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PARAMETRIC EQUATIONS FOR CALCULATION OF MASCROSCOPIC CROSS SECTIONS

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ABSTRACT

Neutronic calculations of the core of a nuclear reactor is one thing necessary and important for the design and management of a nuclear reactor in order to prevent accidents and control the reactor efficiently as possible. To perform these calculations a library of nuclear data, including cross sections is required. Currently, to obtain a cross section computer codes are used, which require a large amount of processing time and computer memory. This paper proposes the calculation of macroscopic cross section through the development of parametric equations. The paper illustrates the proposal for the case of macroscopic cross sections of absorption ($\sum a$), which was chosen due to its greater complexity among other cross sections.
Parametric equations created enable, quick and dynamic way, the determination of absorption cross sections, enabling the use of them in calculations of reactors. The results show efficient when compared with the absorption cross sections obtained by the ALPHA 8.8.1 code. The differences between the cross sections are less than 2% for group 2 and less than 0.60% for group 1.

1. INTRODUCTION

To make de analysis of a nuclear reactor is necessary a neutron modeling of the core. This modeling is based in the theories of transport and diffusion of neutrons [1]. Therefore, is necessary the obtainment of variables presents in those theories like macroscopic cross sections of fission, absorption, scattering, removal and diffusion coefficient.
The nuclear data necessary for the application of the theories of diffusion and transport of neutrons depend on several factors ranging from material composition to state variables, such as fuel temperature , moderator temperature , boron concentration (BC) and burn up (B) [2].
By doing the analysis of the core of a nuclear reactor, one of the most important aspects is the determination of nuclear data , such as cross sections and diffusion coefficient . To determine the nuclear data is required a wide variety of input into computer codes . The codes for nuclear data calculation usually require large amount of processing time and computer memory.
Access to nuclear data values is considered of great importance in the Reactor Physics Area. Therefore, it is important to create alternative methods for calculating such nuclear data , thus having different options to obtain them.
This paper presents an alternative method for generating a library of macroscopic cross sections . In this method , parametric equations (PE) are obtained for describes how the cross

sections vary for different input data. The method of obtaining the PE will be presented in this study and will be exemplified for the case of macroscopic absorption cross section (Σa).
Various analyzes and studies done previously [3,4] using parametric equations to describe the behavior of the cross sections presented different forms of representation. In this paper will be done a study where the most suitable parameters for description of cross sections will be chosen. Furthermore , this work will develop a methodology to the use of polynomial curves to approximate the actual curve of Σa .
This paper aims at the development of parametric equations for calculation of cross sections in order to get the cross sections only by a library of parametric equations . With this, we obtains an alternative manner, fast and dynamic, to calculate the cross sections , facilitating Reactor Physics tasks.

2. THE PARAMETERS OF THE EQUATIONS

To obtain cross sections in a given reactor is necessary a lot of input data into computer codes, from the burn up to the moderator temperature. The PE developed in this paper contain only three of all possible parameters. The chosen parameters were the ones that change more often in a same nuclear reactor.
It was found that the parameters burn up (B), boron concentration (BC) and enrichment (E) are the physical characteristics that changed more often [3,4]. This choice was made based on tests and information from experts in the area during an internship in Eletrobras Eletronuclear company. Furthermore, in a same nuclear reactor, other parameters relating to the core geometry, such as shape of the fuel element, cladding material and flow pump, can be neglect the variation.
These parameters are specific for each nuclear reactor type, so in order to determine the macroscopic cross sections, a reactor type had to be chosen. In this paper was utilized a pressurized water reactor type as the one in Angra I, so the PE obtained in this paper will be valid just for reactors of its kind.

3. METHODOLOGY OF OBTAINING THE PARAMETRIC EQUATIONS

To obtain the PE, a source of nuclear data is required. To this end, a code which can obtain a particular cross section selected for certain parameters is necessary. In the simulation it was varied the three parameters chosen and the others were kept fixed. From these cross sections and parameters values are obtained curves, and the parametric equations were elaborated to describe these curves.
It was chosen the APA - Alpha system / Phoenix / ANC as the source of nuclear data. The APA is a Westinghouse Electric Corporation system developed for various calculations of Reactor Physics [5]. The use of this system was possible during the internship in Eletrobras Eletronuclear company.
The APA performs various calculations of Reactor Physics, and for the calculation of nuclear data, was used the ALPHA 8.8.1 code. This code works with many energy groups, but collapses the cross sections for two groups only.
The PE may be obtained for any nuclear data provided by the code. In this paper was chosen as an example the determination of PE to the macroscopic cross section of absorption, for the second group of energy ($\sum_{a2}$), witch is the group of thermal neutrons. This choice was made

based on the fact that for the type of reactor chosen, the $\sum_{a2}$ is of great importance because the fission will occur in the thermal range of energy.
After getting the data from the ALPHA 8.8.1 code, the points were plotted on a graph in Microsoft Excel 2007 and approximated by polynomial curves through the least squares method.
These polynomial equations are combined to arrive at a final parametric equation that includes the three parameters chosen.

4. THE PE RANGE OF VALIDATION

To obtain the PE is necessary an appropriate choice of values of the selected parameters, because it is only valid in the interval between the maximum and minimum of these values. These intervals of values were chosen based on operation range of Angra I nuclear power plant.
It will be presented below the range of values where the equations are valid for each of the three parameters, as well as all values used as input to generate the cross sections.
For the boron concentration was chosen the interval from 0 ppm to 2500 ppm, and the input values were: 0, 500, 1000, 1500, 2000 and 2500 ppm.
For the enrichment was chosen the interval from 2% to 5%, and the input values used were: 2, 3, 4 and 5%.
For the burn up was chosen a range 0 MWd/Kg to 80 MWd/Kg, and the input values used were: 0, 0.15, 1, 2, 3, 4, 5, 6, 7, 8, 9, 10, 11, 12, …, 78, 79, 80 MWd/Kg. The value 0.15 MWd/Kg was chosen because it is very important regarding the balance of two highly absorbing fission products, Xenon and Samarium, and it can influence a lot on the results of the PE.
As the interval of burn up is very large and were chosen many values within this interval it was decided to make a PE for each 5 MWd/Kg interval to minimize the error, so sixteen PEs were created to describe $\sum_{a2}$.

5. RESULTS AND TESTS

The method described in Chapter 3 was applied and the PEs were generated by combining the approximate polynomial equations.
All sixteen PEs obtained, one for each interval, have the same format, which is:

$$\sum_{a2}(B,BC,E) = \left[\left(f_1E^2 + f_2E + f_3\right)B + \left(f_4E^2 + f_5E + f_6\right)\right]BC + \left[\left(f_7E^3 + f_8E^2 + f_9E + f_{10}\right)B^2 + \left(f_{11}E^3 + f_{12}E^2 + f_{13}E + f_{14}\right)B + \left(f_{15}E^2 + f_{16}E + f_{17}\right)\right] \quad (1)$$

Where;
$\sum_{a2}$ - macroscopic cross section of absorption for group two of energy,
E - enrichment,
B - burn up,
BC - boron concentration,

f_x - constants coefficient of a particular equation.

An example that illustrates the process to obtain $\sum_{a2}$ is shown below, for the following values of the coefficients 'f_x', for PE in range 35 MWd/Kg to 40 MWd/Kg:

Table 1: f_x values

Coefficient:	Range 35-40 (MWd/Kg):
f1	-1.01E-11
f2	7.87E-11
f3	-3.46E-11
f4	3.48E-07
f5	-3.68E-06
f6	1.83E-05
f7	5.03E-13
f8	5.66E-12
f9	1.78E-11
f10	-1.57E-11
f11	-4.22E-08
f12	5.26E-07
f13	-2.01E-06
f14	2.19E-06
f15	-0.00178284
f16	0.02849304
f17	0.03461868

In this range, a Σ_{a2} value was obtained using its PE and then it was compared with the value resulted by the ALPHA 8.8.1 code simulation. The chosen example was calculated by using the following values of burn up, boron concentration and enrichment, respectively: 36,5 MWd/Kg, 850 ppm and 2,5%. The value of Σ_{a2} found by this PE was 0.101347013 cm^{-1} and the one by the code was 0.102135 cm^{-1}, which resulted in a small relative error of 0.77%.
The same procedure was repeated for several burn up values in different ranges in intention to test the validity of the PEs. The Figure 1 shows the relative error between the cross sections obtained by PEs and the ones using the ALPHA 8.8.1 code.

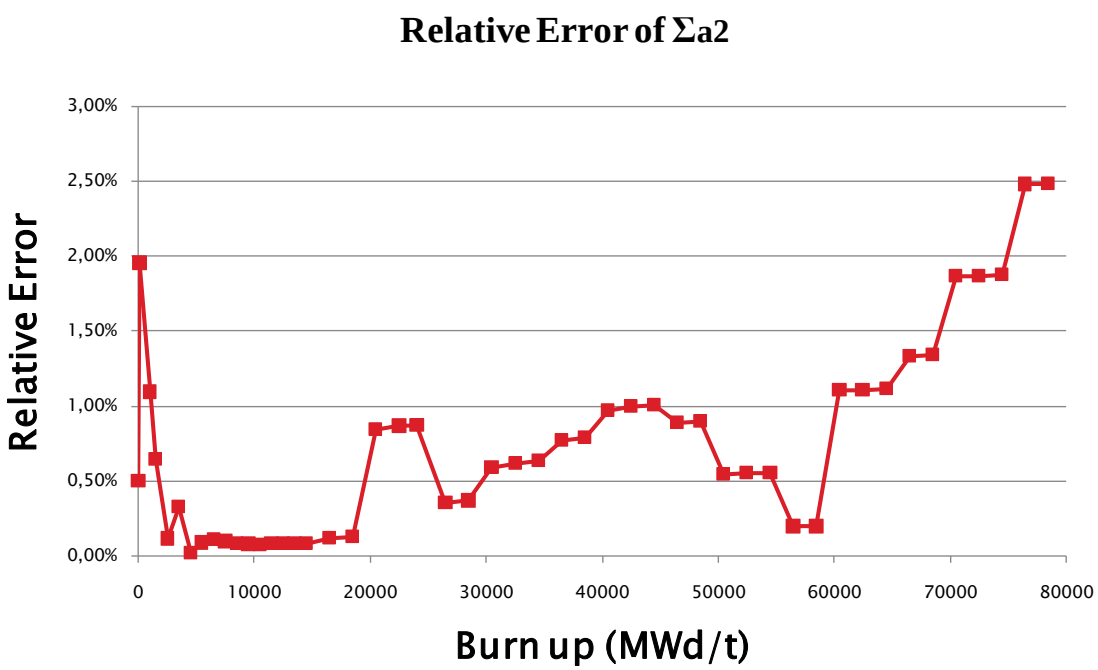

Figure 1: Relative error of Σ_{a2} in function of Burn up

The parameters used to generate these Σ_{a2} values were different from those used to create the PEs, ensuring a proper validation test. The enrichment was 2.5%, boron concentration was 850 ppm and burn up as seen in the Figure 1.
There were error peaks at around 150MWd/t and also for high burn up values. The higher error found was 2.48% at 79500MWd /t. There is a higher amount of errors at the 150 MWd/t area because of the Xenon and Samarium balance, which are not fully represented by this PE.

6. CONCLUSION

In this paper was developed a method of obtaining parametric equations for calculating nuclear data, and it was exemplified for the macroscopic cross section of absorption for the second group of energy. This method basically consists on the generation of nuclear data trough the Alpha 8.8.1 code, and from these data parametric curves were approximated.
The PEs obtained on this paper provides an alternative method, quicker and more dynamic, for cross sections calculation, since it only uses polynomial equations for the approximations. The tests performed in this paper shows good approximations of the cross sections calculated by the PEs in comparison with the ones of the Alpha 8.8.1 code, where the maximum error found was 2.48%. Therefore we can conclude that the PEs created can be used to generate cross sections with good approximations.

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