BENCHMARK BEAVRS-PWR ANALYSIS USING WIMS AND PARCS CODES

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ABSTRACT

Based on the BEAVRS Benchmark (Benchmark for Evaluation And Validation of Reactor Simulations) which contains reference specifications for operational data and measurements for a pressurized water reactor (PWR), this article presents a model for obtaining macroscopic cross sections for the different fuels assemblies of BEAVRS using the WIMS. The microscopic cross section data library used was ENDF/B-VI. For the cross-section calculation method, the multicellular model approach was considered, and the results were compared with other reference works. This methodology aims to validate the model of calculation of the cross sections, and thus, evaluate whether such methodology generates valid results. Also, in this work, the macroscopic cross-sections generated by WIMS were used to simulate the reactor core of a typical PWR in PARCS. The results show a small difference, in the case of the k-eff's values, the maximum standard deviation is 0.00466. All calculations were made for the reactor's Hot Zero Power status.

1. INTRODUCTION

Simulations of the reactor core are essential for the economical and safe operation of a nuclear plant. Currently, several different simulation tools are available and nuclear engineers perform calculations to predict the state of the reactor's core. This work is an example of computational tools applied to perform pressurized water reactor (PWR) simulations. Currently, the complete analysis of the core of a nuclear reactor is still based on the traditional two-step calculation scheme, which has been the standard approach for reactor analysis. These steps consist of: (1) spatial homogenization and condensation of energy groups using cell code, and (2) calculation of the complete core in 3D using few group constants, generated in the previous step. Samchez's work [1] provides a good review of modern homogenization techniques.

The main goal of this work was to calculate the cross sections, to carry out tests and evaluations of the methodology in two stages WIMS-PARCS, with a benchmark problem, based on real data. In addition, the development and verification of the mathematical model of the nuclear reactor PWR was also a goal.

In this work, the benchmark known as BEAVRS was used, which comes from Evaluation And Validation of Reactor Simulations. BEAVRS is a full-scale PWR-based benchmark, containing 193 fuel assemblies (FA) arranged in a 15×15 matrix. These FAs consist of 3 different types of enrichment, that is, 1.6, 2.4 and 3.1 weight percent ^{235}U. These 3

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24 a 26 de maio de 2021
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different enrichment-based FAs may contain different arrangements of burnable absorbers (BAs). Each FA is a 17 × 17 array of fuel pins. A total of 8 grid spacers (6 intermediate, one above and one below) are present in each FA (see Fig. 1).
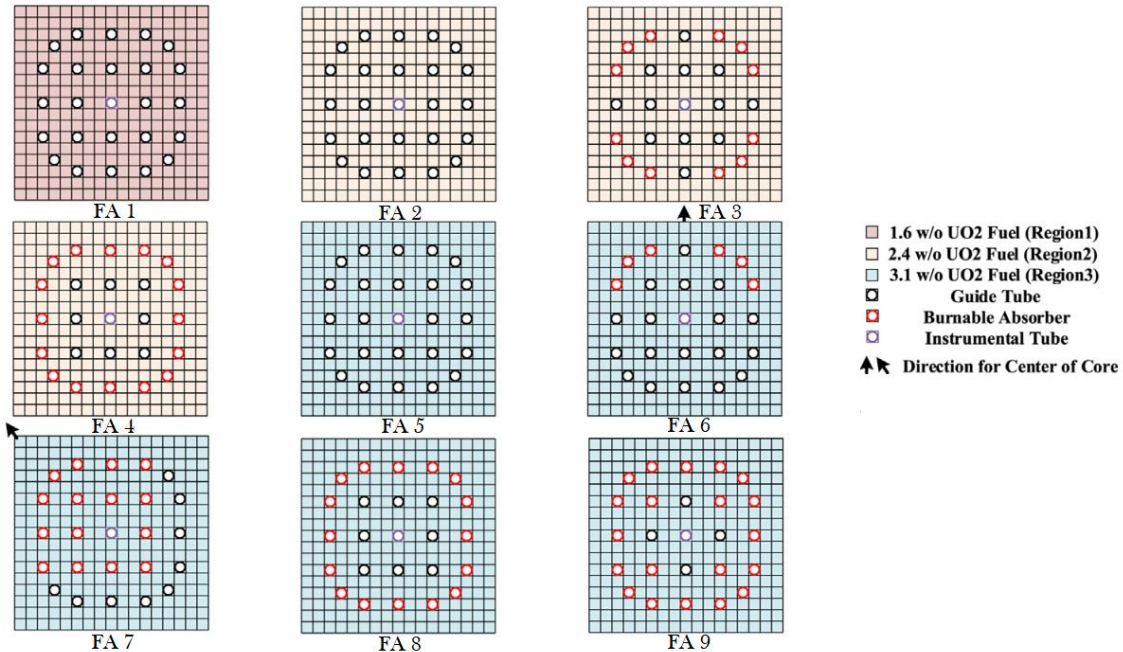

Fig. 1. Specification of FA for BEAVRS cycles 1, adapted from [2].

BEAVRS was published in 2013 by the Computational Reactor Physics Group of the Massachusetts Institute of Technology (MIT), and has been updated several times. Its purpose is to allow comparison of various computer codes of reactor physics applied to full-core calculations with real data. The reference specification contains a detailed description of an unknown (real) nuclear power plant for a 4-circuit Westinghouse reactor located in the USA. The documentation contains details of the power measurements and operating conditions of the reactor for the first and second fuel cycles. More details on BEAVRS can be found in the original reference document [3]. Several research reports on BEAVRS solutions have been published for a wide variety of simulation methodologies [4-8]. The model developed in this work was compared with benchmark data available for the first fuel cycle presented in the publications [3].

The methodology consists of two stages. In the first one, the macroscopic cross sections of the fuel elements will be calculated using the WIMS-ANL code. In the second one, the cross sections obtained will be used to simulate the entire reactor with the PARCS code; these simulations are made for the reactor's Hot Zero Power (HZP) state.

2. METHODOLOGY

2.1 Obtaining macroscopic cross sections

The WIMS code family allows the calculation of neutron flux distributions, k-inf and k-eff values in a variety of reactor. The main objective of the code is to calculate

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24 a 26 de maio de 2021
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homogenized diffusion cross sections for representative areas of the reactor core, which are later used in calculating the power of the core. The WIMS-ANL version has been revised and improved error checking and handling of output. The code can accommodate cross section libraries with more than 69 energy groups. The ENDF/B-VI cross section library with 69 and 172 energy group structures is used with the code. The code is compatible with FORTRAN 77 [9].

The WIMS code can use three geometry models, which are:

- **Pin Cell** representing an infinite network of identical cells,
- **Cluster** that can represent a set of fuel, or a macrocell with annular regions,
- **MULTICELL** is a set of cells or groups connected by probabilities that a neutron leaves one of the cell types (or groups) and enters another type of cells (or groups).

In this work, the MULTICELL method will be used. The MULTICELL approach is only an alternative but offers an additional possibility that any of the cells in Fig. 2a can also be a group of pin cells by themselves. Any of these cells, which for some reasons are considered different from each other, can be calculated independently. The calculation is performed separately for each cell type, but the usual boundary condition is replaced by "intracellular boundaries" expressed in terms of the probabilities that a neutron will leave one cell type and enter another. This is expressed schematically in Fig. 2b, where the cells were separated for the calculations and their interconnections must be provided from outside. Fig. 2c illustrates that cell types can differ not only in composition, but also in dimensions [10].
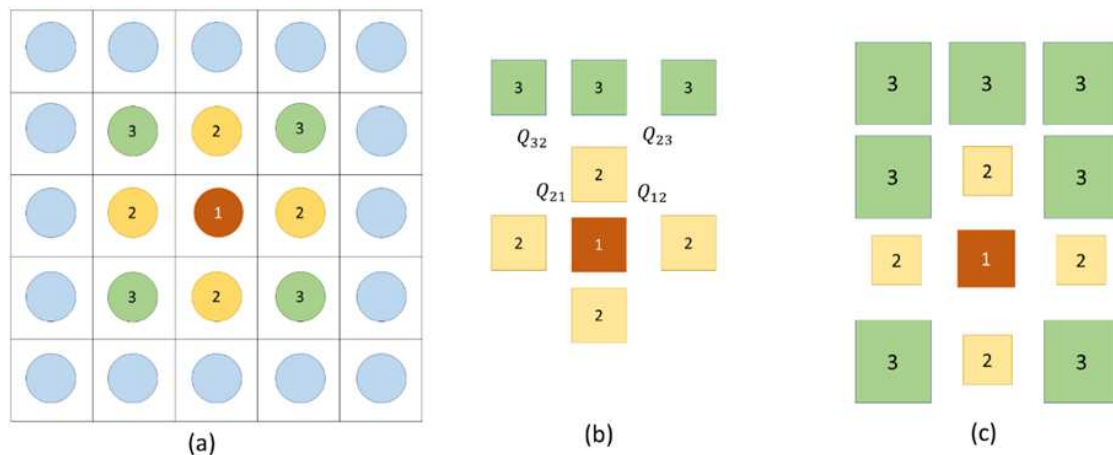

Fig. 2. Schematic interpretation of the MULTICELL approach (a) - Macrocell is calculated chosen cell types (b) - The calculation of the spectrum of 69 groups is carried out separately for each type, (c) - The cells may differ in composition and geometry.

Two basic probabilities are presented:

$P_{i\mathbf{A},j\mathbf{B}}$: probability that a neutron born in region i of cell A will have its first collision in region j of cell B, and

Q_{AB}: probability that a neutron leaving the boundary of cell A will have a collision in cell B.

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24 a 26 de maio de 2021
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For these probabilities, it can be write:

$$P_{i\mathbf{A},j\mathbf{B}} = \delta_{\mathbf{AB}}p_{ij} + p_{ib}^{\mathbf{A}} \cdot Q_{\mathbf{AB}} \cdot \frac{p_{bj}^{\mathbf{B}}}{(1 - p_{bb}^{\mathbf{B}})} \tag{1}$$

where
p_{ij}: probability that a neutron born in region i will have its first collision in region j.

$p_{ib}^{\mathbf{A}}$: probability that a neutron born in region i of cell A will reach the limit of the cell without collision,

$$p_{bb}^{\mathbf{B}} = 1 - \sum_{j=1}^{J} p_{bj}$$

$p_{bb}^{\mathbf{B}}$: probability that a neutron will appear at the cell boundary to have its first collision in region j.

To determine Q_{AB}, one can assume that $Q_{AB} = S_{AB}$ is the probability that a neutron leaves the surface of cell A, and enters the surface of cell B. So S_{AB} is purely geometric, that is, its elements are fractions of surfaces A and B, that is, it is a fraction of cell surface A contiguous to B.

In the WIMS output file, one can find the cross sections for each fuel assembly, in which one can simulate the complete core. These results do not include the boundary conditions, that is, the core reflector is not considered. To have a complete model, it is necessary to obtain the cross sections of the reflector. Due to the need to include the reflector, it was adopted the methodology suggested by the user manual WIMS-ANL [9], which consists of homogenizing the fuel elements neighboring the reflector (See Fig. 3). This homogenization allows the WIMS to use the spectrum of neutrons from the fuel assembly to obtain only the homogenized macroscopic cross sections of the reflector.
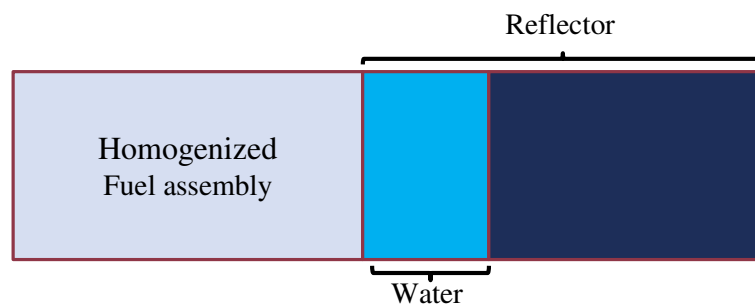

Fig. 3: Schematic model to obtain the macroscopic cross sections of the reflector.

The homogenization logic is similar for all fuel assemblies neighboring the reflector. This logic is shown in Fig. 4. In this figure it shows the homogenization of the FA 9. It can see that, for each region of different color (Fig. 4a) corresponds to a subdivision of different materials (Fig. 4b).

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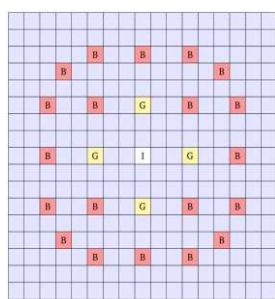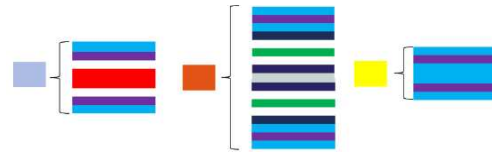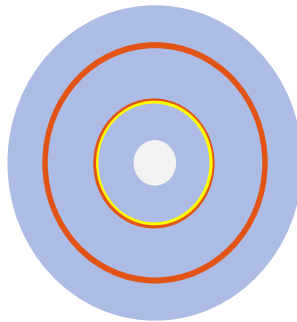*24 a 26 de maio de 2021*
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Fig. 4a
Fig. 4b

Fig.4: a) shows the homogenization of FA 9, b) shows the subdivisions of the homogenization regions of Fig. 4a.

2.2 Reactor core model with PARCS

The model of the reactor core with PARCS, will have the same radial nodalization as the standard load of the reactor core (BEAVRS core), a node for a FA and a total of 257 nodes. The axial nodalization consists of 19 axial levels of active core of equal length and two reflecting nodes, upper and lower (See Fig. 5b).

Another model has been added for the axial and radial reflector. The suggestions made in the WIMS user manual for simulating reflectors were used [9]. In order to simulate the radial reflector, the FA in gray (see Fig. 5a), was applied as an additional square region, having the same size as the other FA (Fig. 5a). And it is filled with 2 regions: water and steel (Stainless Steel). In addition, for the sake of simplicity and to reduce computational effort, the spacer grid models were not considered.
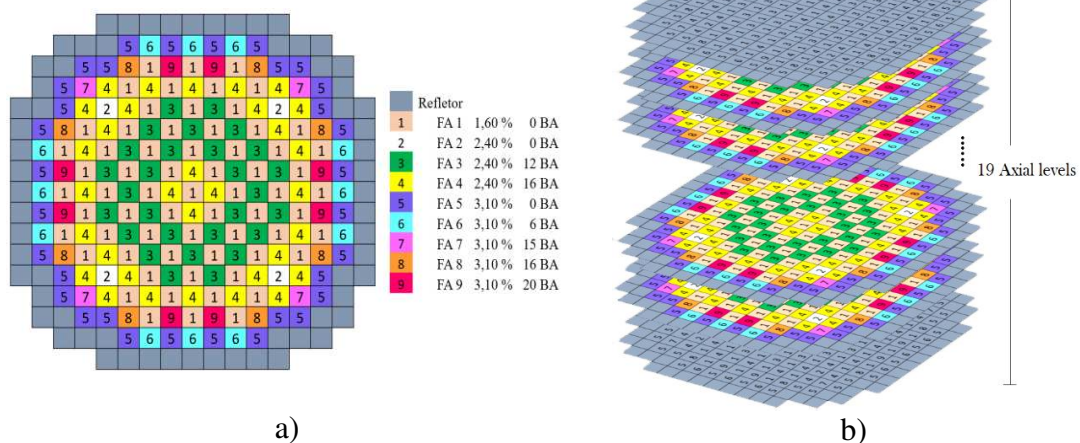

a)
b)

Fig. 5: a) Radial profile modeled on PARCS of BEAVR, b) Axial profile of the core modeled on PARCS.

3. RESULTS

3.1 Macroscopic cross sections

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24 a 26 de maio de 2021
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The first step consisted of obtaining the macroscopic cross sections, for to later use them in the simulation of the complete reactor with the PARCS code. Due to the large amount of data from the cross sections, in this section it will present only the k-eff values, for showing that the values are consistent, some of which have been compared with other studies. In this first stage, the configuration of cycle 1 [3] was simulated, which consists of FAs with different enrichments and different numbers of BAs in each FA, as shown in Fig. 5b. All these values are shown in Tab. 1.

Tab.1. k-eff results for the different FA, obtained with the codes WIMS-ANL, nTRANCER and McCARD [11].

FA	w/o U235	BA	WIMS-ANL	nTRACER	McCARD	Dev.Std
FA 1	1.6	0	0.998173	0.99340	0.99483	0.0024496
FA 2	2.4	0	1.138959	1.13696	1.13751	0.0010326
FA 3	2.4	12	1.013309	1.01318	1.01432	0.0006243
FA 4	2.4	16	0.979513	0.97454	0.97597	0.0025602
FA 5	3.1	0	1.219676	1.21971	1.21971	1.963E-05
FA 6	3.1	6	1.154867	1.16273	1.16313	0.0046595
FA 7	3.1	15	1.072441	1.07835	1.07895	0.0035973
FA 8	3.1	16	1.066354	1.06340	1.06416	0.0015339
FA 9	3.1	20	1.028134	1.02769	1.02886	0.0005906

It can be observed that the k-eff's obtained by WIMS-ANL, are very similar compared to the results obtained by both nTRANCER and McCARD codes [11], this is reflected in the values of the standard deviations. Despite of using different libraries of microscopic cross sections, the results are quite similar; in the case of the work by Ryu et al. the library used is ENDF-B/VII and in this work ENDF/B-VI was used.

3.1 Core model with PARCS

The PARCS core model has the same radial nodalization as the reactor core loading pattern (Fig. 5a), where each node represents a FA, thus totaling 257 nodes. Axial nodalization consists of 19 axial levels of active nuclei of equal length and two reflecting nodes, upper and lower (Fig. 5b). Fig. 6 shows the axial distribution of the detector measurements (orange circles) provided with the BEAVRS specification (Hot Zero Power condition), it also shows the values that were calculated with PARCS (blue circle). It can be seen that these points are quite like each other.

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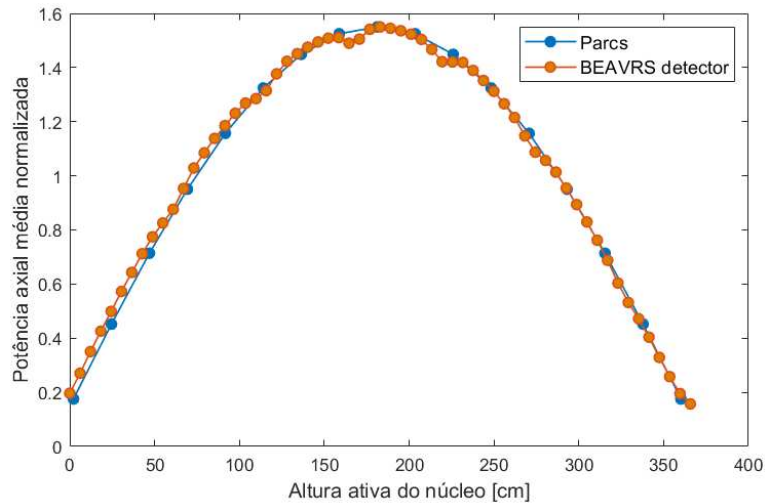*24 a 26 de maio de 2021*
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Fig. 6. Distribution of axial power of BEAVRS (orange dots). Axial power distribution calculated by PARCS (blue dots).

In order to show the behavior of power in a radial plane, Fig. 7a shows the radial power distribution of plane 10. Fig. 7b shows the radial power distribution calculated with CASMO4E by Zhuo Li [5]. It can see that the two figures have a similar behavior.
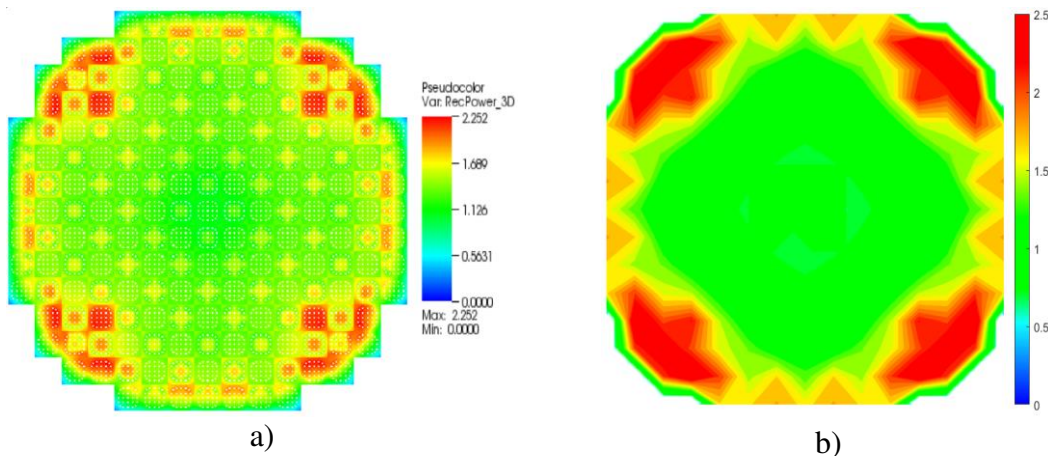

a)   b)

Fig. 7. a) Radial power distribution obtained with CASMO4E b) Radial power distribution obtained with PARCS.

4. CONCLUSION

The data specified in the BEAVRS were used to simulate the reactor core, which consisted of two stages. In the first one, it consisted in calculating macroscopic cross sections, these data were obtained by the WIMS-ANL code. With the model used in WIMS-ANL, the k-eff of each fuel assemblies were calculated; the maximum standard deviation of the results was 0.00466. In the second one, it is used the cross sections in the PARCS input file, in order to obtain the distribution of axial, radial and k-eff power. The results obtained can be considered as expected. One can conclude that the model presents values that can be considered valid.

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24 a 26 de maio de 2021
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