Application of ATHLET/DYN3D coupled codes system for fast liquid metal cooled reactor steady state simulation

V Ivanov[1], A Samokhin[1], I Danicheva[1], N Khrennikov[1], J Bouscuet[2], K Velkov[2] and I Pasichnyk[2]

[1]Scientific and Engineering Centre for Nuclear and Radiation Safety (SEC NRS), Malaya Krasnoselskayast, 2/8, bld. 5, 107140, Moscow, Russian Federation
[2] Gesellschaft für Anlagen- und Reaktorsicherheit (GRS), Forschungszentrum, Boltzmannstraße 14, 85748 Garching bei München, Germany

E-mail: asamokhin@secnrs.ru

Abstract. In this paper the approaches used for developing of the BN-800 reactor test model and for validation of coupled neutron-physic and thermohydraulic calculations are described. Coupled codes ATHLET 3.0 (code for thermohydraulic calculations of reactor transients) and DYN3D (3-dimensional code of neutron kinetics) are used for calculations. The main calculation results of reactor steady state condition are provided. 3-D model used for neutron calculations was developed for start reactor BN-800 load. The homogeneous approach is used for description of reactor assemblies. Along with main simplifications, the main reactor BN-800 core zones are described (LEZ, MEZ, HEZ, MOX, blankets). The 3D neutron physics calculations were provided with 28-group library, which is based on estimated nuclear data ENDF/B-7.0. Neutron SCALE code was used for preparation of group constants. Nodalization hydraulic model has boundary conditions by coolant mass-flow rate for core inlet part, by pressure and enthalpy for core outlet part, which can be chosen depending on reactor state. Core inlet and outlet temperatures were chosen according to reactor nominal state. The coolant mass flow rate profiling through the core is based on reactor power distribution. The test thermohydraulic calculations made with using of developed model showed acceptable results in coolant mass flow rate distribution through the reactor core and in axial temperature and pressure distribution. The developed model will be upgraded in future for different transient analysis in metal-cooled fast reactors of BN type including reactivity transients (control rods withdrawal, stop of the main circulation pump, etc.).

1. Introduction
The next generation of fast reactors is under development in Europe. Implementation of advanced calculation tools for transient simulation is important to provide safety analysis of such systems. To improve computational accuracy, a coupled multi-physics approach is used. Such approach allows performing static and dynamic analyses integrating all relevant physical phenomena: neutronics, thermal-hydraulics and structural mechanics. Current tendency of performing realistic steady-state and transient simulations is to couple best estimate thermo-hydraulics codes with 3D neutronic, thermo-mechanic, burnup calculation and other physical codes. Such approach appears to be more consistent especially for Beyond Design Basis Accidents.

In this work the first results obtained with coupled version of 3D diffusion code DYN3D and ATHLET for fast sodium cooled reactor transient simulations are represented. DYN3D is a reactor physic calculation code for Steady States and Transients developed at Helmholtz-Zentrum Dresden-Rossendorf in Germany [2]. It uses the multigroup neutron diffusion approximation for hexagonal or quadratic fuel assemblies. Coupled version was successfully used and verified for PWR and WWER reactors transient calculations, including Kalinin-3 benchmark [3].

ATHLET is an advanced best-estimate code which has been initially developed for the simulation of design basis and beyond design basis accidents (without core degradation) for light water reactors [1]. Then the ATHLET 3.0 version was upgraded with properties of metal coolants such as sodium, lead-bismuth, lead and helium. But it is still in validation phase. These updates allow made possible to use ATHLET for metal cooled fast reactors calculations.

Previously an attempt to perform comprehensive validation of ATHLET and DYN3D for sodium cooled reactors was made. Preliminary verifications of DYN3D multigroup diffusion code were performed on the example of the BN-type international benchmarks [4]. ATHLET developer team has performed standard thermo-hydraulic tests for sodium coolant [1]. As far as verification showed plausible results for both ATHLET and DYN3D stand-alone versions it is possible to perform first test coupled calculations with ATHLET/DYN3D codes.

BN-800 reactor was chosen for simulation. It is 2100 MWt sodium cooled power reactor designed in Russia by OKBM. BN-800 represents the next generation of BN-reactor types and has been in operation at the Beloyarskaya NPP site since 2015. Test core model was developed in accordance to BN-800 type reactor design and based on public open source materials. The resulting test model for coupled calculations uses simplified approach where only core with upper and lower plenums are modeled. Such approach allows simulating main BN-800 core design features and it is an important step towards coupled version verification as well as first step in development of BN-800 model for safety analysis. Steady-state calculations were performed as part of the testing.

2. Modeling approach for coupled calculations

The next approach was used for the first sodium reactor simulations with coupled ATHLET/DYN3D version: the designed model is limited by core, lower and upper plenum; fixed boundary conditions are used for inlet flow and temperature, and for outlet coolant pressure and enthalpy. The resulting so called "open core" proved to be good as prior step before the full scale model development (core, loops, technological channels) and is applicable for the analysis of transient behavior in the core.

The BN-800 core layout [5] which was planned for the start of operation ("fresh" fuel in the core) was used for the simulation. The simplified approach was used for the modeling: every assembly is homogenized, core and blanket areas have the same axial dimensions, only one type of assemblies is assigned to every fuel zone. Simplified axial assemblies description is based on [6]. This description was used both for neutronic and thermo-hydraulic modeling.

For the testing model only two thermal-hydraulic feedbacks are considered: the fuel temperature and the coolant density. For that case parameterized cross section library was prepared for 4 fuel temperatures and 4 coolant temperatures. Linear approximation was used to calculate reactivity feedbacks during coupled calculations.

Homogeneous macroscopic cross sections are required for each assembly for DYN3D calculation. They were prepared by the lattice code NEWT from SCALE 6.1 package. The work [4] gives an overview of the applied studies and methods for generation of parameterized few-group macroscopic cross sections with SCALE code, adapted to the test transient simulation.

The described approach allows performing test steady-state calculations. The calculated thermo-hydraulic parameters can be used to validate coupled version results by comparing with BN-800 design features.

2.1. Neutron model

The model consists of only the core region (including radial, upper and lower reflectors). The model consists of low, medium, high enrichment sub-cores and MOX-sub-core. Each sub-core is assigned as one thermal-hydraulic channel group.

The core (figure 1) consists of three uranium dioxide fuel regions i.e., inner (low enrichment zone, LEZ), middle (middle enrichment zone, MEZ), and outer (high enrichment zone, HEZ) core regions which have 136, 94 and 139 fuel assemblies, respectively. Uranium zone is surrounded with layer of MOX zone. The MOX core zone is bounded by radial blanket zone (RB) followed by one row of radial reflector (RF) assemblies. To ensure correct axial neutron leakage, top and bottom assembly's technological parts were homogenized to be used as axial shielding.

The fuel is uranium dioxide with the enrichment of 17, 21, 26 % for inner, middle and outer cores, respectively. The MOX fuel is considered as a mixture of natural uranium with 17% plutonium-239. Radial and axial blankets are composed of natural uranium. For the core structural material austenitic (fuel cladding) and ferritic/martensitic (assembly cladding and structure) steels are used [6].
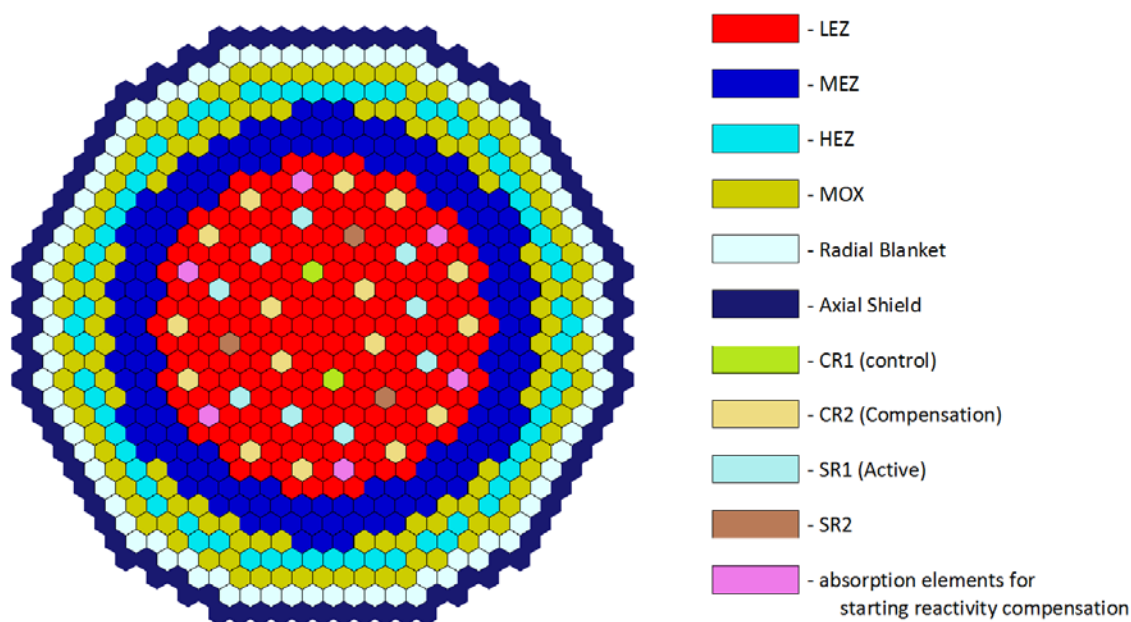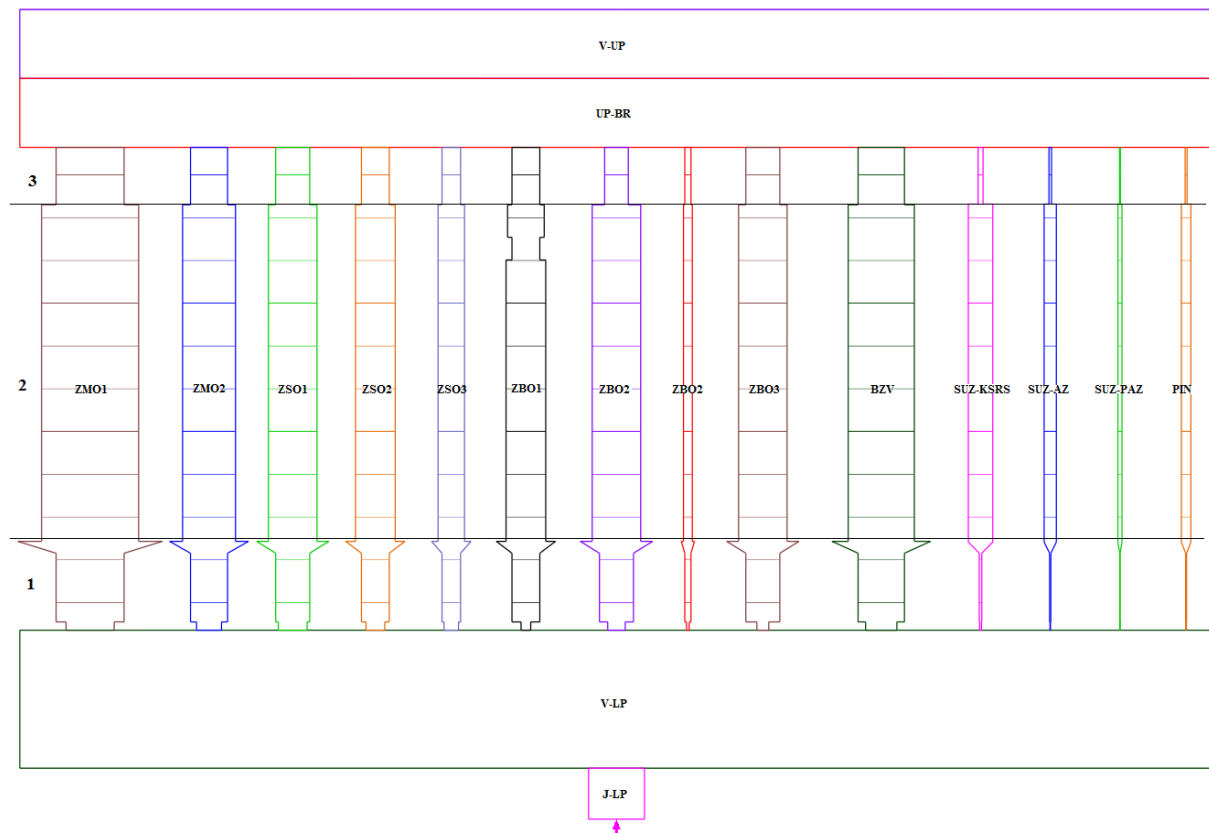

Figure 1. Core layout

2.2. Thermo-hydraulic model

The core hydraulic nodalization scheme consists of 6 hydraulic zones: one zone for each fuel zone (HEZ, LEZ, MEZ and MOX), one zone for radial blanket and one zone for channels with control rods. The assemblies of each hydraulic zone have difference in cross-section area of its inlet part. This difference is necessary for correct coolant mass-flow rate core profiling. The corresponding value of hydraulic cross-sectional area was chosen on the reactor radial power distribution. To achieve suitable mass flow in every hydraulic zone the normalized radial power distribution obtained from standalone DYN3D calculation (with critical positioning of control rods) was used. With accordance to power distribution, hydraulic cross-sectional area distribution was selected.

ZMO1 –ZMO2 – LEZ assemblies, ZSO1 – ZSO3 - MEZ assemblies, ZBO1 - ZBO3 – HEZ assemblies, BZV – blanket assemblies, SUZ-KSRS, SUZ-AZ, SUZ-PAZ, PIN – control rods, J-LP – core mass flow rate boundary condition,V-LP – lower plenum, UP-BR – upper plenum, V-UP – pressure and enthalpy boundary condition.

1 – inner part of assemblies, 2 – middle part of assemblies, 3 – outer part of assemblies

Figure 2. Core hydraulic nodalization scheme

3. Steady state calculation results

As first step, reaching of stabilized reactor parameters at nominal power (steady state conditions) is required to get the correct neutronic and thermo-hydraulic starting parameters before transient calculations. The calculation process by A/D coupled code system consists of three phases: «zero transient» - the model initialization, «steady state» - the stabilization of reactor parameters and «transient» - the calculation of a transient. In the present model, the stabilization of the reactor parameters was achieved after 60 seconds. The transient calculations can be started after this stabilization period, but no any transients were considered in this article. The main reactor parameters [6] had the following values after stabilization: maximum fuel temperature - 1260°C; maximum fuel cladding temperature - 480°C; total power - 2100 MWt and inlet/outlet core coolant temperature - 354/547°C, respectively. Graphics of reactor total power, mean outlet core coolant temperature and maximum fuel temperature and fuel rods cladding temperature during the withdrawal are presented in Figures 3-6.
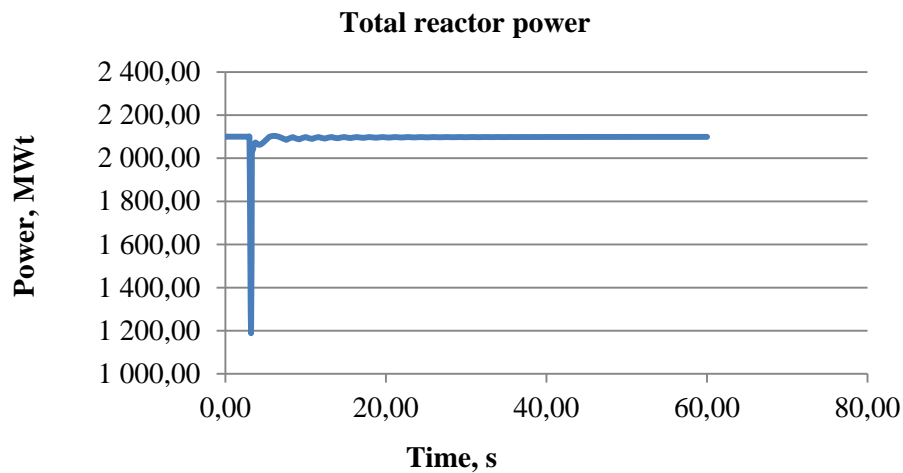
Total reactor power

Figure 3. History of the total reactor thermal power.
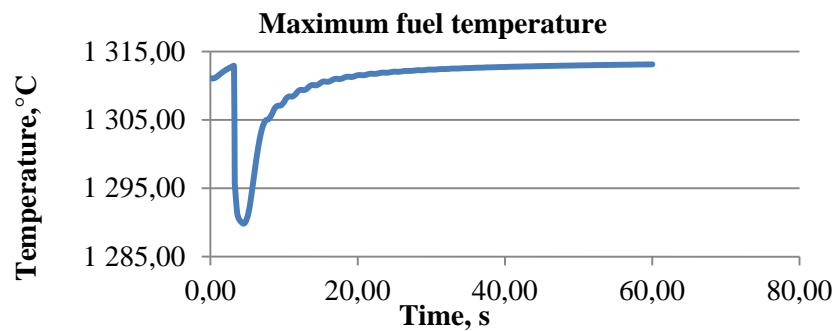
Maximum fuel temperature

Figure 4. History of the reactor maximum fuel temperature.

So, as it follows from Fig.3-6, the calculation by A/D coupled codes provides physically plausible results. That means that data transfer between coupled codes is correctly. So, it can be concluded that A/D coupled codes and developed model can be applied for calculation of different transients on fast metal cooled reactors.
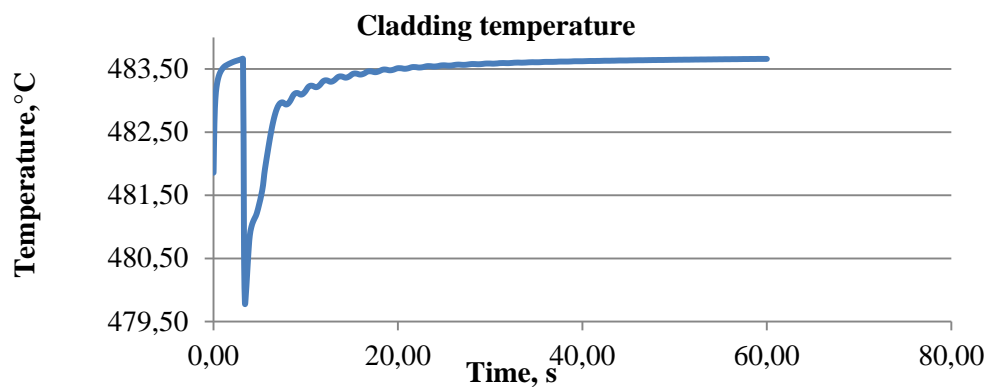
Cladding temperature

Figure 5. History of the reactor maximum cladding temperature.
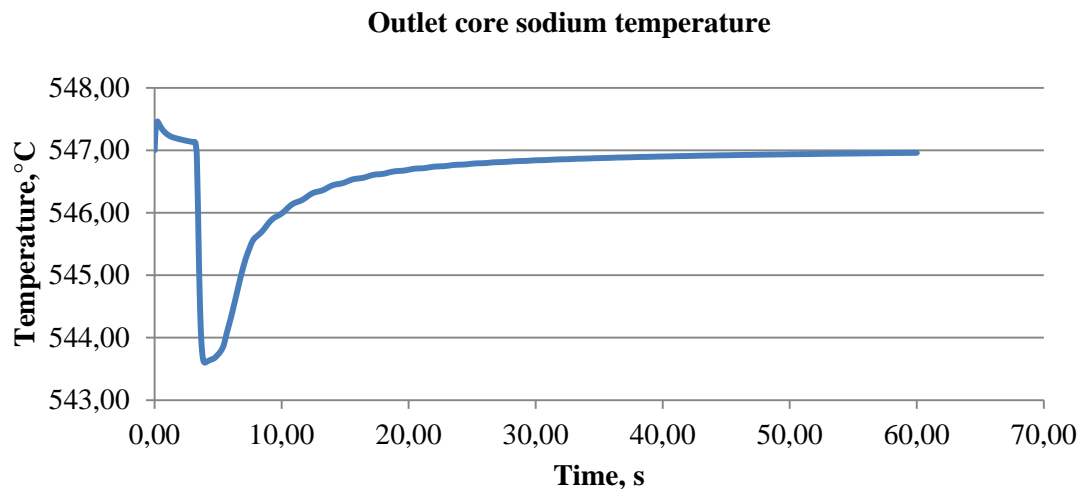
Outlet core sodium temperature

Figure 6. History of the core outlet sodium temperature.

4. Conclusion

The paper describes the developed model of BN-800 type core and the simulation of steady state reactor parameters at nominal power performed by using of the coupled code system A/D.

The steady state calculations showed that data transfer between coupled codes is correct, obtained results are acceptable and developed model can be used as basis for future investigations.

This work is the first step toward liquid metal cooled reactor independent safety analysis with A/D coupled codes. Further developments of the applied methodology will be continued and will include the model updating to the whole BN-800 reactor description for calculation of different transients, basis and beyond basis accidents It is also planned to model the thermal expansion of different core construction parts and to research it's influence on the reactivity.

References

[1]   ATHLET Mod 3.0 Cycle A. User's Manual, (2012). GRS - P - 1/ Vol. 1 Rev. 6 – GRS, Garching bei München, Germany

[2]   Rohdeet U. al., (2016). The reactor dynamics code DYN3D - models, validation and applications, Progress in Nuclear Energy 89, 170-190

[3]   Kozmenkov Y., Kliem S., Rohde U., (2015). Validation and verification of the coupled neutron kinetic/thermal hydraulic system code DYN3D/ATHLET. Ann. Nucl. Energy 84, 153–165.

[4]   Bousquet J., Ivanov V., Danitcheva I., Khrennikov N., Velkov K., Klim S., Samokhin A. (2015). Assessing Reactor Physics Codes SCALE, SERPENT and DYN3D to Simulate Fast Reactors on the Example of the BFS-62-3ABenchmark. In: IPPE Nyutronik conference, Obninsk 12-16 October 2015. –p. 20. (In Russian);

[5]   Vasiliev B., Farakshin M., Belov S., Kuznetsov A. (2016). Prospects of development of BN-800 reactor core. In: 10th International Scientific and Technical Conference, Safety Efficiency and Economy of Nuclear Power Industry, Moscow 25-27 May 2016

[6]   I. Kuznetsov, V. Poplavski, (2012). Safety Aspects of NPP's with Fast Reactors, ISBN 978-5-86656-257-2, Moscow (In Russian)