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Article

RTP TRIGA Reactor Kinetics Parameters from OpenMC

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Abstract

The current work uses Iterated Fission Probability (IFP) routine that was recently implemented in OpenMC to calculate reactor kinetics parameters. IFP is calculated from the product of the multiplication factors tracked across the $L+1$ generations of fission progenies. Since IFP is an excellent estimator of adjoint flux, it is able to calculate Λ_{eff}, β_{eff} and β_i of the reactor. OpenMC calculation of the reactor itself has $k_{\text{eff}} = 1.01687$ with an effective mean neutron generation time, $\Lambda_{\text{eff}} = 44.82\,\mu\text{s}$. The effective delayed neutron fraction, β_{eff} that we get is 0.007235 or 723.5 pcm. Other calculations of β_{eff} using prompt methods for reactors with similar designs gave us values between 724 pcm to 752 pcm. Our own calculations using the prompt method in OpenMC gave us an effective delayed neutron fraction of 734.1 pcm. The group β_i that we obtain is 24 pcm, 131.4 pcm, 124.1 pcm, 284.0 pcm, 112.7 pcm and 47.4 pcm respectively. If we strip away the influences of β_{eff} on β_i, by looking at only the abundances of each delayed neutron group, a_i; we are able to see that the a_i is similar to the abundances of just ^{235}U in the six group abundances of ENDF/B-VIII.0 evaluated cross section library. When we adopt a different evaluated cross section library in OpenMC, changes in β_i is due to the different β_{eff} and λ_i adopted in these libraries.

Keywords: nuclear fission; OpenMC; delayed neutron fraction; iterated fission probability; mean neutron generation time; reactor kinetics; TRIGA reactor

1. Introduction

A realistic simulation of a nuclear reactor requires not only a complete picture of the reactor spatial neutron distribution. More important is how the reactor power changes with time as a response to the movement of control rods. In Malaysia, efforts to model our TRIGA MARK-II research reactor relies on the point kinetics equation as a basis [1] for further approximations. Unfortunately, the available effective delayed neutron fraction, β_{eff} available for the simulator developer are restricted to the design values calculated in the first Safety Analysis Report [2]. Since then, various fuel designs have been added but most old fuels remain in the reactor. Hence, it is unrealistic to expect the β_{eff} of the reactor to remain the same. In the current work, we will use OpenMC to obtain kinetics parameters such as effective delayed neutron fraction, β_{eff} and effective mean neutron generation time, Λ_{eff}. Additionally, with the current Iterated Fission Probability routine in OpenMC, group parameters such as β_i could also be derived. With this group delayed neutron fraction, a more accurate modeling of the reactor power simulation could be developed in the future.

OpenMC is an open source continuous energy Monte Carlo code [3] typically used to solve steady state neutron transport problems. This meant that the solution of monte carlo code tends to focus on the spatial solution of neutron transport such as neutron flux distribution across the reactor for example. Reactor kinetics on the other hand solves neutron transport problems in the temporal domain. Typically, the reactor kinetics equation seeks to calculate the changes in reactor power with regards to the amount of reactivity inserted and the influence delayed neutrons on the reactor operation.

As such it seems counterintuitive to derive reactor kinetics parameters from steady state solution of the neutron transport. However, by solving the neutron transport equation we will be able to

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technically obtain the flux solutions (adjoint flux) from the initial neutron flux fed to the equation. From this point, it is possible to derive reactor kinetic parameters due to the changes we see as the neutron generation changes from one generation to the next.

Complicating things further however is the fact that the monte carlo approach only tracks the passage of neutron, giving us the densities of neutron distribution in the reactor. As such, the flux that we see in monte carlo is not the adjoint flux but something called the forward flux. However, it is possible to calculate neutron importances using Monte Carlo approach. Neutron importance is a measure of how strong the response of a neutron placed at the phase space position $\grave{} = (\vec{r}, E, \hat{\Omega})$ would have on the production of the next generation of neutrons. Even though neutron importances is not equivalent to adjoint neutrons, it was shown to be proportional to it [4]. This allowed for the development of iterated fission probability (IFP) approach [5] that was eventually introduced in OpenMC [6].

In the current work, we use OpenMC IFP routines to calculate *Reaktor TRIGA PUSPATI* (RTP) kinetics parameter. RTP is Malaysia's TRIGA MARK-II research reactor that has been in operation since 1983, with the first criticality at June of 1982. It has a maximum power of 1 MW but the lack of fresh fuel availability in the market constraints the range of operations. At present, the reactor is restricted to just the standard operation modes at power less than 1 MW. The first operational core is loaded with 83 TRIGA fuel elements. The number of fuel element has increased at present, where almost all 126 fuel locations are filled. This fuel designed by General Atomics is an $UZrH_{1.6}$ alloy. It was fabricated by hydriding UZr metal. All Uranium are enriched by ^{235}U to 20 wt% but the absolute amount of Uranium in the each fuel is in either the 8 wt%, 12 wt% or 20 wt% portions. Typically, the fuel with 8wt% Uranium content is called ST8 fuel. The 12 wt% Uranium content is called ST12 fuel. The 20 wt% Uranium content is called the LEU fuel. In the first core configuration however, the reactor is filled only with ST8 TRIGA fuel.

2. Theory and Methodology

2.1. Point Kinetics

The textbook nuclear reactor physics formalism [7] for the changes of neutron flux, ϕ as a function of position, r and time, t can be described by one speed, v neutron diffusion equation taking into account delayed neutron precursors as a source of neutron can be written as

$$\frac{1}{\mathrm{v}}\frac{\partial\phi(r,t)}{\partial t} - D(r,t)\nabla^2\phi(r,t) + \Sigma_a(r,t)\phi(r,t) = (1-\beta)\nu\Sigma_f(r,t)\phi(r,t) + \sum_{i=1}^{6}\lambda_i\hat{C}_i(r,t) \tag{1}$$

with the delayed neutron precursor concentration, $\hat{C}$ given as

$$\frac{\partial\hat{C}_i(r,t)}{\partial t} = \beta_i\nu\Sigma_f(r,t)\phi(r,t) - \lambda_i\hat{C}_i(r,t) \tag{2}$$

where $i = [1,\ldots,6]$ represents each delayed neutron groups. Here we assume that for neutrons produced in fission, the fraction of delayed neutrons is β. Each precursors group has an average half-life, λ_i and the fraction of neutron contributed by each precursor group is given by β_i.

If we assume that the flux in Equation (1) and precursor concentration in Equation (2) are separable,

$$\phi(r,t) = \mathrm{v}n(t)\psi_1(r) \tag{3}$$

$$\hat{C}_i(r,t) = C_i(t)\psi_1(r) \tag{4}$$

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The variable $\psi_1(r)$ is the fundamental solution for

$$\nabla^2\psi_n + B_g^2\psi_n = 0 \tag{5}$$

Defining the reactivity, $\rho(t)$ as the deviation of multiplication factor $k(t)$ from criticality,

$$\rho(t) = \frac{k(t)-1}{k(t)} = \frac{\nu\Sigma_f - \Sigma_a(1+L^2B_g^2)}{\nu\Sigma_f} \tag{6}$$

Given that the neutron diffusion length is defined as

$$L^2 = \frac{D}{\Sigma_a} \tag{7}$$

and mean neutron generation time is given as

$$\Lambda \equiv \frac{1}{\mathrm{v}\nu\Sigma_f} \tag{8}$$

Given that the multiplication factor is equivalent to,

$$k \equiv \frac{\nu\Sigma_f}{\Sigma_a}\frac{1}{1+L^2B_g^2} \tag{9}$$

It is possible to rearrange Equation (1) and Equation (2) to obtain the point kinetics equation

$$\frac{dn(t)}{dt} = \frac{\rho(t)-\beta}{\Lambda}n(t) + \sum_{i=1}^{6}\lambda_i C_i(t) \tag{10}$$

$$\frac{dC_i(t)}{dt} = \frac{\beta_i}{\Lambda}n(t) - \lambda_i C_i(t) \tag{11}$$

In the current work, we will try to derive the reactor kinetic parameters Λ, β and precursor groups β_i.

2.2. Kinetic Parameters

The fundamental delayed neutron fraction, β_0 is defined as the ratio delayed neutron production rate and total neutron production rate [8],

$$\beta_0 = \frac{P_d}{P} \tag{12}$$

The fundamental delayed neutron fraction is usually given for the fission of a particular fissioning isotope by a given neutron energy. For example a conventional nuclear reactor would produce most of its energy, hence its neutrons (delayed and prompt); from the fission of ^{235}U at thermal energies. At fast energies, the ratio Σ_f/Σ_a for ^{238}U improves but the reactor neutron energy spectrum are heavily skewed towards thermal neutron energies. If the effective delayed neutron fraction takes into account the production of delayed neutron from fission contribution coming from various locations, angles and energies [6],

$$\beta_{\text{eff}} = \frac{P_d^{\text{eff}}(\vec{r},E,\hat{\Omega})}{P^{\text{eff}}(\vec{r},E,\hat{\Omega})} \tag{13}$$

Defining implicitly that the flux and its adjoint is a function of $(\vec{r},E,\hat{\Omega})$, with χ, ν, Σ_f is a function of E, we can write the effective delayed neutron production as

$$P_d^{\text{eff}} = \int_V \oint \int_0^\infty \phi^+\chi_d\nu_d\Sigma_f\phi dE d\hat{\Omega} d\vec{r} \tag{14}$$

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and the total neutron production is,

$$P^{\text{eff}} = \int_V \oint \int_0^\infty \phi^+ \chi \nu \Sigma_f \phi(\vec{r}, E, \hat{\Omega}) dE d\hat{\Omega} d\vec{r} \tag{15}$$

At the same time, the mean neutron generation time is defined as the ratio of neutron effective removal time, ℓ_{eff} over effective multiplication factor, k_{eff},

$$\Lambda_{\text{eff}} = \frac{\ell_{\text{eff}}}{k_{\text{eff}}} = \frac{\int_V \oint \int_0^\infty \phi^+ \frac{1}{\text{v}} \phi dE d\hat{\Omega} d\vec{r}}{\int_V \oint \int_0^\infty \phi^+ \chi \nu \Sigma_f \phi dE d\hat{\Omega} d\vec{r}} \tag{16}$$

Here, the adjoint flux ϕ^+ is often described as the weight function determined by the importance of certain location and neutron energy towards the production of neutrons. However, the adjoint flux can also be described as the new flux distribution after a fission reaction. For further clarity, the probabilities expressed in Equation (13) are often written using Quantum Mechanics bra-kets. The total production of neutrons from fission can be expressed using quantum mechanical bra-kets as $\langle \phi^+ | \chi \nu \Sigma_f \phi \rangle$ or just $\langle \phi^+ | \hat{F} \phi \rangle$. Here $\hat{F}$ is the fission operator. The contribution from prompt neutrons can be similarly expressed using quantum mechanical bra-kets as $\langle \phi^+ | \chi_p \nu_p \Sigma_f \phi \rangle$ or simply $\langle \phi^+ | \hat{F}_p \phi \rangle$ with the prompt fission operator, $\hat{F}_p$. Because we can define the total neutron production as the sum of delayed and prompt neutrons, then

$$\frac{\langle \phi^+ | \hat{F}_d \phi \rangle}{\langle \phi^+ | \hat{F} \phi \rangle} + \frac{\langle \phi^+ | \hat{F}_p \phi \rangle}{\langle \phi^+ | \hat{F} \phi \rangle} = 1 \tag{17}$$

$$\beta_{\text{eff}} + \frac{\langle \phi^+ | \hat{F}_p \phi \rangle}{\langle \phi^+ | \hat{F} \phi \rangle} = 1 \tag{18}$$

$$\beta_{\text{eff}} = 1 - \frac{\langle \phi^+ | \hat{F}_p \phi \rangle}{\langle \phi^+ | \hat{F} \phi \rangle} \tag{19}$$

Effective delayed neutron fraction described by Eq: (19) is also called the prompt method [6,9,10]. In the prompt method, we can further expand Equation (19) by defining k_{eff} as,

$$k_{\text{eff}} = \frac{\langle \phi^+ | \hat{F} \phi \rangle}{\langle \phi^+ | \hat{L} \phi \rangle} \tag{20}$$

with $\hat{L}$ being the loss operator. This allow us to write Equation (19) in terms of k_{eff},

$$\beta_{\text{eff}} = 1 - \frac{\frac{\langle \phi^+ | \hat{F}_p \phi \rangle}{\langle \phi^+ | \hat{L} \phi \rangle}}{\frac{\langle \phi^+ | \hat{F} \phi \rangle}{\langle \phi^+ | \hat{L} \phi \rangle}} = 1 - \frac{k_{p,\text{eff}}}{k_{\text{eff}}} \tag{21}$$

In OpenMC v0.15.3, the settings can be setup to prevent delayed neutron creation by assigning

```
openmc.settings.create_delayed_neutrons=False
```

This will grant us effective multiplication factor from prompt neutrons, $k_{p,\text{eff}}$ when the eigenvalue calculation was engaged.

2.3. Iterated Fission Probability

Describing the phase space position as $\grave{} = (\vec{r}, E, \hat{\Omega})$, the initial fission source can be described by $\grave{}_0$. The estimator for the iterated fission probability, I_{FP} [5,6] is given by

$$I_{\text{FP}}(\theta_0) = k_0^{(1)} k_0^{(2)} \cdots k_0^{(L)} k_0^{(L+1)} = k_0^{(1)} k_0^{(2)} \cdots k_0^{(L)} k_{\text{eff}} \tag{22}$$

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where the iterated fission fission probability is defined as the product of $L+1$ number of progenies. At $L+1$ the progenies multiplication factor are assumed to converge to k_{eff}. Due to the proportionality of iterated fission probability with adjoint flux [4], we can simply replace ϕ^+ with I_{FP} yielding us

$$\beta_{\text{eff}} = \frac{\langle I_{\text{FP}}|\hat{F}_d\phi\rangle}{\langle I_{\text{FP}}|\hat{F}\phi\rangle} = \frac{w_d}{w} = \frac{S_d^{(L+1+\lambda)}}{S^{L+1+\lambda}} \tag{23}$$

The variables in this equation are described by,

$L+1$ number of progenies until convergence.

λ is the number of generations from the initial neutrons from its birth until to the time it was tallied.

w_d importance weighted expected number of descendant fission neutrons that originates from delayed neutron ancestor.

w importance weighted expected number of descendant fission neutrons that originates from both delayed and prompt neutrons.

S_d the number of descendants from delayed neutrons.

S total number of descendants.

Effective mean generation time [11] can be described just as easily as,

$$\Lambda_{\text{eff}} = \frac{1}{\text{v}} \frac{\langle I_{\text{FP}}|\phi\rangle}{\langle I_{\text{FP}}|\hat{F}\phi\rangle} \tag{24}$$

The implementation of IFP can be tricky because it required the user to track the movements of the neutron and its progenies. Luckily, this routine have been implemented recently in OpenMC as for the latest stable version 0.15.3 of OpenMC. Once the model of the reactor is established in OpenMC, we can simply use

```
core=openmc.model.Model.from_xml()
core.add_kinetics_parameters_tallies(num_groups=6)
```

in order to invoke the IFP methods for calculating Λ_{eff}, β_{eff} and the six groups $\beta_{\text{eff},i}$. With this routine, the default $\lambda = 10$ is used.

2.4. OpenMC Models

There are several OpenMC models built for the current purpose. Some for the sake of measuring the kinetics parameters with a different method, while others are modified to accommodate the different data libraries used. The basic geometries, fuel loadouts, material temperatures and control rod positions however are similar. Figure 1 depicts the loadout of the fuel in the first operational core configuration of the reactor.

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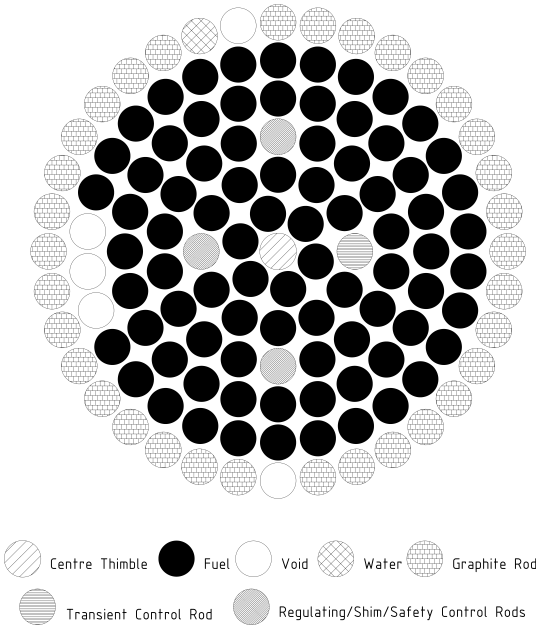

Figure 1. The loadout for the first core configuration of the RTP TRIGA MARK-II research reactor

As illustrated in the figure, ST8 TRIGA fuels filled the inner six annular rings of the core. At the center or at A-ring is the Center Thimble. The thimble could be voided for the purpose of experiment and sample irradiation but its often left flooded otherwise. As such in our model, the Center Thimble is often filled with water. The first ring of fuels is the B-ring. It's filled with six ST8 fuels. Then, the next ring is the C-ring. The first two control rods are placed here.

Two types of control rods are used in the reactor. The first one is called the Regulating control rods. It consist of two parts, B_4C at the top and a fuel component at the bottom. This fuel component of the Regulating control rod has approximately 160 g of Uranium enriched up to 20 wt%. On the opposite side of the Regulating Control rods, Transient control rods are raised and lowered using air pressure. When raised, the upper B_4C part of the rods are raised, allowing nuclear reactions to occur at the bottom part of the controls that were filled only with air.

Pulsing operations are enabled in the reactor due to the ability of the Transient rods to withdraw rapidly. Control rods with B_4C at the top and fuel at the bottom is also called a fuel follower control rods (FFCR). Transient control rods meanwhile are called an air follower control rods (AFCR) because the top of the control rods components is B_4C rods but the bottom are just filled with air.

In the D-ring, ST8 fuels filled up 16 position of the available 18 positions. The other two positions are filled with FFCR control rods. The two FFCR rods are placed opposite to each other, the bottom one in D-ring in Figure 1 is the Shim rod and the top ones is the Safety rod. In the model, the Transient rod is raised 41.72 cm and the Regulating rod is raised 27.49 cm. Both Shim and Safety rod in the D-ring is raised to around 26.4 cm. With these rod positions, the reactor is able to be operated at 1 MW. At these conditions, the bulk water temperature is around 32 °C.

The E-ring 24 positions are fully filled with ST8 fuels. F-ring has 30 positions, three of the positions are void inside for the pneumatic rabbit system for rapid insertion and withdrawal of samples into the reactor core. The G-ring are filled with graphite rods except for two void and one water locations. Using the fission tally, actual number of fission at 1 MW could be calculated. The scaling factor is obtained from the ratio of total number of fission at 1 MW over the sum of all the fission tally. By multiplying the number flux tally with the scaling factor, we can estimate the actual neutron flux in the reactor as shown in Figure 2.

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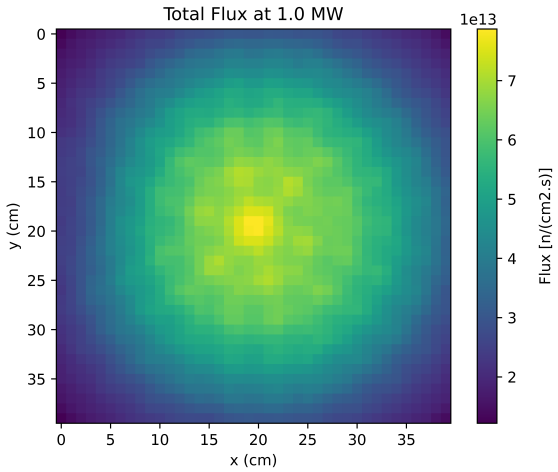

Figure 2. Flux profile of OpenMC model of RTP first core configuration at 1 MW.

3. Results

In the preliminary safety analysis report [2] for RTP, reactor parameters are calculated using $\beta_{\text{eff}} = 0.007$ and $\Lambda_{\text{eff}} = 39\,\mu\text{s}$. The design assumes operational loading of approximately 80 fuel elements with $k_{\text{inf}} = 1.3427$ and $k_{\text{eff}} = 1.0085$ at average fuel temperature of 400 °C and average water temperature of 23 °C.

Previous work [10] gave us $\beta_{\text{eff}} = 0.00724$ for the MCNP model of the RTP on the same first core configuration, calculated using a slightly modified prompt method. On the other hand, we obtained $\beta_{\text{eff}} = 0.007341$ with our naive implementation of the prompt method using OpenMC as described in Equation (19). These results are summarized in Table 1.

Table 1. RTP kinetics parameter calculation with OpenMC† compared to design parameters in the Safety Analysis Report [2,12] and prompt method results [10,13].

Parameters	Λ_{eff} [μs]	β_{eff}	k_{eff}
Core Excess Exp.			1.03512
Design (400 °C Fuel) [2]	38.67	0.00700	1.0085
Design (23 °C Fuel) [2]	37.28	0.00700	1.0460
Prompt Method [10]		0.00724	1.00280
Prompt Method [13]	42.00	0.00752	1.00588
Prompt Method†		0.007341	1.01687
IFP†	44.82	0.007235	1.01687

†Current calculation results with OpenMC

The discrepancy seen in comparison to design β_{eff} might be due to the differences in nuclear data used. More importantly, the exact loading of the fuel, specific design feature of the reactor and the temperature of the materials from model to model also contributes the discrepancy of the results. For example, if we compare with the results from the Slovenia TRIGA MARK-II reactor [13], notable differences could be seen in the values of β_{eff} due to different specific model being adopted.

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Table 2. Calculated six groups, $i = 1 \cdots 6$, effective delayed neutron fraction, $\beta_{\text{eff},i}$ and the corresponding abundances, a_i. Calculation are performed using the evaluated ENDF/B-VIII.0 cross section data.

i	$\beta_{\text{eff},i}$	a_i	$^{235}\text{U}\ a_i^{\dagger}$	$^{238}\text{U}\ a_i^{\dagger}$
1	0.000240	0.03314049	0.035008	0.013938
2	0.001314	0.18159754	0.180698	0.112797
3	0.001241	0.17151105	0.172510	0.131027
4	0.002840	0.39251749	0.386782	0.385143
5	0.001127	0.15570326	0.158575	0.253992
6	0.000474	0.06553017	0.066427	0.103103

† Abundances data from ^{235}U from ENDF/B VIII.0 [14].

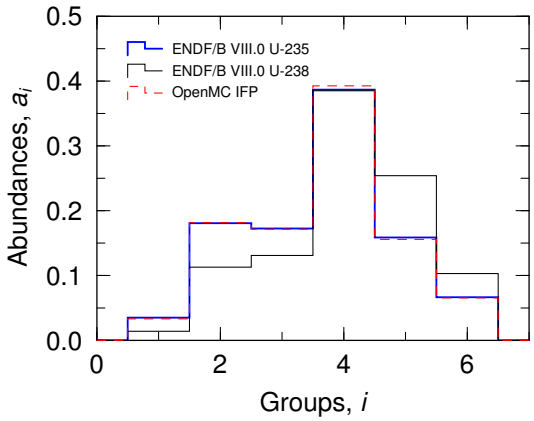

Figure 3. In (——) is the a_i for ^{235}U and (——) is the a_i for ^{238}U taken directly from the evaluated nuclear data [14]. Evaluated nuclear data is compared with (– – –) calculated using OpenMC IFP with ENDF/B-VIII.0 for the reactor model of RTP first core configuration.

One advantage the IFP approach has over prompt method in OpenMC is that we are able to extract the six group values of the effective delayed neutron fraction, $\beta_{\text{eff},i}$ and the corresponding group abundances. In Table 2, we use $\beta_{\text{eff},i}$ from our OpenMC model to calculate the abundances of the group with the relationship,

$$a_i = \frac{\beta_{\text{eff},i}}{\beta_{\text{eff}}} \tag{25}$$

In this table, the abundances a_i is compared with ENDF/B-VIII.0 nuclear data for ^{235}U and ^{238}U. This is further visualized in Figure 3, where we can see that the abundances from our OpenMC IFP results are very close to the a_i of ^{235}U. While it was not exact, the remarkable closeness of the calculated results from the OpenMC model and evaluated nuclear data for ^{235}U indicates that the neutron flux spectrum are dominated by thermal spectrum. This is because ^{238}U+n will only give significant number of fission when $E_n > 0.5\,\text{MeV}$ [15] are available in the reactor.

As a sanity check, we compare the results of our calculations using ENDF/B-VIII.0 with other evaluated nuclear data libraries. In Table 3, we see a small yet definite deviations when comparing with ENDF/B-VII.1 evaluated cross section libraries. The k_{eff} calculated with ENDF/B-VIII.0 gave a higher value of k_{eff} than calculations made with ENDF/B-VII.1 but against calculations made with other evaluated data; the k_{eff} is lower. Other than k_{eff} and β_2, all other parameters calculated with ENDF/B-VII.1 gave a higher value for the corresponding parameter. Our calculation with JENDL 5.0 on the other hand while it also give a higher value for k_{eff} and it overestimates β_2, β_3 and β_4.

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Table 3. Reactor kinetics parameters result calculated using various evaluated OpenMC cross section data.

Parameters	ENDF/B-VIII.0	ENDF/B-VII.1	JENDL 5.0
k_{eff}	1.01687	1.01518	1.02134
Λ_{eff}	44.82 µs	45.46 µs	44.45 µs
β_{eff}	0.007235	0.007270	0.007214
$\beta_{\text{eff},1}$	0.000240	0.000255	0.000230
$\beta_{\text{eff},2}$	0.001314	0.001254	0.001559
$\beta_{\text{eff},3}$	0.001241	0.001252	0.001405
$\beta_{\text{eff},4}$	0.002840	0.002866	0.002854
$\beta_{\text{eff},5}$	0.001127	0.001159	0.000838
$\beta_{\text{eff},6}$	0.000474	0.000484	0.000327

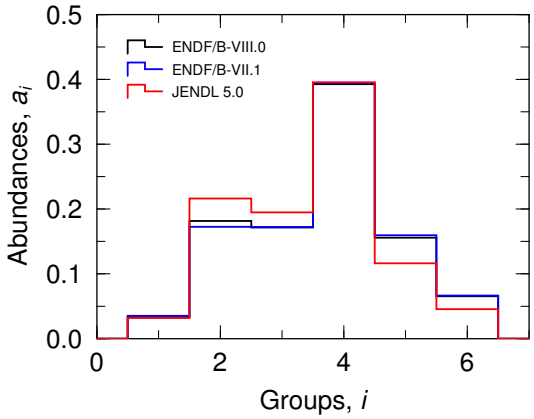

Figure 4. Group abundances calculated from OpenMC IFP routines using ENDF/B-VIII.0 (——), ENDF/B-VII.1 (——) and JENDL 5.0 (——) cross section libraries.

Comparing β_i calculated with the various evaluated cross sections libraries are complicated because it is dependent on the value of its sum, β of each evaluated cross section library. Thus, if we divorce the coupling between β_i and β_{eff}; we can see a clearer picture of the influence of the various groups on the kinetics parameters by looking at the abundances of each groups. Figure 4 plots the abundances of each groups calculated using OpenMC IFP routines using all four evaluated cross section libraries. By looking at just the abundances, its clear that there is only minute discrepancy between results calculated with ENDF/B-VIII.0 and ENDF/B-VII.1 evaluated cross section libraries. Thus, the deviations between these two libraries that we see in Table 3 are completely explained by the differences in β_{eff} values of ENDF/B-VIII.0 and ENDF/B-VII.1 cross sections only. The errors of a_i between these two cross sections are less than 6 %.

Comparing the abundances calculated using ENDF/B-VIII.0 with JENDL 5.0, we see that JENDL 5.0 results for β_2, β_3, β_5 and β_6 are very different from ENDF/B-VIII.0 with differences between the two reaching as much as 30 % for β_6. JENDL 5.0 overestimate ENDF/B-VIII.0 abundances for β_2 and β_3 and underestimates the abundances for β_5 and β_6. Most of the deviations appear due to the differences in the group half-lives averages. In the case of the isotope ^{235}U ENDF/B-VIII.0, the average group half-lives, $T_{1/2,i}$ are 51.98 s, 21.17 s, 5.739 s, 2.289 s, 0.816 s and 0.243 s. The current author are not able to find the $T_{1/2,i}$ for ^{235}U in JENDL 5.0, but from JENDL 4.0, $T_{1/2,i}$ are 55.72 s, 22.70 s, 6.222 s, 2.300 s, 0.610 s and 0.230 s.

4. Conclusions

In the current work, kinetic parameters such as Λ_{eff}, β_{eff} and the respective group β_i are calculated using OpenMC's recently introduced Iterated Fission Probability (IFP) routines. We note that the exact model implemented has a huge impact on the exact value of β_{eff} and Λ_{eff} that we can expect. Design values [2,12] assign $\beta_{\text{eff}} = 0.0070$ for fuel loading of around 80 fuel elements. However, if the fuel elements are swapped with with an aluminum clad, $\beta_{\text{eff}} = 0.0073$ is given [12]. The current IFP

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results of $\beta_{\text{eff}} = 0.007235$ when viewed with respect to design β_{eff}; is still within expectation given the differences in exact loading. In fact, given that the OpenMC model geometry was constructed with no foreknowledge of other monte carlo models [10,13] geometries and assumptions, attempting to calculate the same parameter, the level of agreement in β_{eff} values is fascinating.

One advantage of the IFP approach as opposed to the prompt method is the access to the respective group delayed neutron fraction. With this information, it is possible implement these parameters for use in point kinetics equation. There is no design values that we can compare for group delayed neutron fraction, so comparison between evaluated cross section libraries were performed. What we can conclude is that the differences in β_i boils down to the differences in β_{eff} and λ_i.

With regard to the current work, we have been successful in calculating β_{eff}, Λ_{eff} and the respective group β_i values. Using the control rod calibration data obtained in the early years of the reactor, we can extract information on how much reactivity available to the reactor for a given control rod positions. As such, it is possible to simulate the power of the reactor using point kinetics equation to confirm the power movements of the reactor seen in the early years. Later on, by performing depletion calculation for each successive core configurations; it is possible to derive the kinetics parameters of these core configurations. We hope eventually, we will be able to develop a reactor power simulation for all these core configurations.

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