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Research Article

The Concept of the Heat Removal System of a High-Flux Research Reactor

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Achieving high neutron fluxes in research pressurized water reactors is directly related to the intensity of the coolant flow through the core and the pressure in it, which provides an increased saturation temperature and a margin to critical heat flux. Therefore, it is practically impossible to provide very high neutron fluxes in pool-type reactors, especially in the case of downward movement of the coolant in the core. At the same time, vessel-type research reactors (for example, SM-3 and HFIR) make it possible to achieve neutron flux densities up to 4×10^{15} n/(cm^2 s), but at the same time, the risks of core degradation in case of violations in the heat removal system become quite high. The proposed concept of a heat removal system for a high-flux reactor facility combines the increased reliability of safe heat removal from the core and the convenience of handling irradiation cells, for example, in the production of isotopes. The concept provides for the location of a compact core in a pressurized vessel and the placement of a neutron reflector around the vessel in the reactor pool. Cooling of the reactor core in the housing and the irradiation channels in the neutron reflector is carried out by different systems of forced circulation of the coolant. At the same time, at the shutdown reactor, after opening the natural circulation valves, safe heat removal from the reactor core and the neutron reflector can be carried out by the water of the reactor pool. However, even with a complete failure of all forced circulation circuits, the evaporation of water from the surface of the pool makes it possible to safely remove the residual heat from the fuel assemblies and from the irradiation devices in the cells of the reflector.

1. Introduction

The need for safe, simple, and reliable research reactors encourages developers to look for optimal and competitive design solutions to provide a wide range of research in the areas [1]:

(i) Nuclear physics
(ii) Solid-state physics
(iii) Radiation materials science
(iv) Neutron activation analysis
(v) Neutron radiography of various products
(vi) Radiation doping of silicon
(vii) Production of isotopes for medical industrial purposes

Without carrying out a broad program of fundamental and applied research at nuclear research installations (NRI), it is impossible to justify the safety of nuclear power facilities.

However, we should remember that research nuclear installations are sources of nuclear and radiation hazards. Despite the lower power and, accordingly, the smaller amount of radioactive substances generated during the operation of nuclear facilities, their potential danger to the population and the environment is still high due to a number of specific features [2]:

(i) High frequency of transient modes during operation (start-ups, shutdowns, power changes in a wide range, and dynamic experiments), in which most often incidents occur in the operation of nuclear facilities
(ii) Frequent reloading of cores and constant movement of irradiated products (for research, to spent fuel pools, for long-term storage, for disposal, etc.)
(iii) High cyclicity of loads on the main equipment of the cores and the primary circuit due to the large number of short duration campaigns

(iv) High neutron flux density in the cores of research reactors, leading to a rapid increase in the limiting fluence for the elements of the cores and an increase in the probability of their failures

(v) The presence of highly enriched fuel exacerbates the problem of nonproliferation of nuclear materials and requires effective systems for their accounting and physical protection

(vi) Equipment with experimental devices and related features of operation

(vii) Fewer physical barriers than those in power reactors that prevent the spread of fission products, especially in pool research reactors and critical assemblies

For reactors over 1 MW, the maximum scope of safety requirements contained in the rules and regulations should be applied. The reduction in the scope of requirements is possible only for research reactors with a power of up to 1 MW and critical and subcritical facilities of almost zero power, which do not require forced emergency core cooling systems. These requirements are reduced based on their self-protection properties and based on specific safety justifications submitted by their owners for independent examination.

The task of the presented concept of a hybrid high-flux reactor plant is to substantiate the operability of a three-circuit heat removal scheme and self-protection properties in case of failures in the heat removal system.

2. Description of the High-Flow Reactor Plant

As an example of the application of the proposed concept of a three-loop heat removal system for nuclear facilities, a hybrid high-flux reactor plant with a capacity of 50 MW with a reactor core consisting of 30 VVR-KN fuel assemblies and six CPS working bodies is considered. In the central part of the core, there is a central moderating cavity (CMC) with irradiation cells, in which a maximum neutron flux of more than 4×10^{15} n/(cm^2 s) is provided. To reduce the absorption of neutrons, it is proposed to use a zirconium alloy as the material of the reactor pressure vessel. The irradiation cells in the CMC can be cooled both by a separate circuit of the loop channel located in the CMC area and by the primary coolant in the absence of a loop channel. The coolant movement in the core is ascending.

The lower part of the reactor vessel is made of zirconium alloy based on a core diameter of 480 mm. Based on the rapid increase in the limiting fluence for structural elements in the core area due to the high density of the neutron flux, it is necessary to provide for the possibility of replacing periodically a part of the zirconium body and the reflector for the service life extension of the reactor plant.

The priority methods of functioning of the heat removal systems of the NRI under consideration are the passive principles of operation of the coolant circulation systems in the cooling circuits set out in [3,4]. The only exceptions are the core cooling circuit and the pool cooling circuit when the reactor is operating at the nominal power level. In this case, the intensity of heat removal is ensured by pumping systems, which makes it possible to achieve maximum levels of energy release in fuel assemblies and, accordingly, a high neutron flux density.

The main principles on which the proposed concept of NRI is based are as follows:

(i) The fuel assemblies are cooled by a forced circulation circuit under high pressure.

(ii) Irradiation cells in the CMC can be cooled both by an autonomous circuit of the loop channel located in the CMC area and by the primary coolant, in the absence of a loop channel.

(iii) In the core, fixation of fuel assemblies and irradiation devices is provided for protection against ascent under the action of an upward flow.

(iv) Less energy-intensive elements (neutron reflector and irradiation devices in the reflector) are cooled by a forced circulation circuit under the pressure of the atmosphere and water column in the reactor pool.

(v) In the heat removal system, preference is given to equipment with a passive principle of operation with a low probability of failure.

(vi) Depressurization of any element of the equipment of the heat removal system circuits does not lead to the release of activity into the environment.

(vii) The heat removal system uses only a refrigerant that does not cause deposits on the heat exchange tubes (distillate, atmospheric air).

(viii) The intermediate circuit uses a heat exchange system with the most efficient circulation (natural circulation) and heat transfer (boiling and condensation) mechanisms.

(ix) The heat transfer circuit to the final absorber (atmospheric air) is made in the form of a chimney into which heated air enters after an air heat exchanger with water vapor condensation tubes of the industrial circuit (thermic syphon principle).

(x) The neutron reflector is located in the reactor pool, which makes it possible to provide the most convenient access to the irradiation cells for the replacement of irradiation devices, including replacement at the operating reactor.

(xi) Structural elements of the neutron reflector can be dismantled and replaced with another design and other material (beryllium, heavy water, and lined graphite) for new reactor shutdown tasks.

(xii) The neutron reflector and the irradiation devices located in it are cooled by the downward flow of the coolant of the pool cooling circuit (PCC).

(xiii) When the forced circulation of the coolant in the primary circuit is stopped, the passive valves ensure safe cooling of the core due to the supply of relatively cold water in the volume compensator tank and the transition to natural circulation in the fuel assemblies through the reactor pool.

(xiv) When the forced circulation of the coolant in the PCC is stopped, the passive valve ensures safe cooling of the reflector and the irradiation devices located in it due to the transition to natural circulation through the reactor pool.

(xv) In case of long-term cooldown of the reactor in emergency modes with nonoperating pumps, constant replenishment of water removed from the surface of the pool during evaporation should be provided.

Sketches of a three-dimensional model of the heat removal system of a 50 MW research pressure vessel reactor with a beryllium reflector located in the pool are shown in Figure 1.

As can be seen from Figure 1, the minimum amount of equipment is involved in the reactor heat removal system, most of which uses the principle of passive operation, which reduces the risks of failures and increases the safety of RNI.

2.1. Main Parameters of the Reactor Plant. The main thermal and neutron parameters of the 50 MW research reactor facility under consideration with the calculated characteristics substantiated are given in Table 1.

2.2. Reactor Core. As a reactor core with a central moderating cavity (Figure 2), a design of 30 fuel assemblies of the VVR-KN type, consisting of 24 8-fuel fuel assemblies (1) and six 5-fuel assemblies (2) with working bodies located in the center, is considered CPS (3). In the central part of the core, there is a moderating cavity with irradiation cells, in which a maximum neutron flux of more than 2×10^{15} n/(cm^2 s) is provided. Table 2 presents the characteristics of fuel assemblies taken for the calculation analysis.

The neutron-physical calculation of the reactor is carried out for the geometry presented in Figure 3 using the MCU program [6] for a reactor power of 50 MW. For this variant, heavy water was taken as a reflector. In the figure, the numbers indicate the position 4 control points, the calculated parameters of the neutron flux for which are indicated in Table 3.

Figure 4 shows the calculated distribution of the specific energy release over the height of the maximum thermally stressed fuel element used to determine the maximum temperatures in the fuel assemblies.

3. Description of Heat Dissipation System

The system of heat removal from nuclear facilities consists of two subsystems:

(i) Heat removal system from the core

(ii) Heat removal system from the reactor pool coolant

Each of these subsystems to prevent the release of radionuclides into the environment is made according to a three-loop scheme. The primary circuit cools the core or the reflector; the intermediate circuits are based on the principle of a thermic syphon. The secondary circuit transfers heat from the primary circuit to the third circuit through the system containing a steam generator and steam condensator. This third circuit is made in the form of a natural circulation chimney. The use of a chimney allows the use of atmospheric air as the final absorber, which solves the problem of water treatment to prevent sediments in the tubes of heat exchangers in the case of cooling towers or similar irrigation systems with water evaporation.

3.1. Core Heat Sink Circuit. The location of the reactor pressure vessel in the pool makes it possible to ensure the transition from forced circulation through the core to natural circulation in the passive mode (Figure 5).

3.2. Secondary Circuit. An important role in the implementation of the concept of a safe reactor is played by an intermediate circuit between the primary circuit and the circuit for transferring heat to the final heat sink. The conception of the secondary circuit is based on the principle of a thermic syphon (Figure 4).

This system is a closed gravitational system made of a steam generator in the lower part and a heat exchanger-condenser in the upper part, interconnected by a large steam pipeline and a condensate return pipeline under the action of gravitational forces (Figure 6).

The presence of this circuit prevents the risk of release of radioactive coolant into the environment in any situation with a rupture of pipelines of the cooling circuits.

A secondary circuit conception is capable of transferring large heat capacities at low-temperature gradients. It is a sealed structure, partially filled with a liquid coolant. In the heated part 1 (in the heating zone, or evaporation zone), the liquid coolant evaporates with the absorption of heat 2, and in the cooled part (cooling zone, or 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 pressure difference of saturated steam, determined by the temperature difference in the zones. The return of the liquid to the evaporation zone is carried out through pipeline 5 due to gravity.

For the considered parameters of the reactor plant, the design parameters of the steam generator for heat transfer from the reactor are given in Table 4 and Table 5.

3.3. Heat Transfer Circuit to the Final Heat Sink (Atmospheric Air). The heat transfer circuit to the final heat sink is a conventional large chimney in which cooling air is circulated through the air heat exchanger-condenser due to the difference in hydrostatic pressure of the heated air in the pipe and atmospheric air (Figure 7). It should be noted that only nonradioactive media (secondary steam, condensate, and atmospheric air) circulate in the air heat exchanger housing.

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

(i) Pipe height of 300 m

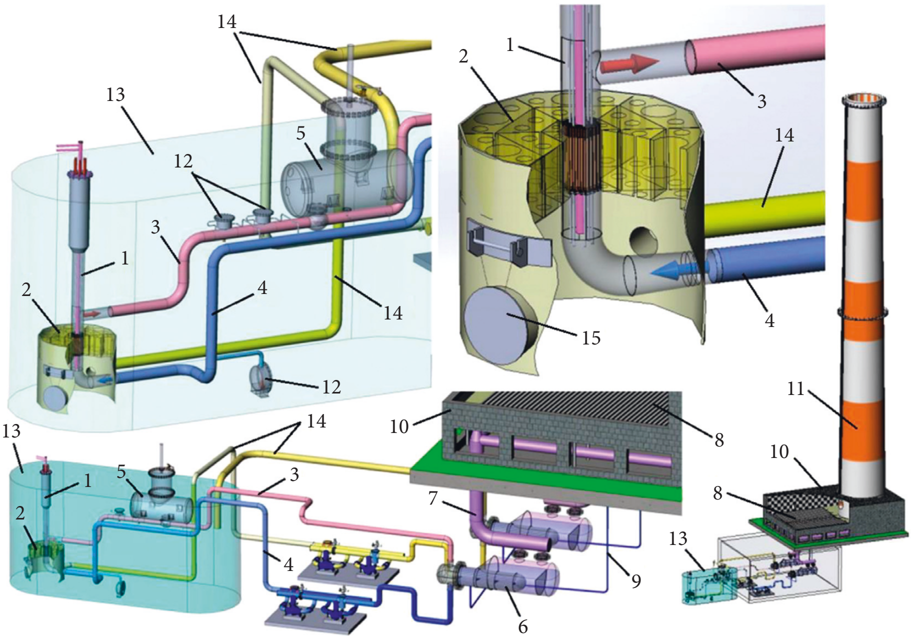

Figure 1: Sketches of a three-dimensional model of the heat removal system of a 50 MW hybrid nuclear facility with a beryllium reflector located in the pool: (1) reactor vessel; (2) neutron reflector; (3) outlet (hot) pipeline; (4) supply (cold) pipeline; (5) volume compensator; (6) steam generator; (7) steam pipe; (8) air heat exchanger; (9) condensate return; (10) air heat exchanger housing; (11) chimney; (12) valves of the shortened natural circulation circuit; (13) reactor pool; (14) pipelines of the pool cooling circuit; (15) natural circulation valve through the reflector.

Table 1: Main parameters of a high-flux RNI.

Reactor characteristics	Value/meaning
Reactor type	Pressurized water reactor with intermediate neutron spectrum and central trap
Power, MW	50
Maximum neutron flux density, $cm^{-2}s^{-1}$	2.2×10^{15}
Fuel	Uranium dioxide, 20% U-235 enrichment
Core geometry	Cylindrical shape with a neutron trap in the center
Number of cells for fuel assemblies, pcs	30
Fuel assembly type, pcs	VVR-KN
Including the following	
5-pipe, pcs	6
8-pipe, pcs	24
Number of cooling systems, pcs	2
Number of cooling circuits, pcs	3
Coolant of the 1primary circuit	Light water
Core diameter, mm	Ø480
Core height, mm	600
Coolant flowrate in the primary circuit, $t\,h^{-1}$	1208
Inlet temperature of the fuel assembly, °C	124
Outlet temperature of the fuel assembly, °C	159
Coolant heating in the core, °C	35
Maximum temperature of fuel elements, °C	229
Head loss in the core, kPa	270
Pressure at the core outlet, Pa	4.7×10^{6}
Hydraulic diameter of circulation pipes mm	400
Number of primary circuit circulation pumps, pcs	2
Number of circulating pumps, pcs	2

Table 2: Characteristics of fuel assemblies VVR-KN [5].

Parameter	Value/meaning
Enrichment 235U, %	19.7
Density of uranium, $g \times cm^{-3}$	2.8
Mass of 235U in fuel assemblies, g	
8-pipe	250
5-pipe	199
Number of fuel rods	
8-pipe	Eight
5-pipe	5
Fuel rod thickness, mm	1.6
Core thickness, mm	0.7
Shell thickness, mm	0.45
Heat transfer surface area, m^2	1.34

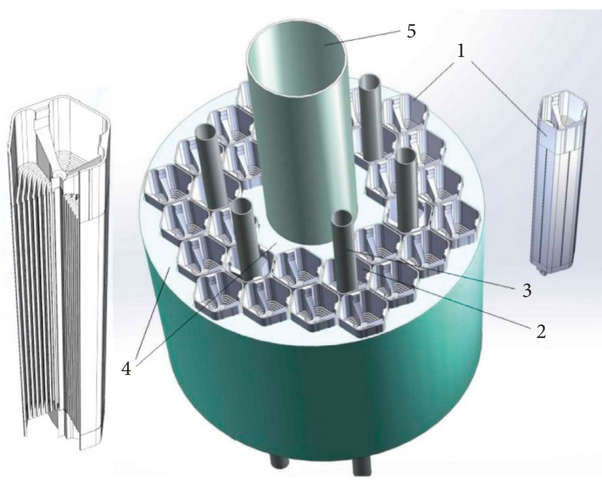

Figure 2: 3D model of the reactor core: (1) 8-pipe fuel asemblies of the VVR-KN type; (2) 5-pipe fuel assemblies of the VVR-KN type; (3) CPS channel; (4) displacer; (5) central retarding cavity.

(ii) Bottom flow section diameter of 12.6 m

(iii) Upper flow section diameter at the top of 10.4 m

For these chimney parameters, calculations by Solid Works/Flow Simulation [7] for a heat sink power of 50 MW gave the results shown in Table 5.

3.4. Reflector Cooling Circuit in the Reactor Pool. The applied value of a reactor plant is determined, first of all, by the possibilities of irradiation and production of isotopes. The presented concept provides a wide choice for optimizing the design of irradiation channels and reflector material. If necessary, you can change the design of the hoop volumes and change the reflector blocks since they are located directly in the pool, and therefore access to them is open (Figure 8). The high density of neutrons in the core causes a significant level of heat release in the reflector and in the irradiation channels located in it; therefore, cooling is provided by a forced downward flow of coolant, which is then fed into the cooling system, the pool cooling circuit (PCC), and the cooled water is returned to the reactor pool.

Heat transfer from RNI to the final heat sink, for example, to atmospheric air, can be organized similarly to heat removal from the primary circuit using the same chimney.

3.5. Passive System for Emergency Heat Removal from the Core. An important component of the proposed concept is the transition to natural cooling of the core with the use of passive safety systems. The basis of these systems is natural circulation valves placed directly in the pool of the reactor facility and opening when the flow rate through the core decreases (valves on the coolant outlet pipeline from the reactor), as well as when the pressure in the primary circuit decreases (valves on the pressure pipeline for supplying coolant to the reactor).

The view and location of these valves in the primary circuit are shown in Figure 9. The upper left part of Figure 9 shows the valves that operate to reduce the flow rate through the core. The principle of their work is based on the fact that under the action of higher pressure in the water supply pipeline to the reactor, the piston moves to the upper position and the saddle connected with it closes the outlet in the lower part of the pipeline at the outlet of the reactor. With a decrease in flow rate, the pressure drop across the reactor becomes insufficient to keep the piston in the raised state, so it lowers under the action of its own weight and the pressure drop on the seat, thereby opening a channel for the coolant to exit the primary circuit into the reactor pool. This leads to the emptying of the pressure compensator with a large volume of relatively cold water (acting as an emergency reactor cooling system, ECCS), the supply of coolant to the core, and a rapid pressure decrease in the primary circuit. After equalizing the pressure in the primary circuit and in the reactor pool, the valve shown in the lower right part of Figure 9 opens. In essence, this valve is a check valve and can be replaced with a standard design; however, in order to prevent jamming, the shut-off element is made in the form of a suspended-on-cable disk that closes the hole when the pressure in the primary circuit exceeds the pressure in the pool.

4. Analysis of Operating Modes of the RNI Cooling System

The results of the computational analysis for the three-loop model of the reactor plant in the RELAP5/MOD3.2 code [8] showed the efficiency of the proposed cooling system and the achievability of a high level of reactor thermal power (50 MW), which corresponds to the maximum neutron flux density in the central moderating cavity above $2 \times 10^{15}\ cm^{-2}s^{-1}$. Due to the limited possibility of a detailed presentation of the calculated results, only some data on the dynamics of changes in thermodynamic parameters in the heat removal system from the core in the steady-state mode and in the case of blackout are given below. The results of calculating the thermal-hydraulic parameters of the heat removal system from the reflector and the pool are not presented as less significant.

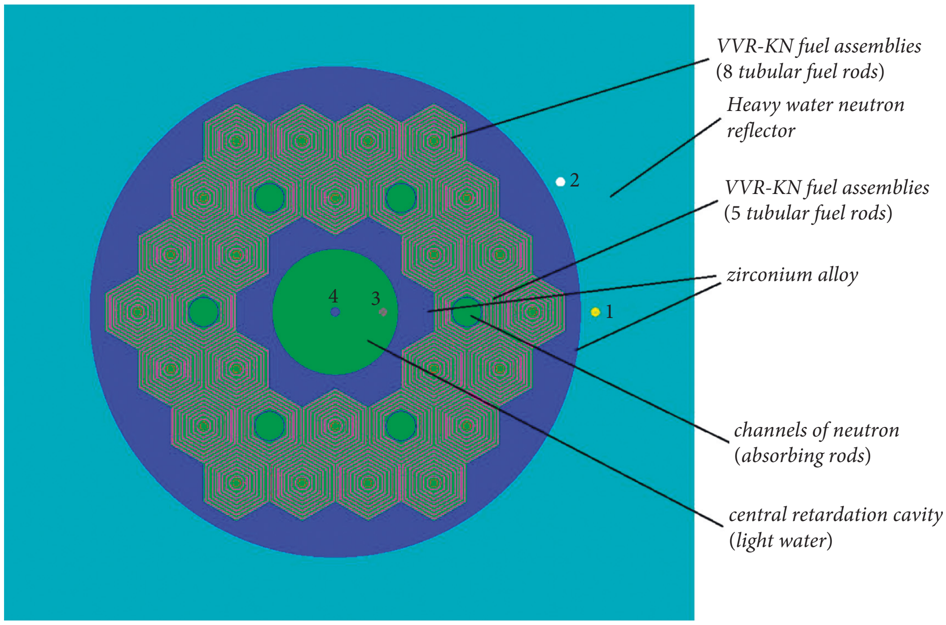

Figure 3: Diagram of core geometry with an indication of materials for neutron-physical calculation.

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

Detector number	Neutron flux density					Calculation error			
	Over 0.1 MeV	0.1 MeV to 1 keV	1 keV to 0.5 eV	Less than 0.5 eV	General flow	Over 0.1 MeV	0.1 MeV to 1 keV	1 keV to 0.5 eV	Less than 0.5 eV
1	2.19 $E+14$	2.15 $E+14$	2.33 $E+14$	6.07 $E+14$	1.28 $E+15$	1.5%	1.5%	1.5%	1.0%
2	1.91 $E+14$	2.01 $E+14$	2.33 $E+14$	6.04 $E+14$	1.23 $E+15$	1.6%	1.6%	1.5%	1.0%
3	3.56 $E+14$	1.69 $E+14$	2.76 $E+14$	1.26 $E+15$	2.06 $E+15$	1.1%	1.5%	1.2%	0.9%
4	2.46 $E+14$	1.20 $E+14$	2.18 $E+14$	1.76 $E+15$	2.34 $E+15$	1.3%	1.8%	1.4%	0.8%

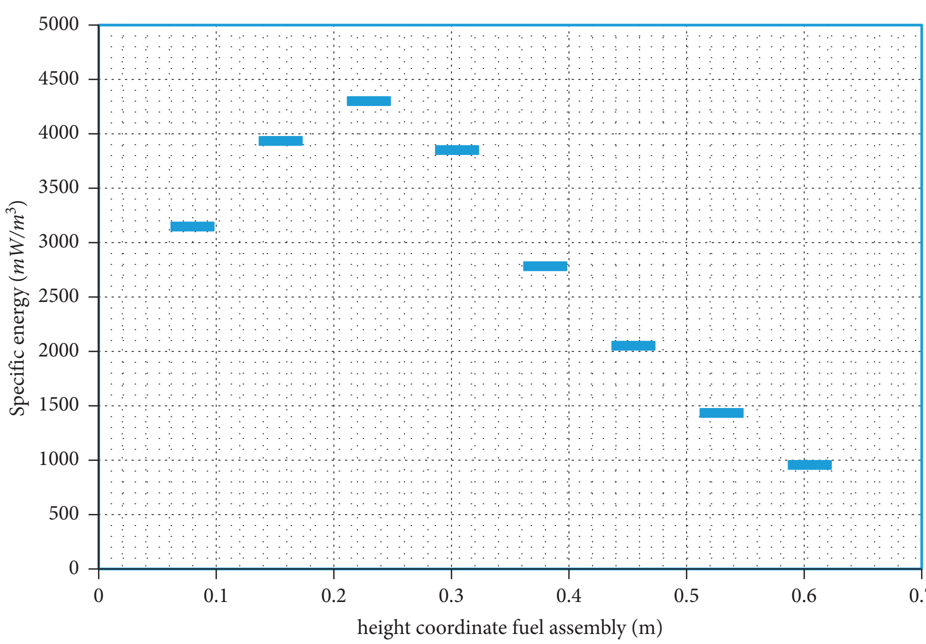

Figure 4: Estimated distribution of specific energy release over the height of the maximum heat-stressed fuel element.

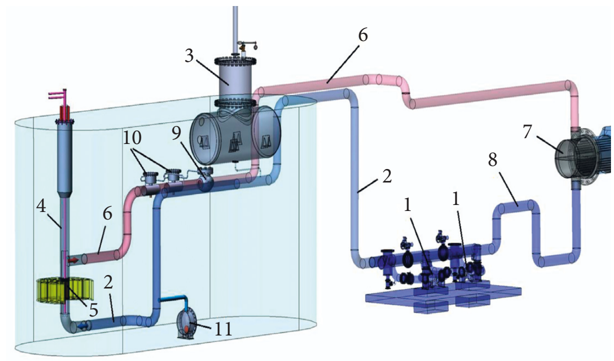

Figure 5: External view of the first core cooling circuit of a high-flow reactor plant. (1) circulation pumps; (2) pressure "cold" pipeline; (3) pressure compensator; (4) reactor vessel; (5) core; (6) outlet "hot" pipeline; (7) steam generator; (8) suction pipeline of pumps; (9) check valve; (10) exhaust valves of natural circulation; (11) inlet valve of natural circulation.

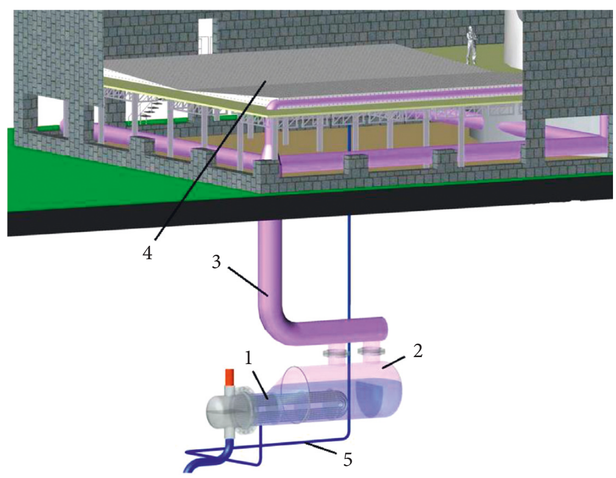

Figure 6: 3D model of the secondary circuit: (1) tubular steam generator; (2) steam generator body; (3) steam pipeline; (4) air heat exchanger-condenser; (5) condensate return pipeline.

Table 4: The main design parameters of the steam generator.

Parameter	Value/meaning
Pressure in the steam generator, kPa	205
Saturation temperature, °C	121
Heat carrier consumption (water), [$t \cdot h^{-1}$]	111.6
Return distillate temperature, °C	65
Heat exchange tubes diameter, mm	25
Heat exchange tubes thickness, mm	2
Heat exchange tubes number, pcs	1600
Heat exchange tubes length, m	13
Steam generator length, m	7.5

Table 5: The main design parameters of the chimney.

Parameter	Value/meaning
Power output, MW	50
Heat exchange surface, m^2	
Air side	33040
Water side	5255
Height, m	0.93
Length, m	50
Width, m	44
Inlet air temperature, °C	24
Outlet air temperature, °C	63.5
Air flowrate, kg/s	1237
Pressure drop across the chimney, kPa	3.5
Pressure drop across the steam condenser, Pa	85

4.1. Design Parameters of the Reactor Plant in the Nominal Mode. After the reactor plant is brought to a nominal power level of 50 MW, steady-state thermal and hydraulic parameters are set in the heat removal system from the core, shown in Figures 10 and 11.

Heat removal from the core in the primary circuit with a pressure of 5 MPa is ensured, for example, by two hermetically sealed GTSEN-146P pumps operating in parallel with a total coolant flow rate of ~1200 t/h. This ensures that the temperature of the coolant at the entrance to the core is 124°C, which heats at the exit from to 159°C.

Heat removal from the primary circuit is carried out using the steam generator of the secondary circuit. The saturated steam generated in the steam generator with a pressure of 0.2 MPa (saturated temperature 124°C) is sent to the air heat exchanger, where it is condensed from the heat

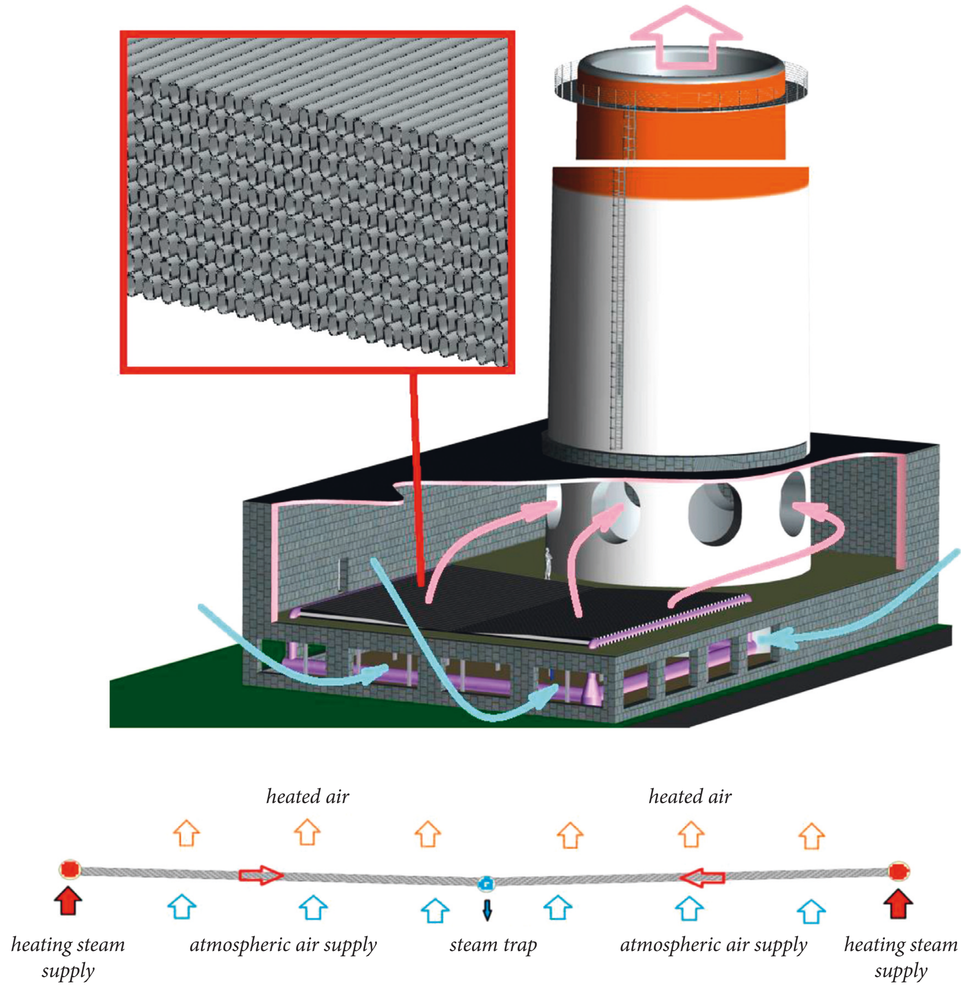

Figure 7: Scheme of heat transfer to the final heat sink, a fragment of a section of the air heat exchanger tube (top), and a scheme of coolant circulation in the air heat exchanger (bottom).

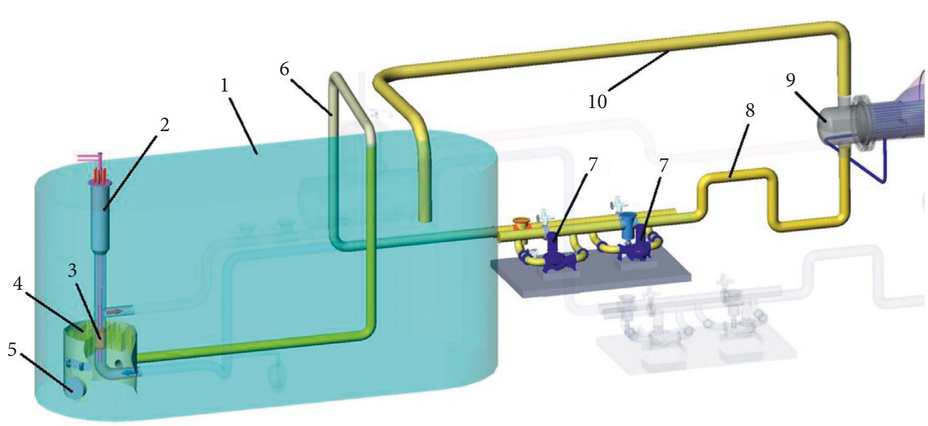

Figure 8: Principle of the pool cooling circuit (PCC), which removes heat from the reflector and irradiators. (1) reactor pool; (2) reactor vessel; (3) core; (4) neutron reflector; (5) natural circulation valve through the reflector; (6) outlet "hot" pipeline; (7) circulation pumps; (8) pressure pipeline of pumps; (9) steam generator; (10) chilled water return pipeline to the pool.

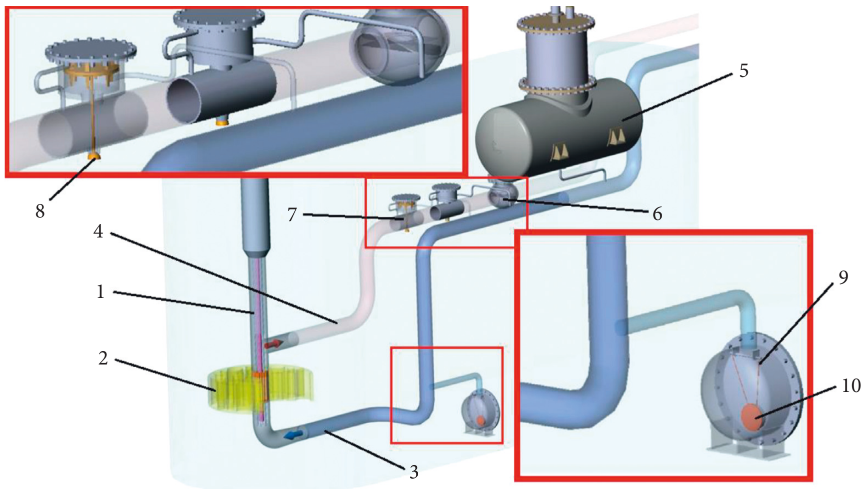

Figure 9: Appearance of equipment for the transition to natural circulation through the pool: (1) reactor vessel; (2) neutron reflector; (3) pressure pipeline of the primary circuit; (4) outlet "hot" pipeline; (5) pressure compensator, ECCS tank; (6) check valve; (7) discharge valve of natural circulation of the primary circuit into the pool; (8) outlet valve coolant outlet channel; (9) inlet valve of natural circulation of the primary circuit; (10) coolant inlet channel from the pool.

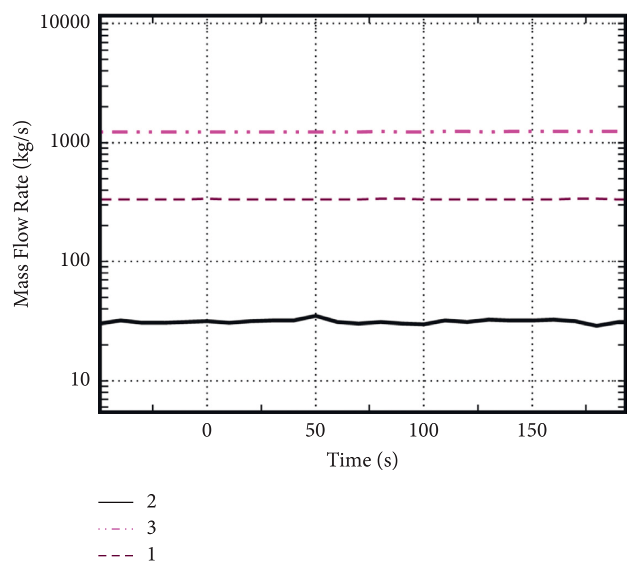

Figure 10: Fluctuations in the flow rate of the coolant in the circuits of heat removal from the core at a power of 50 MW: 1, coolant flow in the I circuit; 2, coolant flow rate in the secondary circuit; 3, coolant flow rate in the heat transfer circuit to the final heat sink.

exchange tubes, and the condensate returns to the steam generator under the action of gravity at a flow rate of 111.6 t/h. When steam condenses in an air heat exchanger, the atmospheric air passing through it is heated on average from 24°C to 64.5°C at an airflow rate through the ventilation pipe of 1237 kg/s.

Thus, the presented system of heat removal from the core at a maximum power of 50 MW provides sufficient heat removal intensity to maintain the temperature regimes of fuel assemblies with aluminum fuel elements in acceptable operating parameters.

4.2. Calculated Parameters of the Reactor Plant with a Complete Failure of Forced Circulation. One of the most important safety characteristics of a reactor facility is the ability to ensure safe heat removal in emergency conditions, for example, in the event of a blackout. In this case, special passive valves transfer core cooling from the forced circulation mode to the mode of natural coolant circulation through the reactor tank. The calculated parameters of the change in coolant flow through the cooling circuits from the moment of loss of power supply are shown in Figure 12. The dynamics of the change in the temperature of fuel rods in the calculated sections of the core during the transition from forced to natural circulation is shown in Figure 13.

Aluminum alloys are used in fuel elements of reactor cores at temperatures not exceeding 250–270°C [9]. The range of maximum operating temperatures of fuel elements in the considered reactor facility satisfies this requirement.

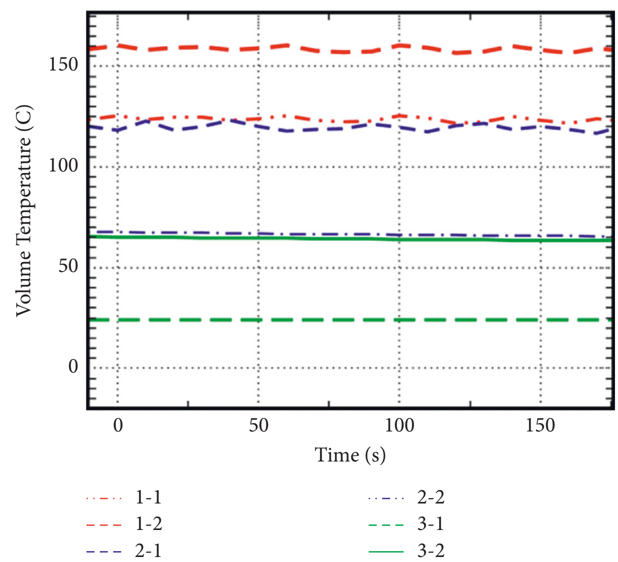

Figure 11: Fluctuations in the flow rate of the coolant in the circuits of heat removal from the core at a power of 50 MW: 1-1, coolant temperature at the core inlet; 1-2, coolant temperature at the core outlet; 2-1, coolant temperature at the steam generator outlet; 2-2, coolant temperature at the steam generator inlet; 3-1, ambient air temperature at the inlet to the air heat exchanger; 3-2, atmospheric air at the outlet of their air heat exchanger.

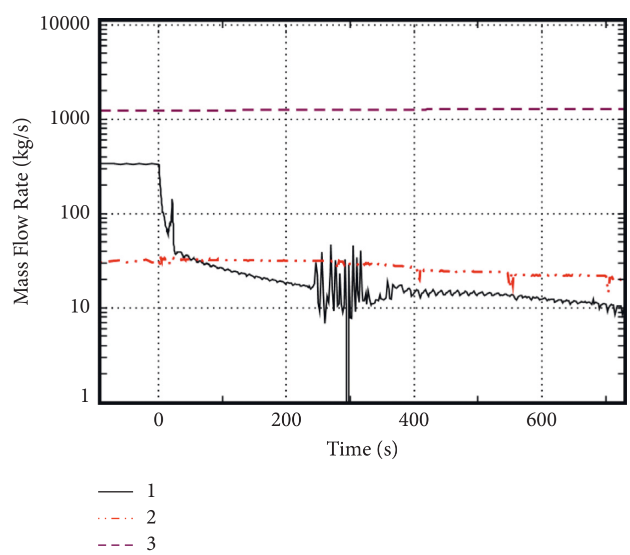

Figure 12: Change in coolant flow in the heat removal circuits from the core in case of a blackout: (1) coolant flow in the primary circuit; (2) coolant flow in the secondary circuit; (3) coolant flow in the heat transfer circuit to the final heat sink (chimney).

4.3. Loss of Coolant Analyses. The calculations of the reactor plant with a break from the LOCA pipeline are not considered in this paper. In the event of a rupture of the supply pipeline and a reverse movement of the coolant flow, it will be necessary to provide an additional system for supplying cold water to cool the core or to ensure the rapid opening of valves with a large flow area at the inlet and outlet nozzles of the reactor. Such a solution will ensure a quick transition to natural circulation through the reactor pool. The disadvantage of such a system is that it cannot be attributed to passive systems and, accordingly, the risks of operator's faults have to be considered.

5. The Discussion of the Results

(i) An idea of the conceptual three-dimensional model of a hybrid high-flux reactor plant with heat removal systems using natural circulation is described in the present article. The advantages of creating a simple and reliable passive system for cooling the core of a research reactor, built on the principle of natural coolant convection through the pool, are substantiated.

(ii) The presence of an intermediate circuit operating on the principle of a heat sink pipe eliminates the risk of radioactive coolant leaks in the environment. The passive operation of this circuit provides high

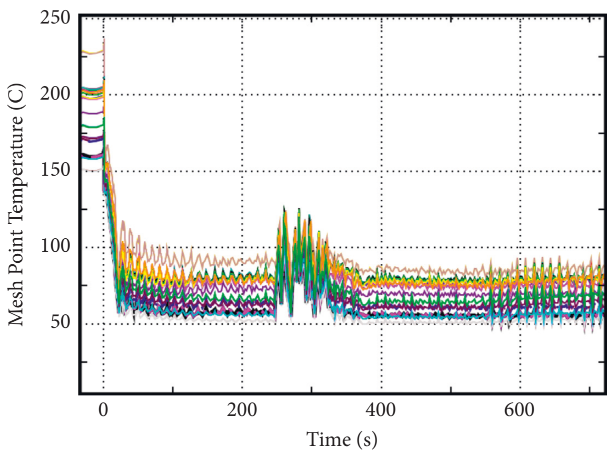

Figure 13: Dynamics of changes in the temperature of fuel elements in various calculated sections of the core during the transition from forced circulation to natural.

efficiency of heat transfer with a simple design and the absence of mechanical moving parts, which determines the reliability of the system.

(iii) 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 heat exchange tubes and the surface of sprinklers in cooling towers.

(iv) The location of the reactor vessel in the pool makes it possible to abandon expensive and complex emergency heat removal systems from the core.

(v) For the considered parameters of the reactor plant at a power of 50 MW, the maximum temperature subcooling on the fuel rods to the water saturation temperature in the core is 36°C.

(vi) Aluminum alloys are used in fuel elements of reactor cores at temperatures not exceeding 250–270°C [9]. The range of maximum operating temperatures of fuel elements in the considered reactor facility satisfies this requirement.

(vii) The design of the core cooling circuit ensures natural circulation of the coolant in the core in the cooldown mode, creating conditions for safe heat removal from the fuel assemblies after shutdown in the absence of forced circulation.

(viii) 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 irradiation devices without shutting down the reactor.

(ix) The experience of using a heavy water reflector in a zirconium tank located in a light water pool in the Australian OPAL reactor showed the high efficiency of such a constructive solution, so such a reflector can also be used to implement this concept.

(x) The proposed concept of a reactor plant with shortened intrabasin circulation valves provides a quick and easy transition to core cooling with pool water, which makes it possible to carry out transport and reloading operations without problems with the reactor vessel cover open.

(xi) Using three-dimensional modeling, the calculation of the main parameters of the equipment of the air cooling circuit with natural air convection, the air heat exchanger, and the chimney was made. The results of thermohydraulic calculation of heat transfer from the reactor core to the final recipient, atmospheric air, are presented.

(xii) The presented results of calculations based on the model in RELAP5 showed the efficiency of the proposed cooling system and the achievability of a high level of thermal power of the reactor (50 MW), which corresponds to the maximum neutron flux density in the central moderating cavity above $2 \times 10^{15}\ \text{cm}^{-2}\text{s}^{-1}$.

(xiii) The presented computational analysis shows that it is realistic to create a reliable functioning 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 plants for various purposes.

Data Availability

No data were used to support this study.

Conflicts of Interest

The authors declare that they have no conflicts of interest.

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