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Technical note

A proposal of a benchmark for β_{eff}, β_{eff}/Λ, and Λ of thermal reactors fueled with slightly enriched uranium

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Abstract

A reactor noise approach has been successfully performed at the IPEN/MB-01 research reactor facility for the experimental determination of the delayed neutron parameters β_{eff}, β_{eff}/Λ, and Λ. In the measurement of the β_{eff} parameter, the reactor power, which is of fundamental importance, was obtained with a very high level of accuracy by a fuel rod scanning technique and a subsequent irradiation of a highly enriched ^{235}U foil for the fission density normalization. The final measured values of β_{eff} and β_{eff}/Λ show very good agreement with independent measurements and can be recommended as benchmark values for thermal reactor applications because their uncertainties are much lower than the target accuracy recommended for β_{eff} calculations (|C-E|/E less than 3%). The theory/experiment comparisons reveal that only JENDL3.3 attends the target accuracy for β_{eff} calculations. This result fully supports the reduction of the ^{235}U thermal yield as proposed by Okajima and Sakurai. The ENDF/B-VI.8 library and its revised version performed at LANL overpredict β_{eff} by as much as 7.2%. The newly released JEFF-3.1 library shows a discrepancy of 4.8% for β_{eff}. For β_{eff}/Λ, the deviations are relatively larger (more than 10%) for all libraries due to the underprediction of the prompt neutron generation time (Λ).

1. Introduction

Although comprising less than 1% of the neutrons emitted by fission, the delayed neutrons play a fundamental role in the reactor physics field. The control and accident analysis of a nuclear reactor and the conversion of reactor period into reactivity require the knowledge of the effective delayed neutron parameters as well as their decay constants. In a nuclear reactor chain there are many fission products (approximately 270) which can be considered potential delayed neutron emitters. However, an experimental characterization of all these emitters is very difficult due to their very low yield and/or low half-lives as well as to their very complex transmutation chain. Yet, it is possible to measure their aggregate behavior and generate a few groups model where the decay constants and abundances are mean values of various emitters with similar decay constant. Among the models utilized in the dynamic behavior of a nuclear reactor, the most common one is the point reactor model. A few number of precursor groups can be considered adequate for this model. To date practically all analysis are performed in a six-group model. The main effective delayed neutron parameters to be used in such set of equations are β_{eff} (the effective delayed neutron fraction) and β_{eff}/Λ, where Λ is the prompt neutron generation time.

There are several in-pile experimental approaches for the determination of β_{eff} (Okajima et al., 2002) and β_{eff}/Λ (Spriggs et al., 1997; Williams et al., 1996). This category of experiment is very important to the point reactor model because it can provide valuable information related to the effective delayed neutron parameters used in such a model. Contrary to the criticality safety problems (Briggs, 2004)

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and shielding problems (NEA; IAEA), where many such experimental results have been evaluated, resulting in thousands of benchmark specifications, the available experimental support for the effective delayed neutron parameters is scarce and in may cases its utilization is not so straightforward and very well established. Concerning thermal reactors fueled with slightly enriched uranium, a literature survey shows that the available experiments related to β_{eff} and β_{eff}/Λ were performed in the following facilities: Stacy (Tonoike et al., 2002), TCA (Nakajima, 2001), MISTRAL-1 (Litaize and Santamarina, 2001), Proteus (Williams et al., 2001), Sheba (Butterfield, 1994), SHE-8 (Takano et al., 1985; Kaneko et al., 1988), and Winco slab tanks (Spriggs, 1977). For the Stacy, Winco, Sheba-II, and Proteus experiments, the reported measured quantity is β_{eff}/Λ. Only the TCA, SHE-8, and MISTRAL-1 experiments report measured values of β_{eff}. The β_{eff} uncertainties range from 1.6% for MISTRAL-1 to 4.6% for SHE-8. The number of experiments related to β_{eff} is quite small and consequently the need of new and accurate experiments for thermal systems in order to have the required degree of confidence in calculations of such systems is completely justified.

The current status of the evaluation and measurements of the microscopic data of delayed neutrons is presented in the WPEC Subgroup 6 report (WPEC Subgroup 6). This report also describes the usefulness of the effective delayed neutron parameters measured in critical facilities (the so called in-pile measurements). Indeed, it is by adjusting the delayed neutron yield to improve the agreement with measured values of β_{eff} that the most suitable data are obtained for inclusion in the nuclear data files. Also, the WPEC Subgroup 6 report recommends a target accuracy of 3% for β_{eff} calculations and also states that this target accuracy will be much harder to be achieved in thermal reactors due to the reduced number of experiments. Therefore, the experiments proposed as benchmarks must have an uncertainty lower than the recommended accuracy for β_{eff} calculations in order to be useful in the nuclear data validation. Finally, this report describes deeply the new 8-group library model incorporated in JEFF-3.1 (JEFF).

Based on these premises, the main purpose of this work is an attempt to fulfill the need of benchmark experiments for β_{eff}, β_{eff}/Λ, and Λ for thermal reactor fueled with slightly enriched uranium. For such a goal, this work presents the experimental determination of these effective kinetic parameters in the IPEN/MB-01 reactor. The core of this facility consists of a 28×26 array of UO_2 fuel rods, 4.3% enriched and cladded by stainless steel (type 304) inside of a light water tank. A complete description of the IPEN/MB-01 reactor can be found in Dos Santos et al. (1999, 2004). The theoretical analyses will consider the verification of the adequacy of the nuclear data of several libraries such as: ENDF/B-VI.8 (BNL), ENDF/B-VI.8 - LANL revision (UEVAL), JENDL-3.3 (Shibata et al., 2002), and JEFF-3.1. A newly developed calculational method based on the coupled NJOY/AMPX-II/TORT (Dos Santos et al., 2000) systems is employed for such a purpose.

The importance of the measurements of β_{eff} is also present in the basic nuclear data libraries such as those used in this work. As stated, the adjustment the delayed neutron yield to improve the agreement with measured values of β_{eff} is a practice that produces very reliable data.

The prompt neutron generation time is not only a very important kinetic parameter, but also pursuits a very important characteristic because from its own definition it is inversely proportional to $\nu\Sigma_f$ where in the case of the IPEN/MB-01 reactor, the ^{235}U fission cross section plays a major role. Consequently, Λ can provide valuable information related to the normalization of the ^{235}U fission cross section in the thermal neutron energy range.

2. Experimental procedure

2.1. Determination of β_{eff} and β_{eff}/Λ

The experimental procedure (Williams et al., 1996; Thie, 1963; Martins, 1992) consists of obtaining the Cross Power Spectral Density (CPSD) from the signals of two compensated ionization chambers (CC-80 Merlin–Gerin) in the frequency ranges $\Lambda_i \ll f \ll \beta_{eff}/\Lambda$ for the measurements of β_{eff} and $f \gg \lambda_i$ for the measurements of β_{eff}/Λ, where λ_i is the decay constant of the ith group of delayed neutrons. A single experimental procedure provides a full frequency ranges which allows both β_{eff} and β_{eff}/Λ measurements. An example of the experimental data is shown in Fig. 2.

In this experiment, the IPEN/MB-01 reactor was made critical in 4.0 W and 100 W as indicated by the control room instrumentation and the control rods were kept in the automatic control mode since their movements do not interfere in the frequency region of interest. Later, these power levels were corrected by the results of the fuel rod scanning technique. The previous technique was based in a series of gold foil irradiations in the moderator and was developed to satisfy a need of the reactor commissioning. The second one, which is more precise than the previous technique, was specially developed to fulfill a need for a more accurate determination of the power normalization. As shown in several publications (Martins, 1992; Avramov

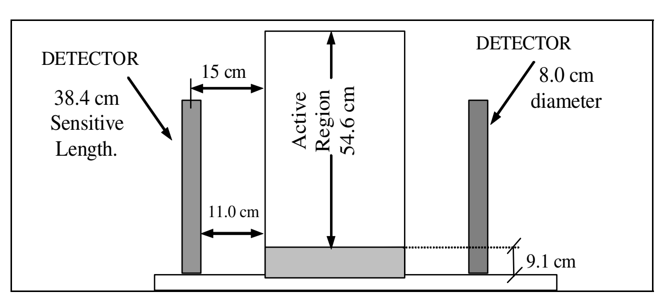

Fig. 1. Side view of the active region and the detectors positioning in the west and east faces of the nucleus.

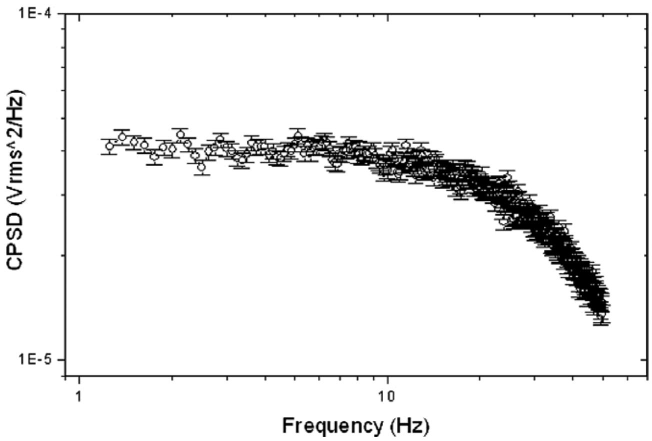

Fig. 2. Experimental CPSD obtained with 500 averages, scan rate of 100 Hz and 800 points of resolution.

et al., 1990), an accurate determination of the power of the reactor is very crucial for an accurate measurement of β_{eff}.

The ionization chambers used for the experimental purposes were placed symmetrically in the core along its west and east faces, approximately 11 cm away from the fuel rods. In this way the detectors were located in the reflector region and about 8.0 cm away from the reflected thermal neutron peak as shown in Fig. 1.

The currents of the ionization chambers were sent to a current-to-voltage converter (Keithley 614 electrometer) and then to a filter-amplifier (036-ZZ-IPEN) which has a low frequency cut-off of 1.0 mHz. The resulting signals, composed only by the fluctuating components (amplified by a factor of 30), were then sent to an Agilent 35670A Dynamic Signal Analyzer which performs the CPSD.

Assuming the point kinetic model in the detectors position, one can write the theoretical expression for the CPSD. For the β_{eff} measurement the CPSD has the form (Martins, 1992):

$$\langle \Phi_{kl} \rangle = \frac{2 I_k I_l G_k G_l F_k F_l D \gamma}{P \beta_{eff}^2} \tag{1}$$

and for the β_{eff}/Λ, becomes (Tonoike et al., 2002)

$$\Phi_{kl} = \frac{A}{(2\pi f)^2 + B^2}, \tag{2}$$

where A is a constant and B is equal to β_{eff}/Λ.

In Eqs. (1) and (2) the k and l indexes refer to both measuring chains, I is the current from the ionization chambers, G is the gain of the filter-amplifier, F is the current to voltage factor, D is the Diven factor, γ is the energy released in fission and P is the reactor power. In both equations the delayed neutrons were disregarded.

From Eq. (1) one can get directly the β_{eff} where, in this case, $\langle \Phi_{kl} \rangle$ is the mean value of the CPSD in the plateau region (from 2 to 9 Hz approximately). On the other hand β_{eff}/Λ is obtained from Eq. (2) through a least-squares fit where A and B are the fitting parameters.

2.2. The reactor power measurement

One of the key parameters in Eq. (1) is the reactor power (Avramov et al., 1990). This parameter was obtained by the determination of the relative power density of each fuel rod of the IPEN/MB-01 reactor. This was accomplished by a set of 100 W reactor operations, as was indicated by the control table, followed by the fuel rod gamma scanning. The fuel rod scanning equipment consists of a HPGe detector coupled to the rod moving system. The HPGe detector discriminates the gamma peak of the fission product ^{143}Ce (293.4 keV). The counting in the photopeak of this fission product corrected by the decay period is proportional to the total fission density at the end of irradiation. The whole active length of the every fuel rod in the core was passed continuously along the 1 cm opening aperture of the detector collimator. Such total counting of ^{143}Ce is again proportional to the fission density in the fuel rod. The fission densities for every fuel rod in the core of the IPEN/MB-01 reactor were afterwards normalized relatively to that of the central fuel rod.

The power normalization was subsequently obtained employing the following approach: every other 1 cm in the axial direction of the active fuel rod length in the central core position was counted in the HPGe detector of the scanning equipment during a specific period of time. The measured data were again the counting of ^{143}Ce photopeak, which are proportional to the fission density at that specific axial point. The necessary corrections for the decay period during the measurements and for the cooling period, which was provided to reduce the detector dead time, were taken into consideration. From the aforementioned results it was constructed an axial profile of the fission density normalized to the central point of the fuel rod. Subsequently, an irradiation of a dismountable fuel rod with a highly enriched ^{235}U metallic foil, sandwiched by two aluminum foils, in its center was performed. Next, the determination of the absolute ^{143}Ce counting of the gamma peak located in 293.4 keV of the ^{235}U metallic foil was determined by a calibrated detection system. The conversion of the ^{143}Ce counting to fission density and subsequently to specific power in the central 1 cm axial region of the central fuel rod was made in a straightforward fashion. The ^{238}U fission contribution to this fuel rod power and the correction factors for the foil perturbations were taken into account by a calculational approach employing MCNP-4C (Briesmeister, 2000). The aluminum foil thickness was chosen so that its correction factors were as close as possible to 1.0. The power of the central fuel rod was subsequently determined from the power of its central axial position by means of a numerical integral approach. From the power of the central fuel rod, the power for all other fuel rods and consequently the reactor power can be obtained in a straightforward fashion. The measured power was 108.7 W with an associated uncertainty of 2.5%. The error propagation incorporates the uncertainties of: the absolute ^{143}Ce counting of the ^{235}U metallic foil, the foil uranium mass and enrichment, the ^{143}Ce yield fraction, the ^{238}U contribution for the total power, the relative power density, and the correction factors for the perturbation of the aluminum and uranium foils.

3. Uncertainty analysis and results

For the uncertainty estimate in the β_{eff} measurements, the following uncertainties were assumed (at 1 σ):

- 1.0% for the ionization chambers currents readings (from Keithley manual).
- 1.0% for the gain of the amplifiers (as measured in this work).
- 1.0% for the current to voltage factor (from Keithley manual).
- 3.0% for the Diven factor (Avramov et al., 1990).
- 1.0% for the energy released per fission (estimated).
- 2.5% for the reactor power (as measured in this work).

The mean value, $\langle \Phi_{kl} \rangle$, of the CPSD in the plateau region, can be obtained by averaging all points in this region and taking the standard deviation of the mean as the respective uncertainty. However, since the CPSD has intrinsic uncertainties (error bars in each frequency point) due to the measurement procedure, it seems better to obtain the mean value by a weighted least-squares fit of a constant. In this way the mean value will have an uncertainty given only by the fitting procedure. For each frequency bin, the error bar is given by (Bendat and Piersol, 1986):

$$\varepsilon(\Phi_{kl}) = \frac{1}{\sqrt{\gamma_{kl} N}} \ (\%), \tag{3}$$

where γ_{kl} is the measured coherence function and N is the number of averages.

Finally, the total uncertainty on β_{eff} is given by standard error propagation through Eq. (1).

For the β_{eff}/Λ measurements, the uncertainty is obtained directly by a weighted least-squares fit of Eq. (2) where the weights are given again by Eq. (3).

Table 1 shows the results for β_{eff} and β_{eff}/Λ measurements at 100 W and 4.0 W. The power levels indicated are already corrected by the fuel rod scanning technique.

The final results for β_{eff} and β_{eff}/Λ can now be obtained through the arithmetic mean of the seven results of Table 1 and the standard error propagation, since these measurements can be considered as independent measurements. Also, the prompt neutron generation time, Λ, can be obtained by making $\Lambda = \beta_{eff} / (\beta_{eff}/\Lambda)$. These final results are shown in Table 2.

The reported uncertainties in the measurements performed in this work are much lower than those of SHE-8 and MISTRAL1 experiments and also much lower than the recommended accuracy (3%) for the β_{eff} calculations. Therefore, taken into consideration the extreme care placed in the design and execution of the experiments and the need of new experiments, it is felt that the measurements performed at the IPEN/MB-01 reactor can be recommend as benchmark values for the effective kinetic parameter validations.

4. Independent verification of the measured values

Regarding the β_{eff}/Λ, Spriggs carried out a Rossi-Alpha experiment at IPEN in 1997 (Spriggs et al., 1997) with two miniatures BF^3 detectors located inside the active core (not in the reflector, as in the case of noise experiments here) and the measured result, $\beta_{eff}/\Lambda = 232.9\ s^{-1}$, differs only 0.8% from the present noise result. In addition to that, a more careful set of Rossi-Alpha experiments is being performed nowadays at the IPEN/MB-01 reactor (Kuramoto and Dos Santos, 2005). In these experiments, the measurements are also performed with the detectors inside the active core but, differently from the previous one, the data include measurements in a state very close to critical (−3.5 pcm) which give a better accuracy and/or precision in the extrapolation to the $\rho = 0$ state. Also, the error bar at each experimental point is taken into account and there are a larger amount of experimental data points than those employed in the earlier experiment of Spriggs. The preliminary result of $(235.12 \pm 1.76)\ s^{-1}$ for the β_{eff}/Λ

Table 1
Results for β_{eff} and β_{eff}/Λ measurements at 100 W and 4.0 W corrected

P (W)	I_k (A)	I_l (A)	$\langle \Phi_{kl} \rangle$ (V^2/Hz)	β_{eff} (pcm)	β_{eff}/Λ (s^{-1})
108.71[a]	11.20E-6	11.25E-6	(9.69 ± 0.13)E-6	740 ± 18	233.71 ± 3.31
108.71[b]	11.20E-6	11.30E-6	(9.74 ± 0.09)E-6	740 ± 18	233.33 ± 1.97
108.71[c]	11.21E-6	11.25E-6	(10.04 ± 0.10)E-6	727 ± 18	232.75 ± 2.30
4.348[d]	456E-9	458E-9	(4.06 ± 0.04)E-5	736 ± 18	226.18 ± 2.17
4.348[e]	458E-9	460E-9	(3.96 ± 0.03)E-5	749 ± 18	228.36 ± 2.45
4.348[f]	456E-9	458E-9	(4.01 ± 0.02)E-5	741 ± 18	234.38 ± 1.63
4.348[g]	457E-9	459E-9	(4.05 ± 0.04)E-5	739 ± 18	228.28 ± 3.16

[a] 200 averages, span = 100 Hz and 800 lines of resolution.
[b] 340 averages, span = 100 Hz and 800 lines.
[c] 500 averages, span = 100 Hz and 400 lines of resolution.
[d] 300 averages, span = 100 Hz and 400 lines.
[e] 300 averages, span = 200 Hz and 800 lines of resolution.
[f] 500 averages, span = 100 Hz and 800 lines.
[g] 500 averages, span = 200 Hz and 400 lines of resolution.

Table 2
Final experimental results for β_{eff}, β_{eff}/Λ, and Λ

β_{eff} (pcm)	β_{eff}/Λ (s^{-1})	Λ (μs)
739 ± 7	231.00 ± 0.94	31.99 ± 0.33

parameter is 1.8% larger than the present noise result and shows a relatively good agreement. These agreements support the experimental results of this work and also show that for this kind of measurements the presence of the detectors in the reflector has negligible impact on β_{eff}/Λ. More precisely, there is no evidence that spatial effects are important for the β_{eff}/Λ experiments performed at the IPEN/MB-01 reactor.

Moreover, it should be noted here that the β_{eff}/Λ measurements through noise analysis do not depend on the magnitude of the CPSD but only on its shape, and thus, it seems to be a trustworthy measurement.

In the case of β_{eff}, the comparison is more restrictive since the independent verification is made with the results of a set of measurements also obtained through reactor noise analysis (Diniz and Dos Santos, 2006) and almost the same experimental conditions as in the present work. However, these measurements can be considered as independent ones. First, because the frequency range of data acquisition is not limited to the plateau region but it is from 0.005 to 50 Hz approximately in order to include the delayed neutrons contribution. Second, the function to be fitted is the complete CPSD of which Eq. (1) is a particular case when the delayed neutrons are disregarded. The fitting parameters are β_i or λ_i ($i = 1,\ldots,6$) and β_{eff} can be obtained from $\beta_{eff} = \sum_{i=1}^{6}\beta_i$. It should be noted, however, that if the parameters to be fitted are β_i, then the decay constants, λ_i, must be fixed as well as the first abundance β_1, the fixed parameters coming from some known nuclear data library. This is a limitation of the method. The advantage of this method relies on the fact that it allows the determination of the effective β without the need of the Diven factor and even the power normalization. The results showed a deviation of 1.6% when the decay constants are from ENDF/B-VI.8 (LANL review), 0.8% in the case of ENDF/B-VI.8 and 0.7% in the case of JENDL3.3 libraries relatively to the β_{eff} value of Table 2. It also should be stressed here that the spatial dependence of the β_{eff} measurements is less restrictive than that of the β_{eff}/Λ because of the lower frequency range of the former case. Therefore, the conclusions reached here are that the present measurements of β_{eff} are completely supported by independent experiments and the final result can be considered accurate enough for the nuclear data validation.

5. Theoretical determination of the effective delayed neutron parameters

For the point reactor model, the theoretical effective delayed neutron parameters to be compared to the experimental values can be defined following a standard mathematical approach (Bell and Glasstone, 1970):

$$\beta_{\mathrm{eff}\,j} = \frac{1}{F}\int\cdots\int \chi_{d_j}(E)\beta_j \nu\Sigma_f(r,E')\phi(r,\Omega',E')\phi^*(r,\Omega,E) \times \mathrm{d}r\,\mathrm{d}\Omega'\,\mathrm{d}E'\,\mathrm{d}\Omega\,\mathrm{d}E, \tag{4}$$

$$\Lambda = \frac{1}{F}\int\int\int \frac{1}{v(E)}\phi^*(r,\Omega,E)\phi(r,\Omega,E)\mathrm{d}r\,\mathrm{d}\Omega\,\mathrm{d}E, \tag{5}$$

$$F = \int\cdots\int \chi(E)\nu\Sigma_f(r,E')\phi(r,\Omega',E')\phi^*(r,\Omega,E) \times \mathrm{d}r\,\mathrm{d}\Omega'\,\mathrm{d}E'\,\mathrm{d}\Omega\,\mathrm{d}E, \tag{6}$$

where all the symbols follow the same meaning as in Bell and Glasstone, 1970.

The effective delayed neutrons fraction is the sum of $\beta_{\mathrm{eff}j}$ given by Eq. (4) for all j and β_{eff}/Λ can be calculated accordingly. The computational approach adopted in this paper will be twofold. At IPEN, the methodology commonly used for reactor calculations and analyses is based on the coupled systems HAMMER–TECHNION (Barhen et al., 1978) for the cross section generation and weighting and CITATION (Fowler et al., 1971) for the neutron diffusion in the reactor system. In an attempt to make a methodology based on transport theory, this work will also employ a novice approach based on the coupled NJOY/AMPX-II/TORT systems. NJOY is employed for the nuclear data treatment. AMPX-II is employed for the calculation of the region dependent few group cross sections. The self-shielding effects of the resolved resonances of ^{238}U and ^{235}U in the fuel unit cell are performed by the ROLAIDS module of AMPX-II. Module XSDRNPM also of AMPX-II is used for the cross section homogenization and condensation in a few group model for the several regions that constitute the IPEN/MB-01 reactor. TORT is employed for the solution of the forward and adjoint neutron transport equation in the reactor system. The advantages of this methodology as mentioned are the solution of the transport equation (forward and adjoint) and the flexibility to use several neutron groups. A specific computer program was written to perform the integrals shown in Eqs. (4)–(6). Both methodologies will be employed in this work, but only the one based on the coupled systems NJOY/AMPX-II/TORT will be used for the analysis of the experimental data.

Initially, Table 3 compares β_{eff}, β_{eff}/Λ and Λ calculated by both methodologies for the ENDF/B-VI.8 case. The same trend was found for the other libraries. Considering the same number of groups and S_N quadrature, there is practically no difference between the values of β_{eff} and β_{eff}/Λ of CITATION and TORT. The difference is pronounced for the β_{eff} case when the number of groups is increased. In this case, β_{eff}, β_{eff}/Λ calculated by TORT show important discrepancies compared to the results of CITATION. Considering the prompt neutron generation time (Λ), the agreement between the methodologies is much better. There is a slight trend with S_N order and to the number of groups. The 16-group structure used by TORT has

Table 3
Calculated results for β_{eff}, β_{eff}/Λ and Λ

Effective parameters	CITATION four groups	TORT		
		Four groups – S_2	Four groups – S_{16}	16 groups – S_{16}
β_{eff} (pcm)	779.85	779.99	773.79	792.41
β_{eff}/Λ (s^{-1})	262.31	261.93	265.18	267.05
Λ (μs)	29.73	29.78	29.18	29.67

five groups in the thermal energy region while the CITATION values consider just one. Therefore, the impact of the number of thermal groups in the determination of the prompt neutron generation time (Λ) is minimal.

Table 4 shows the results for β_{eff}, β_{eff}/Λ, and Λ calculated by TORT (S_{16} and 16 groups) for the nuclear data libraries considered in this work. ENDF/B-VI.8 and its revised version performed at Los Alamos produced essentially the same results. Here and also in Section 6, it is shown only the results of the latter library. Table 4 also shows for comparison purposes the corresponding results of MCNP-4C3 (Van der Marck et al., 2004 for ENDF/B-VI.8, and JENDL-3.3 results; Van der Marck, 2005 for that of JEFF-3.1). The comparison shows good consistency and reinforces the accuracy of the calculated results of TORT and makes the comparison with the experimental values more reliable. In the same table it is shown the total ^{235}U thermal yields of these libraries. β_{eff} and β_{eff}/Λ of the libraries follow roughly the same trend as their ^{235}U thermal yield. The prompt neutron generation time (Λ) has very little sensitivity to the nuclear data library used in the analysis. Therefore, the differences found in the β_{eff}/Λ are mainly due to β_{eff}. As shown in Table 4 the smallest values for the effective delayed neutron parameters β_{eff} and β_{eff}/Λ are given by JENDL3.3 which adopted a lower value of the ^{235}U thermal yield as proposed by Okajima and Sakurai (Yoshida et al., 2001). JEFF-3.1 adopted an intermediate value between JENDL3.1 and ENDF/B-VI.8 (LANL review). In the case of the IPEN/MB-01 reactor most of the fissions (nearly 86%) comes from the thermal neutron energy region where ^{235}U plays a major role. Consequently the results expressed in Table 4 are mainly the ^{235}U effect in the thermal neutron energy region. This aspect is very important for the nuclear data validation since the impact of a specific nuclear data of a fissile nuclide can become very evident. Therefore, the comparison of theory and experiment in the next section can address specific nuclear data needs of ^{235}U for the libraries analyzed in this work.

Table 4
Final calculated results for β_{eff}, β_{eff}/Λ, and Λ given by TORT (S_{16} and 16 groups)

		ENDF/B-VI.8[a]	JEFF-3.1	JENDL3.3
β_{eff} (pcm)	TORT	792.38	774.38	756.16
	MCNP-4C3	781.6 ± 4.1	771.7 ± 4.1	755.6 ± 4.0
β_{eff}/Λ		267.04	261.00	256.60
Λ (μs)		29.67	29.67	29.47
^{235}U thermal yield		1.670×10^{-2}	1.620×10^{-2}	1.585×10^{-2}

[a] LANL review.

6. Theory/experiment comparison

A comparison of β_{eff} and β_{eff}/Λ predicted by ENDF/B-VI.8 (LANL review), JEFF-3.1 and JENDL3.3 with the experimental values, in terms of C/E, is shown in Table 5. This table shows clearly that, for the β_{eff} case, JENDL3.3 library has the best performance while ENDF/B-VI.8 (LANL review) the worst one. JEFF-3.1 is in an intermediate stage. Only JENDL3.3 attends the recommended accuracy for β_{eff} calculations (|C-E|/E less than 3%). As stated, the lower value of β_{eff} of JENDL3.3 is due mainly to the lower value of the ^{235}U thermal yield. Therefore, a reduction of this nuclear data for the ^{235}U case as proposed by Okajima and Sakurai is completely supported by the experimental work of this paper. In contrast, the same performance does not occur for its β_{eff}/Λ. However, this behavior is due mainly to the prompt neutron generation time (Λ).

For the ENDF/B-VI.8 (LANL review) and even for JEFF-3.1, the deviations are relatively large for both β_{eff} and β_{eff}/Λ. The main reason for the β_{eff} discrepancy of these libraries is their higher values of the ^{235}U thermal yield. For the β_{eff}/Λ cases the reason for the discrepancies is due to both β_{eff} and Λ.

As mentioned, the calculated prompt neutron generation time shows very little sensitivity to the methodology employed, also to the nuclear data library used and to the number of groups as well. In a general sense when compared to the experimental value (31.99 ± 0.33 μs) it shows a systematic underprediction of around 8%. This is quite a surprising result since the prompt neutron generation time is a well-defined quantity. This comparison leaves the impression that the calculation of Λ has to be performed in a different way from the traditional multigroup method. Analyzing Eqs. (4) and (5), one may not that the parameter F is common in both equations. Since as shown in the analysis, the discrepancy found in β_{eff} is mainly a nuclear data problem related to the ^{235}U thermal yield, it may be conclude that the parameter F is nearly correct in both equations. Therefore, the source of the discrepancy of Λ may

Table 5
Comparison of the calculated β_{eff} and $\beta_{eff}/\Lambda$with the in-pile noise experiment

	ENDF/B-VI.8[a]	JEFF-3.1	JENDL3.3
β_{eff} (C/E)	1.072	1.048	1.023
β_{eff}/Λ (C/E)	1.156	1.130	1.108

[a] LANL review.

be attributed to the numerator of Eq. (5). The $1/v$ cross section used in Eq. (5) was obtained with the neutron flux as a weighting function. However, in order to preserve Eq. (5) in a multigroup model, the cross section $1/v$ should be weighted with the product of the forward and adjoint fluxes. Since the implementation of such approach is quite laborious in the computer code XSDRNPM, this aspect will be left as a suggestion for a future work.

7. Conclusions

The experimental determination of β_{eff} and $\beta_{\mathrm{eff}}/\Lambda$ of the IPEN/MB-01 reactor employing a reactor noise method has been successfully accomplished. The experimental results are in a very good agreement with independent measurements and the uncertainties are small enough so that the reactor noise method can be considered a good technique for this kind of measurements. Also, the measured values can be recommended as benchmarks to verify the adequacy of the calculational methods and related nuclear data libraries employed in the effective kinetic parameter determinations because their uncertainties are much lower than the target accuracy of β_{eff} calculations. In this aspect only JENDL3.3 attends the target accuracy. The results obtained in this work support the reduction of the ^{235}U thermal yield in order to have a better agreement between theory and experiment. This aspect is clearly seen in the performance of JENDL3.3 which has a lower value for the thermal yield of ^{235}U. In contrast, ENDF/B-VI.8 and its revised version performed at LANL overpredict β_{eff} by as much as 7.2%. JEFF-3.1 shows a performance that is in between JENDL3.3 and ENDF/B-VI.8. $\beta_{\mathrm{eff}}/\Lambda$ shows higher deviation for all libraries analyzed in this work. The main reason for that is the underprediction of the calculated prompt neutron generation time. This quantity shows a systematic underprediction of around 8%. The suggestions of this work are the incorporation of a lower ^{235}U thermal yield in the future versions of the ENDF/B and JEFF releases in order to have a better agreement of β_{eff} with experiments and also to weight the $1/v$ cross section with the product of the forward and adjoint fluxes in the calculation of the prompt generation time.

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