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Reactor Physics Calculations for a Sub critical Core of the IPEN-MB-01 driven by an external neutron source

A. Antunes[1], T. Carluccio[1], A.G. Mendonça[2], A. dos Santos[1], J.R Maiorino[1]

1 Instituto de Pesquisas Energéticas e Nucleares (IPEN / CNEN - SP)
Av. Professor Lineu Prestes 2242
05508-000 São Paulo, SP
aantunes@ipen.br
thiagoc@ipen.br

2 Centro Tecnológico da Marinha em São Paulo – CTMSP
Divisão de Física de Reatores
Av. Prof. Lineu Prestes 2240 – Cid. Universitária
05508-000 São Paulo, SP

ABSTRACT

The IPEN-MB-01 is a Zero Power Reactor (100 W) used for Benchmark of Reactor Physics Parameters. Recently, a possibility to introduce a compact pulsed neutron generator, developed by Lawrence Berkeley National Laboratory in a sub critical core of the IPEN-MB-01 is on discussion. Moreover in the framework of the IAEA Coordinated Research Project (CRP) on Low Enriched Uranium (LEU) utilization in Accelerator Driven System (ADS) included the feasibility study of the evaluation of Reactor Physics parameters. This paper will calculate static parameters, such as k_{eff} (multiplication factor), k_{src} (source multiplication factor), fluxes, and spectrum in a configuration defined in the framework of the CRP. The calculation are being performed using deterministic (TORT) and Monte Carlo (MCNP) codes, and the paper will inter compare the results obtained by these codes.

1. INTRODUCTION

The Brazilian Facility IPEN-MB-01 is a Zero Power Reactor (100 watts), light water tank type, consisting of a 28x 26 rectangular array of UO_2 fuel pins, 4.3 w/0, with a clad of SS-304. The IPEN/MB-01 reactor reached its first criticality on November 9, 1988. Since then it has been utilized for basic reactor-physics research. These critical lattices consist of nearly square, uniform lattices of stainless steel clad cylindrical fuel rods immersed in light water. The pitch of the rods is 15.0 mm, is close to the optimal pitch (maximum k_∞).

The facility is controlled by two control banks, composed by 12 Ag-In-Cd pins. Also there are two banks of Safety Rods, composed by 12 B4C pins, which are kept out of the core.
Since its first criticality in 1988, the IPEN-MB-01 has been used for several reactor physics measurements, such as determination of the spectral index, reactivity coefficients, critical kinetics parameters, spectrum, flux etc. Bitelli[1] summarized the main experiments realized in the facility. Recently, absolute measurements of kinetic parameters in sub critical configuration using a fixed neutron source had been reported by Kuramoto [2]. Presently the facility is included in the NEA International Reactor Physics Evaluation Project (IRPhE)[3].

Although, originally designed with a critical core controlled by rods, it easily can be made subcritical by changing the control rod position, or the number of fuel pins in the core.

The Plasma and Ion Source Technology Group at the Lawrence Berkeley National Laboratory, developed a pulsed compact neutron generator (D-D, D-T or the T-T fusion reaction) [4], which easily could be inserted into a subcritical core of the IPEN-MB-01.

Coupling the compact pulsed neutron generator with the subcritical core, will allow extending the type of reactor physics experiments to be performed in the IPEN-MB-01, mainly kinetics parameters measurements.

Within the framework of the collaborative work on the utilization of LEU in ADS, within the CRP, Analytical and Experimental Benchmark Analysis of ADS [12], we consider in this paper the analyses of a sub critical configuration of the IPEN-MB-01, by removing all control rods, and two rows and lines of the critical configuration as illustrated in figure 1, with a point source of 14MeV located in the middle of the active core (see Fig. 2) in the position M14 (see Fig. 1). The tube guide of the source is to be considered as an empty space. The positions of the control and safety rods are to be considered as tube guides filled with water. The matrix considered for the exercise is a configuration (24x 22) positions in the matrix, as illustrated in figure 1. Also figure 2, illustrates a geometrical modeling and the reference system as given by MCNP

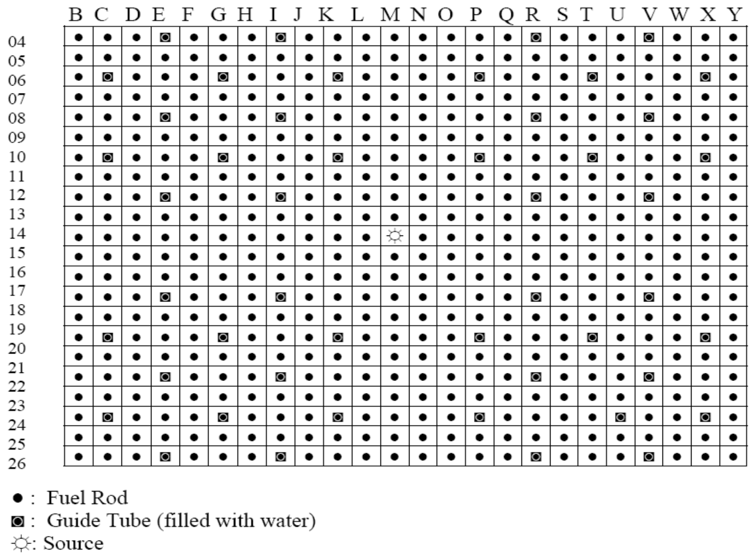

Figure 1. Subcritical configuration considered.

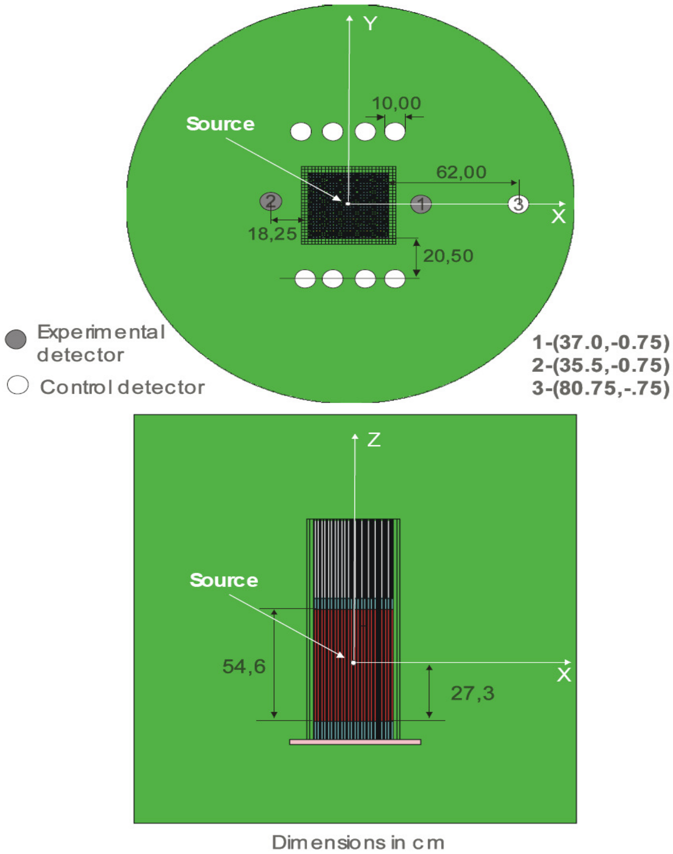

Figure 2. System Model and Detectors where calculations have been performed.

2. METHODOLOGY

In order to compare deterministic and Monte Carlo calculation, we have used the same cross section data base, ENDF\B-VI. In MCNP calculation a point-wise data already processed included in MCNP package [13]. Deterministic codes such as TORT need a special treatment of evaluated nuclear data files to generate group constants. In this paper we are not evaluating the data base, but evaluating the cross section processing for source driven system, and the geometry capability of the deterministic codes.

2.1. Deterministic Discrete Ordinates Calculation

To perform deterministic calculation we used the code TORT [6] (three-dimensional geometric systems) [6], which use the discrete ordinates to treat the directional variable ($\vec{\Omega}$) and the weighted difference, nodal or characteristic methods to treat the spatial variables ($\vec{r}$), in order to solve the static multigroup neutron transport equation .The numerical scheme used in TORT is described in references [10,11]

To generate the cross sections we have used the scheme used at IPEN [9] for criticality calculation, even knowing that it is not adequate for source driven systems. In the first part, the NJOY system was employed to access and to process the nuclear data file in a fine group structure. The cross sections are homogenized in a fine group level. Two sets of fine multi group structure were considered to generate the broad group library: 85 groups and 620 groups (SAND-II structure). These two sets of fine multi group were collapsed to several different numbers of broad groups: 4,10, 14 15 and 16 (in our case), with different number of up scattering. Finally, the broad group library is conveniently formatted to the TORT (3D Discrete Ordinates Code) format using the GIP program [6] . The calculation methodology applied for the generation of cross section is shown in figure 3 [9]. For the problem we solved we have used 16 energy groups, the order of scattering order was P_3, and the order of quadrature was S_{16}.

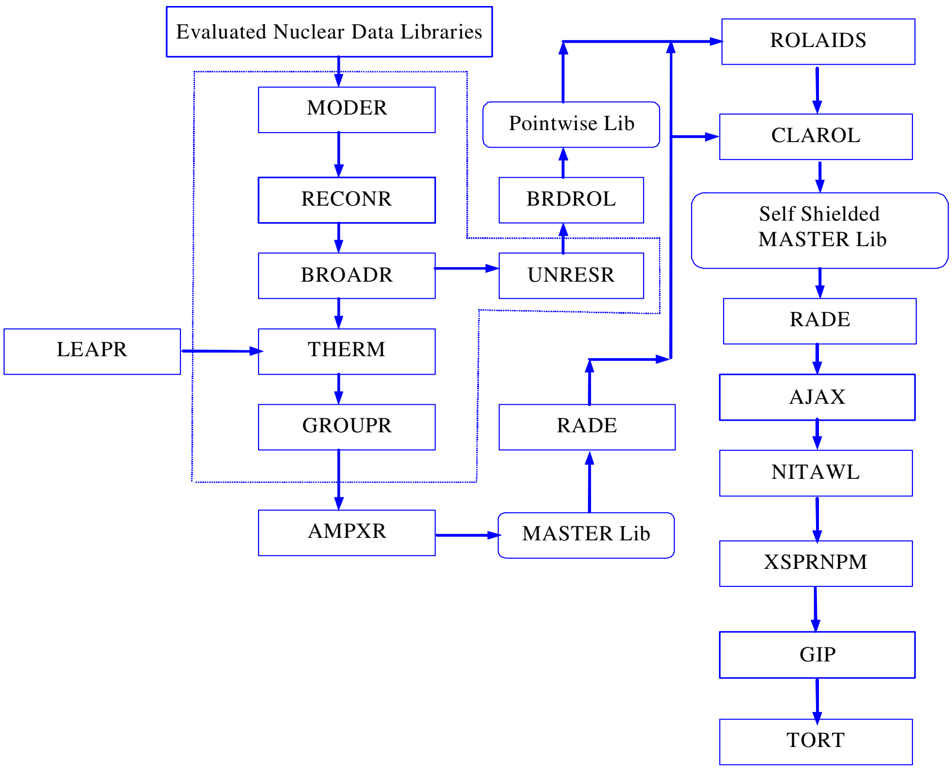

Figure 3. Nuclear Data Processing for Deterministic Calculation.[9]

2.2. Monte Carlo Calculation

The Monte Carlo calculation was performed with MCNP - A General Monte Carlo N-Particle Transport Code - Version 5.1.40 using the default cross section data of MCNP, i.e., point-wise ENDF/B-VI and a S(α,β) treatment for thermal neutrons in moderator.

The calculation of multiplication factor (k_{eff}) and source multiplication factor (k_{src}) has been done with KCODE mode. For k_{src} calculation a 14 MeV neutron of source was used to initialize the KCODE calculation, and in each cycle the source was replaced for the fissions neutrons of generated by neutrons of the last generation. The multiplication of each generation has been used to calculate the total net multiplication of source neutrons as suggested by Meulekamp& Kuijper [7]. The equation used for calculate k_{src} is:

$$k_{src} = 1 - \frac{1}{\sum_{i=1}^{\infty}\prod_{j=1}^{i} k_j}, \qquad (1)$$

Where k_j is the number of neutrons produced by neutrons of past generation, in our case k_j is jth estimative of k_{eff} performed in each cycle of KCODE in MCNP. The number in denominator is the total number of neutrons generated in the fission chain, and since reactor is subcritical, this number will converge, in the same way that k_j will converge to $k_{eff.}$

This approximation is analogous a definition given by [8]:

$$k_{src} \equiv \frac{\langle \mathbf{u} | \mathbf{P\Phi} \rangle}{\langle \mathbf{u} | \mathbf{S} \rangle + \langle \mathbf{u} | \mathbf{P\Phi} \rangle} \cong 1 - \frac{1}{\frac{\langle \mathbf{u} | \mathbf{P\Phi} \rangle}{\langle \mathbf{u} | \mathbf{S} \rangle}} \qquad (2)$$

Where u is an importance function of neutrons with respect to the relevant "observable" of the system, P is a production operator of Boltzmann equation, < | > represents angle-space-energy integration. As u is arbitrary, the choice u=1 recovery the approximation used for k_{src}.

3. RESULTS AND DISCUSSION

The multiplication factor (k_{eff}) calculated by MCNP and TORT agree very well. This mean our cross sections are good for this thermal system without source and our geometric model is similar, once this calculation is source independent. The k_{src} calculated by MCNP use the methodology cited above, although a deterministic evaluation of k_{src} will requires an adjoint calculation. This paper will not reported k_{src} evaluated by TORT since the methodology is on going.

The detectors are out of core in a very thermal spectrum, and both determinist and stochastic calculation agree satisfactorily. In this positions all neutrons, source and fissions, have been moderated.

Table 1. Numerical results to the model problem

	MCNP	**TORT**
Keff	0.9699±0.0001	0.970004
Ksrc	0.9789±0.0002	In development
Flux at Detector 1 [cm^{-2}]	3.55±0.03 E-04	3.2934 E-04
Flux at Detector 2 [cm^{-2}]	6.37±0.05 E-04	6.1457 E-04
Flux at Detector 3 [cm^{-2}]	8.6±0.8 E-07	Out of geometric model

The spectra were normalized in such way that the sum over all groups is one. In this way an error in a group will cause error in the others groups. The major discrepancies are in the fastest and slowest groups. One hypothesis for these discrepancies is that the neutron group structure for TORT is not adequate to deal with the 14 MeV source, mainly because there are only two groups over 1 MeV. The error in moderation process of source neutrons will propagate in all groups.

The major discrepancy was in the flux shape, this also can be attributed to cross-sections due to a wrong treatment of the fastest group (18-3 MeV) increase the leakage of neutrons, increasing the slope of neutron flux in such way that fluxes drops fast. Also in the collapsing process, source neutrons are not taken in account.

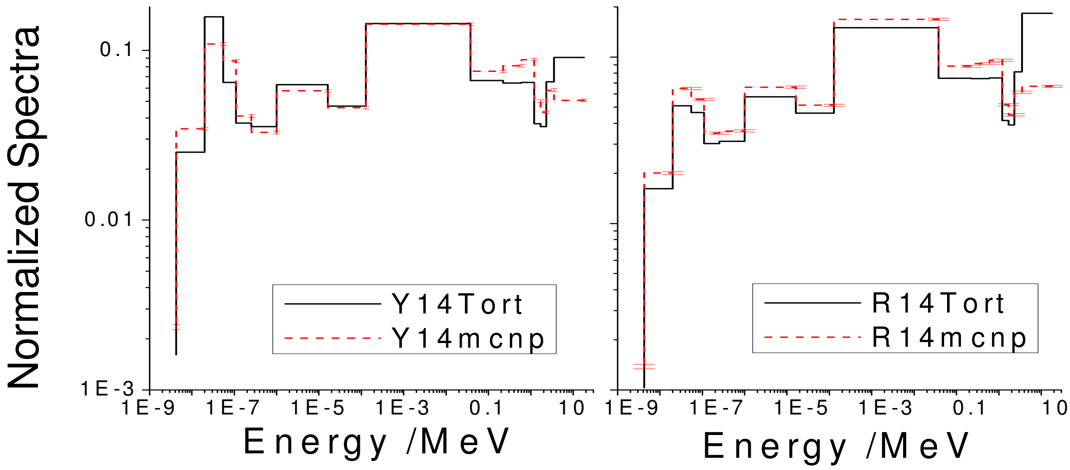

Figure 3. Normalized Z-averaged Spectra of Positions Y14 (a) and R14 (b).

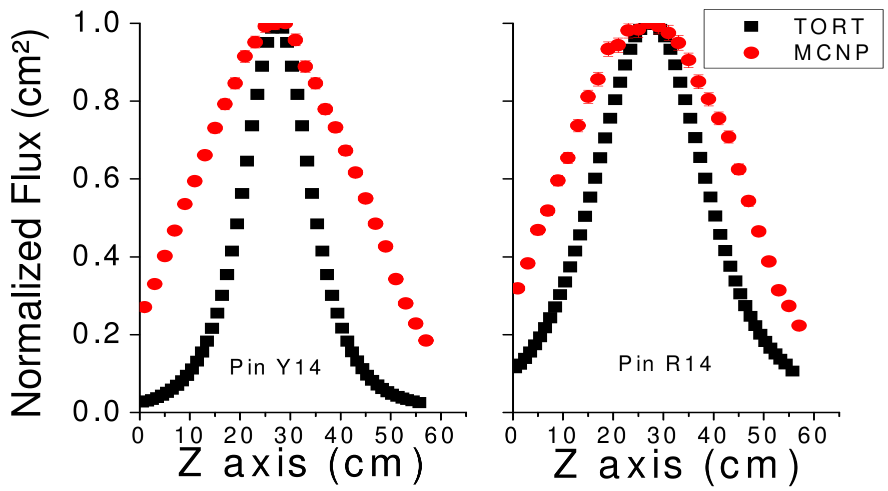

Figure 4 Normalized Flux distribution on Z direction in Positions Y14 (a) and R14 (b).

4. CONCLUSION

These results are preliminary and we are investigating the causes of discrepancies and refining our results. Despite of these discrepancies we may conclude that a source of intensity of 10^9 neutrons s^{-1} will produce a flux of same order of magnitude that the experiments are being conducted in IPEN/MB-01 facility. Also it becomes clear that for deterministic calculation a new scheme to generate cross section must be developed. A suggestion is starting from the fine group structure (BROAD MASTER), and makes a collapse of these set in a one or two dimension using a fixed source problem. Also as already mention the number of fast groups must be increased.

The generation of multi-group cross-sections for source driven subcritical reactors is still an open question and has being investigated by our group.

ACKNOWLEDGMENTS

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