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Research Article

Development of a methodological approach for the computational investigation of the coolant flow in the process of the sodium cooled reactor cooldown[*]

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Abstract

A methodological approach has been developed for the computational investigation of the thermal-hydraulic processes taking place in a sodium cooled fast neutron reactor based on a Russian computational fluid dynamics code, FlowVision. The approach takes into account the integral layout of the reactor primary circuit equipment and the peculiarities of heat exchange in the liquid metal coolant, and makes it possible to model, using well-defined simplifications, the heat and mass exchange in the process of the coolant flowing through the reactor core, and the reactor heat-exchange equipment. Specifically, the methodological approach can be used for justification of safety during the reactor cooldown, as well as for other computational studies which require simulation of the integral reactor core and heat-exchange equipment.

The paper presents a brief overview of the methodological approaches developed earlier to study the liquid metal cooled reactor cooldown processes. General principles of these approaches, as well as their advantages and drawbacks have been identified.

A three-dimensional computational model of an advanced reactor has been developed, including one heat-exchange loop (a fourth part of the reactor). It has been demonstrated that the FlowVision gap model can be applied to model the space between the reactor core fuel assemblies (interwrapper space), and a porous skeleton model can be used to model the reactor's heat-exchange equipment. It has been shown that the developed methodological approach is applicable to solving problems of the coolant flow in different operating modes of liquid metal cooled reactor facilities.

Keywords

Liquid metal cooled reactor, integral layout, methodological approach, CFD, FlowVision, reactor core, LMS, reactor cooldown, interwrapper space

Introduction

The integral layout of a sodium cooled fast neutron reactor (BN), in which all of the primary circuit's key components are installed inside of the reactor vessel, and the specific conditions of heat exchange in liquid metals, which make them different from the rest of the coolant types used in reactors, require dedicated methodological approaches to be developed to model processes of heat and mass exchange in current domestic computational fluid dynamics (CFD) codes. The developed methodological approaches are used for all kinds of computational studies into the thermal-hydraulic processes taking place in sodium cooled fast neutron reactor facilities (BN). The results of such studies are used, e.g., to justify the BN-type reactor cooldown systems.

Major difficulties arising in a computational investigation of thermal-hydraulic processes are connected with simulation of the reactor core and heat-exchange equipment. The structural complexity of fuel assemblies (FA), as well as of intermediate (IHX) and autonomous heat exchangers (AHX), explained by a large number of components these comprise, require development of a methodological approach which would make it possible to calculate with an acceptable accuracy, using well-defined simplifications and dedicated models, the thermal-hydraulic processes taking place in BN reactor facilities.

At the present time, one of the possible options under consideration for cooling down a sodium cooled fast neutron reactor facility (BN) in modes, which require decay heat removal, is the reactor cooldown through the space between the reactor core FAs referred to as the interwrapper space (Fig. 1). This reactor cooldown
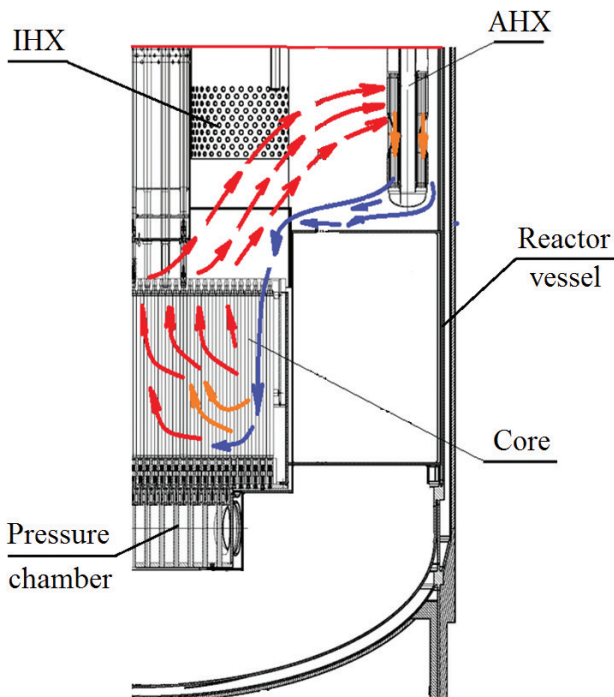

Figure 1. Diagram of the coolant circulation through the interwrapper space during the BN reactor cooldown.

option suggests that decay heat is removed from the core through forming a circulation circuit, in which the AHX-cooled primary coolant flows around the FAs on the outside, while removing heat via the FA jackets. It is assumed that some of the sodium enters the FA from the core's pressure chamber.

A number of research papers have been published for the past decade which deal with the computational analysis of the liquid metal cooled reactor cooldown using CFD codes. In (Pialla et al. 2015), an overview is presented of the studies to simulate the loss of the PHENIX fast reactor heat removal, caused by the shutdown of the primary circuit pumps leading to natural circulation developed in the primary circuit. The numerical simulation was based on system codes (CATHARE and ATHLET) and CFD codes (TRIOU and OPENFOAM). In (Rakhi and Velusamy 2017), reactor calculations in a two-dimensional axisymmetrical statement are presented, and the model employed is verified based on results obtained using other software tools (cross verification), as well as using experimental data.

The most sophisticated numerical model of the reactor core is presented in a report by the Argonne National Laboratory (Merzari et al. 2015). The FAs are modeled here using a computational grid which resolves the distances between them. A great deal of attention is given to the computational grid generation.

Studies have been underway in this country to simulate fuel assemblies and the reactor as the whole. In (Korsun et al. 2012), a calculation is presented for the BREST reactor core based on a porous body model using the APMod software module, and a cross verification with the CONV-3D DNS (Direct Numerical Simulation) code is provided.

The cooldown of the PFBR reactor, India, via the core's interwrapper space was calculated using the ANSYS-CFX and STAR-CD commercial codes (IAEA-TEC-DOC-1691 2012). For these calculations, the reactor core FAs and heat exchangers were simulated using a porous body model. Heat was delivered and removed in the computational model through the FAs and the heat exchangers respectively. Heat exchange through the FA jackets was also taken into account.

In 2015, the STAR-CD code was used to calculate the coolant flow in the Korean innovative reactor KALIMER (Park Jong-Pil et al. 2015). The simulation involved the following assumptions: the core and the heat exchangers were given using a porous body model, and no interwrapper space and FA geometry were simulated.

A number of general principles can be pointed out after considering the above methodological approaches; more specifically, the core and the heat exchangers were simulated using a porous body model, and the full FA geometry is substituted by a volume with equivalent thermophysical properties. It should be also noted that results were obtained in these studies using either domestic precision codes, which do not currently allow solving engineering analysis problems, or based on

multipurpose foreign CFD codes using RANS/URANS turbulence models which are applied to solve an extensive class of problems. Insufficient attention is given in these codes to the peculiarities of heat transfer in sodium-based liquid metal coolants, so a computational fluid dynamics (CFD) code, FlowVision, was chosen to develop the methodological approach for the computational investigation of the coolant flow in the sodium cooled reactor cooldown process. The choice of the FlowVision code is explained by the LMS (Liquid Metal Sodium) model it implements to calculate turbulent heat transfer in sodium coolant, as well as by a porous body model which makes it possible to calculate heat transfer in heat-exchange equipment. The LMS model used as part of FlowVision was verified using an extensive experimental framework (Rogozhkin et al. 2014, 2017b). The code has been certified to model thermal-hydraulic processes taking place during the sodium coolant flowing in a BN reactor (Certification Passport of the FlowVision Code 2019). However, no issues involved in the simulation of FAs and the core as the whole have been considered.

Description of the methodological approach and the computational model

A 3D computational model and a methodological approach were developed based on the FlowVision code to investigate the heat and mass transfer processes and to justify the BN reactor safety as related to the reactor cooldown through the interwrapper space.

The geometrical reactor model (Fig. 2) includes portions of the reactor core, the pressure chamber, the central rotary column (CRC), the IHX, the AHX, and the reactor coolant pump (RCP) with a pressure pipeline. The model represents a fourth part of the actual reactor structure and reproduces its only heat-removing loop.

Typical dimensions of the structural components (FAs with the interwrapper space, AHX, IHX, RCP) are much smaller than the dimensions of the structure's main part. The flow resolution in these leads to a major increase in the number of the calculation cells and to the need to use high-capacity computational resources for the calculation. To optimize the time and the computational resources spent, as well as to simulate correctly the sodium flow in the reactor components, a FlowVision gap model was used (Ozturk et al. 2019) which makes it possible to solve problems of a fluid flow in narrow channels, as compared with the main geometry, without further computational grid refinement. A gap model suggests that there is a steady-state plane flow in the narrow channel for which the resistance forces, depending on the Reynolds number, are known.

To simplify the reactor computational model, the core FAs were represented as hexagonal elements in which the coolant hydraulic resistance and heat-up were simulated.
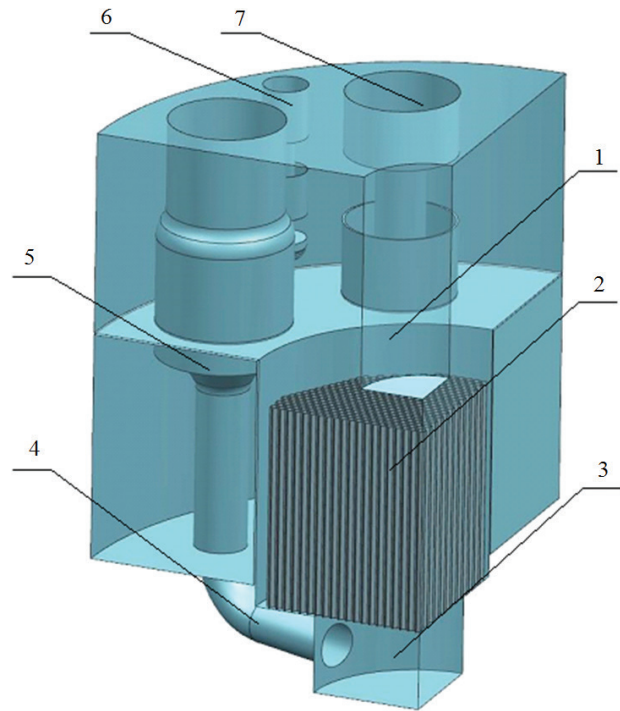

Figure 2. Geometrical model of a BN-type reactor: 1 – CRC; 2 – core; 3 – pressure chamber; 4 – pressure pipeline; 5 – RCP; 6 – AHX; 7 – IHX.

The resistance is assumed to be uniformly distributed throughout the assembly height, and the power density concentrates within the fuel (active) portion (Fig. 3).
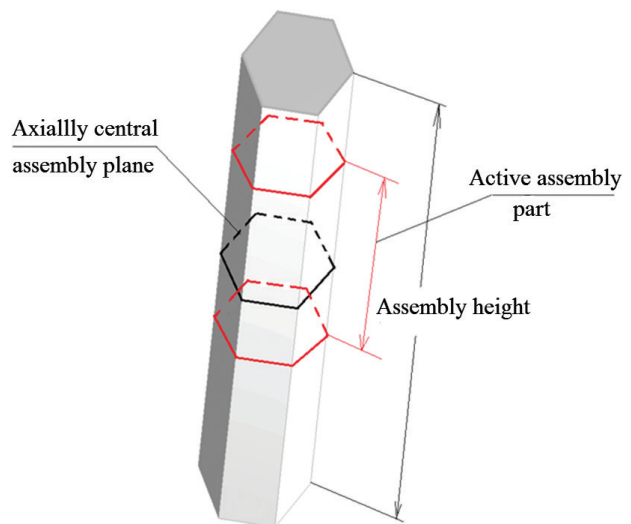

Figure 3. Computational model of an FA.

The FA model represents a metal casing in the form of a hexagonal tube on which the conditions of conjugation with the sodium surrounding the assembly and with the sodium filling the FA are given. A porous skeleton, which simulates the fuel assemblies, is given inside of the FA.

The reactor core simulation took into account the hydraulic profiling (HP) areas, including eight HP areas in the installation region of the main (central) FAs, four HP areas in the side blanket assemblies, and areas with the re-

flector, biological shielding and spent FA assemblies. The HP areas are characterized by different power density and coolant flow rate values.

A model, consisting of three FA types with different power densities, (Fig. 4) was calculated to study the effects of the grid cell dimensions on the calculation results and the possibility of using the FlowVision gap model (Ozturk et al. 2019) for the interwrapper space simulation (Fig. 4).
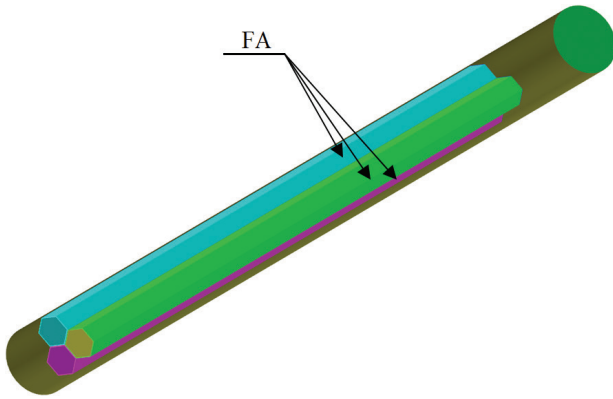

Figure 4. Geometry of a model with three FAs and an interwrapper space.

The interwrapper space in this model is geometrically divided from the FA inner volume. To take into account the heat transfer from the sodium inside of the FAs to the sodium in the interwrapper space, all assembly jackets were grouped to form separate analytical subregions, conjugated with the flow path through heat exchange, in which the equation of heat conductivity in a solid body was calculated.

A mode was considered in which the coolant with the given flow rate entered the computational region and was distributed over the FA bundle and the interwrapper space. The coolant heated in the FA entered the cavity behind the core and moved through the computational region's outflow boundary. The reference quantities were the FA pressure drop, the coolant flow rate through each FA, the coolant flow rate through the intewrapper space, and the FA bundle outlet coolant temperature values.

Simplified and detailed assembly models were calculated for two computational grids. The simplified computational grid (Fig. 5a) used an irregular initial grid condensed at the FA bundle coolant inlet. The simplified grid was locally refined in the FA jacket and interwrapper space region to obtain the detailed grid (Fig. 5b). It should be noted that, unlike the detailed grid, the simplified grid simulated the sodium flow in the interwrapper space by means of a gap model. The number of the computational cells was 37 thousand for the simplified grid and 25 million for the detailed grid.

The simplified and detailed grid calculations were compared with each other. The solution obtained for the detailed computational grid was taken as the reference. The comparison of the calculation results showed that the simplified grid results, obtained using the gap model, differed from the detailed grid results, obtained without
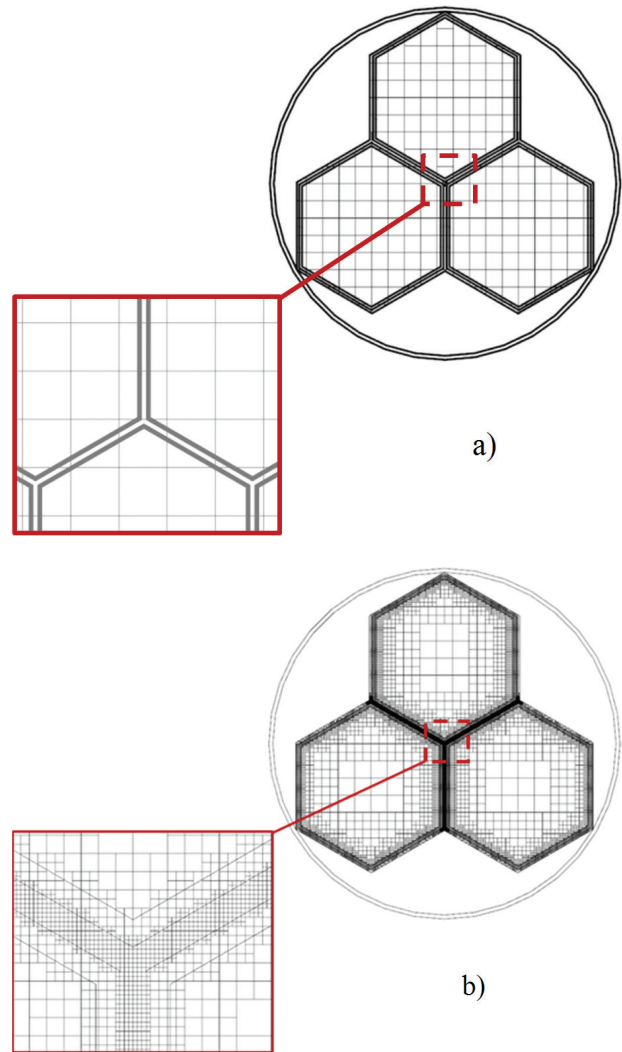

Figure 5. Computational grids (top view): simplified (**a**), detailed (**b**).

the gap model, by not more than 7%, this indicating the rightfulness of using the adopted core calculation methodology with no detailed interwrapper space resolution.

A FlowVision porous skeleton model was used to take into account the heat transfer into the IHX and the AHX. The heat exchanger tube bundles were substituted for annular elements in which the presence of simultaneously two fluids (primary sodium and secondary sodium) was given in the form of continuous and dispersed (porous skeleton) phases. The dispersed phase represents a rigid skeleton which fills a particular share of the computational region and allows simulating the irregularity of the BN reactor secondary circuit temperature distribution. The heat exchange between the circuits was taken into account by defining the thermophysical and thermal-hydraulic characteristics in the continuous and dispersed phases with the aid of user variables (Rogozhkin et al. 2017a). These variables were given with regard for the heat exchanger inlet sodium temperature in both circuits, the heat exchange surface area, the coefficient of heat exchange between the circuits, the mass flow rate, and the specific heat capacity of the conditional dispersed medium.

The porous skeleton model made it possible to evaluate the irregularity of temperature distribution in both circuits. This approach was considered using an axisymmetrical calculation of a sodium flow through the test heat exchanger as the example (Fig. 6).
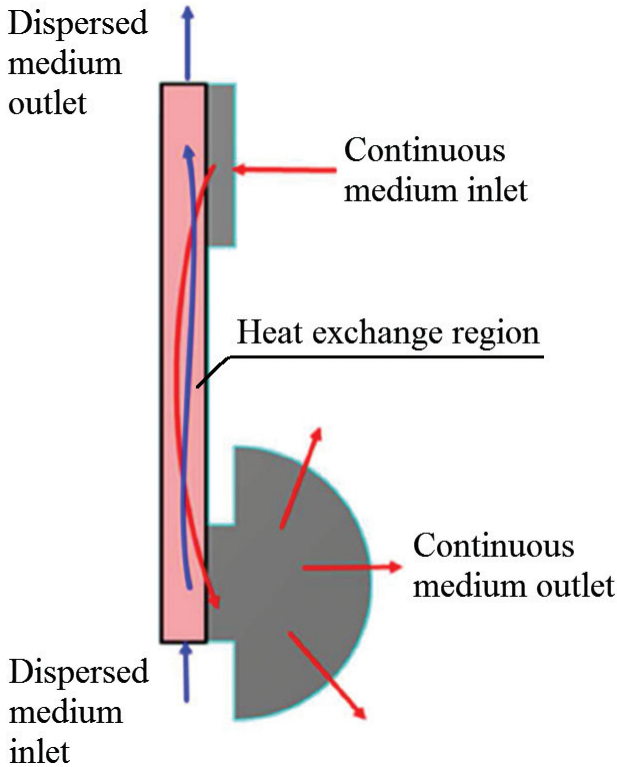

Figure 6. Geometry of the test heat exchanger computational region.

The system of equations for the mathematical heat exchanger model consists of several equations solved numerically by the FlowVision code (FlowVision User Guide 2020), including complete Navier-Stokes equations, a continuity equation (mass conservation law), an energy conservation law equation written through the specific enthalpy, and turbulence k-ε model equations. To take into account the porous skeleton, a term of the Darcy force type is added to the Navier-Stokes equation in the form

$$-F\mathbf{V} - D\mathbf{V}(\mathbf{V}\mathbf{n}_v),$$

where $\mathbf{V}$ is the fluid velocity, m/s; $\mathbf{n}_v$ is the unit vector in the velocity direction; and F [kg/m^{3}s] and D [kg/m^4] are the coefficients responsible respectively for the linear and velocity quadratic parts of the resistance force. These coefficients are determined either empirically or through detailed calculations of the fluid movement inside of the FAs. The porous skeleton model adds also the fluid displacement as a result of the skeleton being present in the fluid-occupied volume.

It was found as a result of the heat exchanger test calculation that the heat exchanger parameters determined by the code differ from the design parameters by not less than 1%.

Calculation of the reactor rated operation mode

A three-dimensional coolant flow in a BN reactor in the rated power operation mode was calculated to test the developed methodological approach and the computational model.

The coolant flow and the heat exchange in the reactor flow area was described in FlowVision using an ideal liquid turbulent flow model with regard for coupled heat transfer. The RCP sections in the pressure pipeline region are taken as the computational model boundaries so that to define the coolant inlet and outlet conditions. The normal component of the coolant mass flow rate, the coolant temperature, and the initial flow turbulence level is given for the model inlet. A free exit condition was given for the computational region outlet with giving the zero static pressure value.

An irregular initial grid condensed towards the FA region was used. Additionally, the initial grid was locally refined (adapted) in the FA wall region for a more detailed simulation of the coolant flow. The number of the computational cells in the model was about 5.6 million. The small number of the computational cells for such class of problems is explained by the fact that one of the reactor's four heat-removing loops was calculated and a gap model was used.

A comparison of the results, obtained by numerical simulation, with the design data has confirmed that the developed computational model gives a good agreement between the obtained results and the design. The deviation of the average coolant in-core heat-up value, obtained in the calculation, does not exceed 3% of the design value, and the error of the core average pressure drop value is not more than 7.4%. The main sodium coolant circulation line in the reactor rated operation mode has the RCP – pressure chamber – core – IHX – AHX direction. The maximum coolant temperature is at the core outlet.

Conclusions

A methodological approach has been developed for the computational investigation of the coolant flow in a reactor during cooldown via the interwrapper space using the FlowVision computational fluid dynamics (CFD) code.

The developed methodological approach and a reactor model were used to study computationally the rated mode of the coolant flow in the reactor in a three-dimensional statement.

A comparison of the numerical simulation results with the design data has confirmed that the methodology to calculate the coolant flow in the reactor core's interwrapper space, using the FlowVision gap model, gives a good agreement of the obtained results with the anticipated parameters. No need for resolving the gaps between the FAs using a detailed computational grid leads to a considerable economy of the computational resources and time required for the calculations.

Simulation of the AHX and IHX heat exchangers using a porous skeleton model makes it possible to calculate the temperature distribution for the EHRS primary and intermediate circuits with regard for the heat retention capability of structural materials.

The developed methodological approach allows solving, with the required accuracy, complex problems of the coolant flow in BN-type reactors both in normal operating conditions and during anticipated operational occurrences requiring the EHRS connection.

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