KINETIC PARAMETERS OF THE DALAT NUCLEAR RESEARCH REACTOR WITH LEU FUEL USING MCNP6 AND JENDL-5 LIBRARY

by

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Kinetic parameters of a nuclear reactor are essential in reactor dynamic and safety related characteristics. Kinetic parameters of the Dalat nuclear research reactor were evaluated using the MCNP6.3 code and a new nuclear data library (JENDL-5). Numerical calculations were performed for the core configuration consisting of 92 low-enriched uranium fuel bundles for obtaining the effective delayed neutron fraction $\beta_{\rm eff}$, the neutron generation time Λ and the prompt neutron lifetime $l_{\rm p}$. Two methods were used to calculate the $\beta_{\rm eff}$: the adjoint weighted method based on perturbation theory and adjoint flux, and the prompt method using the ratio of prompt and total fission neutrons. To calculate the Λ and the $l_{\rm p}$, the adjoint weighted method and the $1/v$ absorber insertion method were used. The values of $\beta_{\rm eff}$, Λ and $l_{\rm p}$, at the beginning of cycle and the end of cycle obtained with the adjoint weighted method are 760 and 708 pcm, 86.99 and 82.41 μs, 79.01 and 82.76 μs, respectively. Comparing among the methods, the kinetic parameters were predicted with the discrepancy of about 1.0-1.6 %. The kinetic parameters will be utilized in the safety analysis report of the Dalat nuclear research reactor.

Key words: Dalat nuclear research reactor, low-enriched uranium fuel, kinetic parameter, MCNP6, JENDL-5

INTRODUCTION

Kinetic parameters of a research reactor such as effective delayed neutron fraction ($\beta_{\rm eff}$), neutron generation lifetime (Λ), and prompt neutron lifetime (l_p) are among important characteristics related to the reactor safety, kinetic and transient analysis [1-9]. Delayed neutrons in a reactor core are generated during the decay process of radionuclides which are the products of fission reactions. For example, neutrons emitted during the beta decay of iodine, bromine and other nuclei. The temporal delay of neutrons is dependent on the lifetime of fission products. Though the delayed neutrons occupy less than 1 % of the total neutron population, they have a significant effect on the safety related characteristics of the reactor core [9, 10-12]. Kinetic parameters are influenced by burnup, fuel type, core configuration and other reactor operation conditions [3, 5, 10, 11]. Thus, precise evaluation of the kinetic parameters contributes to improving the safety and the effective operation of the reactor. Several studies were conducted to estimate the kinetic parameters of various research reactors using continuous-energy Monte Carlo transport codes [1-3, 8, 10, 13-21]. Kinetic parameters of the 20-MW D_2O-moderated research reactor (NBSR) were estimated using the MCNP5 code and the ENDF/B-VII library, which yielded the most accurate estimates for the delayed neutron fraction and the prompt neutron lifetime [22]. Assessment of the kinetic parameters was conducted for the initial core of the Indonesian 30-MW Multipurpose Research Reactor using continuous energy Monte Carlo codes (MVP3 and MCNP-6.2) and the nuclear data libraries of JENDL-4.0, ENDF/B-VII.0 and ENDF/B-VII.1. The results indicated that the MVP3 calculations with the ENDF/B-VII.0 library agreed well with those obtained using the MCNP-6.2 (JENDL-4.0, ENDF/B-VII.0, and ENDF/B-VII.1) and the MVP3 (JENDL-4.0) results [23]. Kinetic parameters were estimated for the Ghana Research Reactor 1 (GHARR-1) after 19 years of operation using MCNP5. The results aligned closely with the initial Safety Analysis Report [24]. Kinetic parameters based

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on burnup were investigated to improve the performance of a research reactor [3]. It was reported that the β_{eff} decreased with burnup, while the l_{p} exhibited an inverse tendency [3].

The Dalat Nuclear Research Reactor (DNRR) is a 500 kW pool-type research reactor, operated by Dalat Nuclear Research Institute (Dalat, Vietnam). In 1983, the DNRR core was upgraded from the TRIGA-MARK II research reactor with the power of 250 kW and loaded with highly enriched uranium (HEU) fuel of the Russian WWR-M2 type. In 2011, the core was completely converted to low-enriched uranium (LEU) fuel with the ^{235}U enrichment of 19.75 wt. % [25]. The operation of the DNRR is essential for the development of many scientific activities and applications including nuclear physics, nuclear engineering, biology and agriculture, as well as radioactive isotope production for medical and industrial applications [25-30].

To support the operation of the DNRR, core physics and safety-related analysis have been performed based on the ENDF/B and JENDL data libraries. However, the kinetic parameters of the DNRR have not been reported in any publication. Thus, it is beneficial to conduct calculations of the kinetic parameters of the current DNRR core using a Monte Carlo method code and the latest data library. The present work reports the kinetic parameters of the DNRR core with LEU fuel using the MCNP6.3 code [31] and the latest JENDL data library (JENDL-5) [32]. Two methods used to calculate the β_{eff} were the adjoint weighted method and the prompt method. Whereas, to calculate the l_{p}, the adjoint weighted method and the $1/v$ absorber insertion method were used. The adjoint weighted method based on perturbation theory and adjoint flux, and the prompt method using the ratio of prompt and total fission neutrons. The $1/v$ absorber insertion method is based on the assumption of the linear dependence of the reactivity with a small ^{10}B content uniformly inserted into the core. The adjoint weighted method provides a high-fidelity evaluation, while the prompt method and the $1/v$ absorber insertion method serve as comparative baselines. The kinetic parameters were evaluated at both the beginning of cycle (BOC) and the end of cycle (EOC) of the DNRR core in comparison among the methods. By comparing among the methods, the reliability and the accuracy of the calculated kinetic parameters of the DNRR are ensured.

THE DALAT NUCLEAR RESEARCH REACTOR AND CALCULATION MODEL

The Dalat Nuclear Research Reactor

The DNRR consists of a cylindrical aluminum tank of 6.26 m in height and 1.98 m in diameter. The active core has a cylindrical shape with a height of 60 cm and the diameter of 44.2 cm. The core includes 121 hexagonal cells for loading fuel bundles, control rods, irradiation channels and beryllium blocks. The control rod system includes: one automatic regulating rod (AR), four shim rods (ShR) and two safety rods (SR). The absorption length of the control rods is 65 cm to completely cover the active core. The dry and wet irradiation channel are covered by aluminum cylinders with a thickness of 0.5 mm. The neutron trap at the core center is a water cylinder with 6.5 cm in diameter and 60 cm in length surrounded by twelve beryllium blocks. The beryllium block has the same outer shape as the fuel bundle. A beryllium ring is located between the active core and the graphite reflector and serves as an additional reflector. The thickness of the graphite reflector is 30.5 cm. Further description of the DNRR core can be found in [25-30]. Figure 1 shows the core configuration of the DNRR consisting of 92 LEU fuel bundles [1]. Table 1 provides the main design specifications of the DNRR core and the VVR-M2 LEU fuel [25].

Figure 2 displays the cross-sectional view and the design parameters of the LEU fuel bundle having

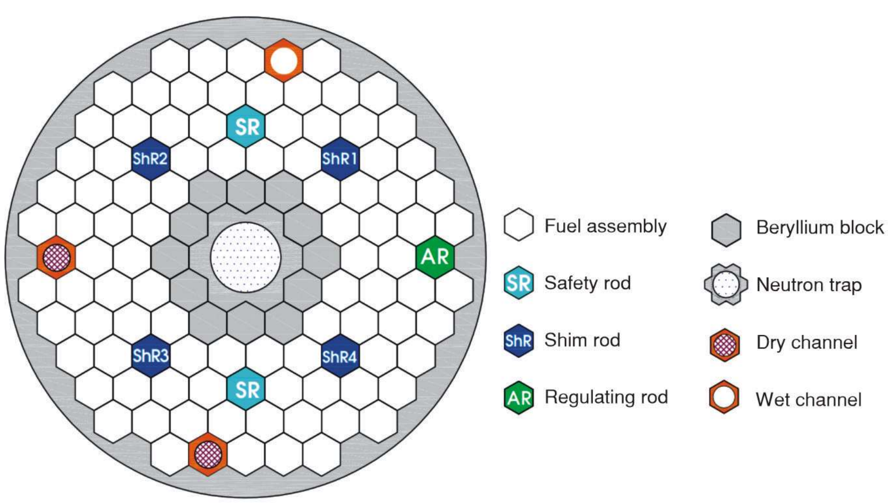

Figure 1. Configuration of the DNRR core with 92 LEU fuel bundles

Table 1. Design specifications of the DNRR core with LEU fuel

Parameter	Value
Reactor	
Reactor type	Pool type
Fuel element type	Russian VVR-M2
Cooling system	Natural convection
Moderator and coolant	Water
Reflector	Graphite, beryllium and water
Nominal power [kW]	500
Core	
Number of fuel bundles	92
Regulating rod (Stainless steel)	1
Shim rod (B_4C)	4
Safety rod (B_4C)	2
Beryllium rod	12
Irradiation channel	4
Fuel	
^{235}U enrichment (wt.%)	19.75
Number of fuel elements in a bundle	3
Outermost element (Hexagonal shape)	1
Inner elements (Circular shape)	2
Fuel element thickness [mm]	2.5
Fuel meat thickness [mm]	0.94
Cladding thickness [mm]	0.78
Fuel clad material	Al
Fuel meat material	Al-UO_2 alloy
Coolant channel width [mm]	2.5-3
Total length of a fuel bundle [mm]	865
Active fuel length [mm]	600

three concentric annular tubes, along with a header and a tail. The outermost fuel tube is hexagonal, with parallel sides of 32 mm in width. The two inner fuel elements have circular shapes, with the outer diameters of 22 mm and 11 mm, respectively. The fuel employed comprises a mixture of uranium dioxide and aluminum (Al-UO_2).

Calculation model

Kinetic parameters of the DNRR core with LEU fuel were calculated using the MCNP6.3 code [31] and the JENDL-5 nuclear data library [32]. Figure 3 displays the vertical and horizontal cross-sectional views of the DNRR core model in the MCNP6.3 code. The model is expanded from the active core to the reactor tank, with dimensions of 198 cm in diameter and 184.5 cm in height. The MCNP6 calculations were conducted with 200 cycles and 10^6 neutron histories per cycle for obtaining the statistical error of k_{eff} less than 20 pcm (1 pcm = 10^{-5}). The kinetic parameters were calculated at the beginning of cycle (BOC) and the end of cycle (EOC) to evaluate the effect of fuel burnup.

KINETIC PARAMETERS

Effective delayed neutron fraction

Radionuclides emitting delayed neutrons are called the delayed neutron precursors. Depending on the half-life of the radionuclides, the delayed neutron intensity and their effect also change with time. Thus, the delayed neutron precursors are divided into six groups based on their half-life. The delayed neutron fraction in a precursor, β_i, is defined as the ratio of the delayed neutrons in this group to the total neutrons

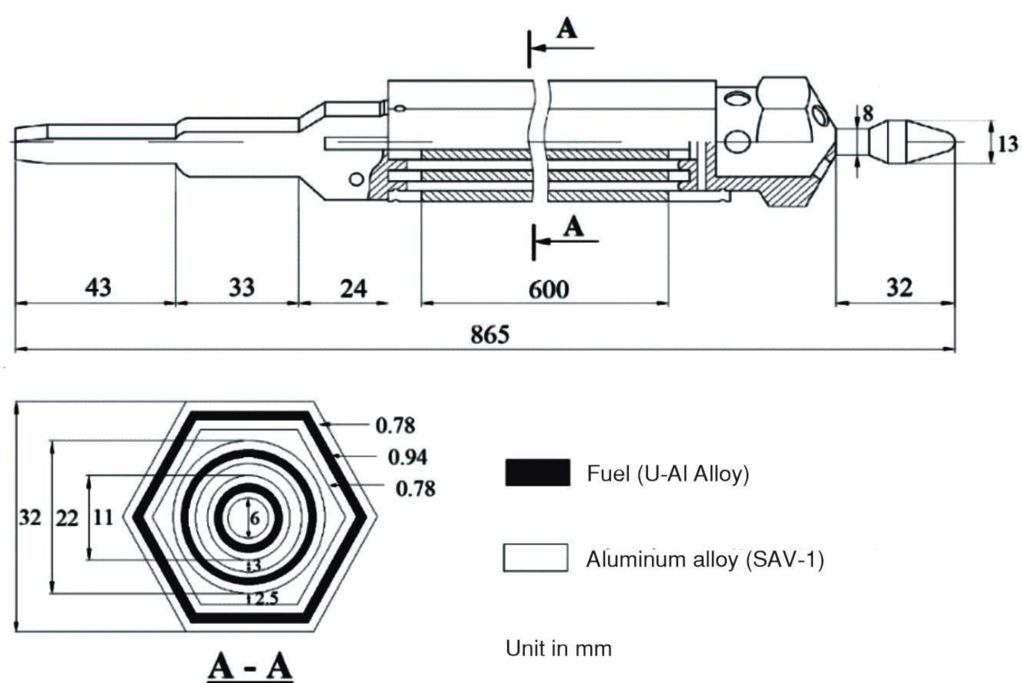

Figure 2. Cross-sectional view of the Russian VVR-M2 type LEU fuel bundle

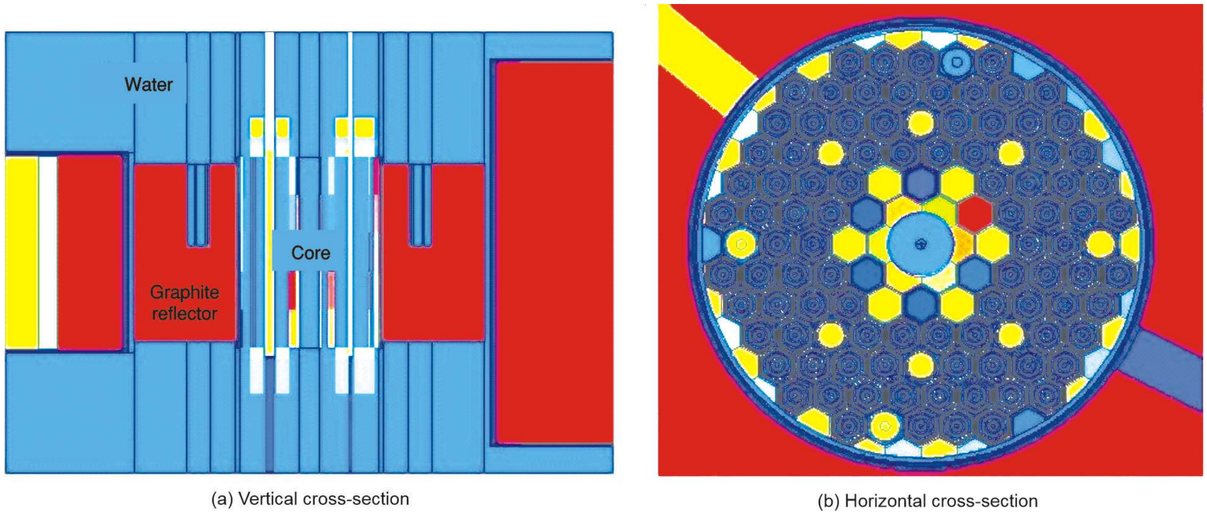

Figure 3. Vertical (a) and horizontal (b) cross-sectional views of the DNRR core model in the MCNP6 code

generated in a fission reaction. The effective delayed neutron fraction, denoted as $\beta_{\rm eff}$, is the sum of the delayed neutron fractions over the six precursors. Two methods employed to calculate the effective delayed neutron fraction $\beta_{\rm eff}$ were the adjoint weighted method and the prompt method.

Adjoint weighted method

The equation to calculate the effective delayed neutron fraction $\beta_{\rm eff}$ is written as [13, 15, 33]

$$\beta_{\rm eff} = \frac{(\phi^+ B\phi)}{(\phi^+ F\phi)} \tag{1}$$

where ϕ and ϕ^+ are the forward and adjoint neutron fluxes, respectively, B – the delayed neutron operator, F – the fission term operator, $\langle.\rangle$ – the integration over space and energy.

The terms in the numerator and denominator of eq. (1) can be elaborated as follows

$$\left\langle \phi^+ B\phi \right\rangle = \int \phi^+(\vec r, E, \vec\Omega) \textstyle\sum_f (\vec r, E')\nu_d(E') \chi_d(E', \vec\Omega' \to E, \vec\Omega)\phi(\vec r, E', \vec\Omega) {\rm d}E'{\rm d}\vec\Omega'{\rm d}E{\rm d}\vec\Omega{\rm d}\vec r \tag{2}$$

and

$$\left\langle \phi^+ F\phi \right\rangle = \int \phi^+(\vec r, E, \vec\Omega) \textstyle\sum_f (\vec r, E')\nu(E') \chi(E', \vec\Omega' \to E, \vec\Omega)\phi(\vec r, E', \vec\Omega) {\rm d}E'{\rm d}\vec\Omega'{\rm d}E{\rm d}\vec\Omega{\rm d}\vec r \tag{3}$$

where $\nu(E')$ and $\nu_{\rm d}(E')$ are the average of total and delayed neutrons at energy E' generated per fission, respectively; $\chi(E', \vec\Omega' \to E, \vec\Omega)$ and $\chi_d(E', \vec\Omega' \to E, \vec\Omega)$ are the spectra of energy and angular distribution $(E', \vec\Omega)$ of the total and delayed neutrons produced by incoming neutrons with $(E', \vec\Omega')$, respectively; Σ_f – the macroscopic fission cross section, $\vec r$ – the neutron position; E – the neutron energy; $\vec\Omega$ is neutron direction. From eqs. (1-3), the effective delayed neutron fraction, $\beta_{\rm eff}$ can be written as

$$\beta_{\rm eff} = \frac{\int \phi^+(\vec r, E, \vec\Omega) \sum_f (\vec r, E')\nu_d(E') \chi_d(E', \vec\Omega' \to E, \vec\Omega)\,\phi(\vec r, E', \vec\Omega) {\rm d}E'{\rm d}\vec\Omega'{\rm d}E{\rm d}\vec\Omega{\rm d}\vec r}{\int \phi^+(\vec r, E, \vec\Omega) \sum_f (\vec r, E')\nu(E') \chi(E', \vec\Omega' \to E, \vec\Omega)\,\phi(\vec r, E', \vec\Omega) {\rm d}E'{\rm d}\vec\Omega'{\rm d}E{\rm d}\vec\Omega{\rm d}\vec r} \tag{4}$$

To simplify eq. (4), the operator $\langle.\rangle$ is used to carry out the integration over the spatial, angular, and energy variables. This integration is expressed as eq. (5) using simplification

$$\beta_{\rm eff} = \frac{\left\langle \phi^+, \chi_d \nu_d \Sigma_f \phi \right\rangle}{\left\langle \phi^+ \chi \nu \Sigma_f \phi \right\rangle} \approx \frac{\langle \chi_d \nu_d \rangle}{\langle \chi\nu \rangle} \tag{5}$$

Spriggs reformulated the $\beta_{\rm eff}$ from the above equation to concentrate only on quantifying the delayed neutron fraction, hence, the k-ratio is subsequently introduced as follows [13, 34]

$$\beta_{\rm eff} \approx \frac{\langle \chi_d \nu \rangle \langle \chi_d \nu_d \rangle}{\langle \chi_d \nu \rangle \langle \chi \nu \rangle} \approx \approx \frac{\langle \chi_d \nu_d \rangle \langle \chi_d \nu \rangle}{\langle \chi_d \nu \rangle \langle \chi \nu \rangle} \approx \beta_0 \frac{\langle \chi_d \nu \rangle}{\langle \chi \nu \rangle} \approx \beta_0 \frac{k_d}{k} \tag{6}$$

The calculation of β_0 using MCNP code is straightforward and is accomplished by counting the total and delayed neutrons generated during fission events, whereas k_d is derived from the delayed neutron energy spectrum.

Prompt method

From eq. (5), one can rewrite the expression for $\beta_{\rm eff}$ as follows [13]

$$\beta_{\text{eff}} \approx \frac{\langle \chi_d v_d \rangle}{\langle \chi v \rangle} = 1 - \frac{\langle \chi v - \chi_d v_d \rangle}{\langle \chi v \rangle} =$$

$$= 1 - \frac{\langle \chi v_d - (\chi_d - \chi) v_d \rangle}{\langle \chi v \rangle} \cong 1 - \frac{\langle \chi_d v_d \rangle}{\langle \chi v \rangle} \tag{7}$$

The estimation in the final phase is founded on the subsequent rationale. The term $(\chi_d - \chi) v_d$ is two orders of magnitude less than the one with χv_p, because v_d is two orders of magnitude smaller than v_p. Due to the same rationale, the shape of χ is almost equal to that of χ_p. At this point, an important step is carried out. The β_{eff} can be estimated as follows

$$\frac{\chi_p v_p}{\langle \chi v \rangle} = \frac{k_p}{k} \to \beta_{\text{eff}} \approx 1 - \frac{k_p}{k} \tag{8}$$

where k is the eigenvalue for all neutrons and k_p – the eigenvalue for prompt neutrons only.

Neutron generation time and prompt neutron lifetime

Adjoint weighted method

The neutron generation time Λ, and the prompt neutron lifetime l_p are calculated based on the following equations [13, 15, 33]

$$\Lambda = \frac{\left\langle \phi^+ \dfrac{1}{\upsilon} \phi \right\rangle}{\left\langle \phi^+ F \phi \right\rangle'} \tag{9}$$

In which

$$\left\langle \phi^+ \frac{1}{\upsilon} \phi \right\rangle =$$

$$= \int \phi^+ (\vec{r}, E, \vec{\Omega}) \frac{1}{v(E)} \phi(\vec{r}, E', \vec{\Omega}') \,\mathrm{d}E' \mathrm{d}\vec{\Omega}' \,\mathrm{d}E \mathrm{d}\vec{\Omega} \mathrm{d}\vec{r} \tag{10}$$

From equations (3), (9), and (10), the neutron generation time (Λ) can be estimated by eq. (11)

$$\Lambda = \frac{\int \phi^+ (\vec{r}, E, \vec{\Omega}) \dfrac{1}{v(E)} \phi(\vec{r}, E', \vec{\Omega}') \,\mathrm{d}E' \mathrm{d}\vec{\Omega}' \,\mathrm{d}E' \mathrm{d}\vec{\Omega}' \mathrm{d}\vec{r}}{\int \phi^+ (\vec{r}, E, \vec{\Omega}) \sum_f (\vec{r}, E') v(E') \chi(E', \vec{\Omega}' \to E, \vec{\Omega}) \, \phi(\vec{r}, E', \vec{\Omega}) \mathrm{d}E' \mathrm{d}\vec{\Omega}' \,\mathrm{d}E \mathrm{d}\vec{\Omega} \mathrm{d}\vec{r}} \tag{11}$$

and the prompt neutron lifetime l_p defined as

$$l_{\text{p}} = \frac{\left\langle \phi^+ \dfrac{1}{\upsilon} \phi \right\rangle k_{\text{eff}}}{\left\langle \phi^+ F \phi \right\rangle} \tag{12}$$

From eqs. (3), (10), and (12), the prompt neutron lifetime (l_{p}) can be estimated by eq. (13) as

$$l_p = \frac{k_{\text{eff}} \int \phi^+ (\vec{r}, E, \vec{\Omega}) \dfrac{1}{v(E)} \phi(\vec{r}, E', \vec{\Omega}') \,\mathrm{d}E' \mathrm{d}\vec{\Omega}' \,\mathrm{d}E \mathrm{d}\vec{\Omega} \mathrm{d}\vec{r}}{\int \phi^+ (\vec{r}, E, \vec{\Omega}) \sum_f (\vec{r}, E') v(E') \chi(E', \vec{\Omega}' \to E, \vec{\Omega}) \, \phi(\vec{r}, E', \vec{\Omega}') \mathrm{d}E' \mathrm{d}\vec{\Omega}' \,\mathrm{d}E \mathrm{d}\vec{\Omega}' \mathrm{d}\vec{r}} \tag{13}$$

where k_{eff} is the effective multiplication factor and v – the neutron speed. The prompt neutron lifetime l_{p} was calculated using the adjoint weighted method, which was implemented in the KOPTS card of the MCNP code [35].

1/v absorber insertion method

The prompt neutron lifetime l_p was calculated using the $1/v$ absorber insertion method [35]. This method involved the introduction of a small addition of ^{10}B as a perturbation in the core materials. The appearance of the absorber content results in a negative change of reactivity. The negative reactivity insertion is calculated as

$$\rho = \frac{\Delta k}{k} = \frac{1}{k_{\text{eff, u}}} - \frac{1}{k_{\text{eff, p}}} \tag{14}$$

where $k_{\text{eff,u}}$ is the effective multiplication factor without the ^{10}B addition (unperturbed) and $k_{\text{eff,p}}$ – the effective multiplication factor with additional ^{10}B content (perturbed).

The $k_{\text{eff,p}}$ is calculated by changing the ^{10}B content added to the core materials. The reactivity insertion due to the addition of the ^{10}B content is

$$-\frac{\Delta k}{k} = N \cdot \sigma_0 \cdot v_0 \cdot l'_p \tag{15}$$

where N is the atomic density of the ^{10}B absorber (atoms/(b.cm)), σ_0 – the thermal neutron absorption cross section in barns (b) (1 b = 10^{-24} cm^2) of the absorber, v_0 – the speed of thermal neutrons, and l'_{p} – the prompt neutron lifetime when ^{10}B absorber is added.

The prompt neutron lifetime l_{p} is calculated when the ^{10}B content approaches zero [14, 35]

$$l_{\text{p}} = \lim_{N \to 0} (l'_{\text{p}}) = \lim_{N \to 0} \left(-\frac{\Delta k}{k} \frac{1}{N \cdot \sigma_0 \cdot v_0} \right) \tag{16}$$

In the present work, the ^{10}B content was added uniformly to all core materials with the concentration from $4.0 \cdot 10^{-9}$ to $15.0 \cdot 10^{-9}$ atoms/(b.cm). The $k_{\text{eff,p}}$ value was then calculated at each step of the ^{10}B contents for determining the linear dependence of the reactivity with the ^{10}B content. The l_{p} is obtained when the ^{10}B content approaches to zero.

RESULTS AND DISCUSSION

Effective delayed neutron fraction

The MCNP6.3 calculations were set up to estimate the k_{eff} of the DNRR with 92 LEU fuel bundles with the standard deviation (1σ) of 15-21 pcm. Figure 4 depicts the dependence of the β_{eff} as a function of burnup obtained from the adjoint weighted method. It can be seen that the β_{eff} decreases rapidly in the first 200 effective full power days (EFPD), which is then followed by a gradual decline. The β_{eff} decreases from the BOC to the EOC (800 EFPD) by about 52 pcm. The adjoint weighted method predicts the β_{eff} values of 760 pcm at the BOC and 708 pcm at the EOC, respectively. Table 2 shows the β_{eff} values of the DNRR core at the BOC and the EOC in comparison between the adjoint weighted method and the prompt method. The prompt method estimated the β_{eff} lower than those from the adjoint weighted method by about 8-12 pcm. Comparing between the two methods, a good agreement was found with the discrepancy less than 1.6 %. Comparing to other research reactors, the β_{eff} of the first core of the RSG GAS using MVP3 and MCNP-6.2 with various data libraries was reported approximately 738.3 pcm [23]. The β_{eff} of the TRIGA Mark II reactor were 800 pcm for a small core (43 fuel rods) and 750 pcm for a full core (90 fuel rods), respectively [5]. Thus, the β_{eff} values obtained for the DNRR are approximate that of other research reactors.

Neutron generation time and prompt neutron lifetime

Table 3 presents the calculated values of neutron generation time and prompt neutron lifetime using the adjoint-weighted and $1/v$ insertion methods. The values of neutron generation time, Λ, at the BOC and the EOC, obtained by using the adjoint weighted method and the JENDL-5 library, are 86.99 and 82.41 μs, respectively. Figure 5 presents the prompt neutron lifetime l_p as a function of burnup obtained from the adjoint weighted method. The l_p increases linearly with burnup due to the decrease of the production rate. The l_p values at the BOC and the EOC are 79.01 and 82.76 μs, respectively. Figure 6 depicts the determination of the l_p values using the $1/v$ absorber insertion method by varying the ^{10}B concentration from $4.0 \cdot 10^{-9}$ to $15.0 \cdot 10^{-9}$ atoms/(b.cm). The l_p values at the BOC and the EOC are 79.72 and 82.98 μs, respectively. Comparing between the two methods, the discrepancy of the l_p values is within 1 %. The tendency of the kinetic parameters agrees with that reported by Atta Muhammad *et al.* [3].

Table 2. Effective delayed neutron fraction of the DNRR with LEU fuel calculated by the adjoint weighted method and the prompt method

Parameter	Adjoint weighted		Prompt	
	BOC	EOC	BOC	EOC
β_{eff} (pcm)	760 ± 4	708 ± 21	748 ± 9	700 ± 18

Table 3. Neutron generation time and prompt neutron lifetime of the DNRR with LEU fuel calculated by the adjoint weighted method and the 1/*v* insertion method

Parameter	Adjoint weighted		$1/v$ insertion	
	BOC	EOC	BOC	EOC
Λ (μs)	86.99 ± 0.11	82.41 ± 0.45	–	–
l_p (μs)	79.01 ± 0.11	82.76 ± 0.15	79.72 ± 0.45	82.98 ± 0.19

CONCLUSIONS

Kinetic parameters of the DNRR with 92 LEU fuel bundles were evaluated using MCNP6.3 and the JENDL-5 library. The β_{eff} decreases gradually with burnup time, and the values at the BOC and the EOC

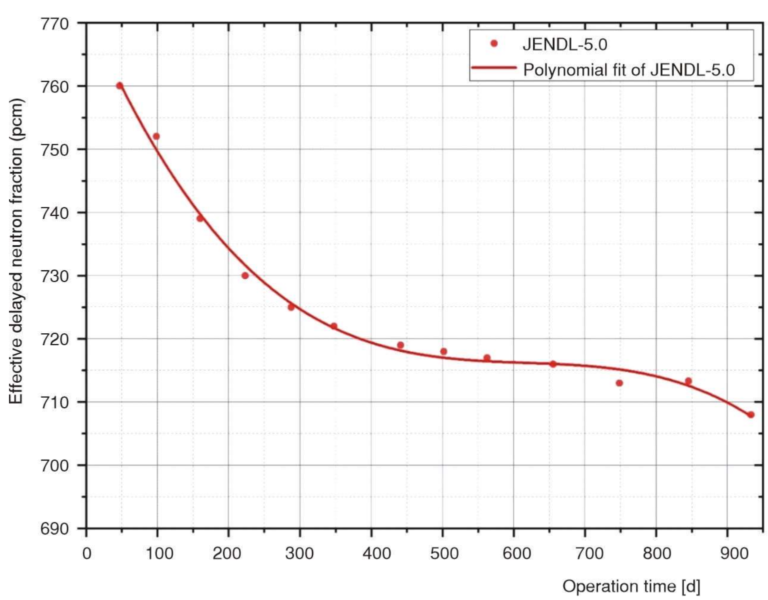

Figure 4. Effective delayed neutron fraction β_{eff} as a function of burnup time obtained from the adjoint weighted method (1 pcm = 10^{-5})

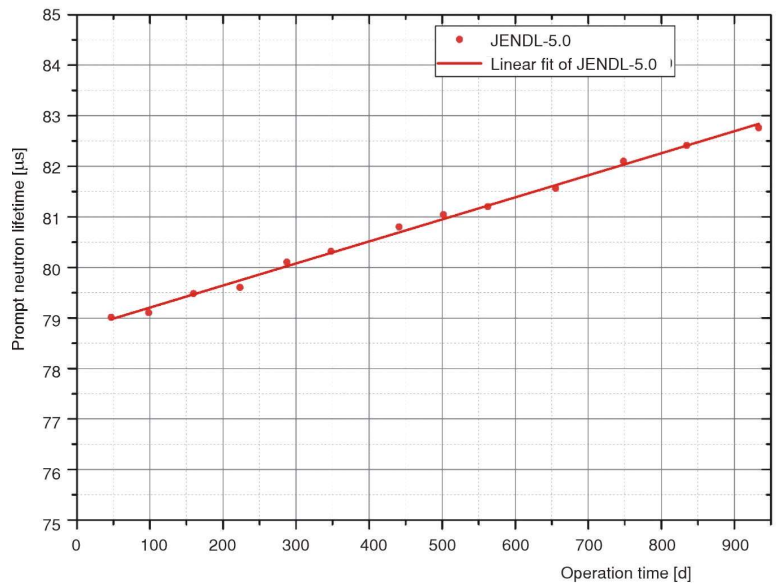

Figure 5. Prompt neutron lifetime (l_p) as a function of burnup time obtained from the adjoint weighted method

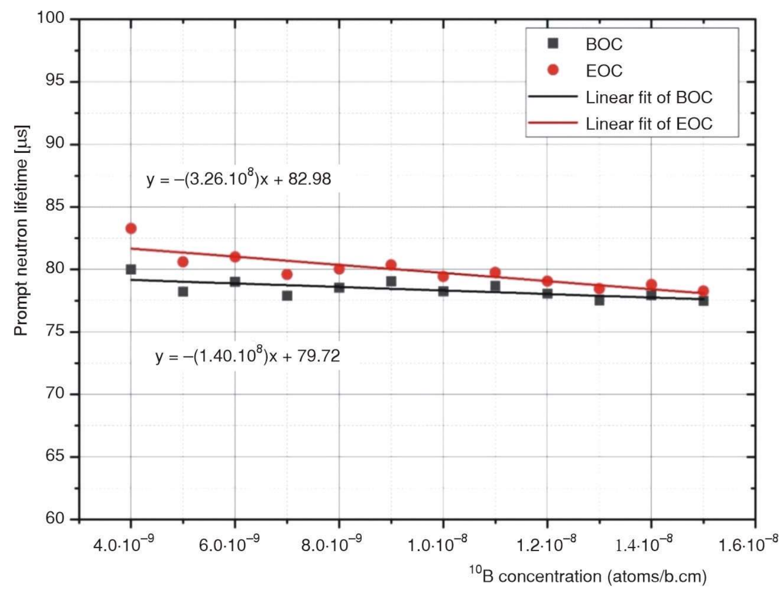

Figure 6. Prompt neutron lifetime as a function of ^{10}B concentration at the BOC and the EOC obtained from the 1/*v* absorber insertion method.

obtained from the adjoint weighted method are 760 and 708 pcm, respectively. The discrepancy between the two methods is within 1.6 %. The adjoint weighted method predicted the neutron generation Λ at the BOC and the EOC as 86.99 and 82.41 µs, respectively. The prompt neutron lifetime l_p values at the BOC and EOC obtained from the adjoint weighted method are 79.01 and 82.76 µs, respectively. The values differ from that obtained with the 1/*v* absorber insertion method by about 1 %. The results indicate good agreements among the methods, confirming the reliability and the accuracy of the calculated parameters. The obtained kinetic parameters could be used in further reactor dynamic and safety-related assessment of the DNRR, and will be incorporated into the safety analysis report.

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AUTHORS' CONTRIBUTIONS

D. T. Dau and H. N. Tran designed the study, analyzed the data, wrote the initial draft of the manuscript, reviewed and edited the final manuscript. N. D Nguyen, K. C. Nguyen, Q. H. Pham, T. N Chu, V. K. Hoang, G. T. T. Phan analyzed the data, reviewed and edited the final manuscript.

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КИНЕТИЧКИ ПАРАМЕТРИ ИСТРАЖИВАЧКОГ НУКЛЕАРНОГ РЕАКТОРА DALAT СА НИСКО ОБОГАЋЕНИМ УРАНИЈУМСКИМ ГОРИВОМ КОРИШЋЕЊЕМ БИБЛИОТЕКА MCNP6 И JENDL-5

Познавање кинетичких параметара нуклеарног реактора неопходно је у динамици реактора и повезаним карактеристикама безбедности. Кинетички параметри истраживачког нуклеарног реактора Dalat процењени су коришћењем кода MCNP6.3 и нове библиотеке нуклеарних података JENDL-5. Нумерички прорачуни извршени су за конфигурацију језгра која се састоји од 92 горивна снопа ниско обогаћеног уранијума ради добијања ефективне фракције закаснелих неутрона β_{eff}, времена генерисања неутрона Λ и времена живота брзог неутрона l_{p}. За израчунавање β_{eff} коришћене су две методе: метода са пондерисаним адјунгованим неутронима заснована на теорији пертурбација и адјунгованом флуксу, и метода брзог прорачуна која користи однос брзих и укупних фисионих неутрона. За израчунавање Λ и l_{p} коришћене су методе адјунговане тежине и уметања $1/v$ апсорбера. Вредности β_{eff}, Λ и l_{p}, на почетку и крају циклуса добијене методом адјунговане тежине, износе 760 pcm и 708 pcm; 86.99 μs и 82.41 μs; и 79.01 μs и 82.76 μs, респективно. У поређењу метода, кинетички параметри предвиђени су са одступањем од око 1.0-1.6 %. Кинетички параметри биће коришћени у извештају о анализи безбедности Dalat реактора.

Кључне речи: истраживачки нуклеарни реактор Dalat, гориво са ниско обогаћеним уранијумом, кинетички параметар, MCNP6, JENDL-5