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CALCULATION OF THE EFFECTIVE DELAYED NEUTRON FRACTION BY TRIPOLI-4 CODE FOR IPEN/MB-01 RESEARCH REACTOR

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ABSTRACT

The effective delayed neutron fraction βeff is an important reactor physics parameter. Its calculation within the multi-group deterministic transport code can be performed with the aid of adjoint flux weighted integrations. However, in continuous energy Monte Carlo transport code, the adjoint weighted βeff calculation becomes complicated due to the backward treatment of the anisotropy scattering. In TRIPOLI-4 continuous energy Monte Carlo code, the βeff calculation was performed by a two-run method, one run with delayed neutrons and second with only the contribution from prompt fission neutrons. To improve the uncertainty of the βeff two-run calculation for the experimental reactors, two simple and fast one-run methods to estimate the βeff in the continuous energy simulation have been implemented into the TRIPOLI-4 code. First approach is an improved one of the Bretscher's prompt method and second one based on the proposal of Nauchi and Kameyama. In these one-run methods, the prompt and the delayed neutrons are first tagged. Their tracking and statistics are separated performed. The new βeff calculations have been optimized in the power iteration cycles so as to estimate the production of prompt and delayed neutrons from the prompt and delayed neutrons of previous generation. To validate the new βeff calculation by TRIPOLI-4, several benchmarks including fast and thermal systems have been considered. In this paper the recent measurements of βeff in the research reactor IPEN/MB-01 have been benchmarked. The basic components of the βeff and the Keff have been also calculated so as to understand the influences of the cross sections and the delayed neutron yields on the reactor reactivity calculations. Three nuclear data libraries, ENDF/BVI.r4, ENDF/B-VII.0, and JEFF-3.1 were taken into account in this study.

Key Words: TRIPOLI-4 Monte Carlo code, effective delayed neutron fraction, Criticality, IPEN/MB-01, ENDF/B-VII.0 and JEFF-3.1

1. INTRODUCTION

Delayed neutrons produced from fission products play a critical role in the nuclear reactor operation and safety. To model reactivity control, both delayed neutron yields and decay constants are important [1, 2]. Based on different experimental and numerical methods to assess these delayed neutron parameters, delayed neutron yields ν_d and energy spectra χ_d are reported in the modern nuclear data files, ENDF/B-VI.r4, ENDF/B-VII.0, and JEFF-3.1. Under multi-group representation, these ν_d data vary from fast fission to thermal fission, from U-235 fission to U-238 fission, and from Uranium fission to Plutonium fission. Using the prompt fission yields ν_p and the ν_d data, the delayed neutron fraction β_0, $\beta_0 = \nu_d / (\nu_p + \nu_d)$, can be calculated.

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As the prompt fission neutrons and the delayed neutrons are different in their energy distribution, the delayed neutron escape probability from the fissile is smaller. According to the fuel and the reflector compositions, the neutron moderation ratio, and the neutron absorption and production in different fuel zones of the nuclear system, delayed neutrons distribution in the experimental reactor cores can also vary significantly. To evaluate the real contribution of delayed neutrons to the reactivity of the fission system, the effective delayed neutron fraction βeff was applied.

To calculate the βeff instead of the β_0 by using the deterministic multi-group adjoint calculation [2], most calculation codes of reactor physics can perform it. The multi-group adjoint method requires the computation of the "backward" model only at the cost of model integrations. In continuous energy Monte Carlo criticality calculation, the adjoint weighted calculation becomes difficult due to the backward treatment of anisotropy scattering cross-sections [3, 4]. To solve this problem to calculate the βeff in a continuous energy Monte Carlo calculation, some approaches to avoid exact adjoint weighted calculation have been proposed and validated [5-8].

To estimate the βeff by TRIPOLI-4 Monte Carlo continuous energy transport code [9] the initial method was to run the code two times: the first run using the standard data library and the second using the data library without the delayed neutrons. Then the βeff can be approximately obtained from the two runs. Another improved method using only one standard library was also introduced into the TRIPOLI-4 but a second run of the code to inactivate the delayed neutrons contribution was still necessary to get the estimated βeff.

As the calculated uncertainty of the βeff can be important due to above two-run methods, we have recently evaluated the published methods and the cost of code implementation and calculation. Two approximations to estimate the βeff have been taken into account together in TRIPOLI-4 based on the separation of the prompt and the delayed neutrons tracking. First approach is an improved one of the Bretscher's prompt method and second one proposed by Nauchi and Kameyama [6, 10]. These two estimations of βeff can be obtained in a single run and they are helpful to do the cross-check of the simulation.

To validate the one-run βeff calculation by TRIPOLI-4, several benchmarks including fast and thermal, metal and solution, UO_2 and MOX systems have been considered. In this paper the new measurements of βeff in the research reactor IPEN/MB-01 [11, 12] have been benchmarked with the TRIPOLI-4 code. Both six groups delayed neutron data from nuclear data library ENDF/B-VII.0 and eight groups data from JEFF-3.1 were taken into account. ENDF/B-VI.r4 library was also applied to compare with the new ENDF/B-VII.0 one.

2. TRIPOLI-4 AND THE EFFECTIVE DELAYED NEUTRON FRACTION

TRIPOLI-4 is the fourth generation of the TRIPOLI family of Monte Carlo codes developed from the 60's by CEA. It simulates the 3D transport of neutrons, photons, electrons and positrons as well as coupled neutron-photon propagation and electron-photon cascade showers. The code addresses radiation protection and shielding problems, as well as criticality and reactor physics problems through both critical and subcritical neutronics calculations. Continuous-energy cross sections from modern nuclear data evaluations, ENDF/B-VI.4, ENDF/B-VII.0, JEFF-3.1, and JENDL-3.3 can be run with TRIPOLI-4 code.

Based on the international benchmarks and the CEA internal measurements, extensive validation studies of TRIPOLI-4 code have been performed on the criticality safety of the fuel cycle, the reactor dosimetry of the PWR, the nuclear heating of research reactors, the radiation damage of reactor grade steels, the neutron activation for decommissioning activities, and the fusion neutronics [9, 13].

For fission reactor physics and criticality calculations with TRIPOLI-4 code, many useful options of the code are available. These options include: interactive I/O display, flexible geometry package including 3D lattices geometry, 3D lattice tally and mesh tally functions, automatic discard of initial cycles for Keff and reaction rate tally, inter-cycles correlation correction on the variance of the scores etc.. [14-16].

To reply the demands from the experimental reactor physics design and the nuclear safety authority, the effective delayed neutron fraction βeff calculation with TRIPOLI-4 is being improved. Instead of running twice to obtain βeff with a relative important variance, two new methods to estimate βeff with different approximations in simulation have been implemented into the TRIPOLI-4 continuous energy run.

The first method based on the contribution from the prompt fission neutrons [10] and the second method based on the contribution from the delayed neutrons [6]. These two methods have been optimized in TRIPOLI-4 to estimate the βeff. With a small additional cpu running time the one-run calculated βeff can be obtained in a Keff run.

In fact both the prompt fission neutron and the delayed neutron have been tagged and simulated in a single TRIPOLI-4 neutron transport simulation. The statistical counting has then been performed to treat separately prompt fission neutrons contribution and delayed neutrons contribution in the power iteration cycles. So the βeff can be simply and approximately estimated with the equations (1) and (2).

$$\beta eff = 1 - (Keff\_p / Keff) = Npd / (Npp + Npd) \qquad (1)$$

$$\beta eff = (Npd + Ndd) / Keff \qquad (2)$$

In the Eq. (1), the Keff_p is the effective multiplication factor Keff for prompt fission neutrons. The previous two-run method is just to compute first the Keff_p and the Keff and then to get the βeff. In present one-run simulation, the ratio Keff_p / Keff can be approximately calculated by the ratio of Npp / (Npp + Npd), where the Npp represents the averaged value of prompt neutrons produced by the prompt neutrons of previous generation, and the Npd the averaged value of prompt neutrons produced by the delayed neutrons of previous generation.

In the Eq. (2), the sum of the averaged values of Npd and Ndd is used to estimate the contribution of the delayed neutrons of previous generation, where the Ndd represents the averaged value of delayed neutrons produced by the delayed neutrons of previous generation.

Because these βeff calculations use the results from the successive generations (cycles) the fission neutron source convergence or the 3D power map convergence must be taken into account after the first discarded generations. In a case with only UO_2 fuel the convergence of prompt neutrons distribution and that of delayed neutrons distribution can be in a similar way. In a case with UO_2 and MOX fuels the convergence of delayed neutron distribution can be slower.

Equations (1) and (2) are two approximations to calculate βeff but they are generally acceptable for civil fast and thermal fission systems. Only in some special fast fission cases an uncertainty of a few percents on βeff can be introduced. This uncertainty of estimation has been also discussed in the reference [17].

To validate these βeff estimations with TRIPOLI-4, several benchmark calculations are being performed on various fissile systems. One of the most interesting βeff measurement benchmarks for light water reactor application is the IPEN/MB-01 research reactor benchmark. Two different benchmark models of IPEN/MB-01 core states have been calculated with TRIPOLI-4 in this study so as to check the βeff measurements and the sensitivity of βeff on the absorber rods positions and the presence of core baffle plates.

3. IPEN/MB-01 RESEARCH REACTOR EXPERIMENTS

The IPEN/MB-01 research reactor facility is located in the city of Sao Paulo, Brazil and it is a zero-power critical facility designed for measurement of LWR reactor physics parameters to be used as a benchmark data for checking calculation codes and related nuclear data libraries. The kinetic parameters experiment was performed between 2004 and 2006 and compiled to International Reactor Physics Benchmark Experiments (IRPhE) Handbook on 2007 [11].

The kinetic parameters experiment was performed by specially developed reactor noise procedures. The macroscopic noise experiment is run for measurement at very low frequency range and a point kinetics model is used to extract the βeff. The microscopic noise experiment employs the traditional Rossi-α techniques but a two-region model is also used to fit the measurement data so as to obtain the βeff [12]. The recommended experimental βeff of the IPEN/MB-01 reactor is 750 +/- 5 pcm [11].

The schematic view of the IPEN/MB-01 core configuration is shown in Figure 1 for radial representation. The reactor core consists of a square lattice of 28 x 26 UO_2 fuel pins and poison pins immersed in a cylinder of water of 100 cm radius. The fuel pins contained 4.34 % enriched uranium and were clad in stainless steel. The active length of fuel pin was 54.84 cm.

Several criticality experiments were also performed inside the water tank and benchmark models have been documented as LEU-COMP-THERM-077 (lct077) in ICSBEP handbook [18]. Other details of the experiments are described in this handbook.

Because the kinetic parameters βeff and Λ (neutron generation time) data were measured experimentally instead of using theoretical calculations of ρ (reactivity) or Λ to deduce βeff from the measured α eigenvalues (α = (βeff – ρ) / Λ). This benchmark becomes one of the most representative experiments for βeff kinetic parameters study [8, 12, 19].

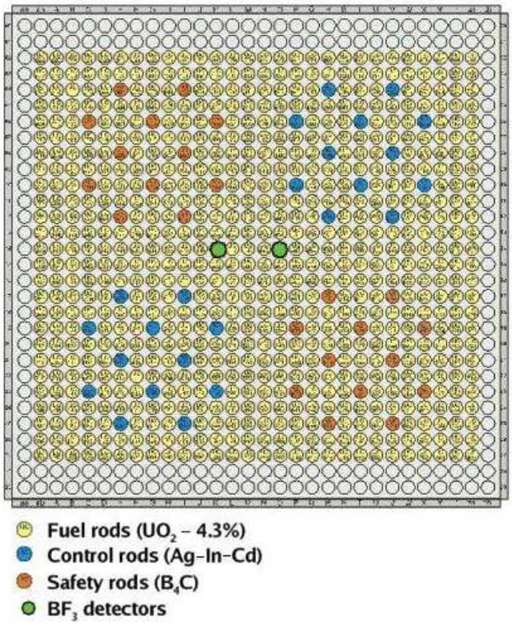

Fig. 1 Schematic view of the IPEN/MB-01 Core [12]

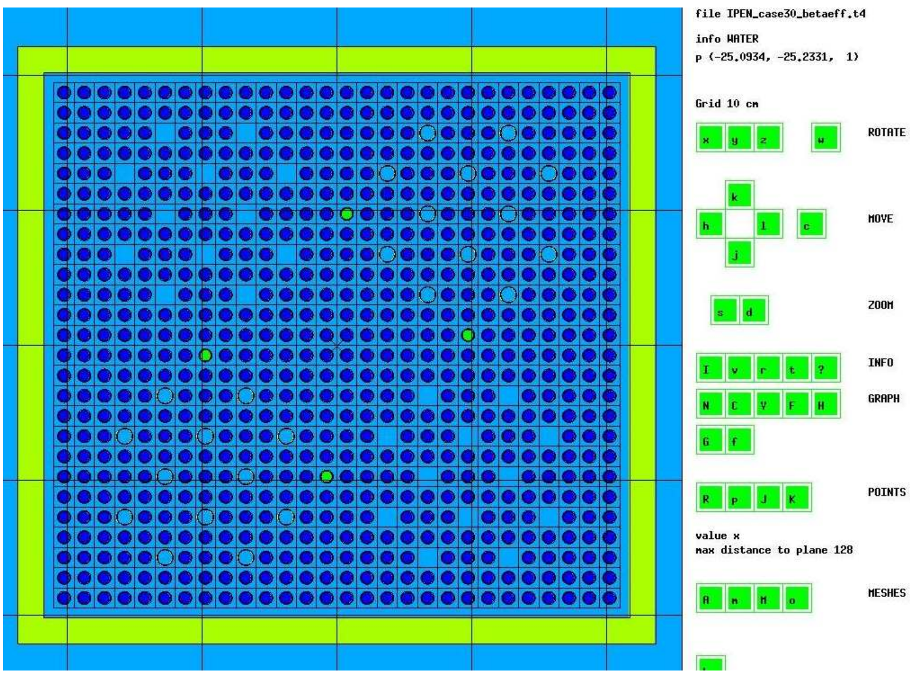

Fig. 2 TRIPOLI-4 modeling of the IPEN/MB-01 reactor core [16, 18] – radial view (Benchmark model : core baffle present, control rods almost withdrawn, and four poison rods inserted)

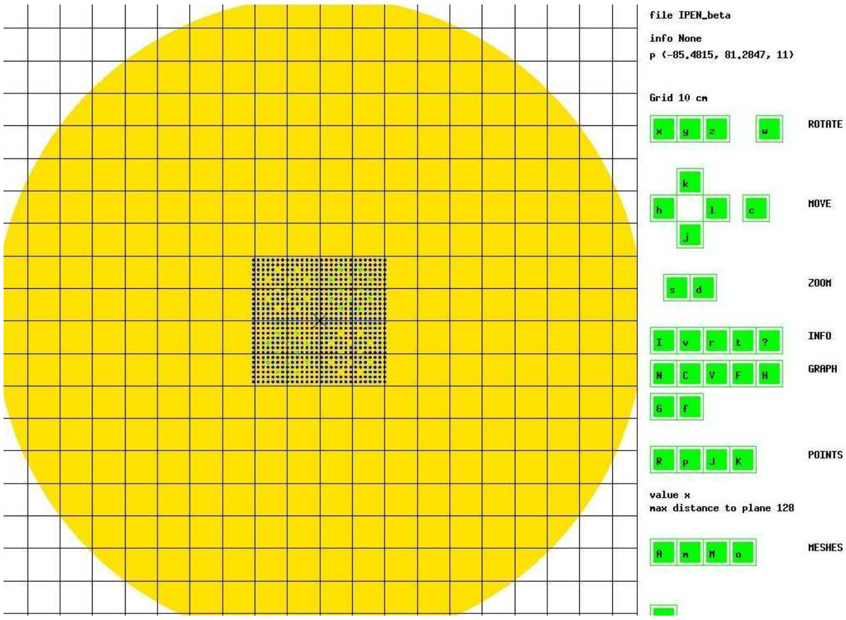

Fig. 3 TRIPOLI-4 modeling of the IPEN/MB-01 reactor core [11, 16] – radial view (Benchmark model : core baffle absent and control rods semi-inserted)

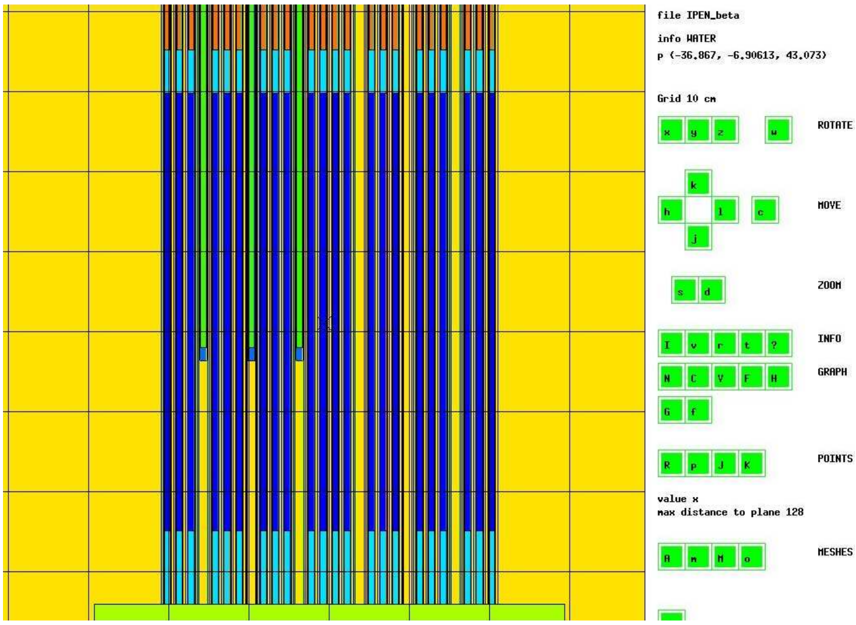

Fig. 4 TRIPOLI-4 modeling of the IPEN/MB-01 reactor core [11, 16] – axial view (Benchmark model : core baffle absent and control rods semi-inserted)

4. TRIPOLI-4 CALCULATIONS AND RESULTS

Several IPEN/MB-01 configurations are available from the references 11 and 18. Two of these reactor cores were selected in this study and they are presented in Figures 2 and 3. The TRIPOLI-4 geometry modeling consists of the lattice representation of the fuel rod array, the control rods, and the safety rods. In Fig. 2, modeling of the core baffle and four poison rods are also included to represent the benchmark model case 3 of lct077 [18]. In Figures 3 and 4, modeling of 24 semi-inserted control rods is included to represent the benchmark model of βeff measurements [11, 19]. Description of the TRIPOLI-4 lattices geometry has been presented in previous study [15].

In this study the continuous-energy cross-section libraries JEFF-3.1 and ENDF/B-VII.0 were prepared by NJOY99-259 processing system with the convergence criteria of 0.1 %. The previous ENDF/B-VI.4 data library was also included to benchmark the evolution of βeff and Keff. The lct077 and βeff experiments were performed at room temperature so the standard 300K cross–section data were used. Fuel and structure material atomic densities were taken from the ICSBEP Handbook.

All TRIPOLI-4 models for the lct077 and βeff experiments were explicit. In each run, 50 initial cycles of 5000 neutron histories per cycle (generation) were skipped in order to obtain converged fission neutron source. With eight CPUs of 2.8 GHz, each parallel run took about 2 hours running time to obtain the tally results with one standard deviation of Keff < 15 pcm and that of βeff < 2 pcm.

4.1 βeff benchmarks with ENDF/B and JEFF-3.1 libraries

Using ENDF/B-VI.4, ENDF/B-VII.0, and JEFF3.1 nuclear data libraries, Table I shows the TRIPOLI-4 calculated βeff values based on equations (1) and (2) for the benchmark model case 3 of lct077. The recommended experimental βeff (Exp-R) from the IRhPE Handbook and the measurements (Exp-RM, Reflector Measurement by Rossi-α two-region model fit, and Exp-FA, Frequency Analysis measurements) from references 12 and 19 are also presented in the Table I.

Using ENDF/B-VII.0, and JEFF3.1 nuclear data libraries, Table II presents the TRIPOLI-4 calculated βeff values based on equations (1) and (2) for the benchmark model of βeff measurements (cf. Figures 3 and 4, and Ref. IPEN/MB01-LWR-RESR-001 [11]).

The TRIPOLI-4 calculated βeff values are clearly nuclear data libraries dependent. The U-235 thermal fission delayed neutron yields are the main cause of the different βeff obtained. Both the ENDF/B-VII.0 and the JEFF-3.1 data libraries produced acceptable βeff values close to the recommended experimental one. Only the βeff calculated from the ENDF/B-VI.4 data library is about 4 % higher than the recommended experimental one.

βeff values calculated with TRIPOLI-4 by the equations (1) and (2) are almost identical. The only difference is their associated uncertainties. When using equation (2) to estimate βeff, the slightly higher standard deviation from the TRIPOLI-4 simulation has been obtained.

Table I. βeff (pcm) of the IPEN/MB-01 reactor calculated with TRIPOLI-4 (Benchmark model : core baffle present and control rods almost withdrawn)

	ENDF/B-VI.4 βeff Std.	ENDF/B-VII.0 βeff Std.	JEFF-3.1 βeff Std.
Eq. (1)	782.2 ± 1.8	744.4 ± 1.6	761.7 ± 1.7
Eq. (2)	781.9 ± 1.9	743.8 ± 1.7	761.5 ± 1.8
^{235}U thermal fission delayed neutron yield ν_d	0.0167	0.0159	0.0162
Exp-R	750.0 ± 5.0	750.0 ± 5.0	750.0 ± 5.0
Exp-RM	754.0 ± 11.0	754.0 ± 11.0	754.0 ± 11.0
Exp-FA	747.0 ± 11.0	747.0 ± 11.0	747.0 ± 11.0

- Eq. (1) and Eq. (2) : TRIPOLI-4 calculations with equations (1) and (2).
- Exp-R : Recommended experimental result [11].
- Exp-RM : Two-region Rossi-α measurement [12].
- Exp-FA : Frequency Analysis measurement [19].

Table II. βeff (pcm) of the IPEN/MB-01 reactor calculated with TRIPOLI-4 (Benchmark model : core baffle absent and control rods semi-inserted)

	ENDF/B-VII.0 βeff Std.	JEFF-3.1 βeff Std.
Eq. (1)	745.5 ± 1.6	765.8 ± 1.7
Eq. (2)	745.3 ± 1.7	765.4 ± 1.8
Exp-R	750.0 ± 5.0	750.0 ± 5.0

- Eq. (1) and Eq. (2) : TRIPOLI-4 calculations with equations (1) and (2).

In Tables I and II, the calculated βeff values are very close when using the same data library. It means that the 1.95 cm thick baffle plates around the 28 x 26 fuel rods and the inserted absorbers (control rods or poison rods) have a very limited impact on the calculated βeff values. It also confirms the observation of the Ref. 11, the βeff values are not very sensitive to the reactivity differences.

4.2 Basic components for βeff estimation and Keff estimation

Table III first presents the TRIPOLI-4 calculated results of the basic components, Npp, Ndp, Npd, and Ndd for βeff estimation of the benchmark model case 3 of lct077. Components Npp, Npd, and Ndd have been introduced in the equations (1) and (2) of section 2. The Ndp is the averaged value of delayed neutrons produced by the prompt neutrons of previous generation.

Table III. TRIPOLI-4 calculated four components for βeff and Keff (IPEN/MB-01 model : core baffle present and control rods almost withdrawn)

Tracking	**ENDF/B-VI.4** Std.	**ENDF/B-VII.0** Std.	**JEFF-3.1** Std.
Npp	0.98730 ± 0.00015	0.99100 ± 0.00015	0.99102 ± 0.00015
Ndp	0.00717 ± 0.00001	0.00686 ± 0.00001	0.00702 ± 0.00001
Npd	0.00778 ± 0.00002	0.00743 ± 0.00002	0.00761 ± 0.00002
Ndd	0.00005 ± 0.00000	0.00005 ± 0.00000	0.00005 ± 0.00000
Keff+	1.00225 ± 0.00015	1.00538 ± 0.00015	1.00567 ± 0.00015

+ Keff obtained from the collision estimator of TRIPOLI-4 calculations.

In Table III, the Npp component from ENDF/B-VI.4 library is about 0.4 % lower than that of ENDF/B-VII.0 but the Ndp component from ENDF/B-VI.4 is about 4.5 % higher than that of ENDF/B-VII.0. The smaller Npp component directly influences on the Keff of ENDF/B-VI.4. The higher Ndp component directly influences on the βeff of the ENDF/B-VI.4. The first term is cross sections related and the second is more delayed neutron yields related (see Table I).

In fact, the sum of components Ndp and Ndd can be served to estimate the β_0 (cf section 1) of the UO2 fuel used in IPEN/MB-01 core. In present study, the U-238 fission in total uranium fission has been calculated by TRIPOLI-4. It is only a small part, 3.51 %. That means the U-238 contribution from delayed neutrons to the IPEN/MB-01 reactor reactivity can be estimated. It is about 10 % of the total delayed neutrons contribution based on the ENDF/B-VII.0 data library.

We can also calculate the sum of these four components in Table III and get an estimation of the Keff based on the neutrons in the generation. In fact the Keff calculated from these components must be identical to the Keff calculated by the standard estimators, collision estimator and track-length estimator, of the TRIPOLI-4 code.

The Keff of the sample IPEN/MB-01 reactor model is about 300 pcm higher when using the ENDF/B-VII.0 or JEFF-3.1 data libraries. This Keff difference has been also observed in the previous study on lct008 low enriched UO_2 lattice in boron water [15]. From Table III, we can also conclude that this difference in reactivity of two versions of ENDF/B libraries is firstly and mainly due to the cross sections effect (Npp) and secondly due to the delayed neutron yields effect (Ndp and Npd).

Between libraries JEFF-3.1 and ENDF/B-VII.0, their disagreement on the Keff of IPEN/MB-01 reactor is small. A slight difference about 2.3% in the Ndp component shows that the impact of the new eight groups delayed neutron data from JEFF-3.1 is very limited for the UO_2 fuel lattice immerged in the light water.

4.3 Neutron balance components and Keff estimation

During the βeff implementation, four extra scores, P (total neutron production by fissions), A (total neutron absorption), L (total neutron leakage), and Nxn (total neutron production by (n,xn) interactions) were also introduced into the TRIPOLI-4 code so as to help users to evaluate the neutron balance components of the simulation and to support an extra estimation of Keff (Keff = (P + Nxn) / (A + L)). In fact the Keff calculated from these components must be identical to the Keff calculated by the standard estimators, collision estimator and track-length estimator, of the code.

Table IV presents these neutron balance components obtained from the TRIPOLI-4 simulations of the benchmark model case 3 of lct077. The main difference among three data libraries is in the total neutron production by fission, P. Slight difference was also found in the neutron leakage, L, and the total neutron production from the (n,xn) reactions. The neutron leakage is around 4%, mainly due to the axial leakage (see Fig. 4).

Table IV. TRIPOLI-4 calculated neutron balance components for Keff estimation (IPEN/MB-01 model : core baffle present and control rods almost withdrawn)

	ENDF/B-VI.4 Std.	**ENDF/B-VII.0** Std.	**JEFF-3.1** Std.
P (neutron production)	1.00227 ± 0.00015	1.00534 ± 0.00015	1.00567 ± 0.00015
A (neutron absorption)	0.99667 ± 0.00009	0.99669 ± 0.00009	0.99672 ± 0.00009
L (neutron leakage)	0.00399 ± 0.00001	0.00404 ± 0.00001	0.00398 ± 0.00001
Nxn (neutron from (n, xn) interactions)	0.00070 ± 0.00000	0.00076 ± 0.00000	0.00071 ± 0.00000
Keff^{+}	1.00225 ± 0.00015	1.00538 ± 0.00015	1.00567 ± 0.00015

+ Keff obtained from the collision estimator of TRIPOLI-4 calculations.

5. CONCLUSIONS

New one-run option to calculate the effective delayed neutron fraction βeff has been implemented into the TRIPOLI-4 Monte Carlo code. Under the continuous energy criticality mode, the delayed neutrons and the prompt fission neutrons are tagged in order to calculate the βeff in a single run with two different methods. The βeff measurements of the IPEN/MB-01 research reactor have been used to validate the new TRIPOLI-4 βeff calculations with three different data libraries. The TRIPOLI-4 βeff values obtained from the new data libraries, JEFF-3.1 and ENDF/B-VII.0, are very close to the experimental one.

This paper also presents the basic components of the βeff and the Keff calculations so as to understand the influences of the cross sections and the delayed neutron yields on the reactor reactivity calculations. An extensive validation work of the TRIPOLI-4 βeff calculations is being performed including MOX fuels and different research reactors. The study on neutron generation time, Λ, will be reported in the future.

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