Detailed Engineering Neutron Codes for Calculations of Fast Breeder Reactors

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Abstract. Detailed engineering neutron codes are designed for calculations of fast breeder reactor cores. In IBRAE a variety of code systems based on diffusion method has been developed and used for different representations of fuel assemblies. One of the codes designed is intended for three-dimensional neutronic calculation in the pin-by-pin approximation. In the first version of the code the fuel assembly represented by a set of regular hexagonal prisms modeling the fuel cluster surrounded by a set of irregular hexagonal prisms which simulate the fuel assembly blanket and the surrounding space. Moreover, boundary microcells belong to two (three for corner microcells) fuel assemblies simultaneously. The irregular five-sided prisms that belong to the only one fuel assembly surround the set of prisms modeling fuel rods in the second version of the code. The first code allows to carry out the calculations of the reactor core in which all fuel assemblies with the same number of fuel elements are presented. The second code allows to calculate the reactor core in which the fuel assemblies with different number of pins are presented. The G7 code uses the hex view of the hex assembly in the form of a hexagonal prism as a central cell surrounded by six trapezoidal prisms. It allows to simulate the assembly shrouds to the external cells but fuel elements and absorber elements to the central cell (pin-cell region). All cells in the models have their material content and size. The paper presents results of calculations for existing and planned reactors obtained by the specified codes.

Key Words: fast breeder reactor core, neutron codes, pin-by-pin model.

Introduction

The development of detailed engineering neutron codes for reactor core calculations caused by the problems of design and operation of fast breeder reactors. Lately researchers and developers have been worked on detailing the following problems:

(a) burnup calculations with the issuance of a large set of actinides and fission product;

(b) calculations on nuclear safety of facilities with fresh and spent nuclear fuel (the calculation with the actual set of assemblies);

(c) calculations of fuel or experimental assemblies pin-by-pin and pin-cell regions descriptions with tracing the assembly characteristics during campaign.

This paper will focus on the problem (c).

1. Code systems

In IBRAE a variety of code systems including GEFEST [1] and EUCLID [2] has been developed and used. From the point of neutron physics view, the basic geometric simplification is a homogeneous representation of fuel assemblies and control rods. This representation is a source of uncertainty in the determination of the control rod worth, the maximum values of the h in the assembly and, accordingly, derivative characteristics based on these values.

The GEFEST used to validate the safety of loading for core of the BN-600 reactor at Beloyarsk Nuclear Power Plant (BNPP). The basis of GEFEST is the code of neutronic calculations in multi-group diffusion approximation for the three-dimensional hexagonal geometry. In GEFEST a set of local characteristics stored in the fuel archive for the assembly is calculated. This fact allows to monitoring the history of each assembly. Two pin-by-pin models were designed and the FUBUKI code was created within GEFEST.

For simulating of the liquid metal cooled fast breeder reactor (BN-1200 and BREST-OD-300) the EUCLID/V1 integral code is used. A feature of these reactor cores is that width across flats of the assembly is almost two times more than for the BN-600 reactor. Solving of the neutronic problem in the approximation of one point on the assembly obviously assumes a large uncertainty in the areas of neutron flux gradient (border of fuel assemblies and assemblies without fuel or control rods) and the larger the assembly size the larger such uncertainty. In addition to the large-sized assembly, the BREST-OD-300 reactor uses the original design of the safety control system. The control rod occupies the central part of the fuel assembly, i.e. there are fuel and absorber at the same time in the assembly. In order to account for these features correctly it was developed the model of assembly partitioned by seven parts (hexagonal prism surrounded by six trapezoidal prisms) and the G7 code.

2. Geometric representation of assemblies

2.1. Models of FUBUKI code

2.1.1. Description of pin-by-pin representation of BN-600 reactor assemblies in the case of 169 cells on assembly

The construction of geometric model for pin-by-pin representation is demonstrated on an example of the fuel assembly in reactor core.

Fuel assemblies of the BN-600 reactor core are a heterogeneous structure and contain 127 pins (see fig. 1). It was suggested to represent the calculated channel cell with fuel assemblies or the control rods by 127 cells that each simulating one pin. These cells are supplemented by cells modeling the shroud of fuel assemblies or control rods, so that the boundary cells of one fuel assembly are the boundary cells of other.

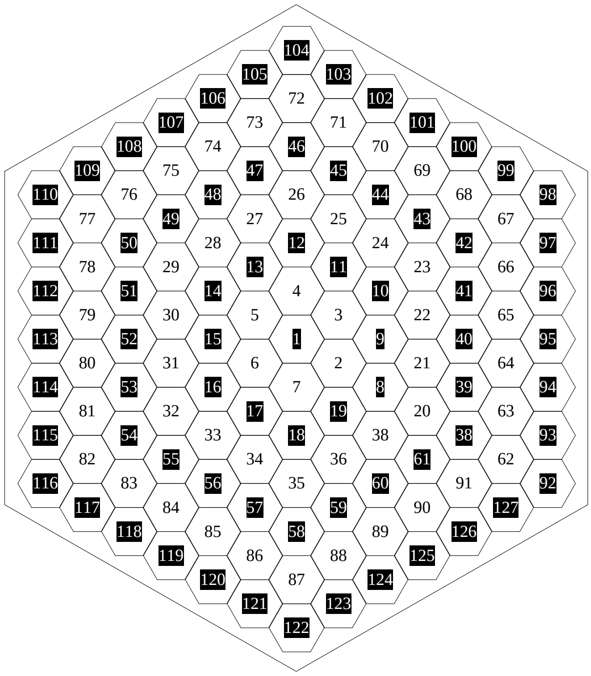

FIG. 1. Simulation diagram of the fuel assembly in the BN-600 reactor core

The hexagonal computational grid for internal cells and irregular hexagons for external ones is obtained in this simulation (see FIG. 2).

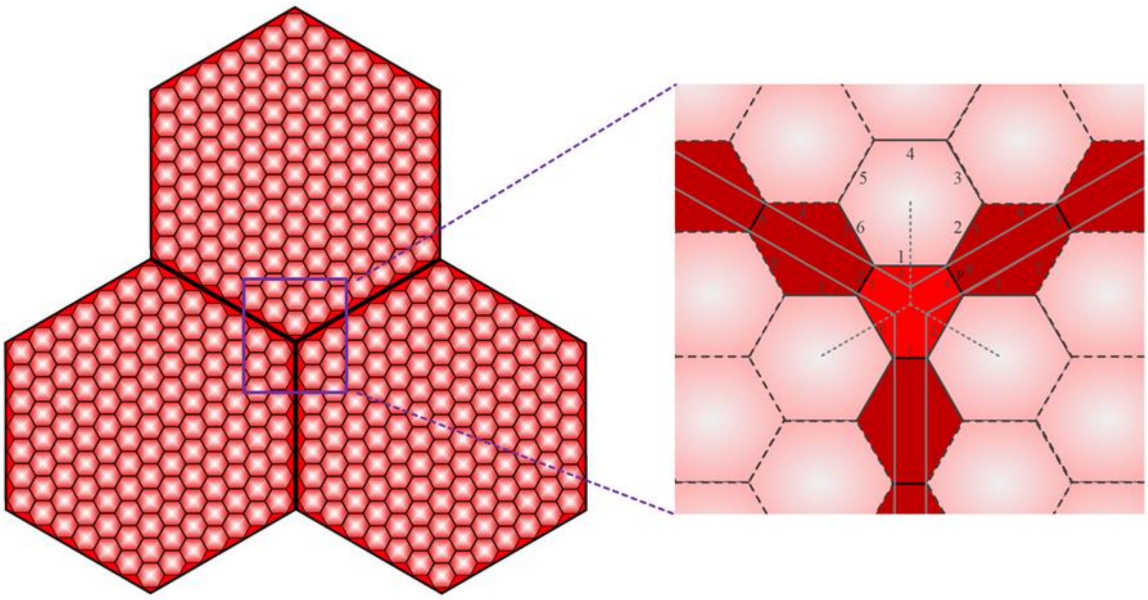

FIG. 2. The three fuel assemblies with the same number of pins

For FUBUKI-1 code it is required to set the values of two geometric parameters that completely describe the computational grid, i.e. h is the width across flats of the internal cell and p is the half-width of the boundary cell.

For all assemblies (except control rods) in the boundary cells it is simulated the shroud of the assembly with an adjacent sodium. Fig. 3 shows a representation of the control rods.

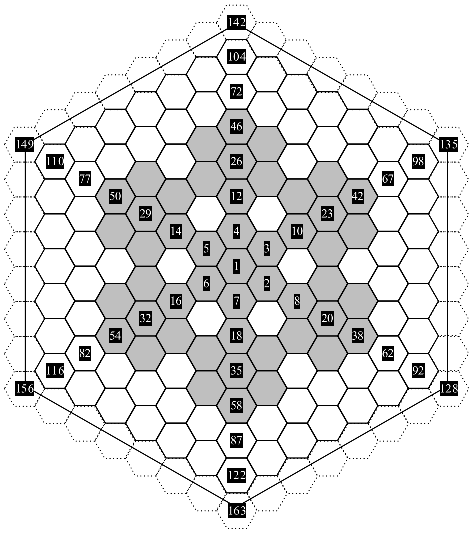

FIG. 3. Cells of the BN-600 reactor model for the control rod (cells with absorber are marked with a dark color)

2.1.2. Description of pin-by-pin representation of BN-600 reactor assemblies in the case of a different number of cells in the assembly

In the BN-600 reactor core there are fuel assemblies with two sets of pins. The fuel assembly with enriched fuel contains 127 pins; the fuel assembly of the blanket contains 37 pins. Also the experimental fuel assemblies with 61 pins are mounted in the BN-600 reactor core and it is planned to install them further. Getting of a correct calculation data of pins in case of fuel assemblies with a variety number of pins has required developing of the model and the computational algorithm called FUBUKI-2. Internal cells of all fuel assemblies are correct hexagonal prisms and contain the one fuel element. H is the width across flats of assemblies, h is the width across flats of assemblies. The boundary cell is a five-sided. Due to the difference in sizes of boundary cells each of them may be in contact with more than one cell of the adjacent fuel assembly (see fig. 4 and 5).

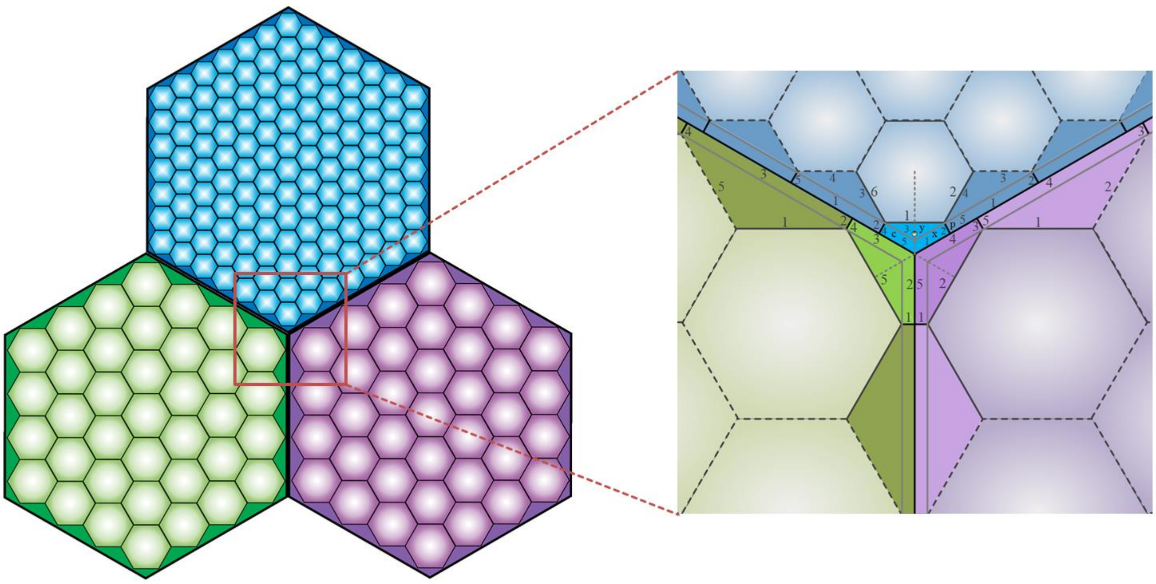

FIG. 4. The geometry of the junction for three fuel assemblies. Two fuel assemblies with 37 pins and a fuel assembly with 127 pins

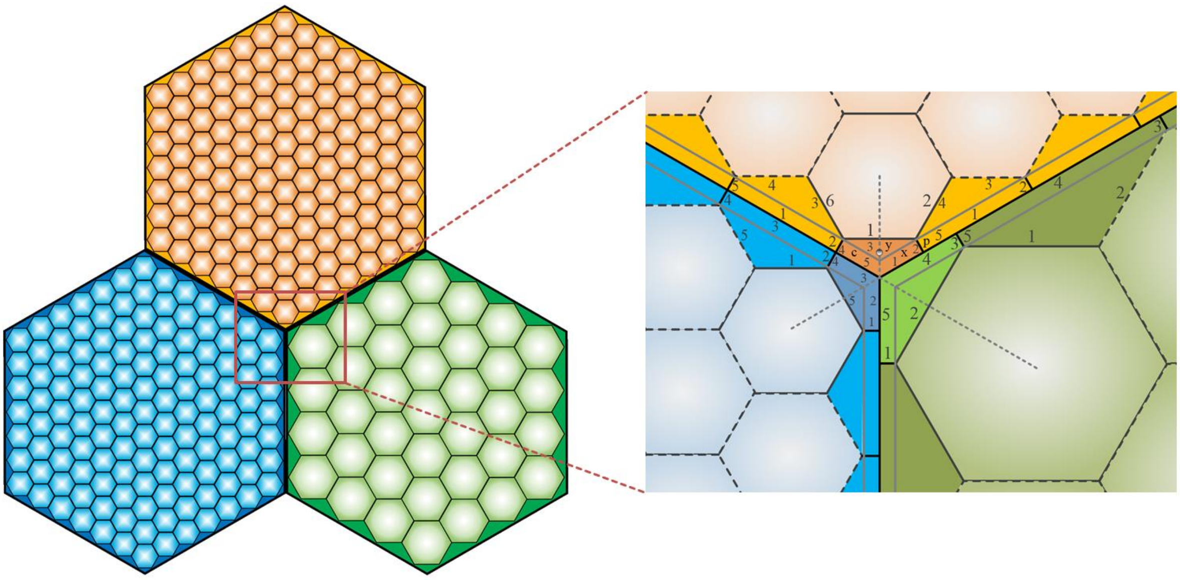

FIG. 5. The geometry of the junction for three fuel assemblies. Two fuel assemblies with 127 pins and a fuel assembly with 37 pins

The boundary cells of all fuel assemblies are homogenized so that their material and macroscopic cross-sections accordingly considered the same for all boundary cells.

2.2. Model of G7 code

The fuel assembly model consists of seven prisms. The G7 code uses the hex view of the hex assembly in the form of a hexagonal prism as a central cell surrounded by six trapezoidal prisms (see fig. 6a). Moreover, a width across flats of the central cell for each of the assemblies may be different (see fig. 6b). In general, each cell (pin-cell region) contains the unique composition of materials.

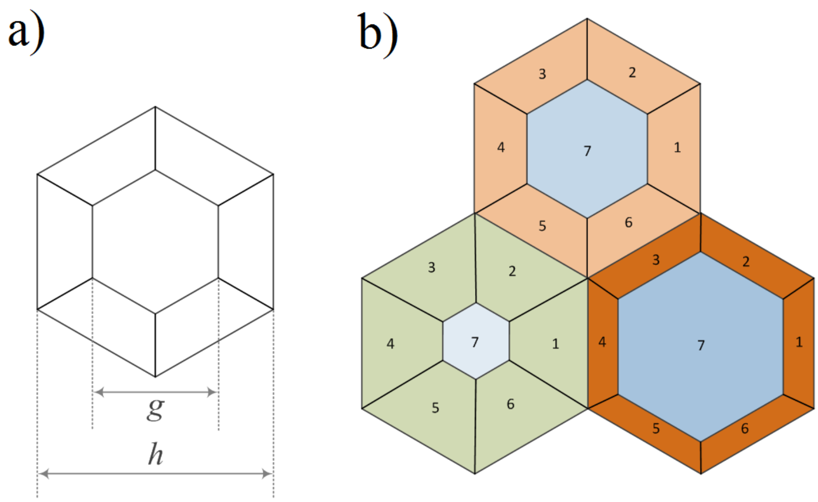

FIG. 6. Assembly model in G7 code.

a) Single assembly;
b) Example of assemblies with different sizes of the central cell

3. Fuel archive

An important part of the code system is the fuel archive. The fuel archive is a file storing characteristics of fuel assemblies, rods and other components for reactor model. It stores the assembly data required for the neutronic calculation taking into account the burnup mode.

The file structure is defined by the geometric parameters of the reactor model. In case of homogeneous representation of fuel assembly the record of the one "average" pin will be stored to the archive. For pin-cell region (see fig. 6) representation the seven records are stored for each fuel assembly. For pin-by-pin (see fig. 4-5) representation the number of records equals the number of pins in fuel assembly.

The fuel archive provides the calculation and the storage of the necessary parameters (nuclear concentrations, geometrical dimensions etc.) and the various characteristics (fluence, power density, radiation burden etc.) for all geometrically selected objects in the reactor model.

4. Some calculated results

4.1.Calculation of the combined experimental fuel assembly using FUBUKI-1

The combined experimental fuel assembly was placed to the BN-600 reactor. It contains 123 pins with uranium oxide fuel and 4 pins with mixed nitride fuel. The scheme (see fig. 7) shows the placement of nitride pins. The combined experimental fuel assembly "looks" at the center of the core by the face with cells 149-156. Nitride pins numbered 82 and 87 are the closest to the center, and nitride pin numbered 122 is the most distant.

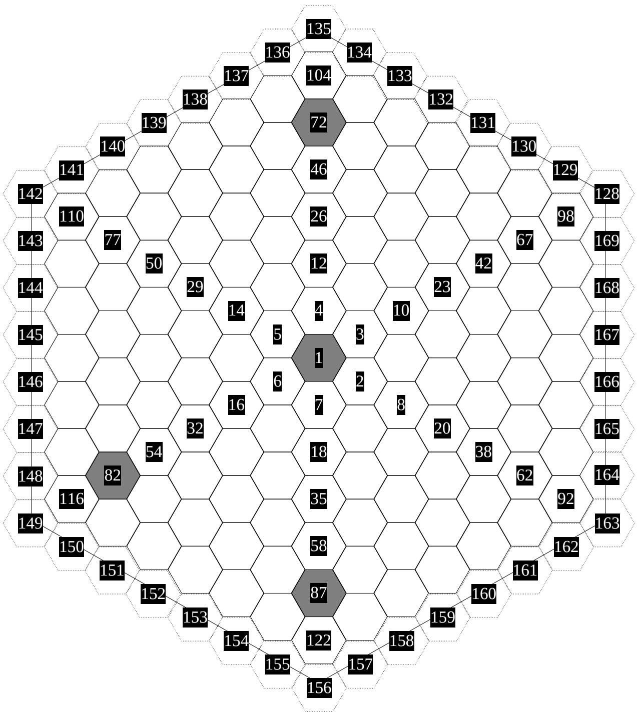

FIG. 7. Combined experimental fuel assembly with nitride pins

Fig. 8 shows the distribution of the relative pin burnup at the end of the campaign for the central layer of the reactor core. Processing of the normalization was performed by the value of the average burnup in the layer.

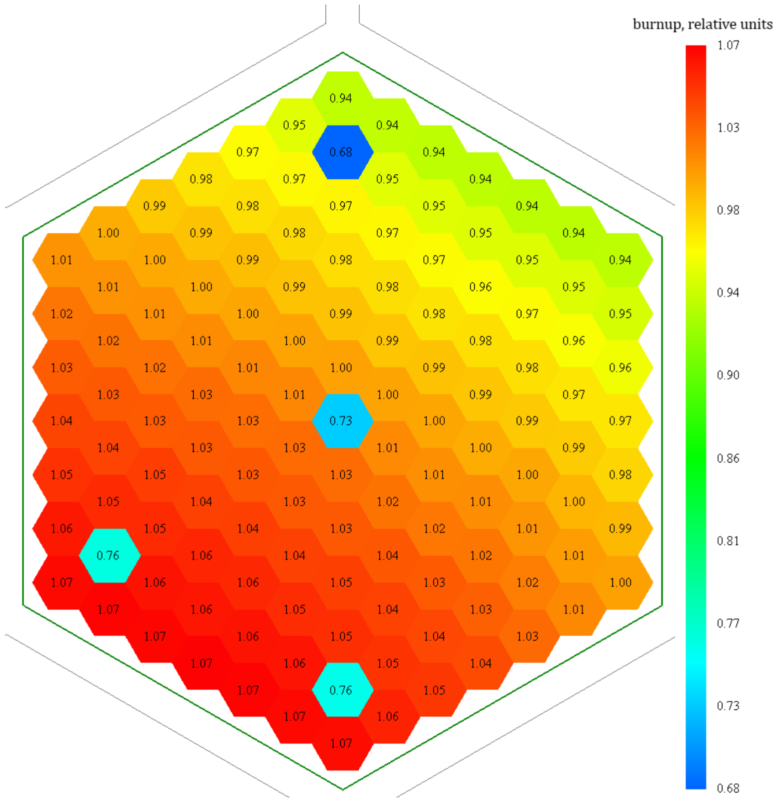

FIG. 8. Distribution of the pin burnup in the combined fuel assembly at the end of the campaign

The FUBUKI-1 code allows to obtaine the calculated characteristic by pin-by-pin approximation directly. It is not necessary to solve the problem of interpolation the neutron flux within the fuel assembly.

4.2 Calculation of the experimental fuel assembly using FUBUKI-2

An experimental fuel assembly was loaded at the BN-600 reactor core. In contrast to the ordinary (standard) fuel assembly, it contained 61 pins with the nitride fuel. The planned campaign of the burnup was simulated by using the FUBUKI-2 code.

Fig. 9 shows the distribution of the relative burnup in the experimental fuel assembly at the end of its campaign for the central layer.

Fig. 10 shows the power distribution of the pins in the experimental fuel assembly at the end of its campaign. The normalization performed by the average value.

Fig. 11 shows the distribution of the relative single-group flux in the experimental and two adjacent fuel assemblies (the number of pins is equal to 127). The adjacent fuel assemblies are closer to the core center. The data are given for the center layer of the reactor core. The normalization is performed by average value for the three fuel assemblies.

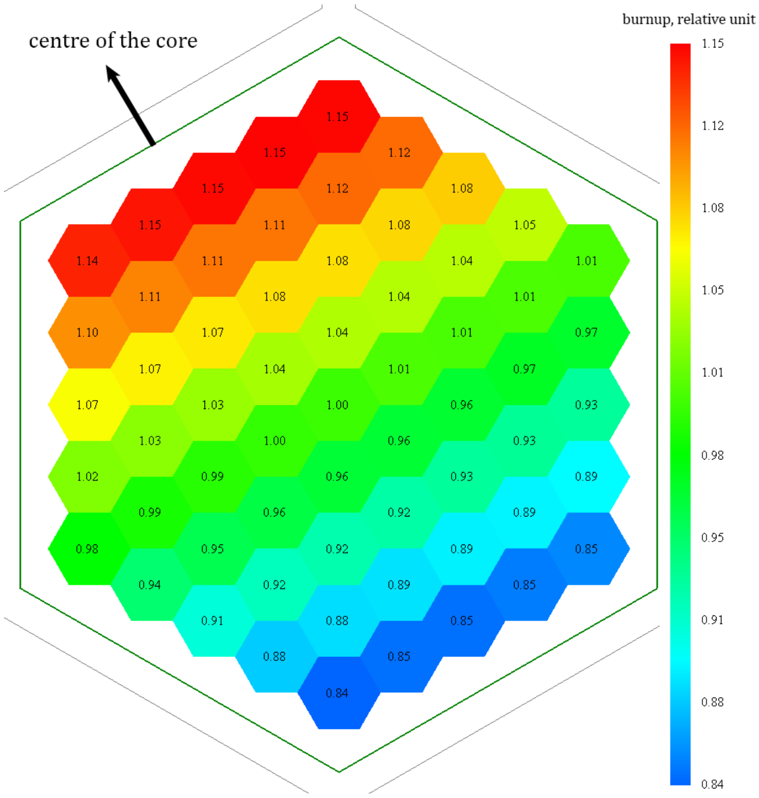

FIG. 9. The distribution of the relative burnup in the experimental fuel assemblies at the end of its campaign

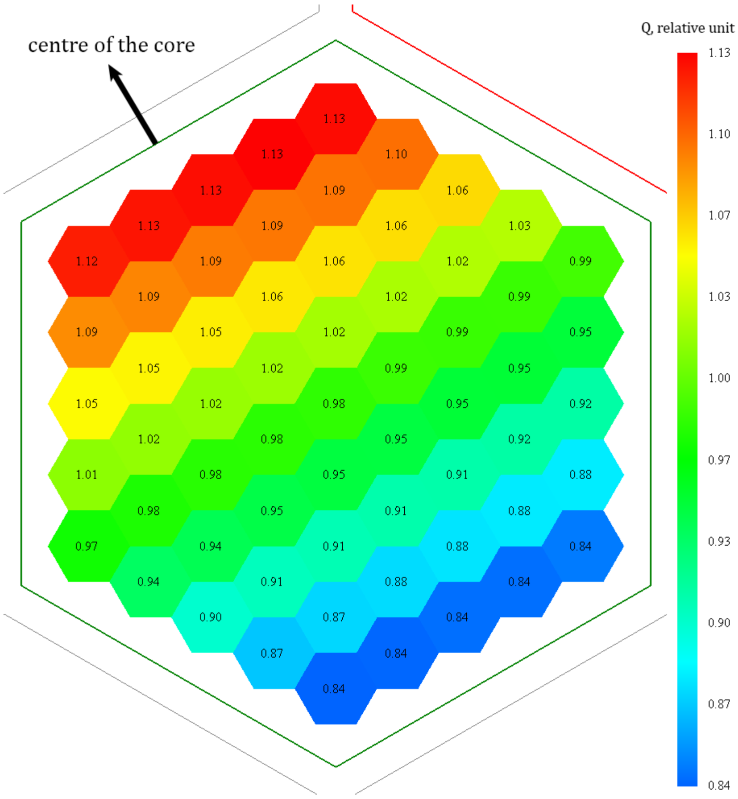

FIG. 10. The distribution of the relative power in the experimental fuel assembly at the end of its campaign

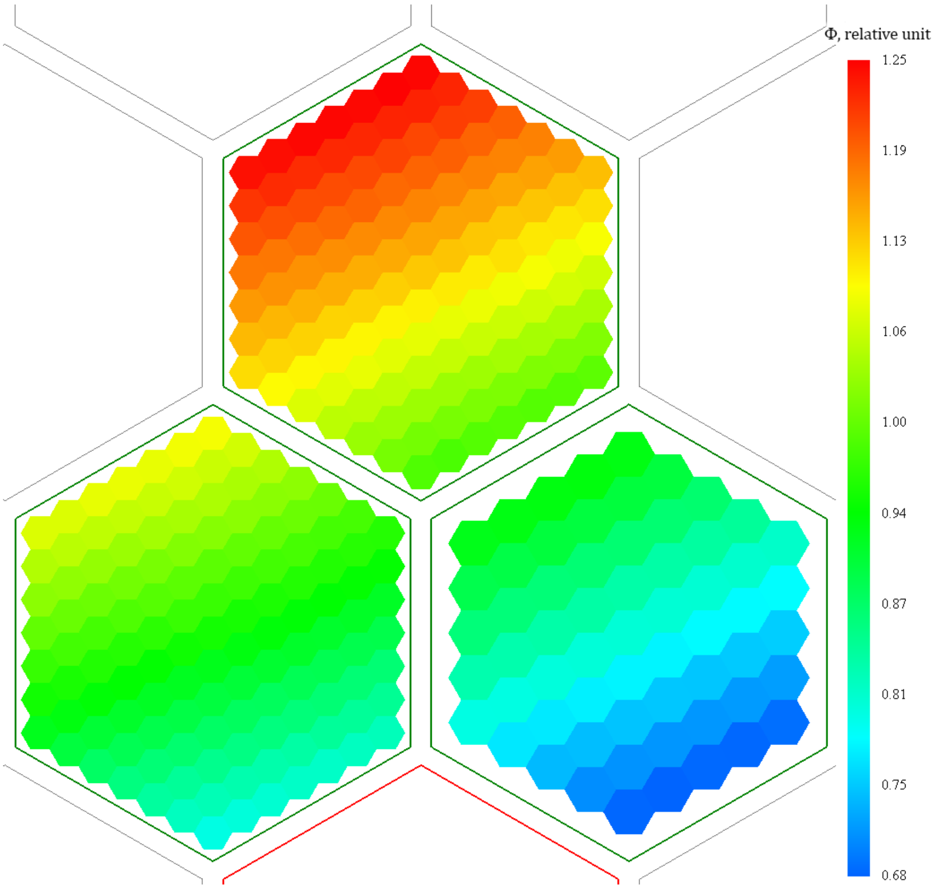

FIG. 11. The distribution of the single-group flux in the experimental and two adjacent fuel assemblies

4.3 Calculation of rod worth of the BREST-OD-300 reactor by the G7 code

The BREST-OD-300 reactor core model in the pin-cell regions was designed. Thus, the fuel and the absorber were separated in the fuel assembly with the control rod. The central hexagonal prism (number 7 on fig. 6) is selected as the absorber; the fuel is located in six trapezoidal prisms (numbers 1-6 on fig. 6).

It was carried out calculation of rod worth of the BREST-OD-300 reactor by the G7 code.

The results were compared with results of calculations for the one-point model in homogeneous approach (G1) and with the calculation in the heterogeneous approach by Monte Carlo (MCU-FR code). The results of comparison are presented in Table I.

TABLE I. COMPARISON OF THE ROD WORTH OF THE BREST-OD-300

Control Rod	G1/MCU-FR	G7/MCU-FR
shutdown	1.106	0.998
compensating	1.098	0.994
automatic regulating	1.112	1.023

The table of results shows the calculation by G7 in the diffusion approximation is in much better agreement with the results of calculation by the Monte Carlo method.

Conclusions

Three detailed models of the representation of fuel assemblies in the reactor core were developed. Algorithms and codes were designed for calculating of these models. The FUBUKI-1, 2 codes allow to perform the pin-by-pin calculation of the fast breeder reactor core and to keep a history of burnup individually for each pin. The G7 code allows more accurately calculating the assembly with a large width across flats and performing the correct calculation of the fuel assemblies containing the control rod (the fuel assembly of the BREST-OD-300 reactor).

It is noteworthy that the researcher have to pay by the calculation time for the detailed neutronic calculation. If we take the calculation time for the one-point model equal to one, the counting time by using the G7 code will be ten, and the time for the FUBUKI calculation will be five thousands.

References

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