UNIVERSAL SYSTEM OF PASSIVE HEAT REMOVAL FROM THE CORE OF A RESEARCH REACTOR

by

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This paper presents the results of an analysis of a universal cooling system for the core of research reactors built on the passive principle of natural convection. A 3-D model, technological and design diagrams of the reactor installation are provided, along with examples of numerical evaluation of transients during the operation of the cooling circuit in normal and emergency modes to substantiate the possibility of using such a cooling system in research reactors of small and medium power. The principal feature of the described passive system is the absence of not only active elements, such as circulation pumps and shut-off and control valves from the cooling circuit, but also of passive elements with moving parts, such as a check valve. The cooling circuit includes only vessels, piping and a heat exchanger. The absence of elements with mechanical moving parts can significantly reduce the likelihood of equipment failures and improve the reliability of such a cooling system while also reducing its cost. The versatility of the proposed system allows it to be used for a wide range of research reactor plants with various capacities, which are nowadays being developed designed to carry out programs in various areas of research and applied usages related to nuclear technologies.

Key words: research reactor, passive system, reactor cooling system, safety of nuclear reactor, natural convection

INTRODUCTION

Currently around 240 research reactors are operating around the world, but there are about 360 of them that are already shut down and decommissioned. One of the reasons for the reduction of the number of research nuclear reactors in the world were not only serious accidents at nuclear power plants, but also an increase of strict requirements for the technical level of reactor safety and personnel qualifications. However, in recent years, the interest to create new research reactors has begun to grow, due to the development of nuclear technologies and the intent of developing countries to have their own national nuclear research centers.

The need for safe, simple and reliable research reactors encourages developers to search for optimum and competitive design solutions that will enable a wide range of research in the following areas: nuclear physics, solid state physics, radiation materials science, neutron activation analysis of the substance, neutron radiography of various products, radiation doping of silicon, production of isotopes for medical and industrial purposes, *etc*.

Research reactors can also be used as training facilities and sources of neutrons for neutron therapy channels.

One of the most important integral systems of research nuclear reactors that affect its neutron-physical characteristics, safety and cost, is a system that ensures reliable heat removal from the core both in normal operation mode and in case of possible emergency situations. With the tightening of safety requirements for nuclear installations as the consequence of serious accidents in nuclear power plants, the complexity and ramification of both safety systems and systems of normal operation significantly increased. This has, on the other hand, led to a sharp increase of the number of possible failures of the elements that might themselves lead to emergency situations and increased the complexity of analyses imperative to justify the safety of such installations.

The creation of simple, safe and reliable research reactor installations with good operational characteristics to a large extent depends on the optimization of heat dissipation systems. The design of such facilities

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requires taking into account all adverse natural (*e. g*. seismicity, flood, *etc*.) and man-made vulnerabilities (*e. g*. lack of reliable power supply systems, *etc*.), but also the possibility of inadequate personnel qualifications, which can lead to errors in the management of reactor installations.

Another important factor in the competitiveness of a project is the cost of the reactor plant and the operating costs of its current maintenance and preventive maintenance.

COMPLIANCE OF A PROPOSED COOLING SYSTEM CONCEPT WITH DESIGN PRINCIPLES OF PROFICIENT RESEARCH REACTORS

In developing of new research reactors in accordance with international standards, it is necessary to focus on safety requirements [1], and also on conceptual provisions and design principles of advanced research reactors for research centers described in [2].

Reliability

Use of technical solutions and equipment tested during reactor operation

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

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 natural circulation over time, even in complex circuits. In addition, great experience has been gained in the creation of systems using natural circulation both with boiling and without boiling of the coolant.

The use of only passive equipment (pipelines, vessels, heat exchanger) in a cooling circuit without mechanical moving parts ensures high reliability of the entire system [1].

Selection of flow rate and pressure drop of the coolant at the reactor core, providing a margin of up to the initial boiling point and the permissible value of the heat safety factor

Adhering to the rate of increase in reactor thermal power provided by regulations, and taking into account the dynamics of the increase in flow rate in the circulation circuit make it possible to reliably ensure that the temperature of fuel elements remains below the boiling point on the 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 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 (by natural processes) self-protection (the negative *void* reactivity effect), boiling of the coolant in the core leads to negative reactivity and a decrease in thermal power.

In this way natural processes on which the principles of operation of a research reactor facility are based allow us to ensure high levels of heat engineering reliability and operational safety.

Safety

Placing the core under high water level

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

Ensuring the preservation of water level above the core with water in case of a leak in the pipelines

Placing the reactor pipeline in a conditionally hermetic vertical channel allows for an elevated water level above the core even in emergency situations when a rupture of circulation pipelines occurs. This channel allows monitoring, collecting, and partially returning of water leaks to the cooling circuit. It will be shown below that an increase in saturation temperature in the core due to pressure of liquid column creates conditions for safe cooling of fuel assemblies during depressurization of the circulation circuit.

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

The prevention of wall boiling on the cladding of fuel rods and core elements during normal operation is achieved by ensuring high efficiency of natural circulation due to low hydraulic resistance of the circulation circuit as a whole and the difference in hydrostatic pressure of the descending and the ascending flow sections, thus creating a driving pressure for natural circulation. This difference is proportional to the height of the contour of natural circulation and the difference in average densities of the coolant in the descending and the ascending flow sections. An increase in the heating of the coolant in the reactor core leads to an increase of this difference in average densities and intensification of circulation. When lifting the coolant in the reactor core, the beginning of the boiling process on the cladding of fuel rods does not affect the stability of natural circulation in the fuel assembly, however, during the descending motion, surface boiling can lead to a reversal of circulation and dangerous deterioration of the heat sink in the fuel assembly [3].

Compliance with the regulations on the speed of reactor output to nominal power level and power limitation allows the fulfillment of the requirement on the absence of boiling on the surface of cladding of fuel elements and core elements, if this requirement is stipulated. But this requirement is not imperative if the materials of fuel claddings and standard parameters of water-chemical regime of the coolant allow surface boiling at maximum heat-stressed fuel elements, as in the case, for example, of research reactor SM-3 [4].

Passive safety systems

An important factor of the safety of a nuclear reactor with natural circulation is that all systems that provide heat removal both during normal operation and in emergency mode 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 concerning reactivity and sufficient efficiency of the elements of control and protection system, this ensures maximum level of safety and reliability of the reactors under consideration.

Flexibility

The universality of the cooling system provides the possibility of implementing various layouts of reactor core for a selected size of reactor vessel. The possibility to vary the number and location of vertical experimental channels is determined by core composition, location of holes on the reactor lid and, in case of a transition to a fundamentally new layout of the core, the reactor lid can be replaced in accordance with the new requirements of experimental volumes.

The number and location of horizontal channels is determined in advance, because, it is in accordance with their purpose and geometry that the reactor vessel is made and the space around the reactor vessel is organized, including the biological shield, horizontal channels and technological rooms for operating them.

Efficiency

In [2], 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 density and its heat output, and
– a variety of experimental spaces.

Simplicity

Extreme simplicity of the circulation loop in a passive cooling system provides easy maintenance of the reactor and the absence of the need for scheduled preventive maintenance of complex equipment of a reactor forced cooling system (pumps, shut-off and control valves, check valves) simply because it does not exist. Reference [3] lists the advantages of a technological scheme with a passive heat sink in the organization of downward coolant movement in the core of a research reactor using natural circulation.

AN EXAMPLE OF THE APPLICATION OF THE PROPOSED CONCEPT OF PASSIVE COOLING FOR A RESEARCH REACTOR UNDER PRESSURE WITH THE NEUTRON TRAP IN THE CORE CENTER

To demonstrate potential capabilities of a research reactor with a passive heat removal system, a vessel-type reactor installation is considered. Its 3-D model is shown in fig. 1.

As an example, a core similar to the core of SM-3 reactor [4] located in SSC RIAR, Dimitrovgrad (Russia), is studied (fig. 2). The main parameters of the research reactor installation with natural circulation of the coolant are shown in tab. 1.

The difference of the considered core from the prototype is that the power is reduced from 100 MW to 10 MW and that there is no profiling of the cells with fuel assemblies for the flow of the coolant.

According to calculated estimates, for the considered reactor using natural circulation in the primary circuit, the level of neutron flux density up to 5 10^{14} $cm^{-2}s^{-1}$ can be achieved in the core without using circulation pumps and valves. During depressurization of the circulation pipe or heat exchanger, the emergency cooling function of the reactor is performed by a Pressurizer, which simultaneously performs the function of the accumulator injection system of the reactor (P-AIS), filled with cold water. Figure 3 shows the diagram of a reactor cooling circuit with natural circulation in a passive circulation circuit having a water treatment system (b) and without such a system (a).

The following results of an assumed state of emergency due to a rupture of the pipeline in the cooling circuit show that cooling of the core of a research

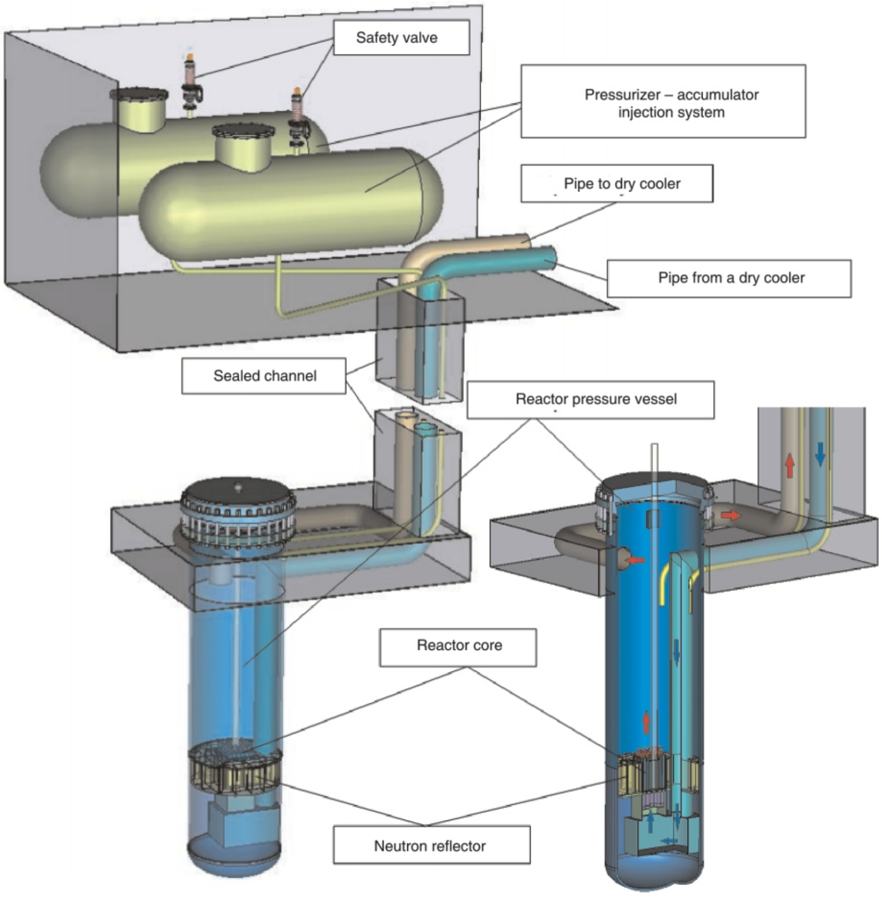

Figure 1. A 3-D model of a research reactor with power of 10 MW and heat removal by natural convection

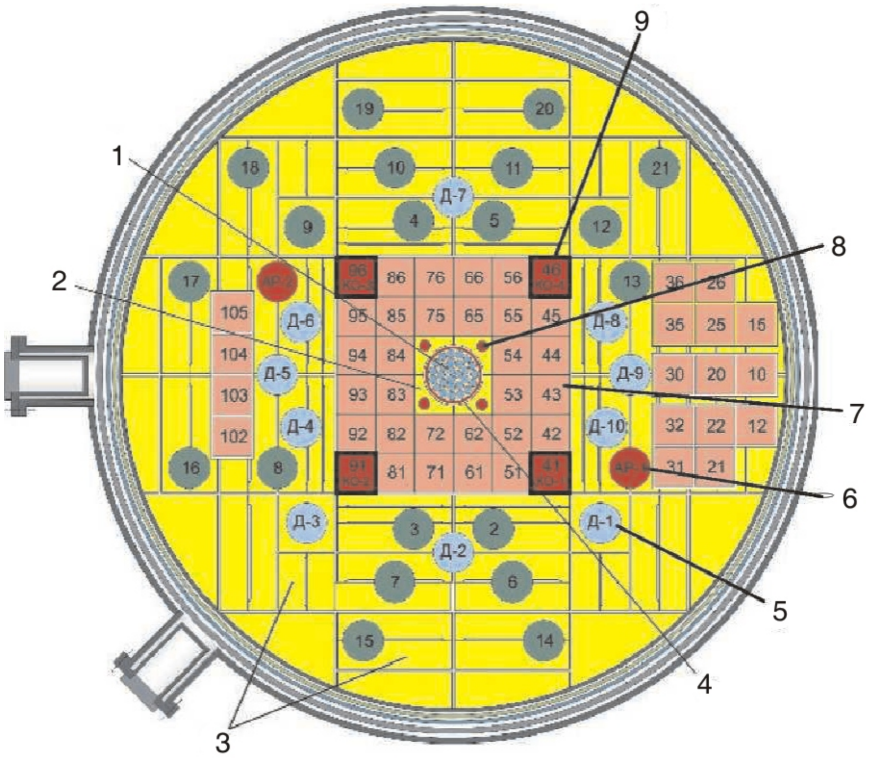

Figure 2. The SM-3 core arrangement

1 – experimental channels in the neutron trap, 2 – beryllium insert, 3 – beryllium block reflector, 4 – central shim rod, 5 – experimental channel cell in the reflector, 6 – control rod, 7 – core cell with fuel assembly, 8 – safety rod, 9 – shim rod

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

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 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 centre
External dimensions of the core [m]	0.42 0.42
The number of cells for fuel assemblies	32
Core height [m]	0.35
Coolant	Light water
Coolant flow rate [$m^{-3}h^{-1}$]	375
Core inlet temperature [°C]	100
Core outlet temperature [°C]	124
Core outlet pressure [Pa]	3.1 10^6
Hydraulic diameter of circulation pipelines [m]	0.35
Height of circuit of natural circulation [m]	40
Maximum fuel campaign duration at nominal power [d]	120

reactor operating at 10 MW is safe. Duplication of the P-AIS system ensures that the reactor is filled with cold water during depressurization of any cooling circuit pipe.

RESULTS OF THE ANALYSIS OF THE FUNCTIONING OF A RESEARCH REACTOR PLANT WITH THERMAL POWER OF 10 MW AND A PASSIVE HEAT REMOVAL SYSTEM

The analysis of reactor cooling systems in transients was made by using RELAP5 [5, 6]. A nodalization diagram of a reactor cooling circuit with thermal power of 10 MW and pressure of 3.5 MPa is presented in fig. 4. The approaches described in [7, 8] were used to analyze the thermo-hydraulic operating modes of the reactor in emergency situations with depressurization of the cooling circuit.

Reactor core cooling circuit with natural circulation

The RELAP5 model of the SM-3 reactor includes the following elements:
– elements of a sealed vessel with the diameter of 1.5 m and height of 7 m, designed for operation under pressure of 3.5 MPa (P-101, Br-12, Br-24),

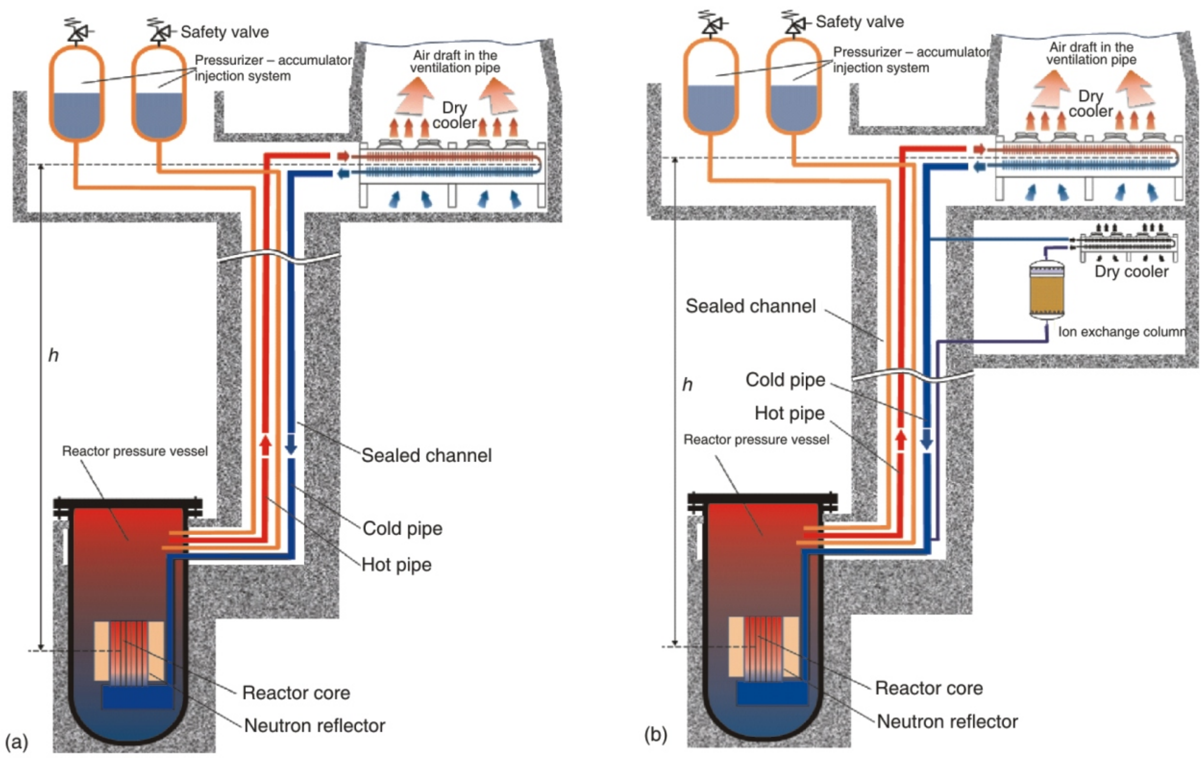

Figure 3. Layout of the cooling circuit of a reactor system with natural circulation of the coolant in a passive circulation loop with a water treatment system (b) and without such a system (a)

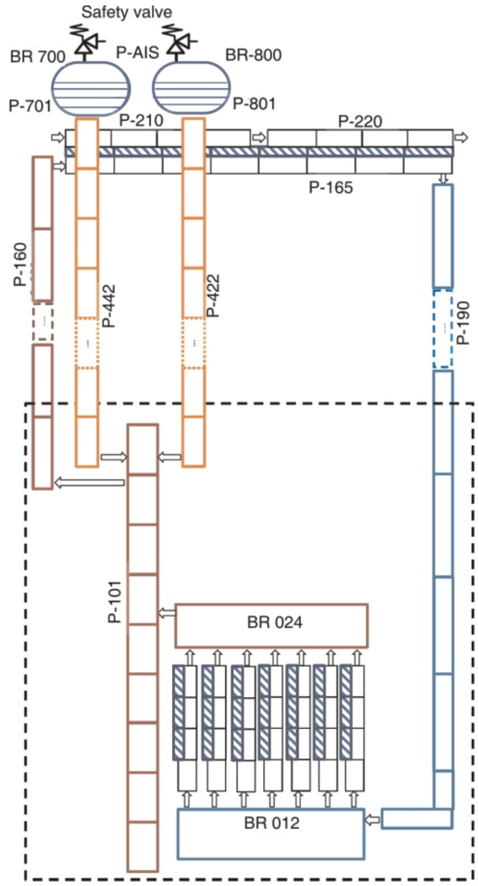

Figure 4. Nodalization diagram of a cooling circuit of a reactor pressure vessel with thermal power of 10 MW

– channels with fuel assemblies, a neutron trap and training channels with thermal structures (P-44, -45, -46, -54, -55, -60, -61),
– inlet cold pipe and discharge hot pipe with an internal diameter of 350 mm (P-190, P-160),
– air heat exchanger with heat structure (P-165, P-220), and
– two P-AIS with cold water injection lines (Br-800, Br-801, P-422, Br-700, Br-701, P-442).

To simulate reactor operation in normal mode, two short campaigns were calculated for 8 hours in two days. The output to rated power of 10 MW was maintained uniformly from the subcritical state for one hour, and the transition of the reactor to subcritical state was reached by using the emergency protection button, *i. e.* at maximum power reduction rate. Residual energy release in the core was realized by a conservative approach based on the calculation of a 10-day campaign when operating at the power of 10 MW, fig. 5.

The results of the calculation of changes of thermal-hydraulic parameters of the reactor plant are shown in the graphs of fig. 6. As the analyses of the graphs show, with fully passive operation of the cooling circuit, there are no difficulties with heat removal from the core in the reactor power output mode during one hour and its shutdown. This points out to high reliability and efficiency of the proposed reactor cooling system and protection against personnel errors due to

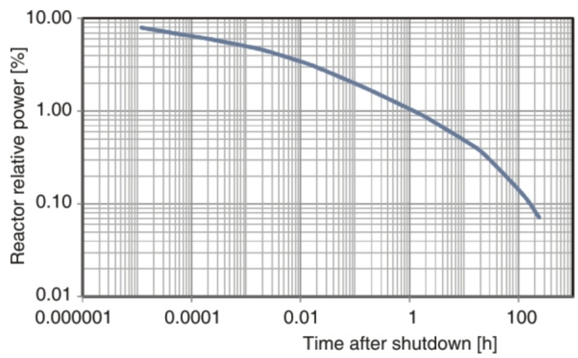

Figure 5. Calculated dynamics of decay heat in the core

the impossibility of operators' intervention in the operation of this system (except by changing the parameters of the pressurizer system) due to the absence of valves and pumps. At the water coolant flow rate of natural circulation of 100 kgs^{-1} established in the cooling circuit, the temperature changes at the core by 24 °C (*i. e.*, increased from 100 °C at the core coolant inlet up to 124 °C at the core coolant outlet).

The analysis of the change of reactor parameters during normal operation is illustrated by graphs of the results obtained with RELAP5. From the dynamics of the change of reactor power, fig. 6(a), it is evident that increasing it leads to an increase of coolant temperature, a decrease of coolant density, and an increase of pressure in the pressurizer and in the core, fig. 6(b). The temperature difference of the coolant between the lifting and the lowering sections of the circulation circuit provides a pressure head of natural circulation of more than 6000 Pa, fig. 6(e). The development of natural circulation provides an increase of coolant mass flow rate up to 100 kgs^{-1}, fig. 6(c), with coolant velocity 1.2 ms^{-1} in circulation pipes with a hydraulic diameter of 350 mm, and in the reactor core up to 1.6 ms^{-1}, fig. 6(d). With power of 10 MW, the maximum values of heat flux density from the cladding surface of fuel elements exceed 1 MWm^{-2}, fig. 6(h), and the temperature of fuel elements reaches 240 °C, fig. 6(f). As in the prototype reactor SM-3, surface boiling occurs for fuel elements with maximum thermal stresses, which increases the heat transfer coefficient from 15 $kWm^{-2}K^{-1}$, to 7 $kWm^{-2}K^{-1}$, fig. 6(g). High velocity of the coolant in the reactor core allows for a large margin from nucleate boiling ratio (over 8.0) in the presence of near-wall boiling on the most heat-stressed fuel rods of the underheated liquid to saturation temperature, fig. 6(h).

Passive system with natural circulation air circuit

The final recipient of released heat of the considered reactor installation can be either evaporated water in an open tank (basin) or circulating atmospheric air. In both cases, the passive system can operate by using

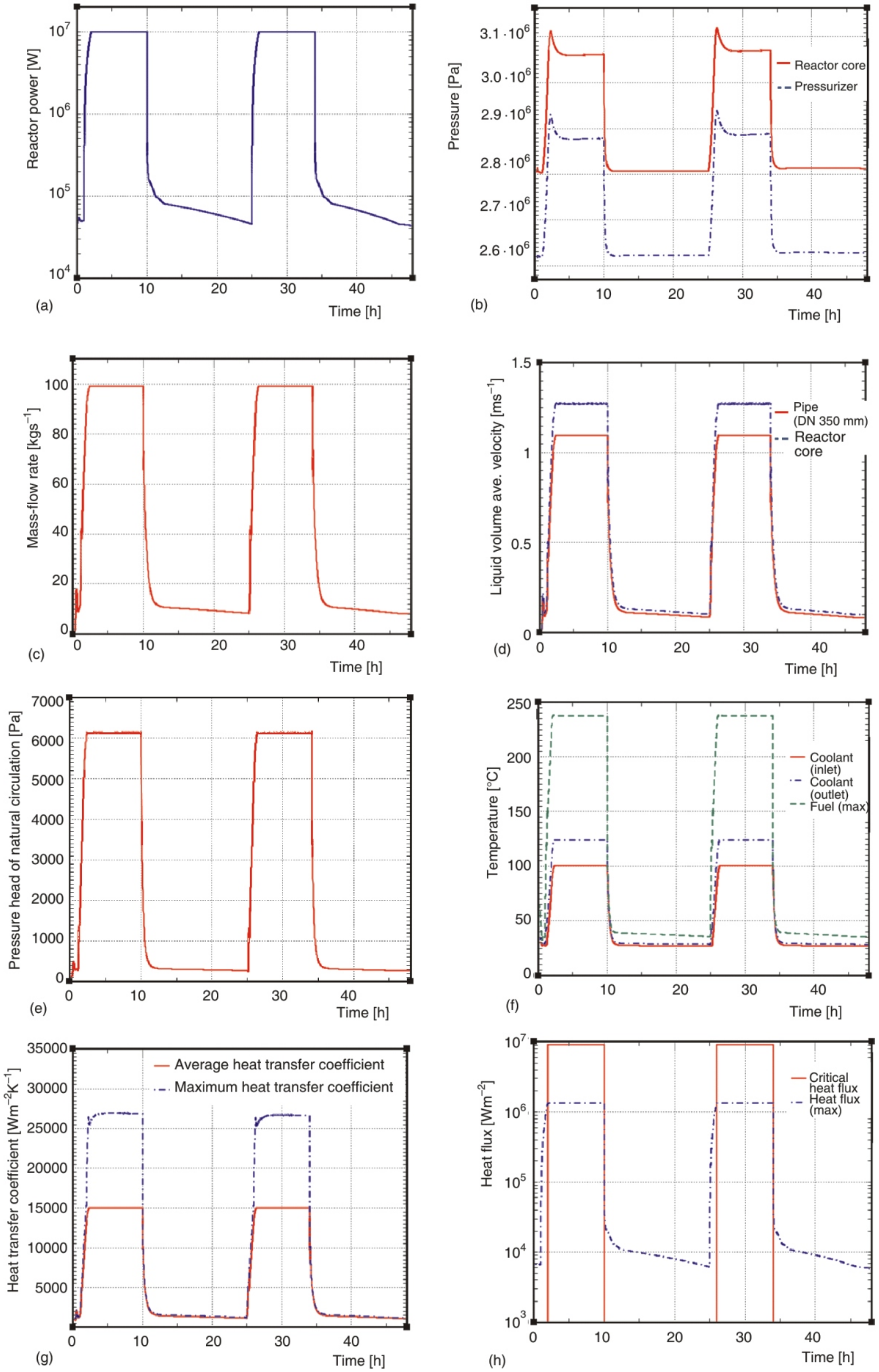

Figure 6. Dynamics of changes in thermal-hydraulic parameters within 48 hours for two campaigns of reactor operation for 8 hours

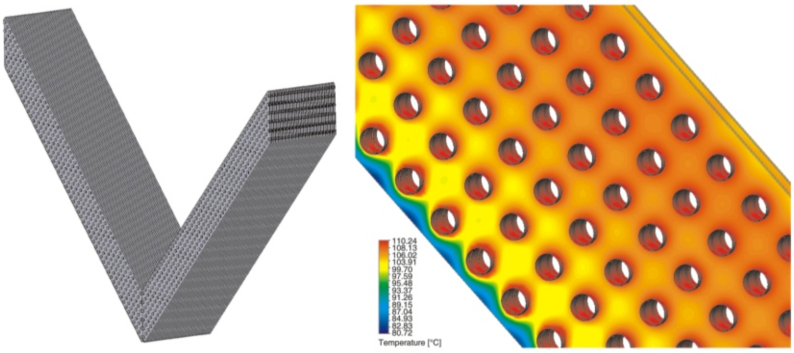

Figure 7. Fragment of the section of the air plate heat exchanger (dry cooler) and temperature distribution over the heat release plate

only natural convection of the final heat sink through the heat exchanger (dry cooler), which greatly simplifies the system of heat removal in the research reactor installation. The disadvantage of using evaporated water in the secondary circuit is the need to constantly replenish water losses, so the system cannot be considered as completely passive. When reactor power is 10 MW, the distillate consumption of evaporation should be about 10 m^3h^{-1}.

When atmospheric air is used as the final absorber, the cooling system can operate in a completely passive mode. The disadvantage of this solution is the relatively low efficiency of heat transfer from the heated coolant to the air; therefore, air heat exchangers have significantly larger dimensions than water-cooled heat exchangers of similar capacity. Dry coolers equipped with fans for circulating cooling air are usually used for these purposes, but in this case, the reactor cooling system is not quite passive either, since it depends on the power supply system. It is interesting to evaluate the parameters of the air heat exchanger when using natural convection not only of water, but of air as well, although in this case it is necessary to ensure the draft of heated air due to the presence of an exhaust pipe.

To estimate the parameters of the air heat exchanger, simulation of air heating was used during air passage through the gaps between stainless steel metal plates that were 1 mm thick, 125 mm wide, and 1100 mm long. The distance between steel plates with holes for tubes with heat carrier was 5 mm, and the outer diameter of the tubes themselves was 10 mm. Heated water in the tubes of an air heat exchanger with a diameter of 10 mm 0.8 mm, fig. 7, transferred heat to the plates and was cooled from 124 °C down to 100 °C. The heating of steel plates was calculated in Solid Works/Flow Simulation [9] and is also shown in fig. 7. The plates were located close to each other at an angle of 45 °C and formed a zigzag structure through which heated air passed in vertical direction. The main parameters of the air heat exchanger are shown in tab. 2. Figures 8-10 show the distribution of temperature, velocity, and air pressure as it passed through the air heat exchanger.

Calculation analysis using SolidWorks/Flow Simulation showed that in order to ensure heat transfer by air convection from water coolant of a research reactor with power of 10 MW, it is sufficient to have an air heat exchanger with the parameters previously listed and an exhaust pipe which is 44 m high and has a bore diameter of 7 m to 5 m, fig. 11(a). Such an exhaust pipe with an air heat exchanger at an ambient air temperature of 20 °C and a difference in the density of air inside and outside the pipe of 0.21 kgm^{-3} provides heating of air increasing its temperature for 63 °C, fig. 11(d), thrust of about 80 Pa, fig. 11(b), and the rate of circulation of heated air in the pipe of about 10 ms^{-1}, fig. 11(c). The mass-flow rate of cooling air is 564 th^{-1}.

Table 2. The main parameters of the air heat exchanger (dry cooler)

Heat exchanger parameters	Value
Material	Stainless steel
The thermal conductivity of the material [$Wm^{-1}K^{-1}$]	16
Transmitted power [MW]	10
Heat transfer surface [m^2]:	
– Air	6608
– Water	1051
Height [m]	0.93
Length [m]	10
Width [m]	8.8
Weight [kg]	23400
Water consumption [kgh^{-1}]	360000
Air consumption [kgh^{-1}]	564000
Water temperature drop [°C]	24
The temperature difference in the air [°C]	63
Pressure drop through the air [Pa]	20

Therefore, by creating a system of natural circulation of the primary coolant and a system of natural convection of cooling air, it is possible to operate a universal and completely passive system for removing heat from the cores of research reactors operating at a relatively high thermal power.

CALCULATION MODELLING OF EMERGENCY PROCESSES USING RELAP5

Functioning of research reactor systems with natural circulation in emergency conditions

The proposed cooling system allows excluding of the following several important events related to

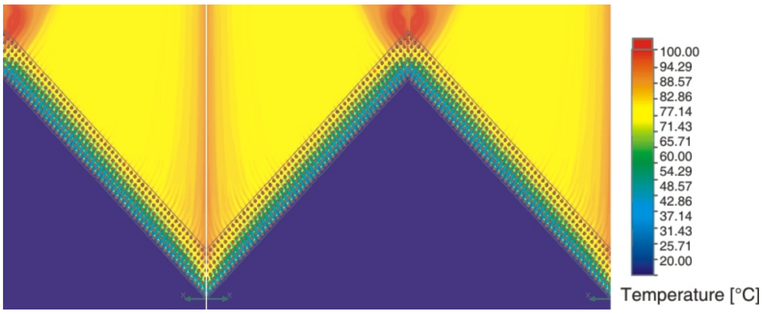

Figure 8. Distribution of air temperature when passing through the air heat exchanger

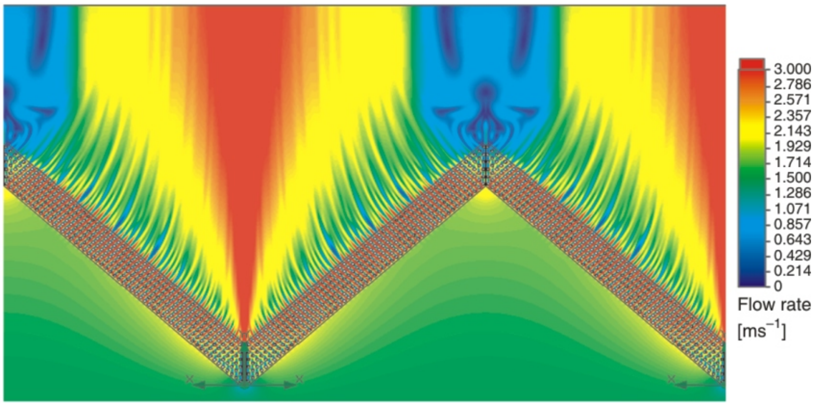

Figure 9. Distribution of air velocity when passing through the air heat exchanger

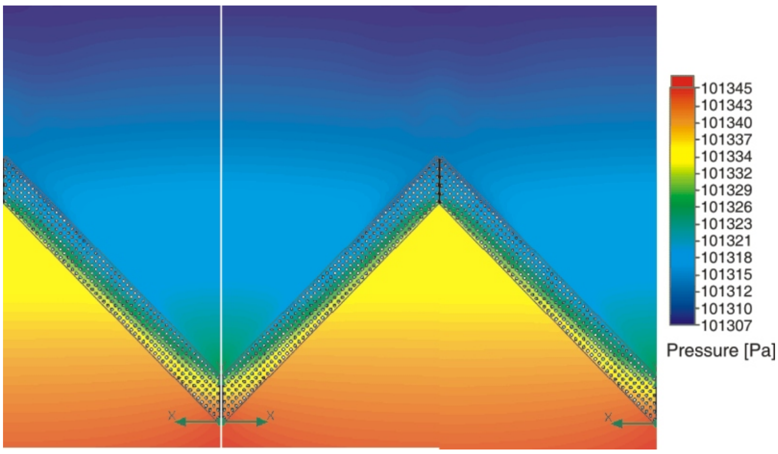

Figure 10. Distribution of air pressure when passing through the air heat exchanger

equipment failures from the recommended list of initiating events in design analysis of design based accidents for research nuclear reactors, due to the lack of such equipment in the system [10]:

– loss of normal electrical power,
– primary pump failure,
– reduction of flow of the primary coolant (*e. g.* due to valve failure or blockage in piping or a heat exchanger), failure of an emergency cooling system, and
– loss of heat sink due to the failure of a valve or pump.

This study does not consider initiating events that lead to the introduction of excess reactivity at the reactor, disruptions in the work with nuclear fuel, natural phenomena and events of anthropogenic origin. The consideration of emergency situations is limited to disturbances related to the operation of the cooling loop of the research reactor, and specifically to the depressurization of this circuit in case of:

– rupture of the main circulation pipe,
– rupture of the pressurizer pipe, and
– failure or depressurization of the gas pressure maintenance system in the reactor cooling circuit.

Figure 12 shows a nodalization diagram for analyzing the development of a situation in case of the listed initiating events related to pipeline destruction of the cooling loop of a research reactor. A rupture of the pipeline is simulated by opening Vel-756 valve, and the room into which the coolant flows is modeled by hydrodynamic component P-760. Since the pipelines are located in a vertical and conditionally leak-tight tunnel, the geometrical dimensions of the P-760 correspond to geometrical parameters of the tunnel.

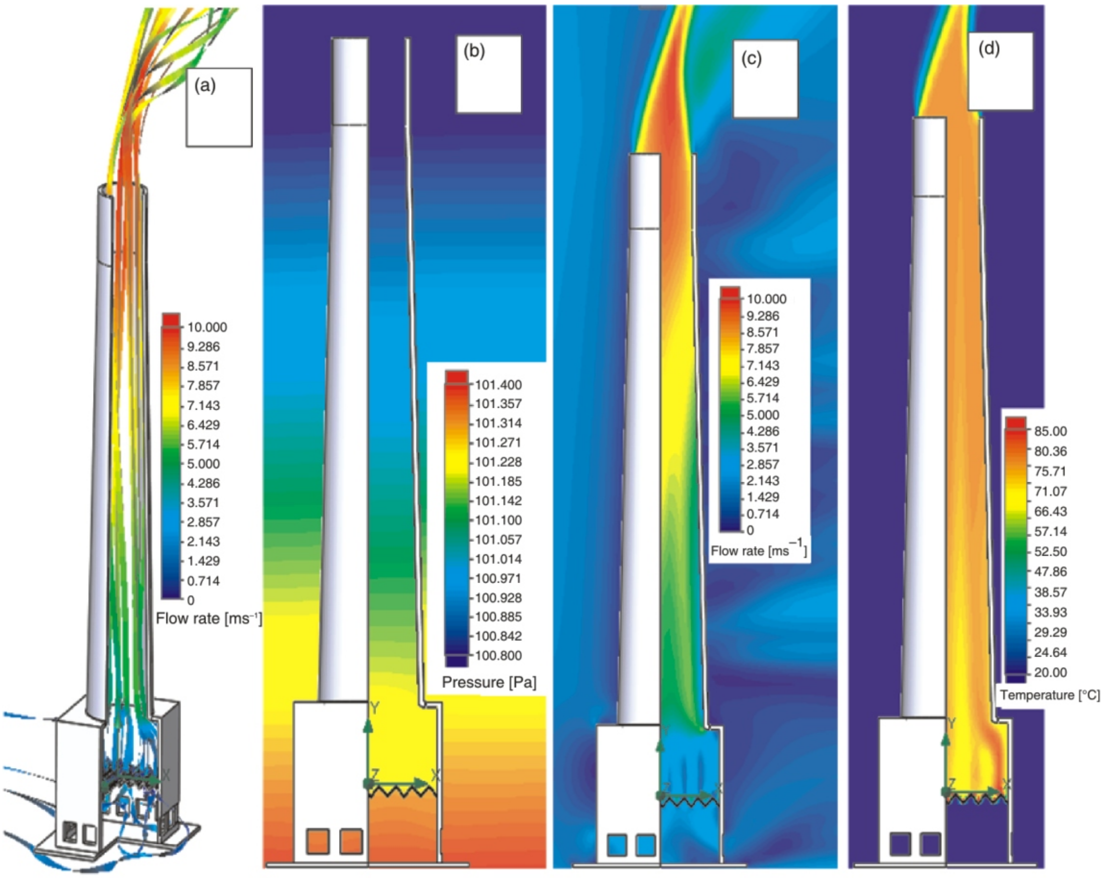

Figure 11. Air cooling system for primary coolant with natural convection

Main circulation pipe rupture

When the pipeline breaks it is assumed that all the pipelines of the reactor cooling circuit and pipelines from the P-AIS are located in a vertical tunnel filled with water.

When the circulation pipe breaks, simultaneously with the flow of the hot coolant into the tunnel the hot coolant in the reactor vessel is partially replaced with cold coolant from P-AIS, which ensures cooling down of the core when the main circulation loop is broken. This is possible because a new circuit of natural circulation is formed between fuel assemblies themselves in the reactor core due to the difference in the average coolant density in the central and the peripheral fuel assemblies caused by a significant difference in the power of residual energy release.

When analyzing RELAP5, it was assumed that the opening of the valve that simulates the destruction of the lifting section of the P-160 pipeline in its lower part occurs in one second, which leads to leakage and filling of the tunnel with water, fig. 13(a), a rapid decrease of pressure in the reactor cooling circuit, fig. 13(b), a short-term (less than one minute) reversal of the circulation in the reactor core, fig. 13(c) and the subsequent establishment of coolant natural circulation between fuel assemblies, in which descending movement is observed in corner cells, and lifting movement is observed in all the others, figs. 13(d) and (e). The partially cooled water in the housing (after emptying P-AIS) is gradually heated due to residual energy release, fig. 13(f). By extrapolating the calculation results of the increase in water temperature in the housing, it can be assumed that without taking into account the heat sink through the walls of the reactor vessel, evaporation and possible addition of cold water, after 18 hours boiling in the reactor can turn into water volume boiling, which creates the risk of overheating and destruction of fuel rods. On the other hand, 12 hours after the reactor shutdown, the maximum heat flux from the surface of fuel elements drops to 5 kWm^{-2}, fig. 13(g), and the temperature of fuel elements reaches 135 °C, fig. 13(h), which is close to the boiling point of water at the depth of 32 m. Such a low heat flux from the surface of fuel claddings suggests the possibility of safe heat removal from the fuel assembly even in the water bulk boiling mode in the reactor vessel.

Pipeline rupture from P-AIS

In order to ensure the safety of heat removal from the reactor core when the pipeline breaks from the P-AIS, two independent P-AIS systems are con-

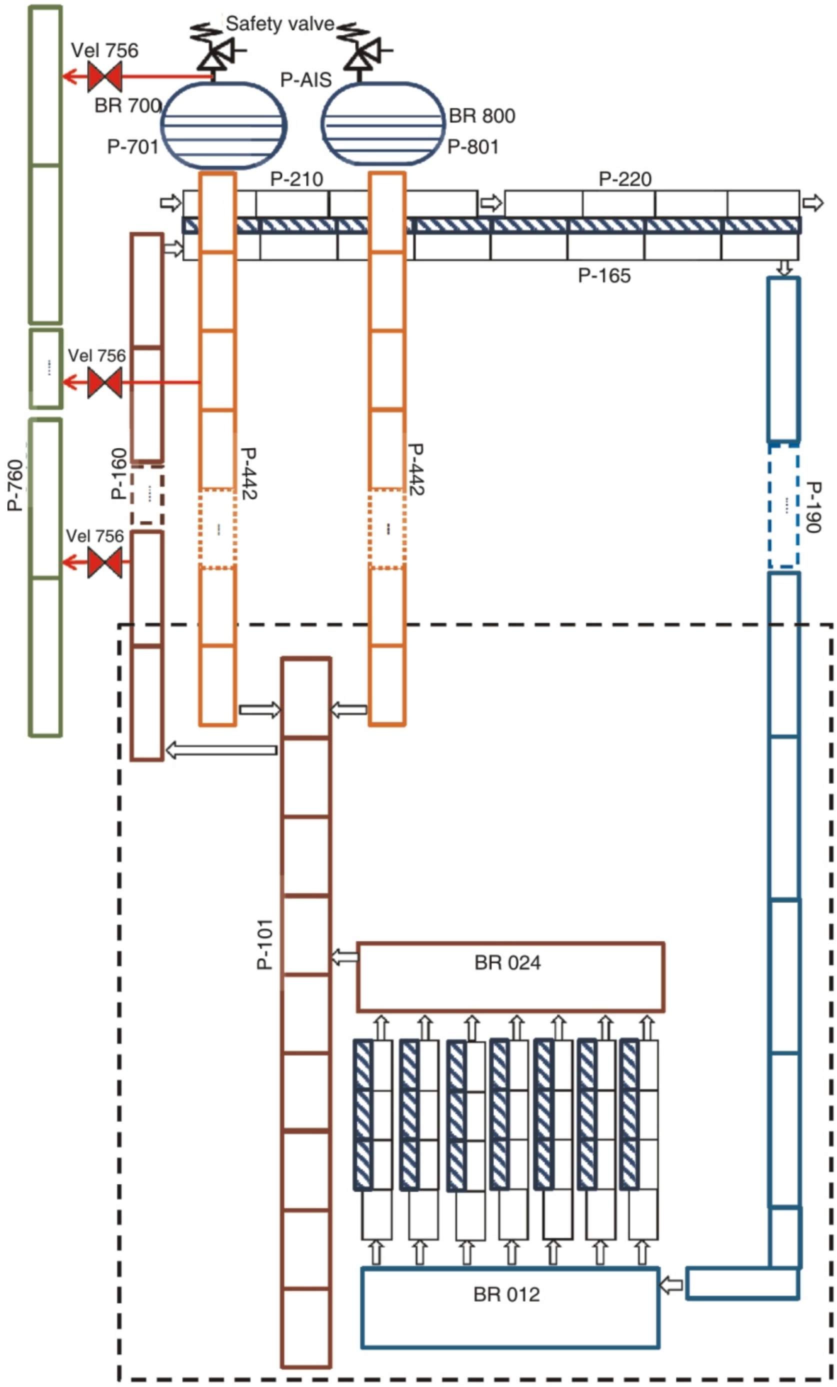

Figure 12. Nodalization diagram for modeling emergency situations with indication of rupture points (Vel-756) for circulation pipelines; P-AIS pipeline; gas pressure system

nected to the reactor vessel. This ensures forced supply of cold water to the reactor vessel, fig. 14(a), and safe cooling of the core, even if one of the pressure compensation system pipelines breaks across its full cross-section and all the cold water from the P-AIS tank connected to it goes into the tunnel. After the pipeline ruptures to the P-AIS, one second after the initial event, the emergency pressure protection trips, and the main natural circulation circuit stops working due to the ingress of gas into the system; the coolant flow rate becomes close to zero. After this, heat removal from fuel rods begins only due to the establishment of natural circulation between the fuel assemblies due to the difference in their power of residual energy release, figs. 14(b) and (c). Maximum temperature of fuel rods at the same time remains at an acceptable level, fig. 14(d).

As evident from the results of the calculation analysis, when the pipelines of the reactor cooling circuit (main circulation pipe and pipeline from the P-AIS system) are destroyed, natural circulation between fuel assemblies is established, which ensures

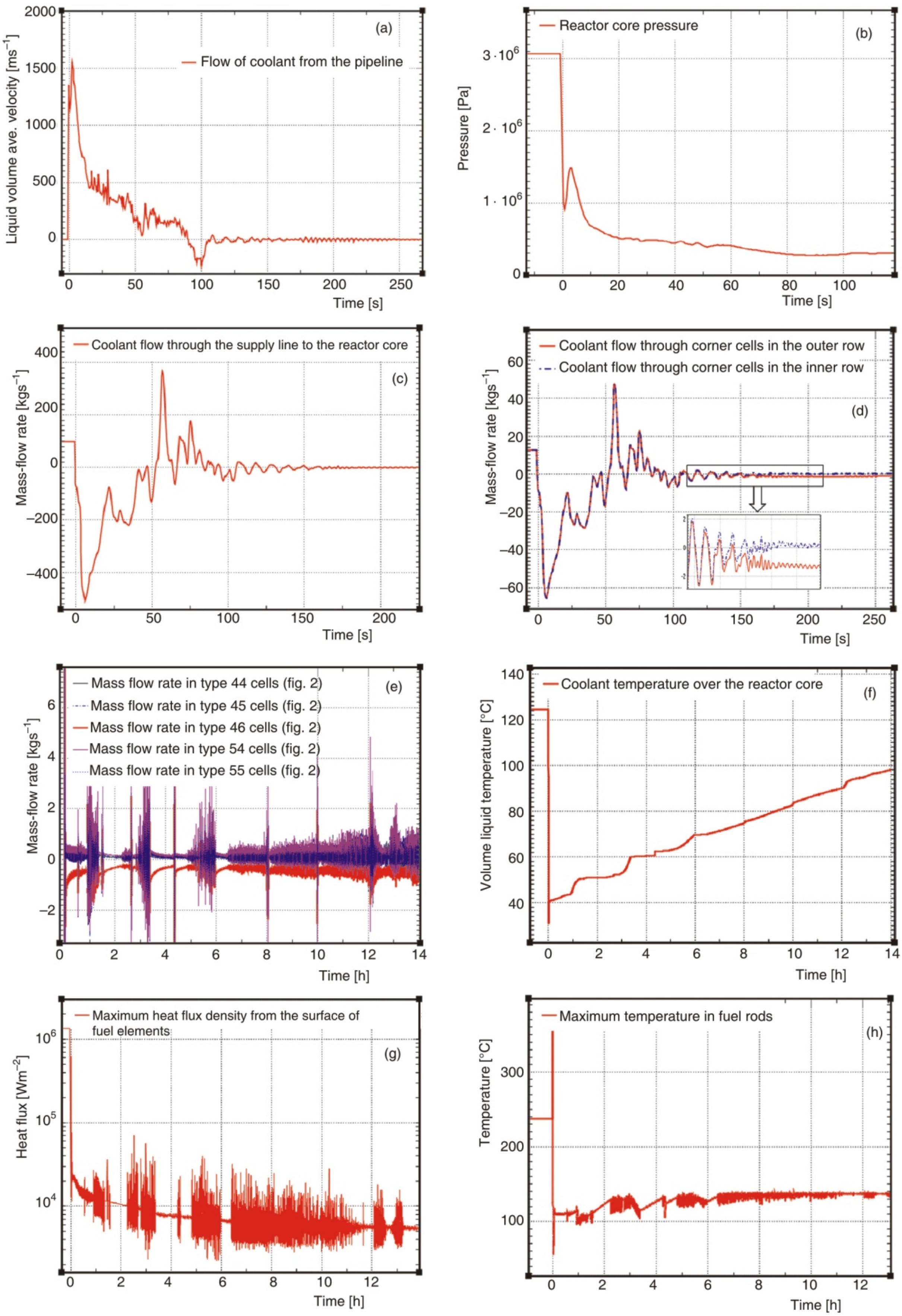

Figure 13. Graphs of changes in thermal-hydraulic parameters of a reactor with natural circulation of coolant in case of destruction of the main circulation pipeline

sufficient heat removal to maintain the tightness of fuel elements.

Depressurization of gas pressure maintenance system P-AIS

Another event that leads to the occurrence of an emergency at a research reactor could be depressurization of gas pressure circuit or the failure of gas pressure maintenance system in the P-AIS. For the analysis of such a situation, gas pipe DN 50 failure was chosen, which in case of an unacceptable increase of pressure ensures the discharge of excessive gas through the P-AIS system safety valve. After the pipe breaks, there is a leak of gas and water from the depressurized tank of the P-AIS system. At the signal of reduced pressure in the tank of the P-AIS system, fig. 15(a), emergency protection system is triggered and the reactor goes into subcritical state with a rapid decrease in thermal power. Compressed gas from the undamaged compensator squeezes cold water into the reactor core, which lowers the temperature of the coolant there and prevents it from boiling up with a decrease in pressure. Coolant circulation along the main circuit is preserved, but the difference between the coolant temperature in the descending and the lifting sections decreases, fig.15(b), which leads to a decrease in the driving pressure of natural circulation, as well as to a decrease in mass flow and coolant velocity in the circulation loops figs. 15(c) and (d). Two short-term reverse circulations at the 40th and 225th minute occur after the initial event, caused by the cold coolant from the P-AIS system entering the *hot* section of the circulation loop pipeline. After coolant heating in the upper part of the reactor vessel, normal circulation in the circuit is restored. The temperature of the coolant and the maximum temperature of the fuel rods in the cells of the reactor core in the situation under consideration gradually decrease figs. 15(e) and (f) ensuring safe heat removal from the core of the research reactor installation.

CONCLUSIONS

This paper aims at substantiating the benefits of creating a simple and reliable passive cooling system

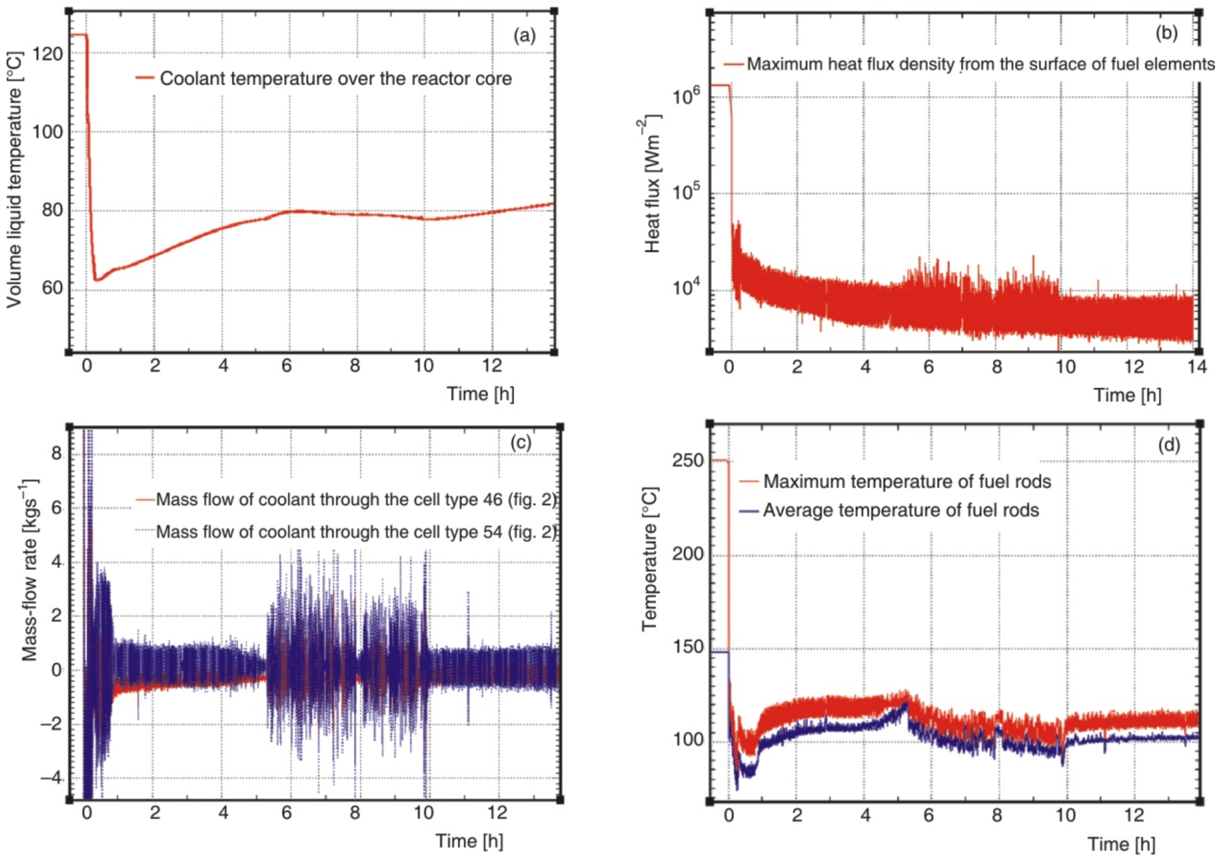

Figure 14. Graphs of changes of thermal-hydraulic parameters of a reactor system with natural circulation of coolant in case of destruction of the supply pipeline from the P-AIS system

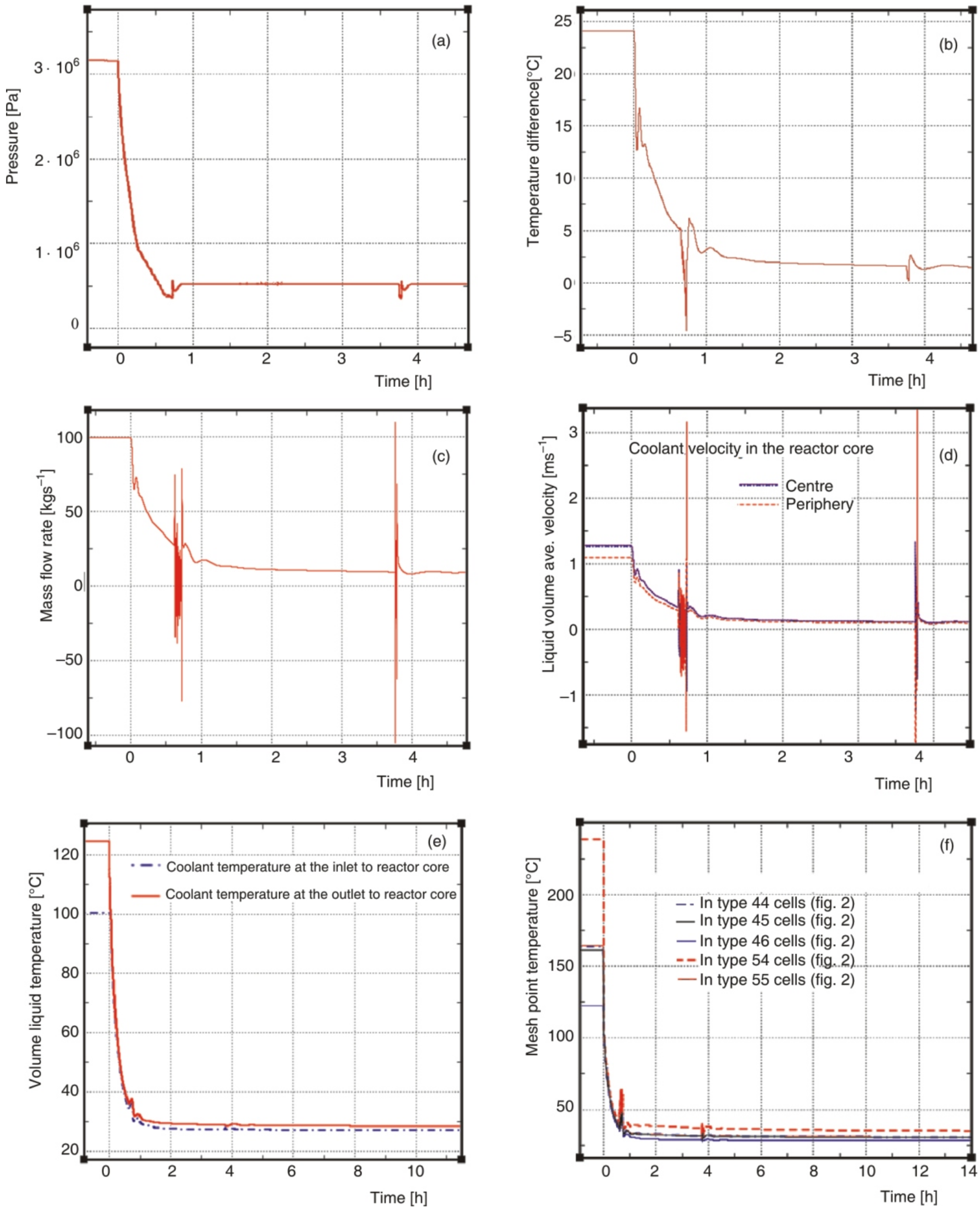

Figure 15. Graphs of changes in thermal-hydraulic parameters of the reactor cooling system with natural circulation in case of rupture in the pipeline of the safety valve of the P-AIS system

for the core of a research reactor built on the principle of natural convection. By using the example of known parameters of SM-3 core and selected parameters of natural circulation circuits of the coolant and cooling air, the RELAP5 model calculations show the attainability of high level of reactor thermal power (10 MW) which corresponds to the maximum neutron flux density of $5 \cdot 10^{14}$ cm^{-2}s^{-1}. The paper presents a conceptual 3-D model of a reactor installation, and technological and design (nodalization) scheme. In RELAP5, the changes in thermal-hydraulic parameters when the reactor reached nominal power level and continued operation in steady-state mode were analyzed, along with an emergency situation of depressurization of the cooling circuit. It was established that in the assumed situation of depressurization of the cooling circuit for the considered parameters of the reactor research installation, a safe heat sink from the core is ensured and the tightness of fuel element claddings is not affected. By using 3-D modeling, main parameters were calculated of the air cooling circuit equipment with natural convection of air, an air heat exchanger and an exhaust vent pipe. The results of thermal-hydraulic calculation of heat transfer from the reactor cooling water circuit to the final recipient – atmospheric air are also presented.

The presented paper shows that it is possible to create a safely functioning and a fully passive system for removing heat from a research reactor core, which can be used as a universal cooling system for a wide range of reactor installations for various purposes.

AUTHORS' CONTRIBUTIONS

The computational models of the RELAP5 were developed by I. Uzikova under the direction of V. Uzikov. The computational models of the SOLID WORKS Flow Simulation model were developed by V. Uzikov. All authors participated in the discussion and analysis of the results. The manuscript was written by I. Uzikova and V. Uzikov, and figures were prepared by V. Uzikov. All authors participated in the preparation of the final version of the manuscript.

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УНИВЕРЗАЛНИ СИСТЕМ ПАСИВНОГ ОДВОЂЕЊА ТОПЛОТЕ ИЗ ЈЕЗГРА ИСТРАЖИВАЧКОГ РЕАКТОРА

У раду су приказани резултати анализе универзалног расхладног система за језгра истраживачких реактора изграђених на пасивном принципу природне конвекције. Приказани су тродимензионални модел, технолошки и структурни дијаграми реакторске инсталације и примери нумеричке процене прелазних појава током рада расхладног круга у нормалним и ванредним режимима, оправдајући могућност коришћења таквог расхладног система у истраживачким реакторима ниске и средње снаге. Основна карактеристика представљеног пасивног система је одсуство у расхладном кругу не само активних елеманата, као што су циркулационе пумпе и запорни и контролни вентили, већ и пасивни елементи са покретним деловима, као што је неповратни вентил. Круг хлађења укључује само капацитивну опрему, цевоводе и измењиваче топлоте. Одсуство елемената са механичким покретним деловима може значајно смањити трошкова. Разноврсност предложеног система омогућава да се користи за широк спектар истраживачких реакторских постројења различитих капацитета дизајнираних за имплементацију програма у различитим областима истраживања и примењених послова везаних за нуклеарну технологију.

Кључне речи: истраживачки реактор, пасивни систем, систем хлађења реактора, сигурност нуклеарног реактора, природна конвекција