Modeling the core of a lead-cooled reactor

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Abstract. This paper presents the results of thermohydraulic modeling of the reactor core with a lead coolant, which is based on the calculation scheme for the ATHLET code, obtained on the basis of open information on the BREST-OD-300 reactor plant. Only the reactor core and methods of its modeling are considered. The work also presents the results on the development of the division of the in-reactor space of the BREST-OD-300 reactor into a system of parallel channels. The subdivision of the in-core space is based on the type and number of different core elements. This way of modeling the core allows you to see changes in different parts of the reactor when calculating transients. The developed model of the partitioning of the in-core space will be further used in the calculation of various transient modes (MCP shutdown, steam generator tube rupture, etc.). Neutron physics is not considered in this paper. There are plans to carry out joint neutron-physical and thermohydraulic calculations using the model from this paper.

1. Introduction
The paper presents the results of calculations for the thermal-hydraulic model of a reactor with a lead coolant, which is based on the calculation scheme for the ATHLET code [1], obtained in [2] based on open information on the BREST-OD-300 reactor plant [3].

In [2], the in-reactor space was actually represented by one channel of variable cross-section in height, the lower part of which connected four independent lowering chambers of each loop, and from the common distributing manifold of the reactor, the flow was directed to each section of the steam generator loop. In [4-7], a model is used in which the downstream sections of each loop are interconnected by transverse hydraulic connections, and the in-reactor space from the lower pressure header of the reactor to the upper distribution header of the reactor is also divided into a system of parallel channels connected by cross connections. In this work, the in-reactor space is divided in accordance with the groups of elements of the reactor core (RC). The simulation results are presented only for the reactor core, since the rest of the reactor plant elements remain unchanged.

2. Modeling tool
To carry out the calculations, the ATHLET improved estimate code was used, which is included in the AC2 software package, officially obtained by the National Research Nuclear University MEPhI on the basis of a license agreement with GesellschaftfurAnlagen-undReaktorsicherheit (GRS) gGmbH, Germany. The ATHLET code is certified in Russia for carrying out stationary and transient modes in reactors with a water coolant; however, the code's capabilities allow it to be used with other types of coolants, including molten lead. This code was chosen for a number of reasons: ATHLET was officially
received by MEPhI from GRS. This code has been certified in Russia for calculations to justify the safety of VVER reactors, but has recently been used for liquid metal reactors, in particular, in the STC NRS. Also in articles [10-12], studies were carried out to verify this code for reactors with a liquid metal coolant.

3. Modeling process

The simulation of the reactor facility was carried out on the basis of open information on the BREST-OD-300 reactor plant [3]. Figure 1 shows a diagram of the BREST-OD-300 reactor plant; figure 2 shows a preliminary developed diagram of the BREST-OD-300.

A detail of the design scheme created in the ATHLET Input Graphic module is shown in figure 3 a) for one loop and the second loop for one SG, and in figure 3 b) for a reactor with two steam generator modules.

This design scheme was created without dividing the in-reactor space into parallel channels. Each separate area of the reactor is a cylinder of the same hydraulic diameter and area. This scheme was created to test the operability of the model and conduct primary calculations with modeling of transient processes in the reactor installation.

Further, a simple core modeling scheme became more complicated. The in-core objects were divided into a system of parallel vertical channels. An example of such detailing is shown in figure 4.

In many modern works, the influence of the reactor core cell splitting on the results obtained is investigated [13-16]. This is an important task that increases the accuracy of the calculations and increases the importance of the results.

In early works [4-7], the difference was investigated as a result of calculating transient processes without splitting and with splitting the in-reactor space into a different number of sections. It was calculated that the difference in the temperature of the coolant with a different methods of simulation can reach up to 10%.

Figure 5 shows the different schemes and stages of reactor splitting. Figure 6 shows the graphs of the temperature change of the coolant in the upper part of the distributing manifold when two MCPs are switched off without splitting and with splitting the in-reactor space.
Figure 3. Detail of the design scheme: (a) one circulation loop with a second loop, (b) a reactor with two steam generator modules.
4. Development of a partition of the in-reactor space

To divide the in-reactor space according to the types of core elements, the core model from [17], shown in figure 7, was used. This core model was divided into groups according to the types of elements, and then individual groups were divided into segments along the radius. In this case, the passive reactivity feedback device (PRFD) bloc and the reflector bloc (RB) are modeled as separate assemblies, which are combined into groups, since these types of assemblies have a cover structure. The fuel assemblies of the central zone, the peripheral zone and the fuel assemblies with control rods of the CPS are combined into 7 regions (1 central and 6 lateral), since in this reactor installation the fuel assemblies are without cover. The in-core
storage (ICS), like the reflector unit, is modeled as separate assemblies combined into groups. Figure 8 shows a diagram of this type of partitioning.

![Diagram of reactor core division](image)

**Figure 7.** Core model used.

![Diagram of reactor core division](image)

**Figure 8.** Division of the reactor core.
As mentioned earlier, it was decided to divide the fuel assembly with fuel and control rods of the CPS into 7 segments. In this case, the assemblies of the passive reactivity feedback device block unit and the reflector unit are combined into groups so that 6 groups of each type are obtained, located around the core. It was customary to divide the in-core storage into 12 groups for more convenient modeling of segments above the core.

For each group, the hydraulic diameter occupied by the coolant was calculated taking into account the number of fuel assemblies, fuel rods, control rods of the CPS, flow area, etc.

The space above the core is modeled in the form of segments, for which the hydraulic diameter is also calculated depending on the position of the segment. Dividing the in-reactor storage into 12 segments is convenient because each pipeline to the SG is modeled as independent. Such modeling shows more accurate results when calculating transient processes, as shown in [5]. Figure 9 shows a schematic diagram of the simulation of the reactor segments above the core.

![Figure 9. Partitioning of the in-reactor space above the core.](image)

5. Conclusion
Thus, the in-reactor space is modeled not by single blocks, which do not allow observing changes during transient modes, but by a system of parallel channels, as shown in figure 10. These channels are independent of each other, which will make it possible to see changes in different parts of the reactor during transient modes.

The developed model of the partitioning of the in-core space will be further used in the calculation of various transient modes (MCP shutdown, steam generator tube rupture, etc.)

In the future, a code will be written that will allow real-time monitoring of changes in the parameters of the core during transient conditions.

There are plans to carry out joint neutron-physical and thermohydraulic calculations using the model from this work. Research devoted to the implementation of joint neutron-physical and thermohydraulic calculations is an extremely urgent task. Much contemporary work is devoted to this and related issues [18-20].
Also, this model will be supplemented with hydraulic connections between parallel channels to ensure mixing of the coolant and energy release, depending on the number of fuel assemblies and fuel elements in each section.

![Figure 10. Comparison of the two models (a) side view, (b) top view.](image)

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