Nuclear Engineering and Technology 55 (2023) 2088–2095

Contents lists available at ScienceDirect

Nuclear Engineering and Technology

journal homepage: www.elsevier.com/locate/net

Original Article

Experimental and theoretical justification of passive heat removal system for irradiated fuel assemblies of the nuclear research reactor in a spent fuel pool

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ARTICLE INFO

Article history:
Received 21 November 2022
Received in revised form
13 January 2023
Accepted 19 February 2023
Available online 21 February 2023

Keywords:
Passive heat removal system
Nuclear research reactor
Irradiated fuel assemblies
Spent fuel pool
Residual heat power

ABSTRACT

The safety of nuclear installations is largely determined by the tightness of fuel elements cladding. As the Fukushima nuclear accident showed, the main task in case of loss of power supply is to ensure reliable removal of residual heat release from spent fuel pool (SFP) with irradiated fuel assemblies (IFAs). The paper presents the results of calculated-experimental studies and thermal-hydraulic modeling of temperature storage modes of IFAs in SFP. Experimental studies of SFP's temperature regime and calculated evaluation of residual heat removal due to the thermal conductivity of building structures surrounding the SFP were performed. To ensure the safe operation of research reactors, it's necessary to know the IFA's residual heat power (RHP) in the reactor and SFP, which is determined depending on the operating time of fuel assemblies (FAs) and the IFAs calculated holding time. The FAs operating time depends on the reactor energy output. The IFAs calculated holding time is determined by the fuel burnup, U-235 mass in the fuel, and reactor utilization factor. The IFAs fuel burnup was calculated using the MCU-PTR program. Also presented are the RHP's calculation results using some of the empirical dependencies. The concept of a passive heat removal system (PHRS) based on thermosyphon's operating principle was proposed.

1. Introduction

The fields of use of atomic energy, including production of electrical and thermal energy at nuclear power plants (NPPs) [1], solving scientific and technical and research problems, production of isotopes for industry and medicine in nuclear research reactors (NRR's) [2,3], etc., are constantly expanding. At the same time, the most important aspect of nuclear installations use is to ensure their nuclear and radiation safety at all stages of their life cycle, including decommissioning [4,5].

After the accident that occurred on March 11, 2011, at the Fukushima Daiichi NPP in Japan, it became clear that, to ensure the safety of nuclear power facilities, it is necessary to take into account all possible risk factors.

According to the normative requirements of NP-033-11 "General safety provisions of nuclear research facilities", the purpose of ensuring the safety of nuclear research facilities (NRFs) is the limitation of radioactive impact on the personnel, people, and environment during normal operation and malfunction, including emergencies.

The safety of a nuclear facility, including a NRF, is ensured by establishing a series of successive physical barriers, such as fuel matrix, fuel element cladding, sealed coolant circulation loop, etc., on the path of the radioactive fission products spread into the environment.

The main safety condition of nuclear installations is the retention of the tightness of fuel element cladding, both during reactor operation and during storage of IFAs in SFPs. To meet this requirement, it is necessary to provide reliable cooling of IFAs both during normal operation and in emergencies, including complete loss of power supply [6].

One of the ways to increase the safety of a nuclear installation is the use of PHRSs for the removal of residual heat from IFAs.

The present work aims to estimate the efficiency of the passive system of residual heat removal from the spent fuel pool of the IVV-2M nuclear research reactor.

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https://doi.org/10.1016/j.net.2023.02.028

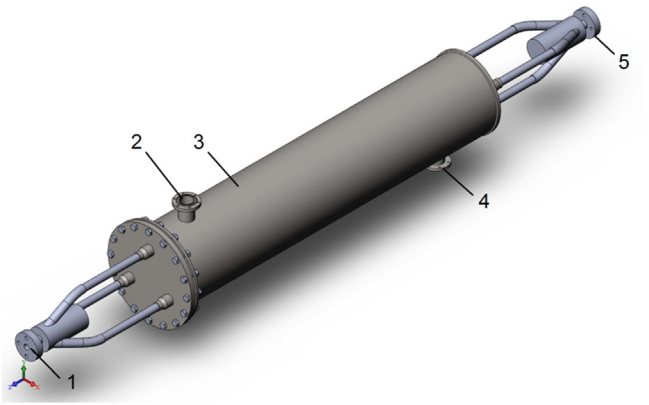


Fig. 1. 3D model of the heat exchanger: 1, 5 – inlet and outlet of the primary heat-carrying agent; 2, 4 – inlet and outlet of the secondary heat-carrying agent; 3 – heat exchanger shell.

To achieve this goal, the following tasks were assigned:

- Simulation of thermal-hydraulic processes in SFP using a computational fluid dynamics application SOLIDWORKS Flow Simulation embedded within SOLIDWORKS 3D CAD;
- Calculated evaluation of residual heat removal from IFAs through the building structures surrounding the SFP;
- Study of the RHP's dependence on the fuel burnup fraction and holding period of IFAs using the MCU-PTR software tool;
- Experimental studies of water temperature changes at characteristic points of the SFP's volume using a special device under various operating modes of the standard cooling system, including its failure due to a power outage;
- Study of the condensing part efficiency of the passive cooling system;
- Development of a principal scheme of a passive system of heat removal from the NRR's SFP.

2. Research object and methods

2.1. Construction of a three-dimensional solid model

The SOLIDWORKS Flow Simulation embedded within SOLIDWORKS 3D CAD was used to carry out computational studies. It is a computational fluid dynamics (CFD) software suit that enables you simulate fluid flows and heat transfer and is widely used in solving problems in the field of nuclear energy application [7–10].

The first stage of the study was the creation of a three-dimensional solid model of the heat exchanger (Fig. 1) of the standard system of the SFP's main equipment cooling and application of SOLIDWORKS Flow Simulation for performing the thermal-hydraulic calculation [11], as well as simulating various modes of temperature and velocity distribution in the heat exchanger [12].

2.2. Estimation of dissipation of residual heat through the building structures

In the next step, a calculated evaluation of residual heat removal from the SFP's IFAs into the building structures was carried out [13]. The SFP is surrounded by a reinforced concrete structure, through which part of the residual heat is dissipated, as illustrated in Fig. 2, where t_1 - t_6 are isothermal surfaces; t_o is outside air temperature; t_{CS} is coolant surface temperature; t_{CT} is coolant temperature; t_W, t_{W1}, t_{W2} are temperatures of the walls of the SFP's tank; t_C is concrete temperature; λ_C is thermal conductivity coefficient of concrete; R_{WA} is thermal resistance of concrete in the SFP's wall area; R_{BA} is thermal resistance of concrete in the SFP's bottom area.

2.3. Experimental studies of temperature changes over the SFP's volume

A special rod with thermocouples (Fig. 3) located at different altitudes was developed and designed for experimental studies of the SFP's temperature regime, taking into account the SFP's design features of the IVV-2M reactor [14]. The safety of these studies was

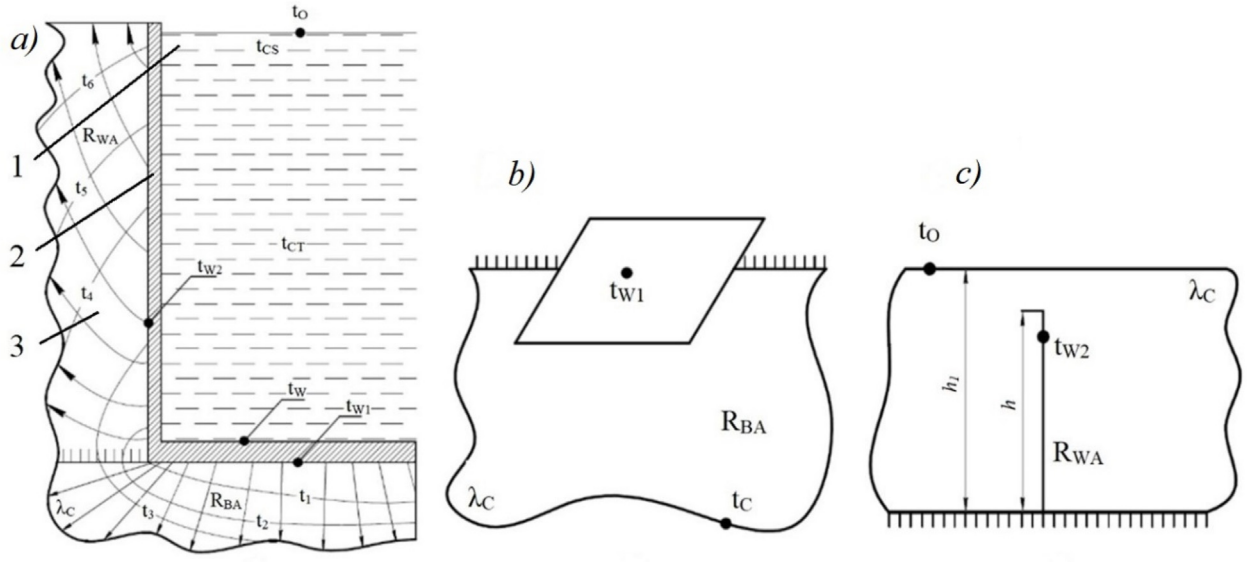


Fig. 2. Calculated scheme of the SFP's tank: *a* – scheme of the SFP's tank with a layer of concrete poured on the outside in all directions; *b* – scheme of the bottom of the SFP's tank, buried in concrete, which is a semi-infinite array, bounded from above by an adiabatic surface; *c* – scheme of the walls of the SFP's tank with height *h*, buried in concrete in the form of an infinite plate with height h_1 and bounded by an adiabatic surface from below; 1 – water; 2 – wall (facing of the spent fuel pool); 3 – concrete.

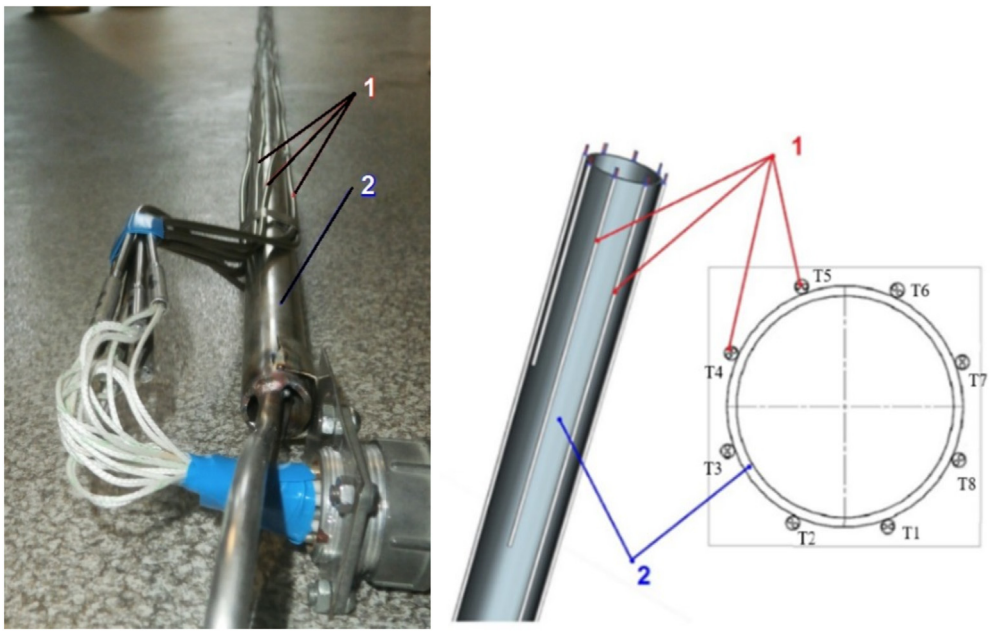


Fig. 3. Rod with thermocouples: 1 – thermocouples, 2 – thin-walled tube; T1 - T8 – designations of thermocouples.

ensured by the development of a special measurement program and the established limit for the increase in water temperature in the spent fuel pool.

2.4. Theoretical study of the burnup depth and residual heat release of IFAs

For the safe operation of a NRR, it is necessary to know the RHP of IFAs. The residual heat or decay heat is a specific feature of nuclear fuel, which is caused by the fact that, after the termination of the fission chain reaction and the usual thermal inertia, the heat production in the nuclear reactor continues for a long time, which creates several technically complex problems directly related to nuclear safety, for example, the need for reliable cooling to maintain the tightness of fuel cladding.

To estimate the quantity of residual heat, two methods were chosen:

I. The Wigner-Way formula [15]:

$$N_{residual} = 6.62 \bullet 10^{-2} \bullet N_0 \bullet \left[\tau^{-0.2} - (\tau + T)^{-0.2}\right], \quad (1)$$

Where N_0 is the reactor thermal power before the shutdown, T is time, in seconds, of the reactor operation on this power level before the shutdown, τ is the calculated holding time of IFAs after irradiation in the reactor, s.

This formula (1) is recommended to be used for the time elapsed since the reactor shutdown, exceeding approximately 10^4 s. The formula gives the approximation of the time dependency of the reactor residual power due to the radioactive decay of fission products.

II. The Untermyer-Weills formula [16]:

$$\frac{N_{residual}}{N_0} = 10 \bullet \left\{(\tau + 10)^{-0.2} - (\tau + T + 10)^{-0.2} - 0,87 \bullet \left[\left(\tau + 2 \bullet 10^7\right)^{-0.2} - \left(\tau + T + 2 \bullet 10^7\right)^{-0.2}\right]\right\} \quad (2)$$

The equation (2) is an empirical formula obtained by Untermyer and Weills. It takes into account the contribution of residual power from the decay U-235 and Pu-239 of fission products.

The operating time of FAs in the reactor core depends on the energy output. The calculated holding time of IFAs is determined by the fuel burnup fraction, the mass of uranium U-235 in the fuel, and the reactor utilization factor. The reactor utilization factor is the ratio of the time the reactor is operating at power to the time it is shutdown.

The fuel composition of a power or special nuclear reactor undergoes continuous changes due to radioactive decays of its components and transformations caused by the interactions of neutrons with these components: radiative neutron capture reactions, neutron multiplication processes (n, 2n) and (n, 3n), and reactions with the emission of charged particles. During the reactor operation, there are produced in uranium fuel new actively fissile heavy isotopes such as Pu-241 and Cm-243 long-lived nuclides, and the atoms which in practice may be considered stable, such as U-233, Np-236, Pu-239, Cm-245. These nuclei contribute to the energy release, and it is necessary to take into account their accumulation to correctly predict the reactor's characteristics at different stages of its operation [17].

The calculation of fuel nuclide composition of the IVV-2M fuel assemblies to be unloaded from the core of the NRR is performed using the MCU-PTR precision code.

The MCU-PTR code [18] was designed for the simulation of neutron, photon, electron, and positron transport processes in systems with three-dimensional geometry, taking into account changes in the nuclide composition of materials during interaction with neutrons. It uses the Monte Carlo numerical method and the evaluated nuclear data.

To set the initial value of the average burnup for FAs of the NRR IVV-2M, the “Methodology for Determining the Burnup of Uranium-235 and Determining the Mass of Nuclear Materials in Spent Fuel Assemblies of the NRR IVV-2M”, MK-09.28/44, was used. According to the methodology, the change in concentrations of nuclei included in this chain of transformations is described by the system of differential equation (3):

$$\begin{cases} \frac{dN^{U5}}{dt} = -R_f^{U5}\left(N^{U5}\right)\cdot N^{U5} - R_{n,\gamma}^{U5}\left(N^{U5}\right)\cdot N^{U5} \\ \frac{dN^{U8}}{dt} = -R_f^{U8}\left(N^{U5}\right)\cdot N^{U8} - R_{n,\gamma}^{U8}\left(N^{U5}\right)\cdot N^{U8} \\ \frac{dN^{Np9}}{dt} = -R_f^{Np9}\left(N^{U5}\right)\cdot N^{Np9} - \lambda_{Np9}\cdot N^{Np9} + R_{n,\gamma}^{U8}\left(N^{U5}\right)\cdot N^{U8} \\ \frac{dN^{Pu9}}{dt} = -R_f^{Pu9}\left(N^{U5}\right)\cdot N^{Pu9} + \lambda_{Np9}\cdot N^{Np9} - R_{n,\gamma}^{Pu9}\left(N^{U5}\right)\cdot N^{Pu9} \\ \frac{dN^{Pu0}}{dt} = R_{n,\gamma}^{Pu9}\left(N^{U5}\right)\cdot N^{Pu9} - R_f^{Pu0}\left(N^{U5}\right)\cdot\phi\cdot N^{Pu0} - R_{n,\gamma}^{Pu0}\left(N^{U5}\right)\cdot N^{Pu0} \\ \frac{dN^{Pu1}}{dt} = R_{n,\gamma}^{Pu0}\left(N^{U5}\right)\cdot N^{Pu0} - R_f^{Pu1}\left(N^{U5}\right)\cdot N^{Pu1} - R_{n,\gamma}^{Pu1}\left(N^{U5}\right)\cdot N^{Pu1} - \lambda_{Pu1}\cdot N^{Pu1} \end{cases} \quad (3)$$

Where $R_f^{U5}(N^{U5})$, $R_f^{U8}(N^{U5})$, $R_f^{Np9}(N^{U5})$, $R_f^{Pu9}N^{U5})$, $R_f^{Pu0}(N^{U5})$, and $R_f^{Pu1}(N^{U5})$, s^{-1}, are the rates of fission reactions for ^{235}U, ^{238}U, ^{239}Np, ^{239}Pu, ^{240}Pu, and ^{241}Pu, presented as functional dependences on ^{235}U concentration. $R_{n,\gamma}^{U5}(N^{U5})$, $R_{n,\gamma}^{U8}(N^{U5})$, $R_{n,\gamma}^{Pu9}(N^{U5})$, $R_{n,\gamma}^{Pu0}(N^{U5})$, and $R_{n,\gamma}^{Pu1}(N^{U5})$, s^{-1}, are the rates of radiative neutron capture reactions by ^{235}U, ^{238}U, ^{239}Pu, ^{240}Pu, and ^{241}Pu nuclei. The decay constants of ^{239}Np and ^{241}Pu are represented in the previous equation by λ_{Np9} and λ_{Pu1}, s^{-1}. The concentrations of ^{235}U, ^{238}U, ^{239}Np, ^{239}Pu, ^{240}Pu, and ^{241}Pu, cm^{-3}, were displayed by N^{U5}, N^{U8}, N^{Np9}, N^{Pu9}, N^{Pu0}, and N^{Pu1}, respectively, being functions of time t.

Fig. 4 shows the results of changes in the burnup fraction of the IVV-2M's FAs during one period between the refuelings with energy output step of 7500 MWh. For each fuel assembly, the calculation of changes in the fuel nuclide composition is carried out for:

- initial loading into the core of the NRR IVV-2M ("fresh" FAs, 0% burnup, nuclide composition is known from the passport data);

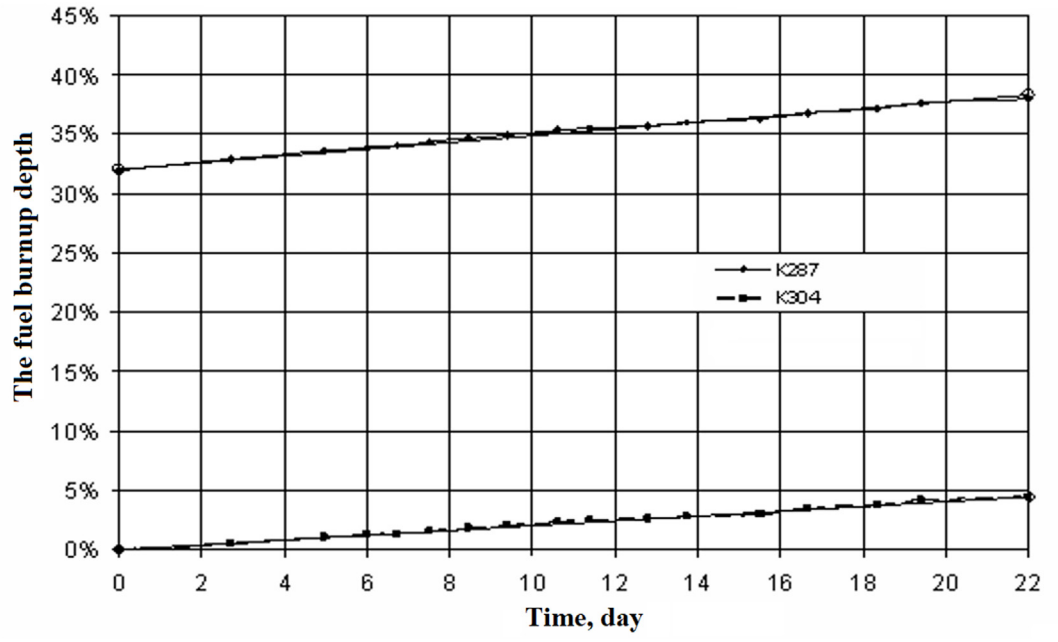


Fig. 4. Change in the fuel burnup depth in FAs of IVV-2M type (K287, K304 – designations of fuel assemblies).

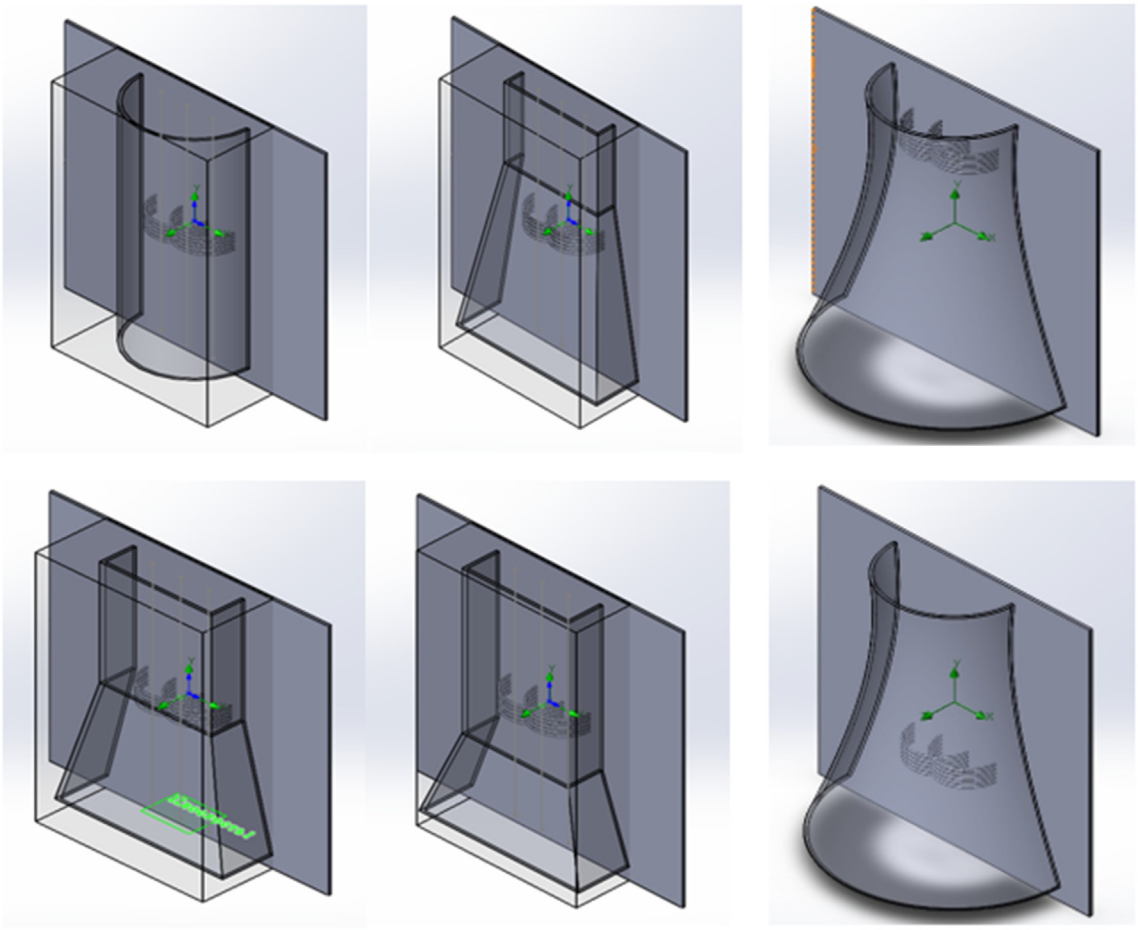


Fig. 5. Configuration of the exhaust tube.

- moving FAs across the core during each refueling (intermediate calculations of the value of fission products accumulation);
- final unloading of FAs from the NRR's core and its transfer to the SFP (IFAs).

The total power of the residual energy output after the reactor shutdown during the final unloading of FAs from the core is the sum of the energies of α-, β-, γ-, or n-radiation emitted during the radioactive decay of the nuclides, taking into account the decay constant of each nuclide formed in the burnup process. Fission products experience mainly β-radioactive decay, which is usually accompanied by the emission of γ-radiation [19].

Thus, the calculation of the isotopic composition of IFAs using the MCU-PTR serves as the initial data for calculating the residual energy output with other codes, such as ORIGEN [20], MURE, CARE [21], etc., or using other methods and tools.

The theoretical calculation of the RHP was performed for the operating time of FAs in the reactor core for 21 days and the holding time of IFAs in SFP in the interval from 0 to 10^7 s (116 days).

2.5. Study of the efficiency of the condensing part of the passive heat removal system

The efficiency of the condensing part of the proposed PHRS was also investigated. The condensing part is an air heat exchanger, cooled by ambient air. The work on the comparison of heat removal from the condensing part using an exhaust tube of different configurations (Fig. 5) and without its use, at different ambient temperatures, was carried out. The calculations considered the ambient temperature range from −31 °C to +35 °C according to meteorological observations at the site of the IVV-2M reactor. The study was performed using the SOLIDWORKS Flow Simulation software suit [22].

3. Results and discussion

After conducting the thermal-hydraulic simulation of the SFP with several IFAs in SOLIDWORKS Flow Simulation, the results in the form of flow trajectories under natural convection conditions were obtained (Fig. 6).

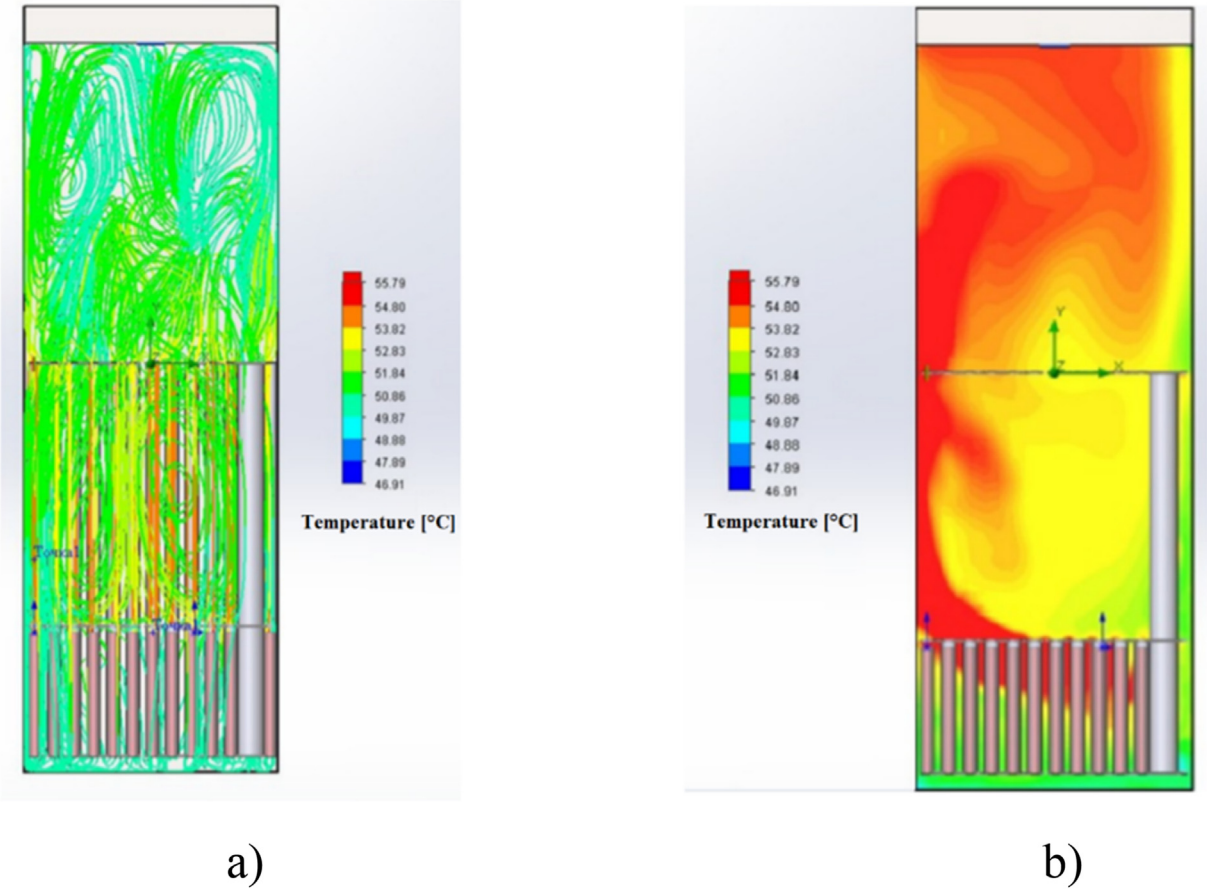


Fig. 6. Flow trajectories under natural convection conditions (a), and temperature distribution in the middle section of SFP for the storage of IFAs (b).

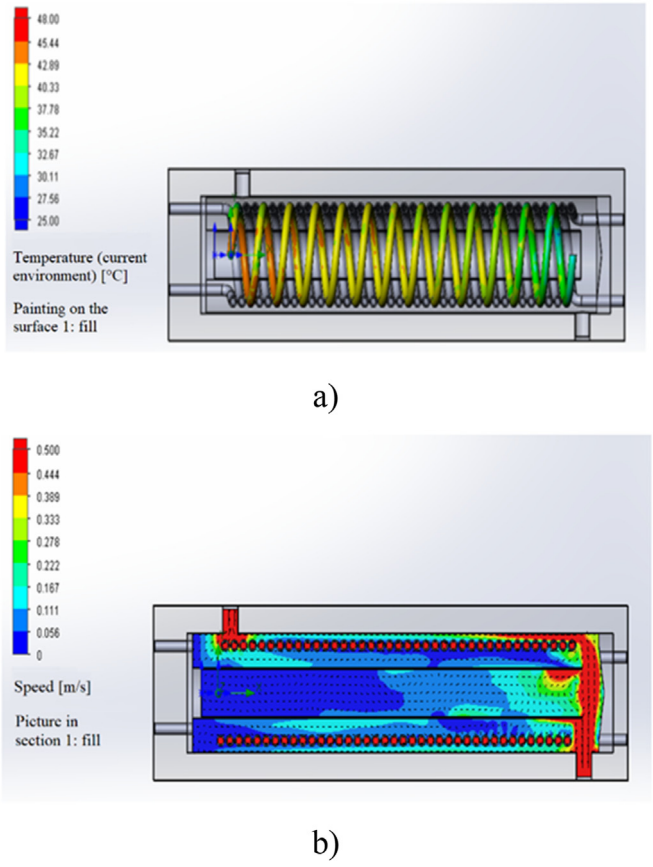


Fig. 7. Flow temperature distribution over the coil tube (a), and velocity distribution over the cooling water of the second circuit (b).

Next, after building a 3D model of the heat exchanger, various modes of temperature and velocity distribution in the heat exchanger were simulated. The results are presented in Fig. 7.

Calculated evaluation of residual heat removal from IFAs in SFP into the building structures showed that at the temperature difference between the coolant and concrete from $\Delta t_1 = 12$ °C to $\Delta t_2 = 87$ °C, the heat losses were as follows:

- In the bottom of SFP: $Q_{B1} = 57.7$ W to $Q_{B2} = 418.31$ W
- In the walls of SFP from $Q_{W1} = 632.36$ W to $Q_{W2} = 10317.98$ W
- Total heat losses of walls and bottom from $Q_{T1} = 690$ W to $Q_{T2} = 10736.29$ W.

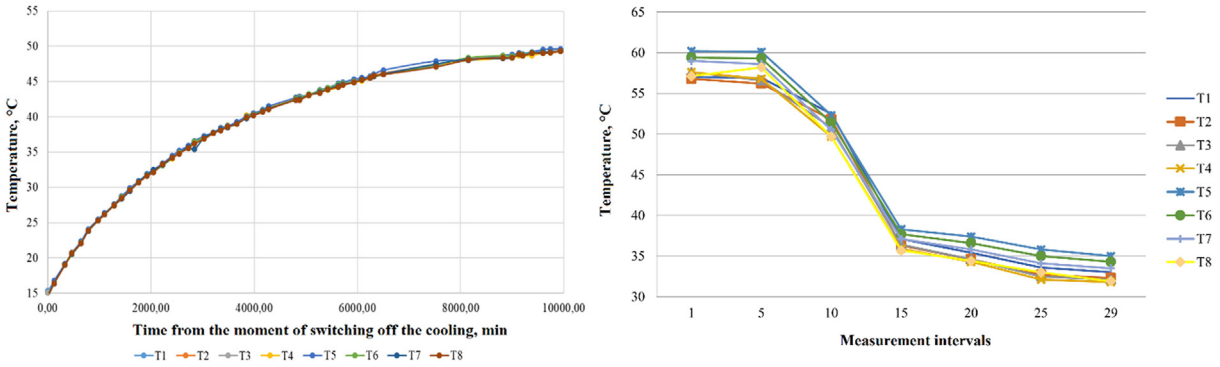
Measurements of water temperature changes in characteristic points of the SFP's volume were carried out when disconnecting all cooling pumps and then reconnecting them, the results obtained are presented in Fig. 8.

The theoretical calculation of the RHP was performed with formulas (1) and (2). Fig. 9 shows the obtained calculation results depending on the holding time of IFAs in SFP.

When comparing the obtained data, it was revealed that the divergence of the results varies from 0 to 80% depending on the holding time of IFAs in SFP. Presumably, this is due to the error of the selected calculation methods, as well as the possible use of formula (2) to estimate the residual heat release of uranium-plutonium fuel.

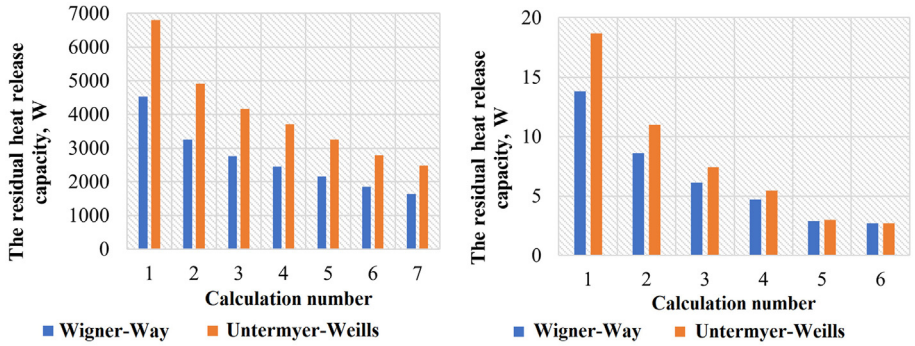
Based on the conducted complex scientific-research works, the principle scheme of PHRS (Fig. 10, a) for NRRs is proposed.

To check the developed system, an experimental installation is proposed (Fig. 10, b), which consists of two coaxial tubes, in the annular gap of which there is a low-boiling liquid. The IFA's model



a) When the SFP's coolant pumps are turned off. b) When the SFP's coolant pumps are turned on.

Fig. 8. Graph of temperature changes in characteristic points of the SFP's volume, time intervals for taking results: **1** – 0 h; **5** – 13 h; **10** – 28 h 20 min; **15** – 43 h 30 min; **20** – 59 h 35 min; **25** – 85 h; **29** – 108 h; T1 - T8 – thermocouples, located by height on the measuring rod.



(a) IFAs holding time interval, s: $6 \cdot 10^3$ to $12 \cdot 10^3$ (b) IFAs holding time interval, s: 10^6 to $1{,}55 \cdot 10^7$

Fig. 9. Calculated values of the residual heat power at different holding time intervals of IFAs in SFP.

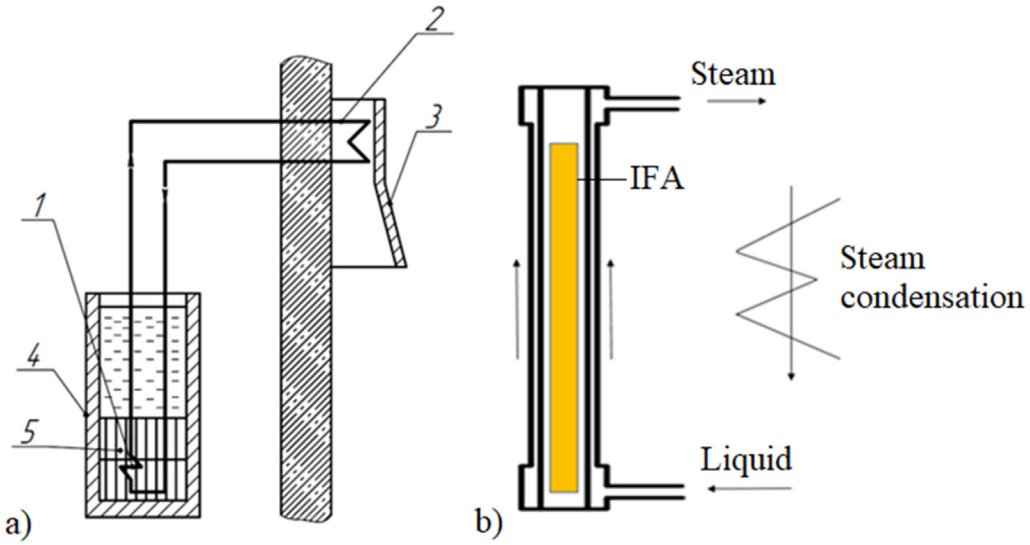


Fig. 10. The principle scheme of PHRS (a), where 1 – evaporator, 2 – condenser, 3 – exhaust tube, 4 – SFP, 5 – IFA; and scheme of the experimental installation represented by subfigure (b).

(the heating element) is located in the inner tube. The space of the inner tube is filled with water, which has a boiling point higher than that of the coolant. Heat transfer from the IFA's model to the low-boiling liquid is carried out through the inner tube wall. In the annular gap, the low-boiling liquid reaches the saturation temperature, evaporates, and in the form of steam, rises to the condenser, where, after condensation in the condensing part, it returns to the evaporative part through the lowering path.

A study was conducted to evaluate the efficiency of the condensing part of the proposed system. Calculation results of the dissipated capacity at different ambient temperatures, as well as with the use of an exhaust tube and without it are shown in Fig. 11.

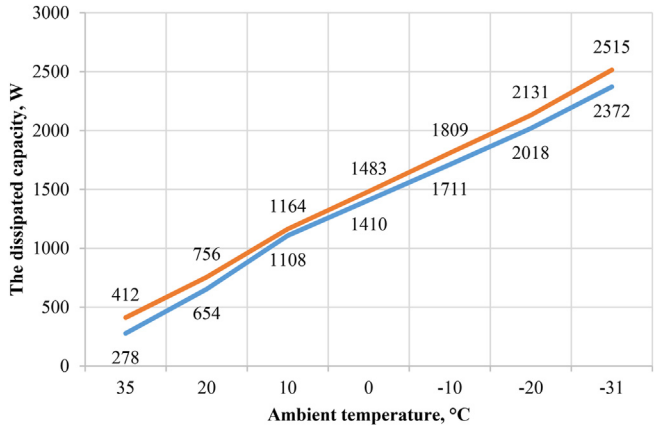


Fig. 11. Calculation results of the dissipated capacity.

4. Conclusion

In the course of scientific-research works on the development and evaluation of the efficiency of PHRS, the SFP's simulation was carried out, the heat losses in the reinforced concrete structures surrounding the SFP were calculated, an assessment of the temperature regime during normal operation of the heat exchanger and when it is turned off was performed, the residual heat power of IFAs in SFP at different loads was calculated, the principle scheme of PHRS was proposed, the calculation of the efficiency of the condensing part was carried out. The results obtained show that the developed PHRS can ensure the safety during storage of IFAs in SFP in any emergencies, including loss of power supply for own needs.

Declaration of competing interest

The authors declare that they have no known competing financial interests or personal relationships that could have appeared to influence the work reported in this paper.

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