# Coupled calculations for the fast reactors safety justification with the EUCLID/V1 integrated computer code

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**Abstract**. The EUCLID/V1 integrated computer code is designed for the safety analysis and justification of the NPPs with liquid metal cooled fast reactors under normal operating conditions, normal operation failure, design basis accidents and beyond design basis accidents. The EUCLID/V1 code includes the system thermohydraulic module, spatial time-dependent neutronic modules, quasi two-dimensional fuel rod module and the modules of burnup and decay heat calculations. In the neutronic modules the improved quasistatic method is employed to solve the transport equation in the multigroup diffusion or discrete ordinates (Sn) forms.

This work is devoted to the simulation results of some of the BREST-OD-300 and BN-1200 normal operation failure and accidents by means of the EUCLID/V1 code. Particularly, the loss of offsite power regime in the BN-1200 reactor has been modeled. In this case the reactor shutdown cooling with the emergency heat removal system is considered. It has been shown that after the control rods drop and reactor pumps shutdown the efficiency of two of four channels of the emergency heat removal system is sufficient to prevent the maximum cladding temperature from exceeding the safety limits. In frame of the test calculations the accident caused by the insertion of the total positive reactivity margin via withdrawal of all control rods from the reactor core at the maximum design speed during full power operation without scram operation (UTOP+ULOF) has been simulated for the BREST-OD-300 reactor facility. The obtained results indicate that after the reactor coolant pumps shutdown the total power decreases due to the thermohydraulic reactivity feedbacks and operation of the passive feedback system. By 100 s of the scenario natural circulation of lead progresses in the primary side, it is equal to 9.6% of the nominal total flow. The core disruption does not happen.

**Key Words**: EUCLID/V1 integrated computer code, safety analysis and justification, BREST-OD-300, BN-1200.

## 1. Introduction

According to the Russian federal rules in the field of atomic energy use and approaches recognized by the international community the NPP project safety must be justified, including numerical modelling of the various operation modes with computer codes that allow performing self-consistent simulations of various physical processes. EUCLID/V1 is an integrated code, designed to simulate the behavior of reactor facilities under normal operating conditions, normal operation failure, design basis accidents and beyond design basis accidents by performing the coupled neutron-physical, thermo-mechanical and thermohydraulic calculations [1]. By the requirements of the Federal Environmental, Industrial and Nuclear Supervision Service of Russia (Rostechnadzor) use of the computational codes for the safety justification of nuclear facilities must be proved, in other words must be justified the software's ability to simulate the process and phenomena occurring in the respective operation modes of the object, with the certain accuracy on the basis of validation and verification (V&V) calculations. The extensive V&V of the single modules of the EUCLID/V1 code have been carried out on analytical and numerical tests, experimental results and benchmarks [2], [3]. To validate a coupled modelling of physical processes in a reactor core and its loops, the data obtained on the BN-600, BN-800 and BOR-60 transient regimes have been used.

In this paper the results of the calculations on the safety justification of the BREST-OD-300 [4 – 7] and BN-1200 [8 – 10] reactor facilities with the EUCLID/V1 code are presented.

## 2. Brief code description

The EUCLID/V1 code consists of the following modules (see FIG. 1.):

- HYDRA-IBRAE/LM system thermohydraulic module [11];
- BERKUT fuel rod module [12];
- DN3D neutronic module;
- modules of the burnup (BPS [13]) and residual heat (OSTB) calculation.

For the preparation of nuclear data the CONSYST code [14] is used with the RESEAU module for the cross-section interpolation.

The HYDRA-IBRAE/LM module [11] is used to calculate the thermohydraulic processes occurring in the loops of fast reactor facilities with sodium or lead coolants. It is also possible to simulate water and air loops using this module. In addition, there are special models for simulation of interloop leaks in case of steam generator tube rupture. The two-phase fluid model is employed with the set of verified and validated closure relations. The gas phase represents a coolant vapour and a set of noncondensable gases for all coolants except for lead. Lead coolant vapours are not modelled due to the low values of saturated vapour pressure at the operating temperature range. Thermohydraulic module HYDRA-IBRAE/LM uses the database, which contains thermodynamic and thermophysical coolants properties. These data are presented in a range of pressures and temperatures required for the calculation of the various operating modes of the BREST-OD-300 and BN-1200 reactors. Thermohydraulic module HYDRA-IBRAE/LM allows simulating thermohydraulic processes in the normal operating conditions, normal operation failure and in case of accidents.

The BERKUT fuel rod module of the integral EUCLID/V1 code is used to simulate the processes occurring in the fuel rods of the fast reactor with mixed nitride uranium-plutonium and mixed-oxide fuel, mononitride and uranium dioxide fuel and cladding made of austenitic or ferro-martensitic steels with gas gap under normal operating conditions, normal operation failure and emergency conditions [12]. The BERKUT module allows solving the heat conductivity task in a fuel rod, simulating the stress-strain state of the fuel pellet and the cladding in case of an open gap or contact between cladding and pellet. In addition, it can be used to calculate data on the fission products release into the gas gap, irradiation and thermal creep strain accumulation, swelling etc. The axial symmetry approximation of boundary conditions and temperature field is used to calculate the fuel rod temperature distribution. Basic relations for the determination of the fuel rod strain-stress state are derived from the equilibrium equations, strain compatibility and physical laws coupling stress and strain.

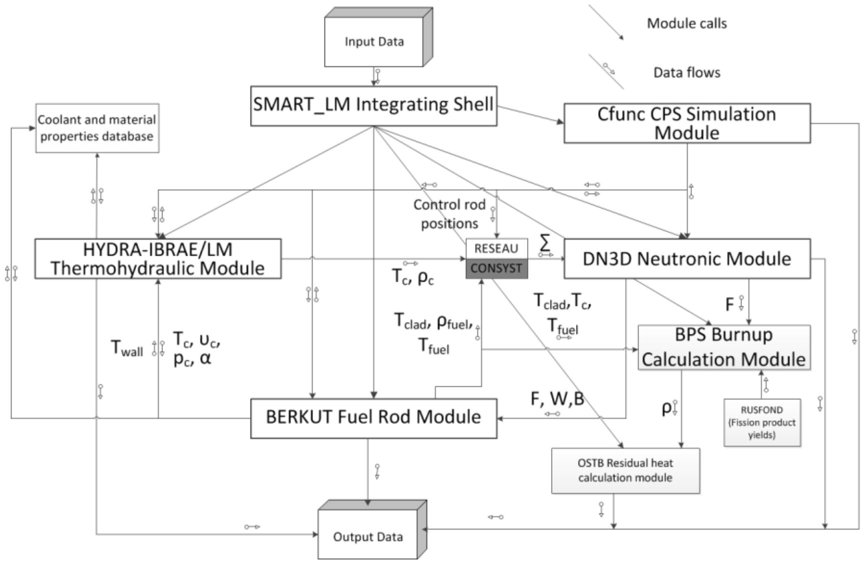



*FIG. 1. Structure of the EUCLID/V1 code: $T_{wall}$ – fuel rod wall temperature; $T_c$ – coolant temperature; $T_{clad}$ – fuel rod cladding temperature; $T_{fuel}$ – fuel temperature; $\rho_{fuel}$ – fuel density; $\rho_c$ – coolant density; $u_c$ – coolant velocity; α – heat exchange coefficient; $p_c$ – coolant pressure; F – neutron flux; W – power; B – burnup; ρ –concentrations of fuel nuclides and fission products; Σ – neutron cross sections*

The DN3D neutronic module is used for the full-scale modelling of neutron physical processes in fast reactors. The following major processes are modelled: changes in neutron flux and power distribution within the core, control rod movement, changes in the reactivity and others. The multigroup diffusion approach to the neutron-transport equation is considered. The DN3D module has the option to calculate neutron flux using mesh, which consists of 7 points per a fuel assembly (so-called G7 module). Thus the G7 module takes into account heterogeneity of the fuel assembly structure. This approach provides the correct estimate of the control rods worth and realistic description of the neutron flux heterogeneity.

As a part of the neutronic module the kinetic option is implemented, which is designed to calculate spatial and energy distribution of the neutron flux and its functionals including problems with cavities, large gradients of neutron field and problems with a high degree of attenuation of radiation. The kinetic option is based on the $S_N$ discrete ordinates method. To account for the scattering anisotropy $P_M$ approximation is used. The approximation of spatial dependence is represented by finite-difference WDD (Weighted Diamond Difference) and nodal schemes. Energy dependence is taken into account in the multi-group approximation. Calculations can be carried out both in a three-dimensional hexagonal (HEX-Z) and combined geometry. In the latter case, the spatial grid consists of hexagonal and tetrahedral prisms.

The BPS module [13] (Burning and Poison calculation System) allows to calculate any nuclide chain transitions for actinides and fission products. As a database BPS uses ROSFOND’s MF-90 (JSC «SSC RF – IPPE») library [15], which contains data on almost 1600 nuclides from the fission products, including tritium from ternary fission.

The contribution to the decay heat from all decaying nuclides in the OSTB module is determined by the equation (1) for actinides and fission products:

$$W = 1.60219 \cdot 10^{-13} \cdot \sum_j \rho_j \lambda_j E_j\ , \qquad (1)$$

where W – decay heat (W); $\rho_j$ – nuclide j density, nuclei/cm$^3$; $\lambda_j$ – decay constant of nuclide *j*, 1/s; $E_j$ – heat from the nuclide *j* fission, MeV/fission.

## 3. BN-1200 simulation

The EUCLID/V1 code allows simulating the following modes of the BN-1200 reactor facility:

- start-up;
- power mode;
- planned shutdown;
- operation with partial number of coolant circulation loops;
- unauthorized insertion of positive or negative reactivity;
- violation of heat removal from the reactor core due to equipment failure or operator error;
- violation of heat removal from the reactor core due to partial overlap of the flow cross section;
- interloop leakage (intermediate heat exchangers, autonomous heat exchangers);
- loss of offsite power.

The EUCLID/V1 nodalization scheme of the BN-1200 reactor facility includes the main equipment of the primary, secondary and water loops:

- core;
- upper plenum;
- pumps of the primary side;
- pipelines of the primary side;
- lower plenum;
- tract of the reactor vessel cooling;
- gas system of the primary side;
- intermediate heat exchangers;
- overflow chambers of the intermediate heat exchangers;
- pipelines of the secondary side;
- coolant pumps of the secondary side;
- steam generators;
- emergency heat removal system.

The core is divided into parallel channels. In each channel fuel and/or absorbing rods are modelled. In the considered BN-1200 model the fuel and absorbing rod cladding is made of EK-164 c.w. austenitic steel, the nuclear fuel is MOX, the absorber material is boron carbide. Fuel and absorbing rods are filled with helium gas.

The BN-1200 core contains 1408 fuel assemblies (FA). FA pitch is equal to 185 mm. The core and baffle map of the BN-1200 reactor is shown in FIG.2. For the BN-1200 simulation the neutron-physical model with seven computational points per a fuel assembly is realized. In the axial direction the BN-1200 model is split into 51 layers.

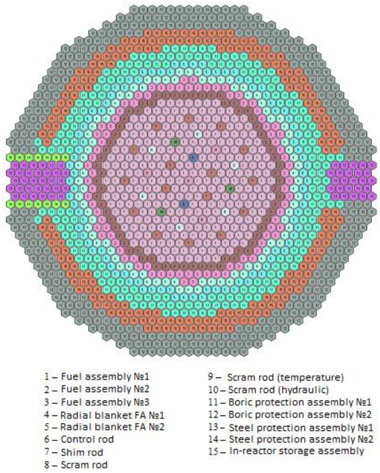


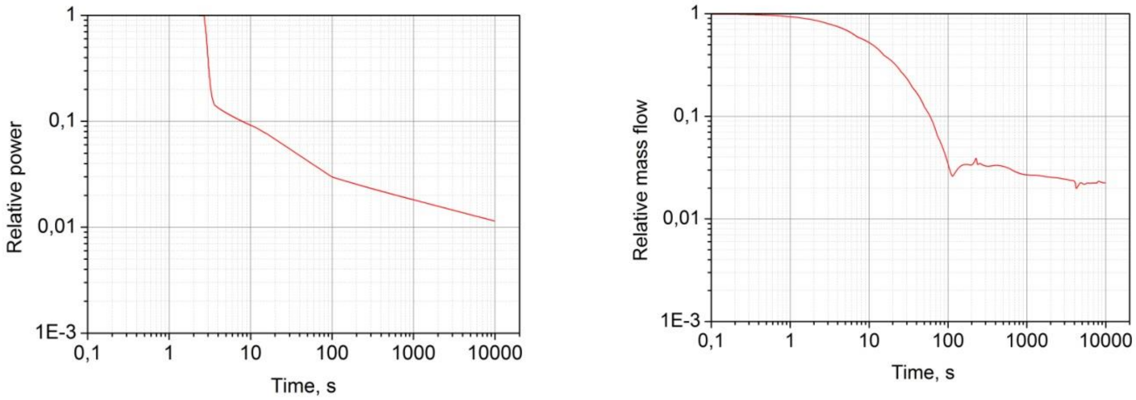

*FIG. 2. Core and baffle map of the BN-1200 reactor model*

The normal operation failure regime «loss of offsite power» in the BN-1200 reactor has been modelled. In this regime the reactor shutdown cooling with the emergency heat removal system is considered. According to the scenario at the beginning the BN-1200 reactor facility is in full power operation. The loss of offsite power leads to the shutdown of the coolant pumps of the primary and secondary sides and loss of feed water flow through the steam generators. To avoid possible severe consequences the following measures are envisaged in the BN-1200 project [16]:

- the scram, control and reactivity compensation rods are inserted into the core;
- the passive scram rods drop after the sodium mass flow through the core decreases up to 50% of the nominal value;
- the emergency heat removal system starts;
- the steam generators are isolated and filled with nitrogen (from the third loop side).

Two of four loops of the emergency heat removal system are conservatively supposed to fail.

In FIG. 3-4 some of the obtained calculation results are presented. The fuel rods cladding temperatures do not exceed the safety limit. The peak sodium outlet temperature is about 580°C, the peak cladding temperature is about 620°C.




a) b)

*FIG. 3. Relative power as a function of time (a) and relative sodium mass flow through the core as function of time (b)*

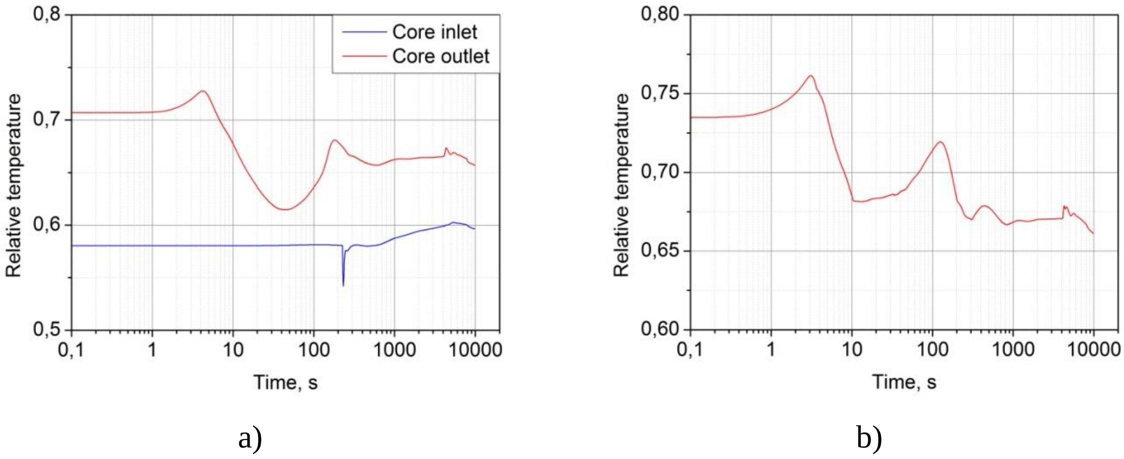



*FIG. 4. Relative sodium temperature as a function of time (a) and relative maximal fuel rod cladding temperature as a function of time (b)*

## 4. BREST-OD-300 simulation

The EUCLID/V1 code allows simulating the regimes of the BREST-OD-300 reactor facility, caused by the following initial events (normal operating conditions, normal operation failure):

- start-up;
- power mode;
- planned shutdown;
- operation with partial number of coolant circulation loops;
- unauthorized insertion of positive reactivity;
- violation of heat removal from the reactor core due to equipment failure or operator error;
- pressure changes in the primary side;
- deterioration in heat dissipation by the secondary side;
- excessive heat dissipation by the secondary side;
- pressure changes in the secondary side.

To simulate the BREST-OD-300 reactor modes the corresponding model has been developed in terms of the EUCLID/V1 input file. The BREST-OD-300 core with 169 fuel assemblies (FAs) is considered. The central zone (CZ) includes 109 FAs:

- 84 ordinary CZ FAs;
- 7 FAs with shutdown rods;
- 14 FAs with reactivity compensation rods;
- 4 FAs with automatic control rods.

The peripheral zone (PZ) includes 60 FAs. The core is surrounded by 3 rows of 144 hexagonal channels of the Pb baffle, 18 of them are channels of PFS (passive feedback system). PFS realizes the passive feedback between the reactor reactivity (power) and the coolant flow (coolant inlet head). The Pb baffle channel pitch is equal to the FA pitch. Behind the Pb baffle 66 blocks of the steel baffle are located. The core and baffle map is shown in FIG. 5.

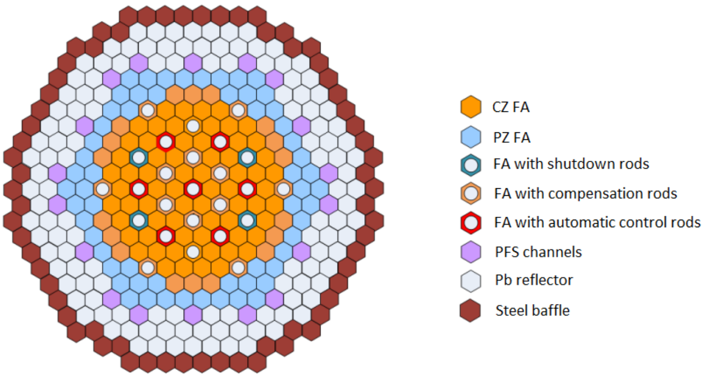



*FIG. 5. Core and baffle map of the BREST-OD-300 reactor model*

The thermohydraulic model includes following elements:

- core, central and peripheral cavities of the reactor;
- steam generators;
- reactor coolant pumps;
- heat exchangers of the emergency heat removal system.

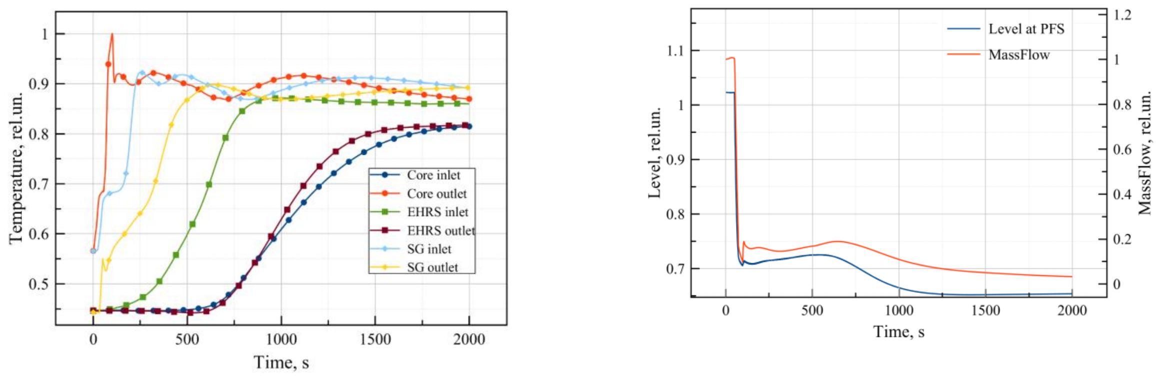
The lead circulation tract is presented with four loops according to the number of reactor coolant pumps. In each loop the independent regions of lead flow from the core to the steam generators and the lead return passes to the downflow part of the reactor central cavity are modelled.

The qualification procedures (geometrical fidelity) [17] of the BREST-OD-300 computational model and steady state parameters have been done. The obtained results indicate the good agreement between the EUCLID/V1 data and BREST-OD-300 design specifications.

In frame of the test calculations the accident caused by the insertion of the total positive reactivity margin via withdrawal of all control rods from the reactor core at the maximum design speed during full power operation without scram operation (UTOP+ULOF) has been simulated for the BREST-OD-300 reactor facility.

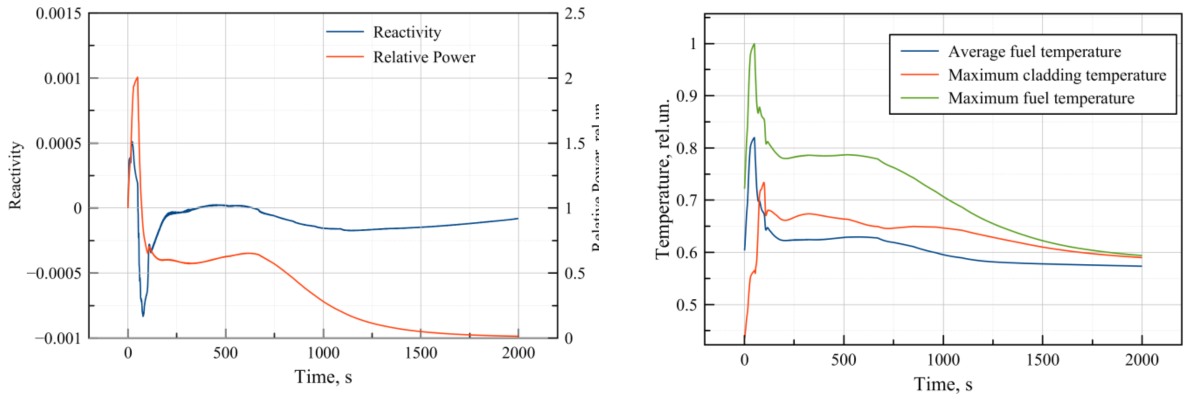
According to the accident scenario the BREST-OD-300 initial state is full power operation. The initial event is the control and protection system failure, when two groups of control rods are withdrawn in turn and the positive reactivity of 0.65 $\beta$ is inserted. The shutdown, emergency power reduction and controlled power reduction systems fail. The aggravating scenario factor consists in the steam generator shutdown because of lead core outlet temperature exceeding 620°C. The coolant pump protection system tripping is envisaged after the steam generator outlet temperature reaches 520°C. With the coolant pump shutdown the forced coolant circulation in the primary loop stops and the passive feedback system introduces the negative reactivity. Two of four loops of the emergency heat removal system starts operation when the lead outlet temperature in the air heat exchanger reaches 450°C. Other two loops of the emergency heat removal system are supposed to fail.

FIG. 6–7 show the simulation results of the scenario described above. The lead inlet and outlet temperatures versus time in the core, steam generators and emergency heat removal system are presented in FIG. 6 a. By 100 s of the scenario natural circulation of lead progresses in the primary side, it is equal to 9.6% of the nominal total flow. (FIG.6 b).The coolant pump shutdown results in the negative reactivity insertion due to the decrease of lead level in the passive feedback system channels (see FIG.6 b and FIG. 7 a). Because of that the reactor transits to the subcritical state.




*a)* *b)*

*FIG. 6. Relative lead inlet and outlet temperatures versus time (a) and relative lead mass flow and lead level in the passive feedback system channels versus time (b)*




*a)* *b)*

*FIG. 7. Relative reactivity and integral power (in rel. un.) versus time (a) and relative average fuel, maximal fuel and maximal cladding temperatures versus time (b)*

The simulation of this accident evidences that the maximal cladding temperature exceeds the design limits, but does not exceed the melting point.

## 5. Conclusion

The loss of offsite power in the BN-1200 reactor facility (normal operation failure simulation) has been modelled using EUCLID/V1 code. It has been found that the fuel rod cladding temperature safety limit has not been exceeded.

The insertion of the total positive reactivity margin via withdrawal of all control rods from the reactor core at the maximum design speed during full power operation without scram operation in the BREST-OD-300 reactor facility have been modelled with the EUCLID/V1 integrated computer code. It was shown, that the maximal cladding temperature has exceeded the design limits but it has been lower than the melting point and the core disruption has not happened.

## 6. Acknowledgements

The EUCLID/V1 code has been developed within the "Codes of a new generation" subproject of the "PRORYV" (or "Breakthrough") project at the expense of the Federal targeted program "Nuclear power technologies of a new generation for 2010 – 2015 and for the future till 2020".

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