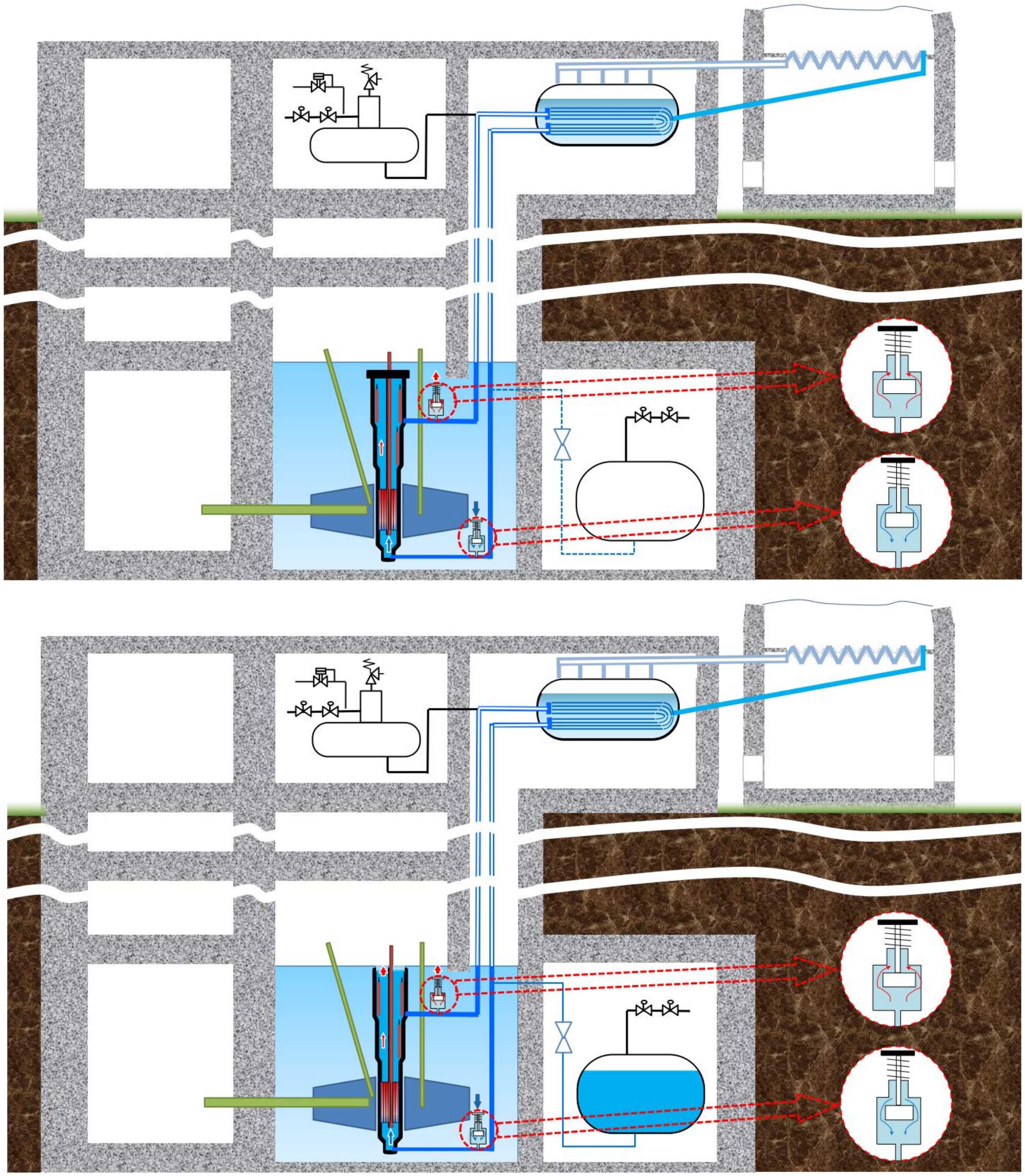
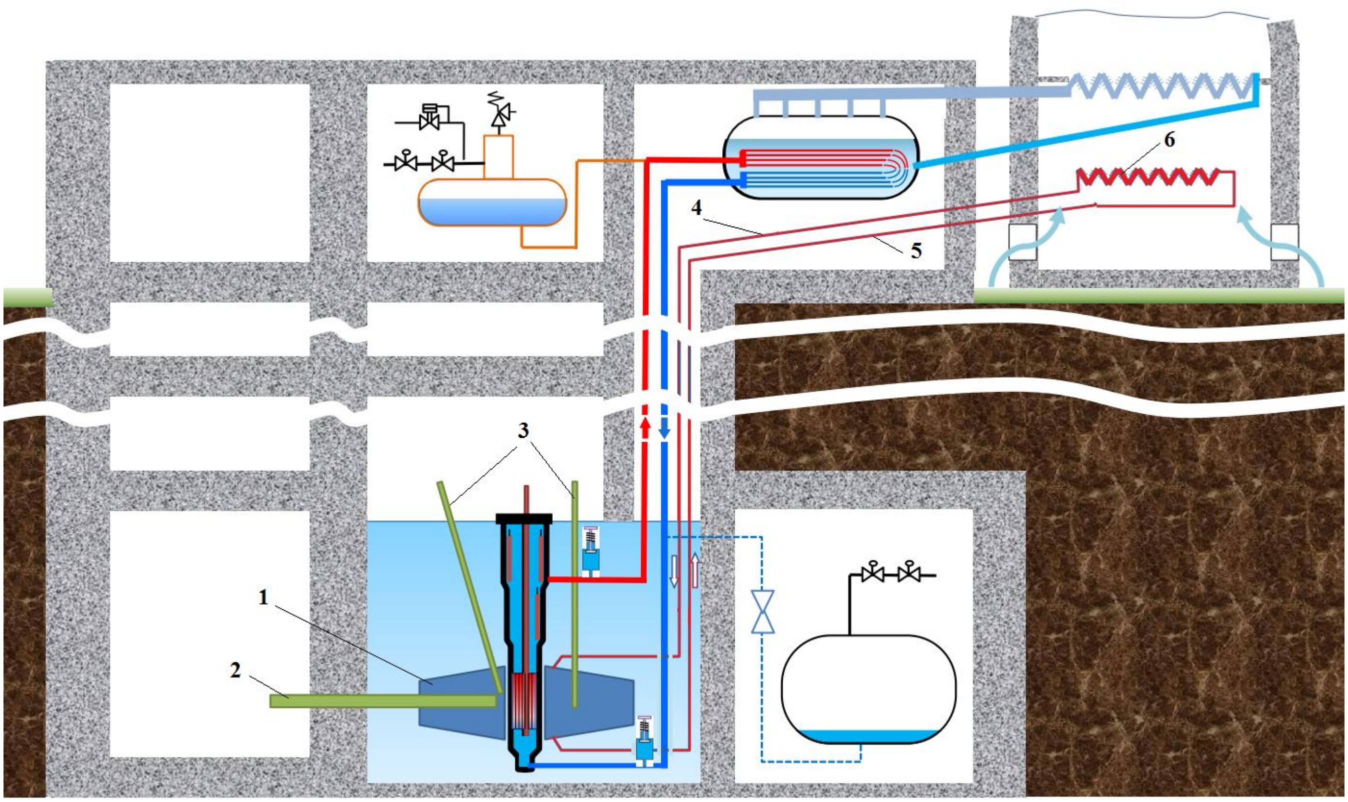


Figure 7 - Scheme of the operation of the intra-pool natural circulation circuit during the reactor refueling: at the top - draining the primary coolant into the pool; below - coolant drain into Monte-jus

3.7 Cooling system for heavy water reflector

As a good external reflector of neutrons, for example, beryllium or heavy water can be used. Due to the rather high neutron flux density during fluence gain, leading to the destruction of the material, the beryllium reflector blocks will have to be replaced quite often. Therefore, the preferred option for the reflector is a heavy-water reflector in a zirconium alloy housing, similar to the OPAL reactor reflector, which has proven itself well in operation. Moreover, this will allow organizing a convenient heat removal from the reflector using the same natural circulation mechanism with the existing chimney (Figure 8).



1 - heavy water reflector in a zirconium housing; 2 - horizontal experimental channel; 3 - vertical experimental channels; 4 - circulation pipeline with heated coolant; 5 - circulation pipeline with cooled coolant; 6 - air heat exchanger

Figure 8 - Diagram of the use of a chimney for organizing heat removal from a heavy-water reflector during the reactor operation at power

4. Calculations to confirm the feasibility of the proposed concept

4.1 Neutron-physical calculation of the core

The neutron-physical calculation of the reactor is carried out for the geometry shown in Figure 9 using the MCU program [5] for a reactor power of 25 MW. Heavy water was used as a reflector. In the Figure, the numbers indicate the position of 4 control points, the calculated parameters of the neutron flux for which are indicated in Table 4

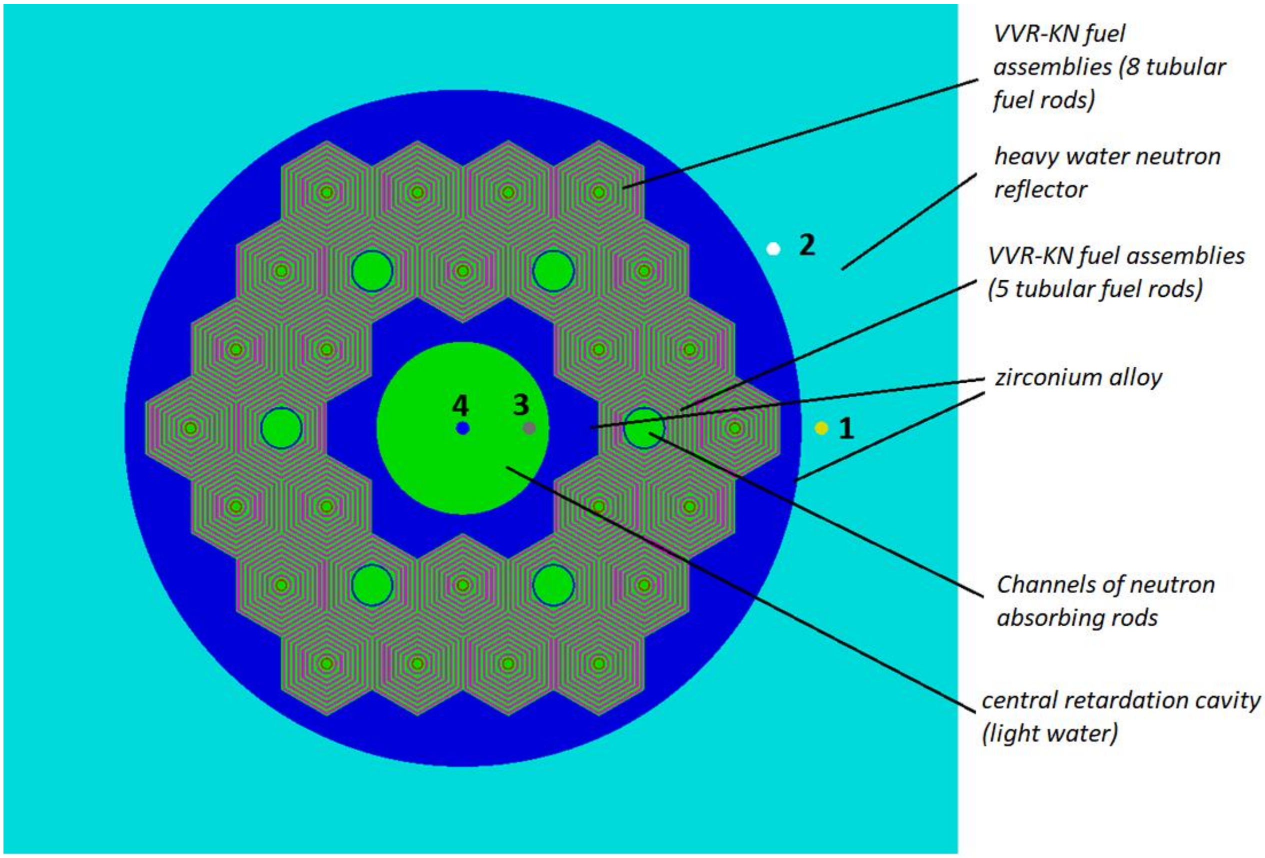


Figure 9 - Diagram of core geometry with indication of materials for neutron-physical calculations

Table 3. Calculated parameters of the neutron flux at control points

Detector number	Neutron flux density					Calculation error			
	over 0,1 MэВ	0,1MeV.. to 1 keV	1 keV to 0,5 eV	less than 0,5 eV	General flow	over 0,1 MeV	0,1MeV to 1 keV	1 keV to 0,5 eV	less than 0,5 eV
1	1,09E+14	1,07E+14	1,16E+14	3,04E+14	6,37E+14	1,5%	1,5%	1,5%	1,0%
2	9,53E+13	1,01E+14	1,16E+14	3,02E+14	6,14E+14	1,6%	1,6%	1,5%	1,0%
3	1,78E+14	8,47E+13	1,38E+14	6,30E+14	1,03E+15	1,1%	1,5%	1,2%	0,9%
4	1,23E+14	6,01E+13	1,09E+14	8,78E+14	1,17E+15	1,3%	1,8%	1,4%	0,8%

Figure 10 shows the calculated distribution of the specific energy release along the height of the maximum heat-stressed fuel element used to determine the maximum temperatures in the fuel assemblies.

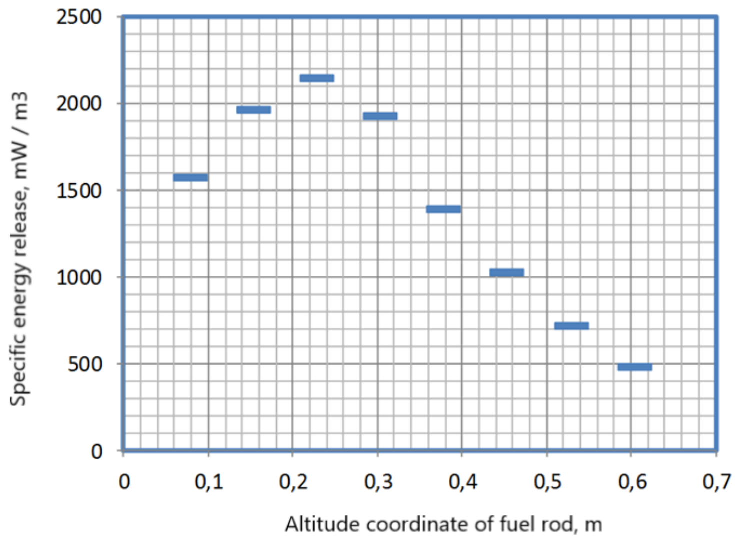


Figure 10 - Calculated distribution of the specific energy release along the height of the maximum heat-stressed fuel element

4.2 Thermal-hydraulic calculation of fuel assemblies in the core using the SolidWorks / FlowSimulation code

The thermohydraulic calculation of the reactor core is carried out for the geometry of fuel assemblies of the VVR-KN type using the CAD software SolidWorks / FlowSimulation [8]. The calculated hydraulic performance is shown in Figure 11.

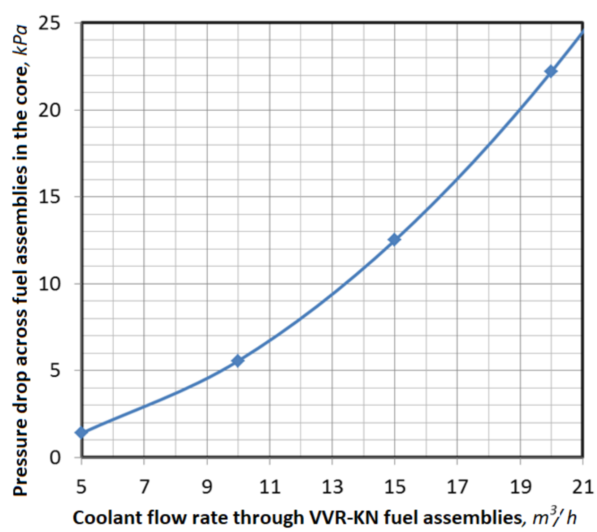


Figure 11 - Hydraulic characteristics of fuel assemblies with 8 rods of the VVR-KN type

For the standard coolant flow through the reactor, calculated using the RELAP5 / mod3.2 code [6, 7] and shown in Table 1, Figure 12 shows the altitude distribution of the coolant velocity and temperature in the maximum heat-stressed fuel assembly.

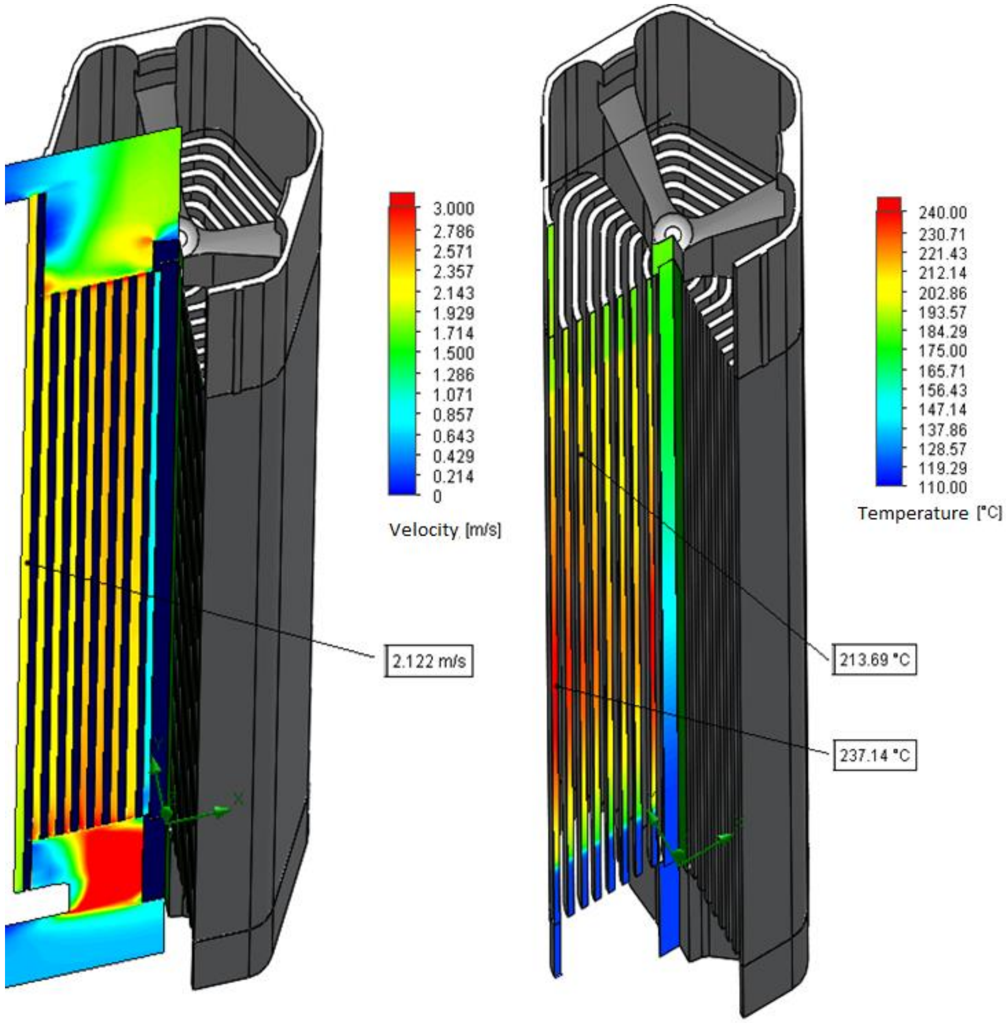


Figure 12 - Calculated distribution of the coolant velocity and fuel element temperature along the height in the maximum heat-stressed fuel assembly

4.3 Thermal hydraulic calculation of the reactor cooling system using the RELAP5 / Mod3.2 code

4.3.1 Nodalization scheme

The nodalization scheme for the calculated thermohydraulic analysis of the development of the situation with a three-circuit research reactor unit according to the RELAP5 / Mod3.2 code [6, 7] is shown in Figure 13.

The primary circuit is modeled by the following components:

- P-150 (vessel);
- descending pipeline « P-190 »,

- ascending pipeline P-160;
- tubulars of the P-165 heat pipe steam generator;
- downstream pipeline P-190,

The core is modeled by :

- low core space BR-012;
- high core space BR-024;
- maximum heat-stressed fuel assembly P-110;
- a group of maximum heat-stressed fuel assemblies - P-111;
- a group of medium heat-stressed fuel assemblies - P-112;
- a group of minimally heat-stressed fuel assemblies - P-115.

The primary circuit pressurizer is modelled by:

- capacity of the BR-800,801 ;
- safety valve V-808;
- pipeline P-803

The intermediate circuit (heat pipe) is modeled by:

- steam generator - BR-590,599:
- steam pipe - P-591;
- pipe space of the air heat exchanger - P-595;
- Condensate return pipeline - P-597.

Air circuit for transferring heat to the final absorber.

- atmospheric air inlet - TV-100, SV-205;
- the annular space of the air heat exchanger - P-210;
- ventilation pipe - P-220;
- hot air outlet from the pipe - TV-101, SV-230.

The valves of the short circuit of natural circulation are modeled by the valve V-756 and the time-dependent connection TV-750.

The nodalization diagram shows the modulated break points on the ascending and descending pipelines. For this, the components of the gate valve V-303, 304 and time-dependent connections TV-301, 302 are used.

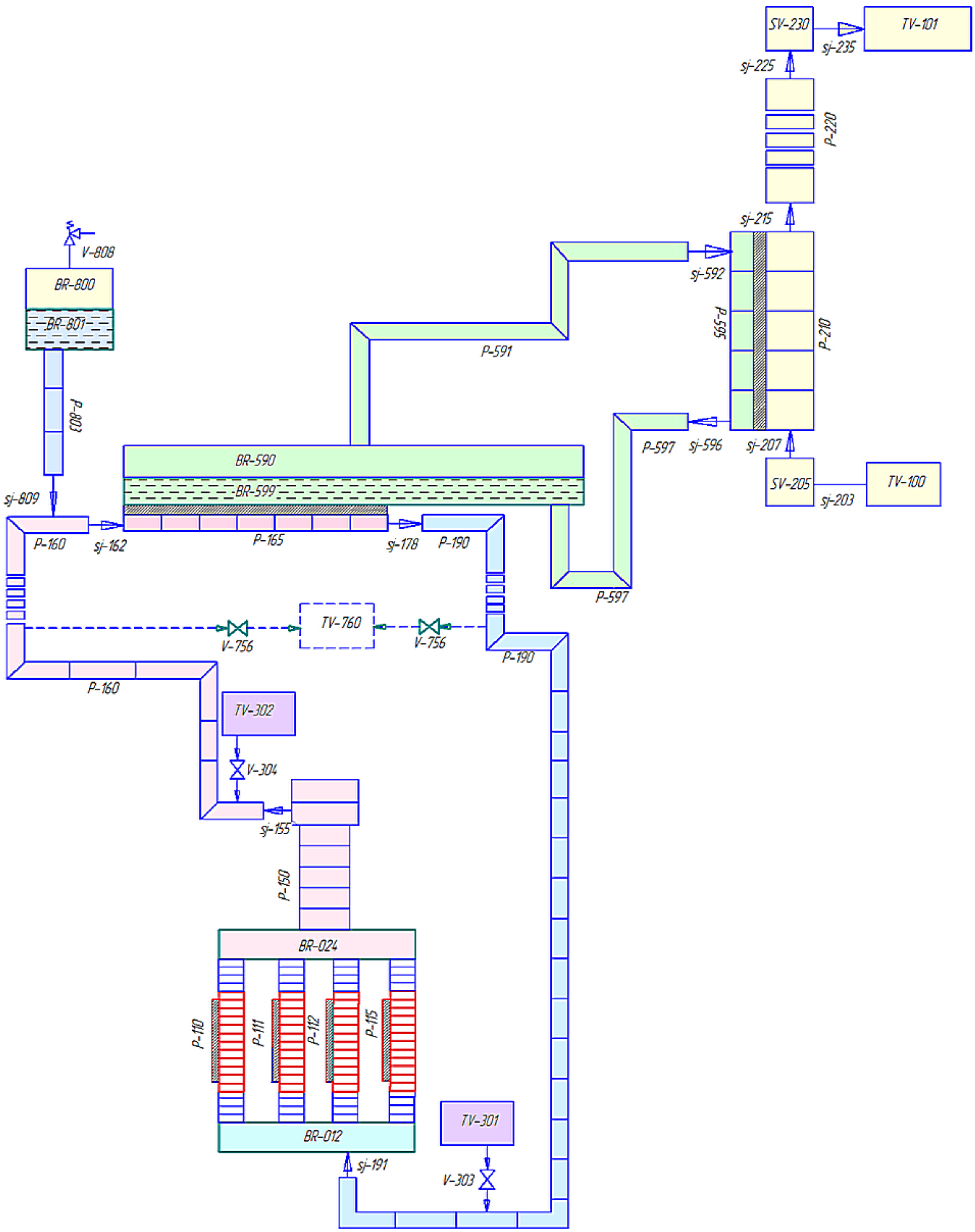
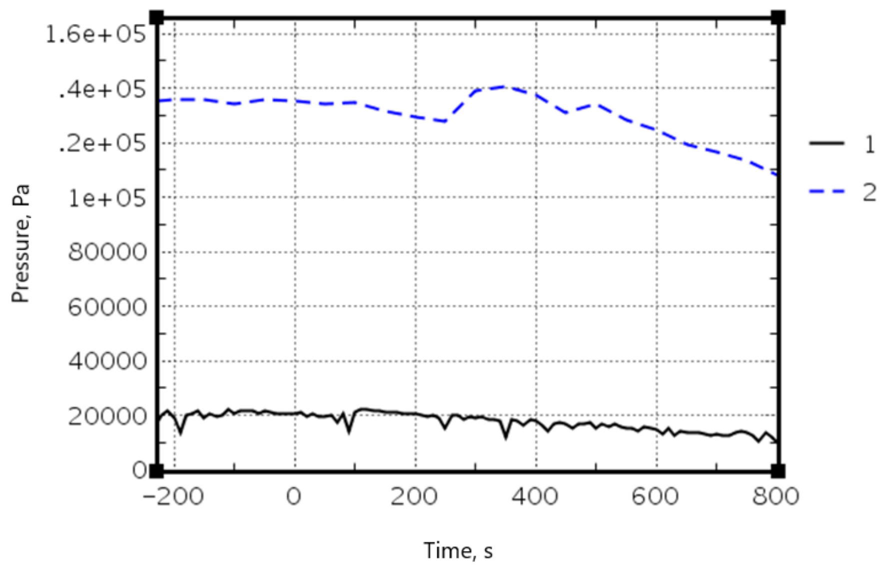


Figure 13 - Nodalization scheme for thermohydraulic calculation of cooling circuits and simulation of accident conditions with indication of break points

4.3.2 Influence of the air presence in the heat pipe on the temperature conditions

The operation principle of a heat pipe (thermosyphon) does not imply the presence of non-condensable gases in a sealed cavity. Therefore, the pressure in it must be determined by the pressure of the saturated vapor of the working fluid. The presence of air significantly increases the operating temperature at which the heat pipe works the most efficiently.

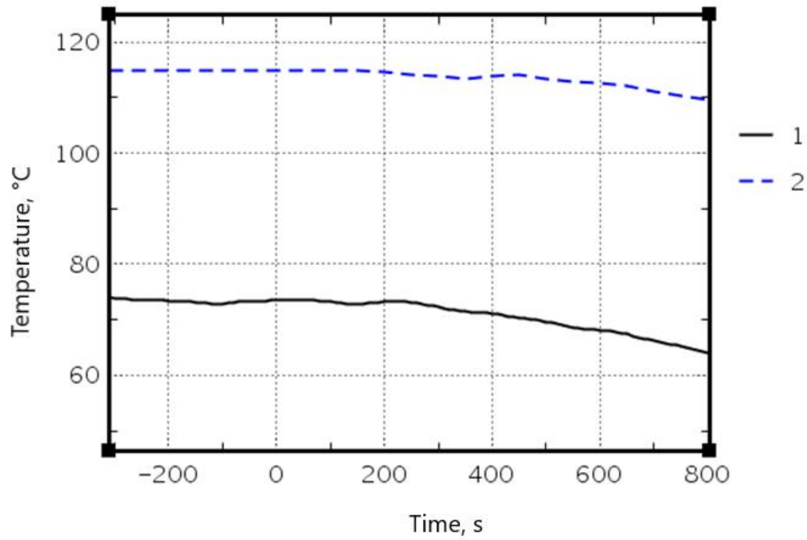
Calculations using the RELAP5 code allow evaluating the efficiency degradation due to the air presence in the sealed circuit. The Figure 14 shows the difference in pressure in the heat pipe with a smooth decrease in power from the nominal level of 25 MW at a rate of 24 kW/s in the presence of air in the heat pipe (2) and in its absence (1).



1 - in absence of air; 2 - in presence of air

Figure 14 - Steam pressure changing in the heat pipe

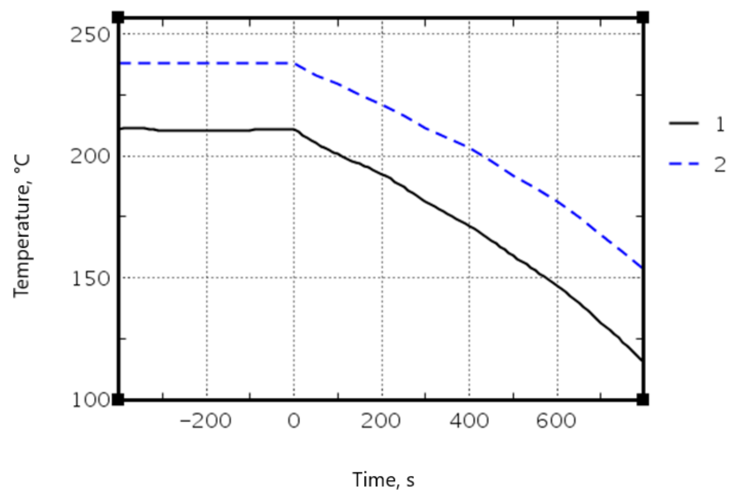
The difference in partial pressure in the heat pipe causes a significant difference in operating temperatures (Figure 15).



1 - in absence of air; 2 - in presence of air

Figure 15 - Steam temperature changing in the heat pipe

The presence of air in the thermosyphon significantly affects the temperature in the primary circuit. In the mode with a smooth power decrease in Figure 16, this effect is clearly visible.



1 - in absence of air; 2 - in presence of air

Figure 16 - Change in the maximum temperature of fuel elements

The analysis shows that the presence of air in the heat pipe significantly (~ 25 ° C) increases the temperature in the primary circuit. Nevertheless, ***for conservative reasons, all further analysis was carried out for the conditions of the presence of air in the sealed circuit of the heat pipe.***

4.3.3 *Calculation of the the transient with the power increase up to the rated power level*

Below are the calculated parameters of the reactor cooling loop with a smooth exit to the rated power level of 25 MW for 600 seconds. Figure 17 shows the change in the heat flux density along the height of the maximum heat-stressed fuel element when reaching the nominal power level.

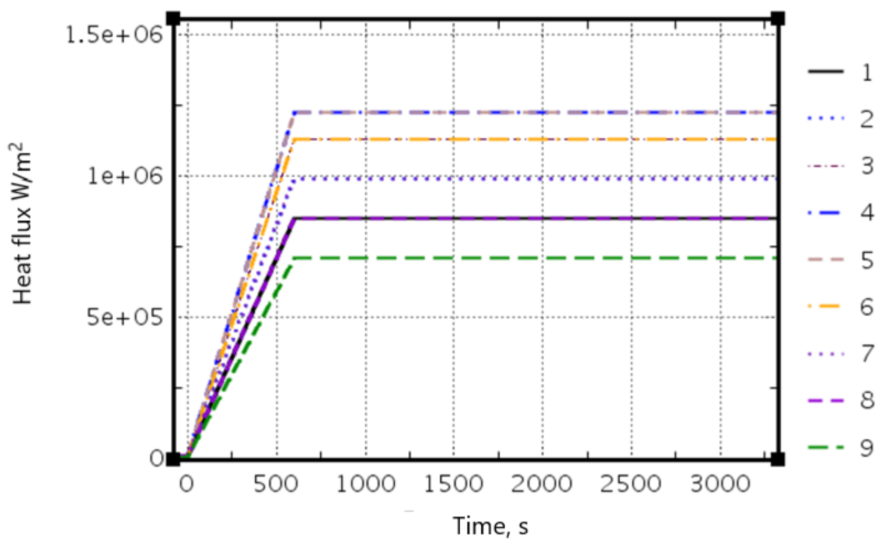
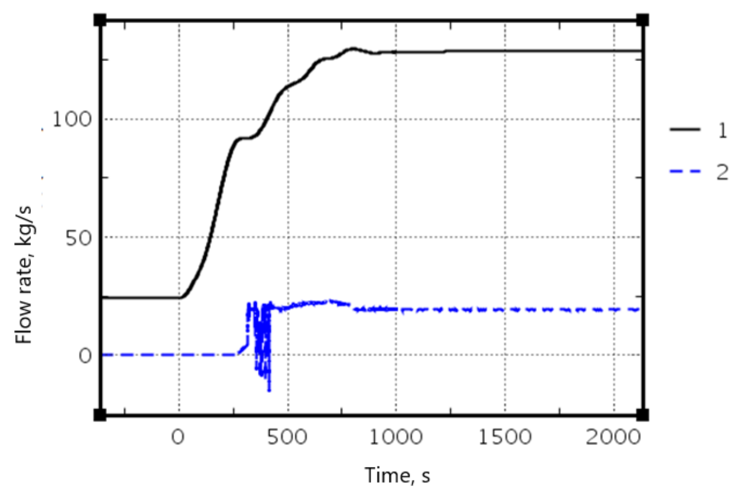


Figure 17 - Change in the heat flux density over 8 altitude sections of the maximum heat-stressed fuel element

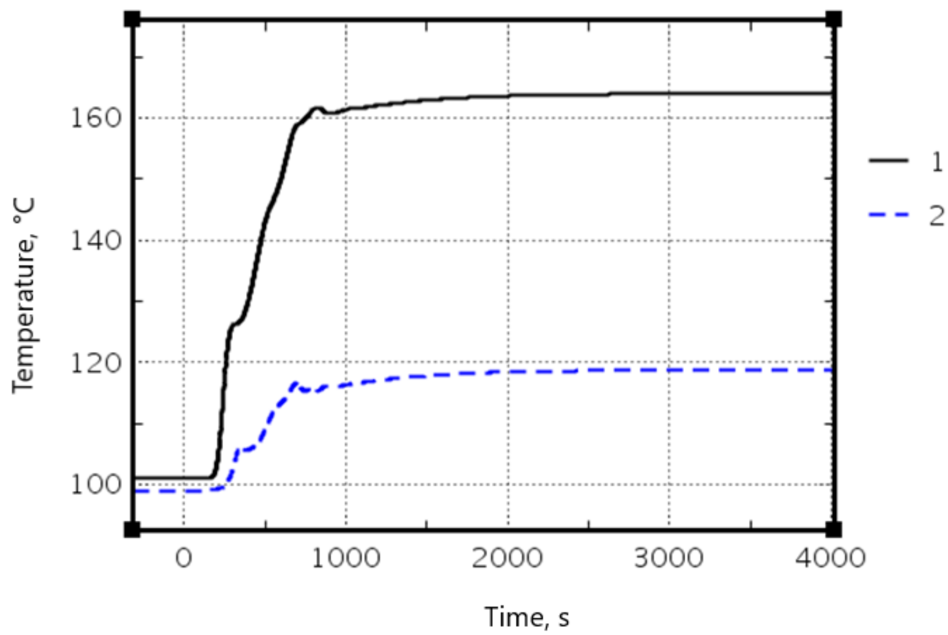
The change in the coolant flow rate in the primary circuit and in the heat pipe is shown in Figure 18.



1 - flow rate in the primary circuit; 2 - flow rate in the heat pipe

Figure 18 - Changing the coolant flow rate

The core inlet and outlet temperatures are shown in Figure 19.



1 - temperature at the reactor inlet; 2 - temperature at the reactor outlet

Figure 19 - Change in the temperature of the coolant in the reactor

The change in the heat flux density along the altitude sections of the maximum heat-stressed fuel element is shown in Figure 20.

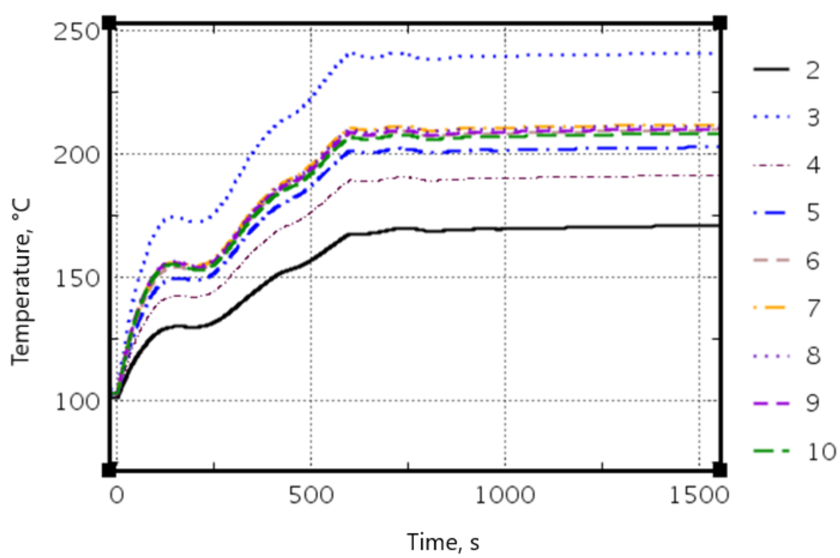


Figure 20 - Temperature changing of fuel elements in 9 altitude sections of the maximum heat-stressed fuel element

4.3.4 Calculation of the reactor shutdown mode when the emergency protection is activated

When the emergency protection is triggered, there is a sharp decrease in power from the nominal level of 25 MW and a corresponding change in all parameters. Figure 21 shows the rate of power decay assumed in the calculation.

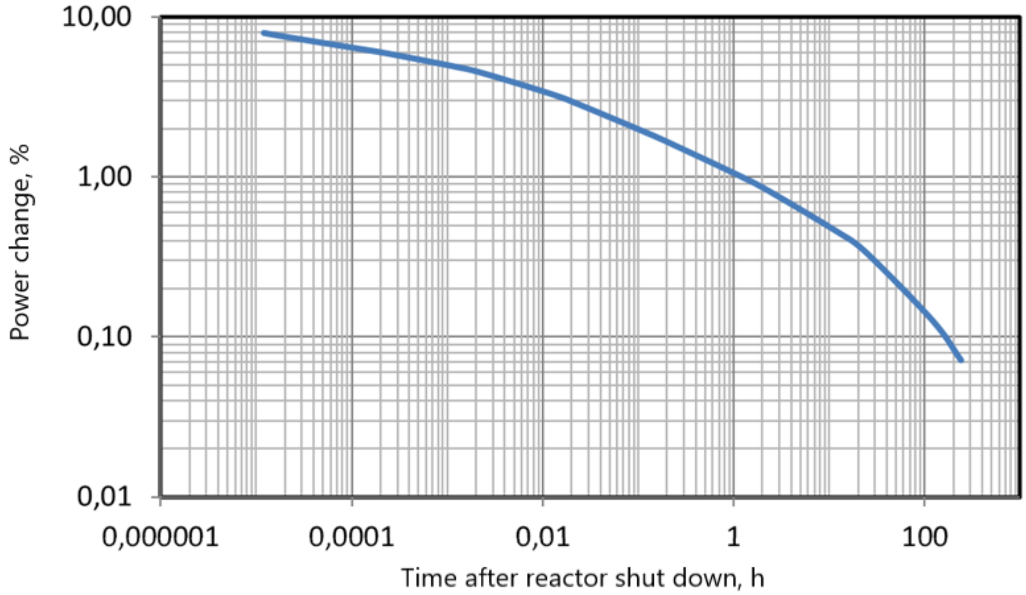


Figure 21 - The rate of power drop taken in the calculation

Figure 22 shows the dynamics of temperature decrease along the altitude sections of the maximum heat-stressed fuel element.

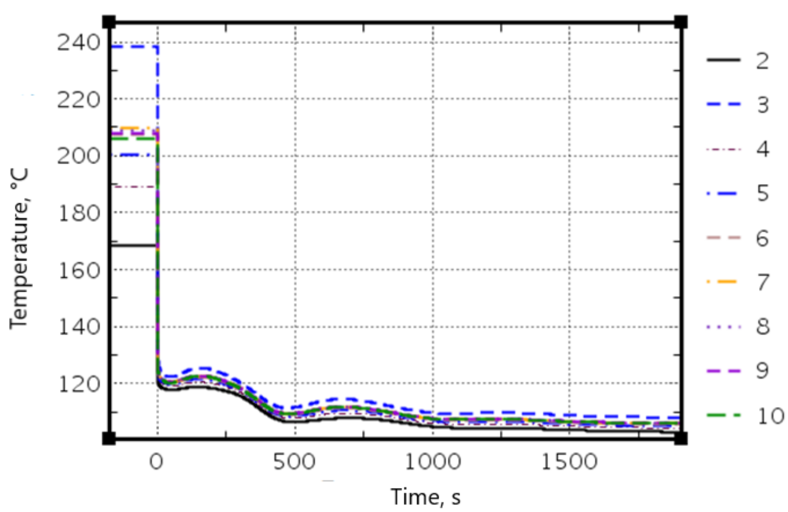
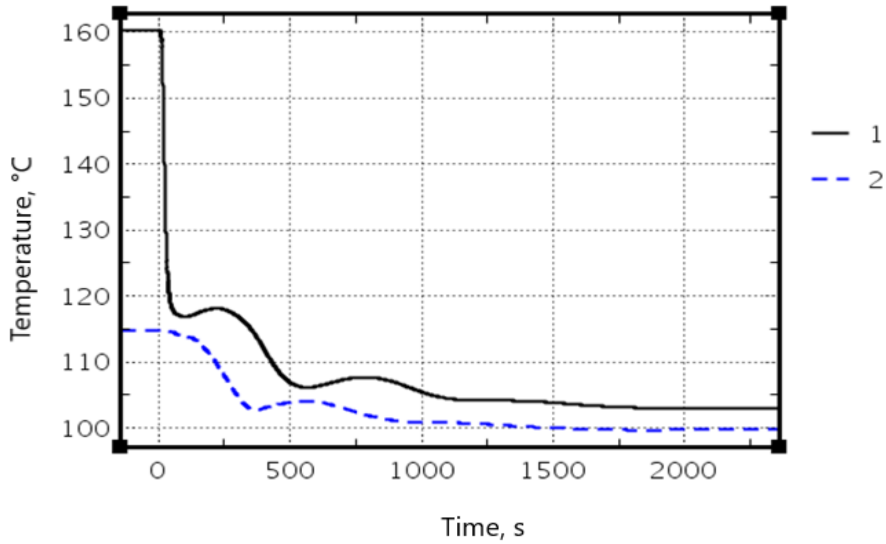


Figure 22 - Temperature decrease in the altitude sections of the maximum heat-stressed fuel element

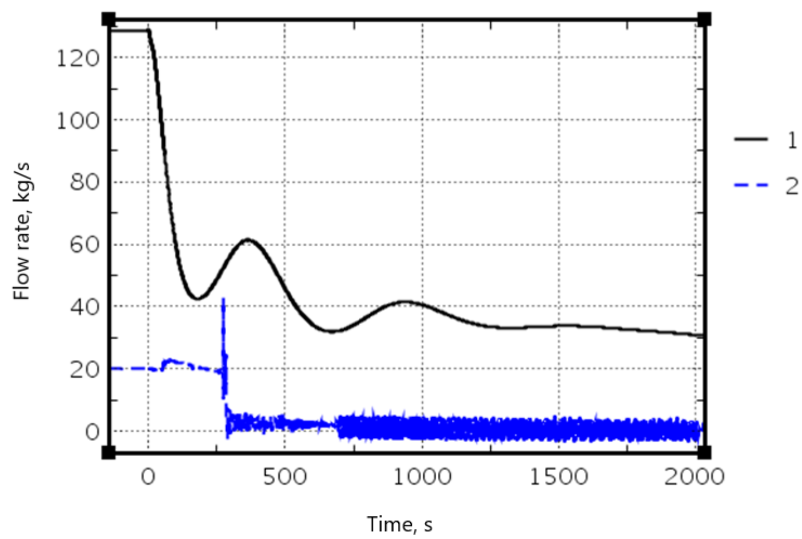
The temperature change at the reactor inlet and outlet when the emergency protection is activated, is shown in Figure 23.



1 - temperature at the reactor inlet; 2 - temperature at the reactor outlet

Figure 23 - Change in the reactor coolant temperature when the emergency protection is triggered

The flow rate change in the heat pipe when the emergency protection is triggered is shown in Figure 24.



-

1 - flow rate in the primary circuit; 2 - flow rate in the heat pipe

Figure 24 - Change in the flow rate of primary circuit coolant and in the heat pipe when the emergency protection is triggered

4.4 Thermal hydraulic calculation of the heat pipe steam generator

The main design parameters of the heat pipe steam generator in the air presence in the circuit (Figure 4) are shown in Table 4.

Table 4 Main parameters of the heat pipe steam generator (presence of non-condensable gases, conservative calculation)

Parameter	Value
Material	Stainless steel
Thermal conductivity of the material, [W m $^{-1}$ K $^{-1}$]	16
Transmitted power, [MW]	25
Coolant Flow rate, [t h^{-1}]	463,3
Temperature at fuel assembly inlet, [° C]	164
Temperature at outlet of the fuel assembly, [° C]	118
Pressure in heat pipe, [MPa]	0,17
Steam temperature in heat pipe, [° C]	115
Steam flow rate in heat pipe, [t h^{-1}]	69,12
Average logarithmic temperature head, [° C]	24
Outside diameter of heat exchange tubes, [m]	0,025
Wall thickness of heat exchange tubes, [m]	0,002
Calculated heat transfer coefficient, [W m^{-2} K^{-1}]	1483
Calculated heat exchange surface, [m^2]	700
Adopted heat exchange surface, [m^2]	765
Heat exchange tube length, [m]	9
Number of heat exchange tubes, [pcs]	1063
Coolant velocity in tubes, [m s $^{-1}$]	0,37
Primary circuit pressure drop, [Pa]	600

4.5 Thermal hydraulic calculation of the air heat exchanger

To assess the parameters of the air heat exchanger (Figure 5), a simulation of air heating is carried out when it passes through the gaps between elliptical stainless steel tubes with a wall thickness of 1 mm, the parameters of which are presented in Table 5.

Table 5 Main parameters of the air heat exchanger

Paramater	Value
Transmitted power, [MW]	25
Material	Stainless steel
Thermal conductivity of the tube material, [W m $^{-1}$ K $^{-1}$]	16
Heat exchange surface, [m^2]:	
- from the air side	1086,4
- from the steam side	1030
Cross-section of elliptical heat exchange tubes, [mm]	10,4×18,4×1,5
Tube length, [м]	9
Number of tubes in a row (right and left)	1120
Number of layers in the heat exchanger, [pcs]	10
Number of tubes, [pcs]	22400
Hydraulic diameter (air side), [m]	0,038
Hydraulic diameter (steam side), [m]	0,014
Steam / condensate flow rate (at 25 MW), [t h^{-1}]	69,1
Air flow rate (at 25 MW), [t h^{-1}]	739
Air heating, [° C]	33,2
Air pressure drop, [Pa]	50

The temperature field in the shell side of the heat exchanger, calculated in the Solid Works / Flow Simulation package, is shown in Figure 25. This figure also shows the distribution of air velocity and pressure as it passes through the air heat exchanger.

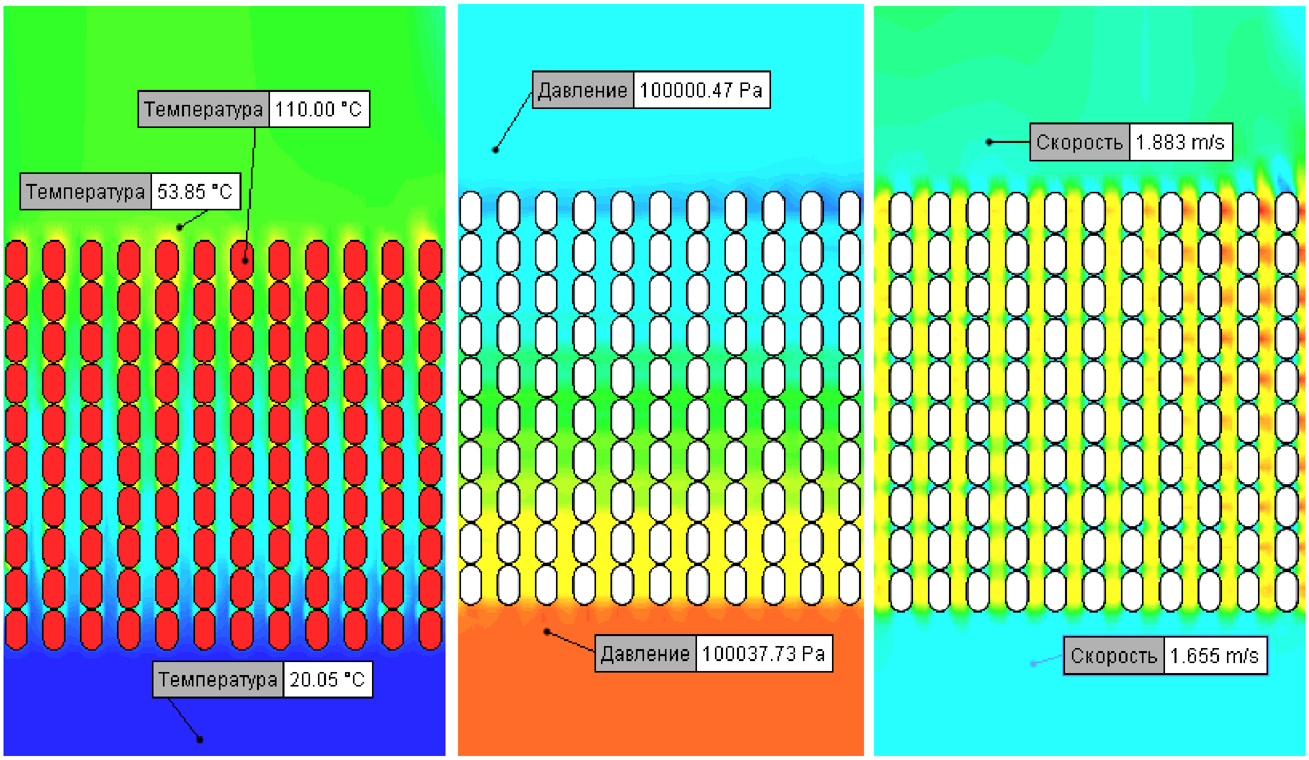


Figure 25 Distribution of temperature, speed and pressure of air as it passes through the shell side of an air heat exchanger

4.6 Hydraulic calculation of the chimney

For the calculated analysis of air cooling, the following parameters of the chimney are taken:

- Pipe height 145 m
- Diameter of the flow area in the lower part 12.6 m
- Diameter of the flow area in the upper part 10.4 m

For these parameters of the chimney, the Solid Works / Flow Simulation calculations for different heat sink power gave the results shown in Table 6.

Table 6 Basic design parameters of the chimney

Dissipated power, MW	10	25	50	100
Inlet air temperature,°C	20	20	20	20
Outlet air temperature,°C	37,3	53,2	73,5	110
Air flowrate, kg/s	571	739	932	1108
Pressure drop across heat exchanger Pa	33	50	69	100

For the assumed parameters of the chimney and the reactor power of 25 MW, Figure 26 shows the distribution of temperature, velocity and pressure of air passing through the chimney.

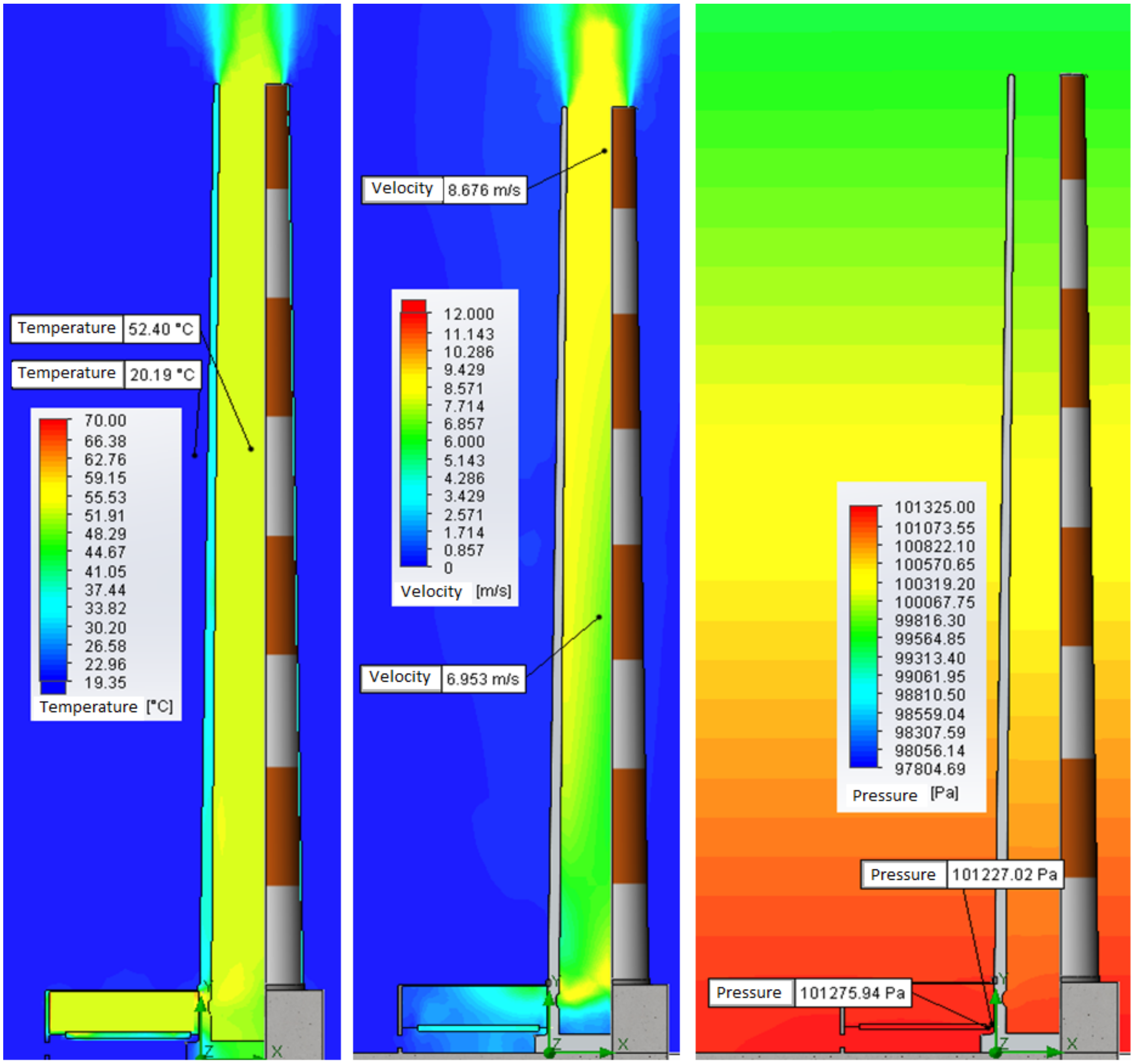
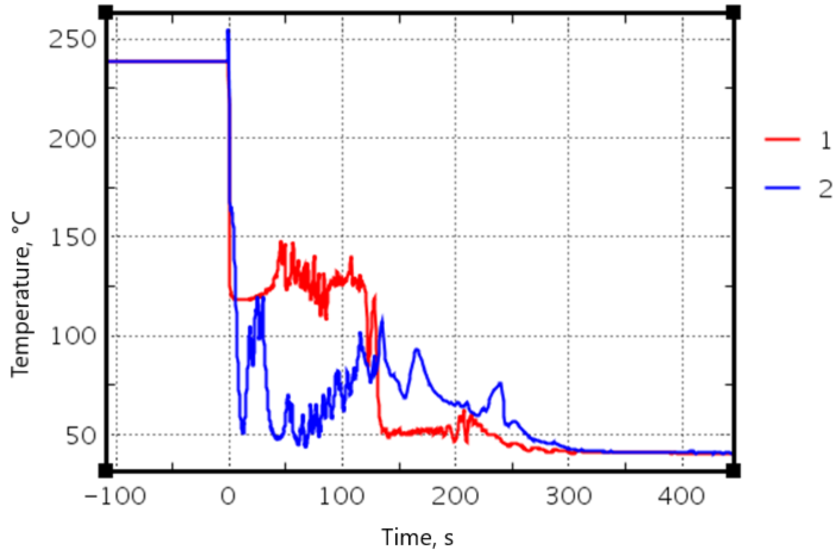


Figure 26 Distribution of temperature, speed and pressure of the air passing through the chimney.

5. Simulation of accidents using the RELAP5 code

The reactor cooling system proposed for consideration makes it possible to exclude most of the items from the recommended list of initiating events for the analysis of design basis accidents for research reactors due to the absence of pumps and shut-off control valves in the system.

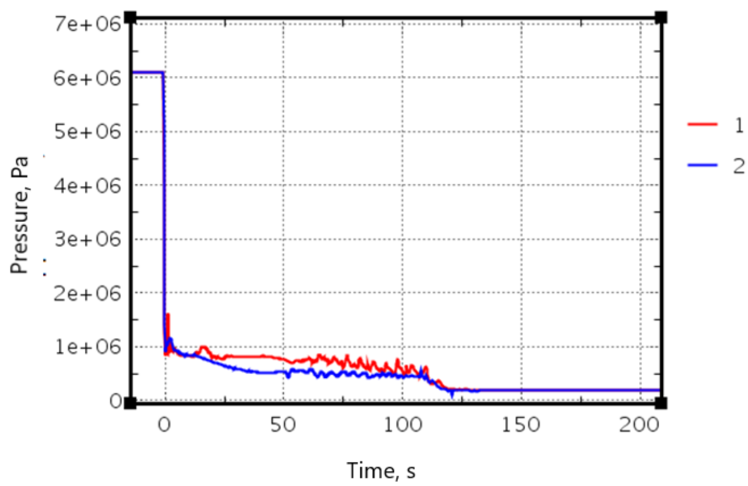
The dynamics of changes in the maximum temperature of fuel elements in the event of a rupture of a hot and cold main circulation pipelines is shown in Figure 27.



1 - rupture of the "hot" pipeline; 2 - rupture of the "cold" pipeline

Figure 27 Change in the temperature of the maximum heat-stressed fuel element in the event of the primary pipelines rupture

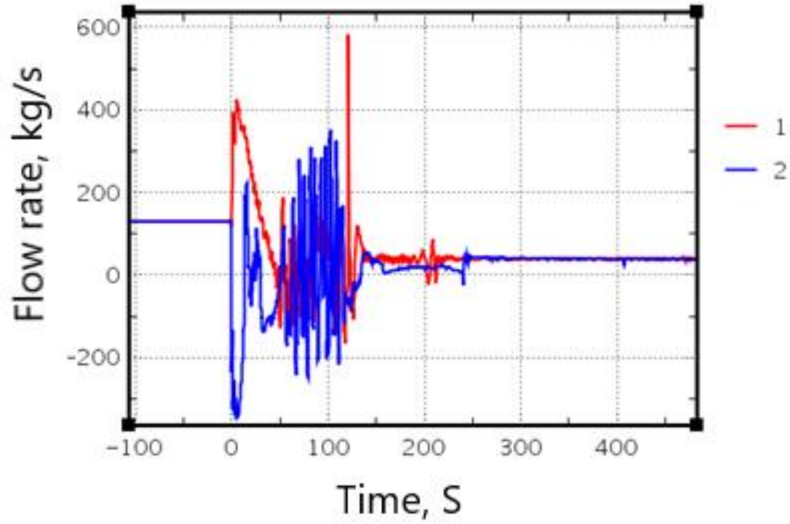
The dynamics of the pressure change in the core during the rupture of the hot and cold main circulation pipelines is shown in Figure 28.



1 - rupture of the "hot" pipeline; 2 - rupture of the "cold" pipeline

Figure 25 - Pressure change in the core upon rupture of the circulation pipelines of the primary circuit

The dynamics of the coolant flow rate change in the core during the rupture of the hot and cold main circulation pipelines is shown in Figure 29.



1 - rupture of the "hot" pipeline; 2 - rupture of the "cold" pipeline

Figure 26 - Coolant flow rate change through the core in the event of a rupture of the primary circuit pipelines

Conclusions

- A conceptual three-dimensional model of the reactor plant and a technological scheme of the heat removal systems based on the natural circulation are presented. The advantages of creating a simple and reliable passive system for the reactor core cooling, built on the principle of natural convection, have been substantiated;
- The presence of the intermediate circuit operating on the principle of a heat pipe (thermosyphon) eliminates the risk of radioactive coolant release into the environment. The operation of this circuit provides high efficiency of heat transfer with a simple design and the absence of mechanical moving parts, which determines the reliability of the system;
- Air cooling of the reactor plant using a chimney eliminates the problem of water treatment of cooling towers to prevent the formation of salt deposits on the heat exchange tubes and the sprinklers surface in cooling towers;
- The location of the reactor deep below the ground level offers advantages over conventional research reactor design:
  - most scenarios of external impacts on the reactor facility are excluded. Therefore, many emergencies resulting from explosions, tornadoes, snow loads, aircraft crashes, etc. can be ignored, there is no need for expensive containment, the physical protection of the reactor facility is simplified;
  - the absence of expensive and complex systems for heat removal from the core can dramatically reduce the cost of the reactor facility;
  - The large depth of submersion of the core under the ground allows creating a simple and highly efficient natural circulation system. At the same time, there are no visible restrictions on an increase in the height of natural circulation and a corresponding increase in the driving pressure through the core;
  - the reactor can operate equally safely and efficiently in a very wide range of capacities based on the assigned irradiation tasks;
  - there is practically no problem of expensive dismantling of the reactor plant;
- Automatic adjustment of the flow rate in the cooling circuits when the power level changes and no need for personnel to reduce the intensity of coolant circulation, eliminates emergencies with deterioration of heat removal from the core;
- Automatic adjustment of the flow rate in the cooling circuits when changing the power level provides the utmost simplicity of reactor plant control and reduces the requirements for personnel qualifications. Therefore, such a reactor can operate in countries where there are no personnel with extensive experience in operating reactor facilities, and the reactor facility can also be used for training purposes;
- Relatively low velocities of the upward coolant flow in the core at high pressure provide a large margin before boiling on the fuel elements surface and do not create problems for the upper location of the control rods (on the reactor head), which greatly simplifies their design;

- For the considered parameters of the reactor plant with a power of 25 MW, the temperature margin before the maximum temperature on the fuel elements to the saturation temperature of water in the core is 32 °C in the presence of air in the heat pipe and 57 °C in the absence of air;
- Aluminum alloys are used in fuel elements of reactor cores at temperatures no higher than 250-270 °C [9]. The range of maximum operating temperatures of fuel elements in the considered reactor plant meets this requirement.
- The design of the cooling circuits ensures continuous natural circulation of the coolant in the core in all modes, creating conditions for safe heat removal from fuel assemblies both during operation at power and after shutdown;
- The use of a neutron reflector external to the reactor vessel in the pool provides easy access to the irradiation volumes and the possibility of replacing the irradiation devices without shutting down the reactor;
- The experience of using in the Australian reactor OPAL a heavy-water reflector in a zircalium tank located in a pool with light water has shown the high efficiency of such constructive solution. The deep location of the pool and the reactor vessel makes it possible to organize efficient cooling of the reflector heavy water using natural convection in the circulation pipelines and an air heat exchanger in the chimney;
- The production of heavy water and the purification of heavy water from the reflector from tritium during the operation of the reactor plant can be carried out using the technology of separating water isotopes into hydrogen atoms, proposed in article [9].
- The absence of fittings and pumps in the cooling circuits ensures smooth changes in heat removal parameters and completely eliminates water hammer;
- The proposed concept of a reactor plant with valves of a shortened intra-pool circulation provides a quick and easy core cooling by the pool water, which makes it possible to carry out transport and handling operations without any problems with an open reactor head;
- The physical principles of operation of the cooling circuits are well studied and do not require additional R&D during the design of the reactor plant;
- Using three-dimensional modeling, the main parameters of the equipment of the air cooling circuit with natural air convection, air heat exchanger and chimney were calculated. Presented are the results of thermohydraulic calculation of heat transfer from the reactor core to the final recipient (atmospheric air);
- The presented results of calculations using the RELAP5 model showed the efficiency of the cooling system using natural circulation and the attainability of a high level of reactor thermal power (25 MW), which corresponds to the maximum neutron flux density in the central moderating cavity over 1×10^{15} cm^{-2}s^{-1};
- The presented results of calculations of emergency situations with ruptures of LOCA pipelines according to the model in RELAP5 show that safe heat removal from the core is ensured and there is no loss of the leaktightness of the cladding of fuel elements and the release of the fuel composition into the coolant;

- The presented computational analysis shows that it is realistic to create a safe and completely passive system for removing heat from the core of a research reactor, which can be used as a universal cooling system for a wide range of reactor facilities for various purposes.

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