Thermal-hydraulic model of the reactor facility with lead coolant in the ATHLET code

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Abstract. This article presents the results of the creation of a thermal-hydraulic model of a reactor facility with lead coolant based on the BREST-OD-300. Neutron physics was not considered. The work carried out calculations and modeling of the stationary process and transient modes, such as: shutdown of one of the MCP at different reactor power, disconnection of one of the feed pumps. The ATHLET code developed in GRS (Society for the Safety of Installations and Reactors, Garching, Germany) is used as a calculation program.

1. Introduction
Every year, electricity consumption in the world is growing, which is associated with the expansion of the scale of production and consumption of electricity for household and industrial needs. Only nuclear power is capable of generating electricity at the required scale. In this case, the use of fast neutron reactors with a liquid metal coolant and a closed nuclear fuel cycle looks more promising [1].

One of such reactors was the design of the BREST-OD-300 reactor facility [2], shown in figure 1. This reactor differs from the existing ones BY using dense mixed uranium-plutonium nitride fuel and high-boiling lead coolant. Also, this type of reactor is distinguished by a new approach to ensuring the safety requirements of future nuclear energy.

A lead coolant reactor has a minimal set of reactor protection control systems, realizing the concept of natural safety. Thus, there is a need to create neutron-physical and thermal-hydraulic models to justify safety when various transient processes occur in the reactor. For example, a rupture of the heat exchange tube in the steam generator, the shutdown of one of the main circulation pumps, or the shutdown of one of the feed pumps of the second circuit [3-5].
2. Calculation tool
To create a thermohydraulic model of the reactor set up in this work, the code ATHLET is used. Thermal-hydraulic system code ATHLET (Analysis of Thermal-Hydraulics of Leaks and Transients), developed in Gesellschaft für Anlagen- und Reaktorsicherheit (GRS). The code is based on the finite volume method for the purpose of analyzing the entire spectrum of operational transients, design and beyond design basis accidents without destroying the core, envisaged in nuclear or non-nuclear power plants [6, 7]. This code was chosen for a number of reasons: ATHLET was officially received by MEPhI from GRS; this code is certified in Russia for calculations to substantiate the safety of VVER reactors, but recently it is also used for liquid metal reactors, in particular, in SEC NRS. Also, in the articles [8-10], studies were conducted on verification of this code for reactors with liquid metal coolant.

3. Creating and researching a model in the ATHLET code

3.1. Testing stationary process
The model includes a reactor, four circulation loops with main circulation pumps (MCP), two sectional steam generator (PG) on each loop, the second circuit consisting of heat exchange tubes and feed pump. Neutron physics is not considered, energy release is set as a heat source with a uniform distribution throughout the core. Loops of the second circuit for each steam generator have a feed pump, which set the heat flow in the loops of the second loop, allowing to simulate the temperature change of the feed water at the inlet to the steam generator. A fragment of the design scheme of a reactor facility with one circulation loop and two SG sections is shown in figure 2.

Steady state is carried out in several stages. Initially, the same temperature is setting up in all objects on the contour, in that case the pump does not work, i.e. coolant and working fluid flow rate along the contour 0 kg/s and energy release in the reactor and SG are absent. Then the pumps is turned on, the power in the reactor area was raised, the temperature controller in the second circuit of the steam generator started to work, to maintain the set temperature at the entrance to the steam generator along the lead circulation loop. Local drag coefficients at the boundaries of adjacent areas are calculated on the basis of their geometry.

As a result of the stationary state calculation of the model, graphs of changes in the parameters of the coolant and working fluid for the main elements of the calculated scheme were obtained, which are presented in figure 3.
Figure 2. Detail of the calculation scheme

Figure 3. Graphs of changes in the parameters of the coolant and the working fluid when entering a stationary mode.
3.2. Transient Modeling
In this work, we simulated four transients:

1) Turning off of the MCP in one of the four loops with preserving the reactor power 100%;
2) Turning off of the MCP in one of the four loops with a decrease in reactor power up to 67%;
3) Turning off of the MCP in one of the four loops with a decrease in reactor power to 0%;
4) Turning off of the feed pump in one of the SG without changing the reactor power.

Figure 4 shows the change in coolant flow rate when the MCP is shutting down in one of four circulation loops. As can be seen from the figure, a reverse coolant current occurs in the emergency loop. This leads to the fact that the cold coolant from the fourth loop is not fed to the input of the reactor core, but it mixed in the transfer manifold after the reactor core. As a result, the temperature of the coolant after the distributing manifold becomes lower than at the exit from the reactor core. But as the total coolant flow through the reactor core is decreased, the final coolant temperature higher than in stationary state. Figure 5 shows the change in the temperature of the coolant when one of the MCPs is turned off while the reactor power is 100%.

![Graph showing coolant flow rate](image1)

**Figure 4.** Coolant flow rate in downstream sections when the MCP is disconnected and the reactor power is 100%

![Graph showing coolant temperature](image2)

**Figure 5.** Coolant temperature at the inlet, outlet from the reactor core and in the distributing manifold when the MCP is disconnected and the reactor power is 100%
When simulating a similar transition state with a reactor power of 67%, a similar mixing of cold coolant in the transfer manifold and a change in its temperature occurs. But due to the fact that the reactor power is reduced to 67%, the final coolant temperature is reduced. Figure 6 shows a graph of the temperature change of the coolant when the MCP is disconnected while the reactor power is 67%.

![Figure 6](image)

**Figure 6.** The temperature of the coolant at the inlet, outlet from the reactor core and in the distributing manifold when the MCP is disconnected and the reactor power is 67%

At the first moment of time, when the MCP is disconnected and the reactor power decreases to 0%, a similar process occurs, but with a further decrease in the coolant temperature to the original one. This process is depicted in figure 7.

![Figure 7](image)

**Figure 7.** Coolant temperature at the inlet, outlet from the reactor core and in the distributing manifold when the MCP and the reactor power are off 0%

When simulating the shutdown of the feed pump in one of the two SGs of the fourth loop, the reactor power does not change and all four MCPs work. In this transitional mode, the coolant in an uncooled SG heats up quickly, that leads to increasing of the coolant temperature throughout the all installation. figure 8 presents a graph of this process. Also, there is an increase in the temperature of the working fluid in the inactive SG, which leads to an increase in the steam content at the outlet from the SG. The
temperature of the working fluid in the remaining operating SGs also increases, but this does not lead to an increase in steam in the heat exchange tubes. These temperature changes are shown in figures 9 and 10.

Figure 8. Coolant temperature at the output of SGs

Figure 9. The temperature of the working fluid in the inactive SG

Figure 10. The temperature of the working fluid in the active SG
4. Conclusion
Since the lead coolant reactor under consideration has a minimal set of reactor protection control systems, realizing the concept of natural safety, there is a need to create neutron-physical and thermal-hydraulic models to justify the safety of a reactor facility when various transient processes occur in the reactor.

In the present work, a multi-loop scheme design was created for the analysis of transient processes of a reactor with a lead coolant based on reactor BREST-OD-300.

The method for establishing of the stationary state of the installation has been developed and tested. Calculations of transient processes were carried out, such as:
1) Turning off of the MCP in one of the four loops with preserving the reactor power 100%;
2) Turning off of the MCP in one of the four loops with a decrease in reactor power up to 67%;
3) Turning off of the MCP in one of the four loops with a decrease in reactor power to 0%;
4) Turning off of the the feed pump in one of the SG without changing the reactor power.

In transients 1, 2, 3 (with stopping MCP), the presence of reverse coolant current in the emergency loop is shown, which leads to a decrease in coolant temperature in the distribution manifold in the upper part of the reactor after reactor core, but due to a decrease in coolant flow through reactor core, the coolant temperature increases at the entrance to the SG in working loops.

In the 4th process (shutdown of the feed pump), it was shown that within 10,000 s from the moment of the beginning of the transition process the reactor installation goes to a new stationary state.

In all the calculations carried out, the scheme showed stable operation and can be used to analyze various emergency and transient modes of an installation of this type.

The following works involve the modernization of the scheme, related to:
1) dividing the descending sections and the in-reactor space into a system of parallel hydraulically connected regions with transverse heat exchange sections of the coolant flow between them, considering the capabilities of the ATHLET code;
2) Development of a model of an emergency cooling system;
3) Modernization of the reactor core and the calculation of neutron-physical processes.

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