Neutronic calculation of fast reactors by the EUCLID/V1 integrated code

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Abstract. This article considers neutronic calculation of a fast-neutron lead-cooled reactor BREST-OD-300 by the EUCLID/V1 integrated code. The main goal of development and application of integrated codes is a nuclear power plant safety justification. EUCLID/V1 is an integrated code designed for coupled neutronics, thermomechanical and thermohydraulic fast reactor calculations under normal and abnormal operating conditions. EUCLID/V1 code is being developed in the Nuclear Safety Institute of the Russian Academy of Sciences. The integrated code has a modular structure and consists of three main modules: thermohydraulic module HYDRA-IBRAE/LM/V1, thermomechanical module BERKUT and neutronic module DN3D. In addition, the integrated code includes databases with fuel, coolant and structural materials properties. Neutronic module DN3D provides full-scale simulation of neutronic processes in fast reactors. Heat sources distribution, control rods movement, reactivity level changes and other processes can be simulated. Neutron transport equation in multigroup diffusion approximation is solved. This paper contains some calculations implemented as a part of EUCLID/V1 code validation. A fast-neutron lead-cooled reactor BREST-OD-300 transient simulation (fuel assembly floating, decompression of passive feedback system channel) and cross-validation with MCU-FR code results are presented in this paper. The calculations demonstrate EUCLID/V1 code application for BREST-OD-300 simulating and safety justification.

1. Introduction

The main goal of development and application of integrated codes is a nuclear power plant safety justification. The codes provide a self-consistent simulation of various physical processes. One of these codes is the EUCLID/V1 [1] integrated code designed for coupled neutronics, thermomechanical and thermohydraulic fast reactor calculations under normal (steady state on the permitted power levels, normal transient conditions) and abnormal operating conditions.

EUCLID/V1 code is being developed in the Nuclear Safety Institute of the Russian Academy of Sciences within the PRORYV (or Breakthrough) project of the Russian Federal Targeted Program "Nuclear Power Technologies of the New Generation for 2010–2015 and until 2020". The integrated code has a modular structure and consists of three main modules: thermohydraulic module HYDRA-IBRAE/LM/V1, thermomechanical module BERKUT and neutronic module DN3D. In addition, the integrated code includes databases with fuel, coolant and structural materials properties.

Neutronic module DN3D [2] provides full-scale simulation of neutronic processes in fast reactors. Heat sources distribution, control rods movement, reactivity level changes and other processes can be
simulated. Neutron transport equation in multigroup diffusion approximation is solved. Neutron field can be calculated for seven radial nodes in each assembly (subunit G7) to take into account absorber heterogeneity in the fuel assembly. This approach provides the control rods worth correct calculation and more detailed simulation of the neutron flux.

Kinetic option is included in DN3D module for calculation the spatial and energy neutron flux distribution and different functionalities in the problems with cavities, large gradients of neutron field and strong degree of radiation extinction. Kinetic option is based on the $S_n$ discrete ordinates method. Database of nuclear cross sections is prepared by CONSYST [3] code.

A fast-neutron lead-cooled reactor BREST-OD-300 transient simulation (fuel assembly floating, decompression of passive feedback system channel) and cross-validation with MCU-FR code results are shown in this paper.

2. Coupled calculations of BREST-OD-300 reactor system

Fuel assembly floating modes calculated for one (control system rods operating/failure) and two assemblies floating (control system rods operating/failure). Reactor power, full reactivity and parts of reactivity are considering. The initial state is steady state at nominal power level. The floating velocity equals 5 cm/s. An additional mode is calculated with one assembly floating at speed of 20 cm/s and control system rods are operating.

Figures 1 and 2 present power changes for different floating velocity, figures 3 and 4 presents reactivity and parts of reactivity.

![Figure 1](image1.png)  
**Figure 1.** Power changes (floating velocity = 5 cm/s).

![Figure 2](image2.png)  
**Figure 2.** Power changes (floating velocity = 20 cm/s).

![Figure 3](image3.png)  
**Figure 3.** The reactivity and parts of reactivity (floating velocity = 5 cm/s).

![Figure 4](image4.png)  
**Figure 4.** The reactivity and parts of reactivity (floating velocity = 20 cm/s).

In case of slow floating (5 cm/sec) control system rods can keep power inside the dead zone. In case of faster floating (20 cm/sec), power decreases fast down to 80% level and then control system rods...
stabilize power at the initial level. Fuel assembly floating insert negative reactivity, which can be compensate by control system rods. Fuel temperature negative feedback is more efficient in case of a fast floating of fuel assembly. The fast power level decrease down to 80% causes fuel temperature change and fuel temperature reactivity effect gives a considerable contribution in reactivity balance. Decompression of passive feedback system channel was simulated. In this simulation control system rods weren’t operating. At the first stage of process the lead level in passive feedback system channel decreases uniformly down to 130 cm, which corresponds to coolant rate decline. At the second stage the lead level increases uniformly from 130 cm up to coolant free level in the reactor. The second stage is actually the decompression of passive feedback system channel. Changes of power level, reactivity and parts of reactivity are shown in the figures 5 and 6.

At the first stage, with the lead level in passive feedback system channel decreasing, negative reactivity is inserted, which can be compensated by fuel temperature feedback. At the second stage, with the lead level in passive feedback system channel increasing, negative reactivity from the first stage is compensated by insertion the positive reactivity, which a caused by lead level increasing, and the positive reactivity can be compensated by fuel temperature negative feedback.

3. Cross-validation with MCU-FR code

At the lack of experimental data cross-validation is very important for systems with lead coolant. The BREST-OD-300 reactor core model is calculated with EUCLID/V1 and MCU-FR [4] codes. MCU-FR code is the version of MUC, based on MCU-6 package and designed for calculations of systems with a fast neutron spectrum.

MCU-FR code uses 3D pin-by-pin fuel assembly nodalization and continuous energy cross sections supplied with the code. The cross-sections are prepared from RUSFOND [5] files and temperature of 600 K is included. EUCLID/V1 code uses homogenization of fuel assemblies and cross sections for 26 energy groups, prepared with CONSYST code based on RUSFOND files.

Four different critical states under normal reactor operating conditions are calculated and neutron spectrum, neutron multiplication factor and power distribution are analyzed. Different control rods positions are defined in the states (table 1).

| Variant № | The control rods position (cm) |
|-----------|-------------------------------|
|           | CR1  | CR2  | RC   | PS   |
| 1         | 50   | 50   | 60   | 70   |
| 2         | 50   | 130  | 60   | 70   |
| 3         | 130  | 130  | 60   | 70   |
| 4         | 50   | 130  | 60   | 180  |
The core of BREST-OD-300 contains protection system (PS) rods, reactivity compensator (RC) rods and control system (CS) rods. The position is set from the bottom of reactor, and control system rods has two groups (CR1 and CR2) with different position for each group.

The neutron spectra for 26 energy groups calculated by EUCLID/V1 and MCU-FR codes are in a good agreement (figure 7) for variants 1 and 4.

![Neutron spectrum](image1)

**Figure 7.** Neutron spectrum for variants 1 (a) and 4 (b).

Difference of neutron multiplication factors in codes EUCLID/V1 and MCU-FR does not exceed 0.1% (table 2) for all variants. The calculation error in MCU-FR code equals 0.0002.

| Variant № | EUCLID/V1 | MCU-FR | Difference (%) |
|-----------|-----------|--------|----------------|
| 1         | 1.0085    | 1.0093 | 0.08           |
| 2         | 1.0065    | 1.0070 | 0.05           |
| 3         | 1.0046    | 1.0053 | 0.07           |
| 4         | 0.9760    | 0.9757 | 0.03           |

The agreement can be explained by different heterogeneous correction factors in CONSYST code.

Power distribution calculations are caused by good agreement of neutron multiplication factors.

Power distributions are considered for variants 1 and 4 with very different control rods positions. Power distributions for variants 2 and 3 are almost the same as for variant 1.

The axial power distributions calculated by EUCLID/V1 and MCU-FR (figure 8) are in good agreement for variants 1 and 4.

![Axial power distribution](image2)

**Figure 8.** The axial power distribution for variants 1 (a) and 4 (b).
The radial power distributions calculated by EUCLID/V1 и MCU-FR in the middle of reactor core (figure 9) are in good agreement for variants 1 and 4.

4. Conclusion
The fuel assembly floating and decompression of passive feedback system channel modes for BREST-OD-300 reactor system was simulated with the EUCLID/V1 integrated code. The calculations demonstrate EUCLID/V1 code application for BREST-OD-300 simulating.

The EUCLID/V1 и MCU-FR result analysis demonstrates that EUCLID/V1 code models allow calculating correct neutron multiplication factors and power distributions. The difference of $k_{eff}$ does not exceed 0.1%. The maximum difference in power distributions equals 5%.

References
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