Monte Carlo Burnup Analysis of the Nuclear Research Reactor IEAR-1 Using MCNP-4B Code

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1. Introduction

Currently, several codes, or combinations of codes have been developed [1,2,3] to perform Monte Carlo depletion analysis. Comparison with other methods based on the homogenization of the fuel elements assesses the importance of a more realistic treatment of the geometry to increase the accuracy.

In this work a new approach for the nuclear reactor parameters calculation and burn up analysis is proposed using the Monte Carlo MCNP-4B code [4]. The present methodology permits a detailed consideration of reactor core geometry and explicit inclusion of the fission products cross sections, serving as basis to perform calculations of others quantities such as neutron flux distributions in the reactor core in each burn up time step.

The general-purpose Monte Carlo MCNP-4B code has been previously utilized at IPEN-CNEN/SP for the simulation of the fuel loading experiment in the IPEN/MB-01 reactor [5], and also, for the determination of the isothermal reactivity coefficient [6]. This code has been used in the present work for the simulation of the neutron transport and reactor parameters calculation in the IEA-R1 reactor.

2 Description of the 5 MW IEA-R1 Reactor

The IEA-R1 pool type reactor is designated to the medicine applications radioisotopes production, material testing, sample irradiations, activation analysis and also perform an important tasks in research in the areas of radiobiology, radiochemistry and physics. The reactor core is constituted by a total of 24 MTR type fuel elements surrounded by reflector graphite elements. Two types of fuel elements are used in the core divided in 20 standard fuel elements and 4 fuel elements containing control absorbers plates, as shown in Figure 1. The standard fuel elements (FE) are constituted by 18 fuel plates and the control elements (CE) are constituted by 12 fuel plates. In both fuel element type the enrichment is 20 % in weight.

Three different densities are used, respectively 1.9 and 2.3 g/cm^3 composed by U_3O_8-Al and 3.0 g/cm^3 composed by U_3O_8-Al.

The dimensions of the FE and CE are shown in Figures 2 and 3. The fuel plate dimensions are 62.5 cm of total height (60 cm of active height) and 6.26 of total width and 0.076 cm in thickness. Control plates are constituted by Ag-In-Cd (80-15-5 %) and positioned inside the control guides located in the control elements (CE). The central element (see Figure 1) is composed by beryllium and designed to the ^{99}Mo production.

3 Methodology

Up to now, calculation of criticality of the nuclear reactor IEAR-1 has been performed by the system of codes LEOPARD-4 - HAMMER-TECHNION – CITATION [7,8]. The CITATION code uses diffusion theory to obtain the fluxes required for reactor burn up analysis. LEOPARD-4 is used for the two group fuel cross section generations and HAMMER-TECHNION is used for the non-fuel regions including the control rods. Nevertheless, this code has serious limitations regarding the consideration of the geometry of the system. Calculations made by this code show an increase in the K_{eff} value in about 2000 pcm with respect to the experiment.

The purpose of this work is the optimization of the method of analysis of criticality of the nuclear research reactor IEAR-1 using a more realistic model of the geometry of the reactor core, and pointwise cross sections which would avoid several limitations typically found in cell codes.

3.1 Monte Carlo MCNP-4B code

The MCNP-4B is a Monte Carlo code, which has a great flexibility in modeling the geometry of the reactor core. It can be used for neutron, photon and electron transport and supports a wide variety of scoring options and modeling features such as the one called Repeated Structure which let us to describe explicitly each of the fuel element components. On the basis of this feature we can model the reactor core based on the repetition of single MCNP cells.

In this sense, each fuel element, composed by 18 slabs containing the meat covered by aluminum and surrounded by water, is divided in rectangular cells. Each cell can be filled with a universe, which represent a lattice. Each universe has an identification number so that every cell belonging to the universe is associated with this number, which in turn is the identification number of the element with a given value of burn up.

The key feature of this approach is that it permits essentially “exact” modeling of all geometrical detail, without resort to energy and spatial homogenization of neutron cross sections.

3.2 Microscopic Cross Sections and Fission Products Concentration Generation

Fission products (FP) poisoning takes an important aspect in reactor burn up analysis and the accurate prediction of neutron absorption by them is extremely important for the fuel cycle strategy. A methodology was proposed by Abe and Santos based on a coupled HAMMER-TECHNION/CINDER-2 system called HAMCIND [9]. In this method, the basic fission product library, as ENDF/B-V is processed by the NJOY [10] and its output is accessed by the HAMCIND which performs the cell burn up calculation for the determination of the concentration of the fission yields at different burn up time steps.

Microscopic cross sections are obtained by NJOY which reconstructs, broadens and formats the cross section data into an appropriate form for the MCNP-4B code. The thermal scattering law $S(\alpha,\beta)$ was obtained from LEAPR module considering bound in water at 293 K.

3.3 Calculational Scheme

Fission products concentrations obtained by the HAMCIND cell burn up calculation are utilized in a linear interpolation scheme according to the burn up values of each fuel element for a specific reactor core configuration, so that, a set of interpolated fission product concentrations for each fuel element is obtained according to the burn up value of the respective FE. For different core configuration a new set of concentration values are generated by interpolation and written in an appropriate MCNP-4B input format. Microscopic cross sections are also provided to the MCNP-4B as described in the previous section.

4 Results

Four reactor core configurations were considered in the present analysis namely: configurations 205, 206, 207 and 210. Each one is constituted by different fuel element burn up values. Table 1 shows the burn up value of each FE according to its location and configuration in the reactor core. The first row values belong to the first configuration, e.g., configuration 205. The second row values belong to the configuration 206 and so on.

Table 1. Fuel Element Burn up values for each reactor core configuration

Configuration 205

23.55	17.23	6.33	17.03	15.25
6.35	4.38	4.55	24.64	7.68
5.33	11.54	3.93	8.54	28.74
6.65	24.65	4.69	4.37	24.81
15.38	17.14	6.49	24.44	13.35

Configuration 206

25.11	19.65	8.82	19.39	17.00
8.85	7.78	7.89	27.61	10.20
0.00	10.48	10.98	15.14	2.630
11.60	30.72	11.29	10.91	18.16
19.06	21.82	11.35	25.80	16.96

Configuration 207

26.77	21.91	11.18	21.66	18.71
11.19	10.94	11.05	30.47	12.65
0.00	10.48	10.98	10.91	18.16
11.60	30.72	11.29	10.91	18.16
19.06	21.82	11.35	25.80	16.96

Configuration 210

19.77	0.00	12.90	0.00	19.97
12.91	13.26	13.35	0.00	0.00
2.00	12.97	------	13.48	4.54
13.38	0.00	13.64	13.25	17.49
14.45	0.00	13.10	0.00	18.28

Basically, two parameters have been calculated with the present methodology using the MCNP-4B code and compared to those obtained by the standard methodology using the CITATION code: a) the multiplication factor, K_{eff1}, at the core criticalization and b) the multiplication factor, K_{eff2}, without the control absorbers plates from which the excess of reactivity, R, was obtained by $R = (K_{eff2} - 1.0)/ K_{eff2}$. The results were compared to measured values available. Table 2 shows the results using the different methods and the excess of reactivity are compared to experimental values.

Table 2. Multiplication factor and excess of reactivity for several core configurations and conditions.

	Configuration 205		
K_{eff1}	1.00290±0.00060	1.02652	1.00
K_{eff2}	1.04879±0.00033	1.07523	-----
Excess of Reactivity (pcm)	4652	6997	4414
	Configuration 206		
K_{eff1}	1.00490±0.00033	1.02621	1.00
K_{eff2}	1.04556±0.00033	1.07102	-----
Excess of Reactivity (pcm)	4357	6631	4363
	Configuration 207		
K_{eff1}	1.00393±0.00033	1.02503	1.00
K_{eff2}	1.04348±0.00033	1.06954	-----
Excess of Reactivity (pcm)	4167	6502	4053
	Configuration 210		
K_{eff1}	1.01084±0.00035	1.02703	1.00
K_{eff2}	1.06736±0.00034	1.09151	-----
Excess of Reactivity (pcm)	6311	8384	6088

Preliminary MCNP-4B values were obtained by the simulation of 50 millions source neutrons histories for each case analyzed. The maximum standard deviation for the multiplication factor values was 0.06 %. For the experimental core critical situation, e. g., $K_{eff}=1$, results obtained by the simulation using the present methodology are, in general, in good agreement, with a criticality over-prediction of 290 pcm, 490 pcm and 393 pcm, respectively, for the core configurations 205, 206, 207. However, a significant discrepancy of about 1000 pcm has been found in the configuration 210, which can be attributed to the presence of the beryllium irradiator element in the center of the core which does not exist in others configurations. The same results provided by the standard methodology give significant discrepancies of about 2500-2700 pcm. The main reason for those discrepancies, however, comes from the cross sections generation processing which is under verification and revision at moment.

5 Conclusions

In this work, an approach for burnup analysis for criticality calculation and excess of reactivity of the nuclear research reactor IEAR-1 was developed through the use of a more realistic description of the nuclear core and fuel elements which permit the consideration of the effect of exact geometry in the problem. The agreement with the experimental values obtained by this method and its difference with the previous calculation based on the homogenization of the fuel elements shows the importance of a non-homogenized treatment of this problem.

The approach developed in this work and the geometrical model designed for the IEAR-1 reactor serves as basis for future analysis of other quantities such as neutron flux distributions in any particular region of the nuclear core.

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