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Study of Kinetic and Neutronic Parameters for HEU and Potential LEU/MEU Fuels in a Typical MNSR

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ABSTRACT

The MTR-PC package, which combines WIMSD and CITATION codes, was used to calculate the effective delayed neutron fraction and prompt neutron lifetime for a Miniature Neutron Source Reactor (MNSR). In the Reduced Enrichment for Research and Training Reactors (RERTR) program, these kinetic parameters were calculated for various potential LEU and MEU fuels. The effect of composition changes due to fuel depletion on these parameters was studied for HEU and potential LEU/MEU fuels. The results of kinetic and other neutronic parameters for the HEU MNSR core are in agreement with the values in the literature. The delayed neutron fraction and prompt neutron lifetime showed minor variations over the 200 Effective Full Power Days (EFPD) burnup cycle of the MNSR cores for each fuel. The maximum change in delayed neutron fraction (β_{eff}) over the burnup cycle was found for UO_2 fuel with Zircaloy-4 clad system, showing the maximum ^{239}Pu production among all fuel types. The maximum neutron generation time (Λ) increase of 0.4μs was found in UAl_4 90.3% enriched core.

Keywords: *Kinetic parameters, MNSR, LEU fuels, Neutronic analysis, MTR-PC*

1. Introduction

Before 1978, nuclear reactors for research and isotope production were mainly based on High-Enriched Uranium (HEU) fuels. This created a high proliferation risk as the fuel used in those reactors contained a large amount of ^{235}U isotope [1]. Consequently, in the light of the RERTR program, many reactors have been successfully converted to LEU fuels provided their performance remains within the acceptable criterion with minimum economic penalties [2].

The Miniature Neutron Source Reactors (MNSRs) are low-power reactors based on HEU fuel. A few MNSRs operate in different countries, mainly for research and training purposes. To a certain extent, other researchers have already done the conversion studies for some of them. The UO_2 as LEU fuel has been considered one of the study analyses for Syrian MNSR [3]. Similar studies were carried out for Ghana Research Reactor with UO_2, U_9Mo and U_3Si_2Al fuels as potential LEU fuels [4]. The analysis of core lifetime and inventory of isotopes for a typical MNSR was conducted for U-9Mo-Al and UO_2 as potential LEU fuel [5]. Uranium silicide and uranium molybdenum dispersed in aluminum matrix were included in another work for four potential LEU fuel options and Medium Enriched Uranium (MEU) fuels for similar calculations [6].

Besides core lifetime and isotope inventory calculations in respect of HEU to LEU/MEU conversion of typical MNSR core requires the estimation of the kinetic parameters for transient, accidental and stability analysis. However, the calculations of the kinetic parameters for alternative fuels for different MNSRs have been performed only at Beginning of Cycle (BOC). A. researchers used MCNP-4C computer code to estimate the kinetic parameters for a Syrian MNSR [7]. The prompt neutron lifetime for Nigeria MNSR (NIRR-1) reactor is calculated for both HEU and LEU cores [8]. In another study of core conversion of Ghana MNSR, the delayed neutron fraction is calculated for UO_2 fuel with Monte Carlo runs of the MCNP code [4].

The delayed neutron fraction (β_{eff}) sets the allowable limit of reactivity insertion to avoid prompt criticality. The neutron generation time (Λ) determines the power change rate on reactivity insertion. Delayed neutron fraction and neutron generation time depend on the fuel type and core arrangement. They are subjected to two kinds of changes: Firstly, the short-term changes that happen during a transient, a typical example is when reactor power is increased. This will harden the spectrum and lead to an increase in fast fission of ^{238}U. The change in neutron spectrum can also affect the magnitude of kinetic parameters in the core. The delayed neutron fraction is affected by fission in ^{238}U, and the neutron generation time is sensitive to the average speed of neutrons and the fission cross section of the fissile isotope. The hardening of the spectrum also changes the neutron generation time. The changes above are usually neglected to simplify the mathematical model. The long-term changes in the core isotopic composition about fuel burnup alter the effective delayed neutron and neutron generation time.

This work focuses on estimating kinetic parameters for conventional HEU and potential LEU/MEU fuels for typical MNSRs. The parameters β_{eff} and Λ are computed at different burnup steps to see the variation of these parameters. The MTR-PC neutronics calculation package, which includes computer codes WIMSD/4 and CITATION. These were used for the calculation of kinetic parameters of MTR type research reactor [9].

2. Brief introduction of the reactor

The reactor is a typical tank in pool type with Highly Enriched Uranium (HEU) fuel. The rated thermal power is 27kW and has self-limiting characteristics with large negative moderator temperature coefficient. Its core is almost square cylinder with 344 fuel pins and 10 non fuel pins (6 dummy pins and 4 tie rods). The only cadmium control rod is situated at the center of the core. The core has annular and bottom beryllium reflector. It also has a shim tray at the top to add Be reflector plates for reactivity compensation. Water is used for neutron shielding and reflection. The vertical and core-mid-plane horizontal cross sections shown in Fig. 1 and the important design specifications are provided in Table 1.

Table 1: Some important design parameters of the reactor [10]

Parameters	Description
Reactor type/class	Tank-in-Pool/MNSR
Power (nominal/self-limiting; kW_{th})	27/87
^{235}U core loading (g)	994.8
^{235}U enrichment (%)	90
Material (fuel/clad/mod./reflector)	UAl_4/Al (303-1)/H_2O/Be
Control rod (meat/clad/length, cm/worth, mk)	Cd/S.S./23/6.7
Core (height/diameter; cm)	23/25
Fuel (meat-dia., cm/number of pins)	0.43/344
Excess reactivity (mk; cold, clean)	4.0
Number of irradiation sites (inner/outer)	5/5

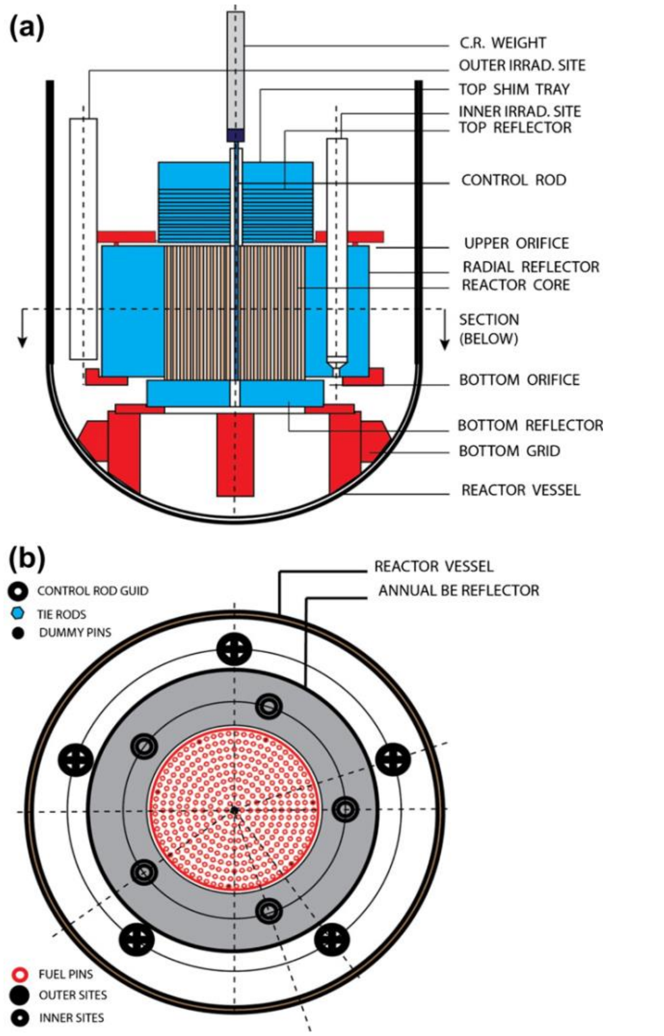

Fig. 1: Cross-sectional views of the MNSR system (a) vertical section and (b) horizontal section at the core mid-plane [6].

3. Theory and computational modeling

The MTR PC package has the ability to efficiently manage the transfer of the data among the lattice cell calculation code WIMSD4 [11] and diffusion theory based code CITATION through BORGES computer program [12]. The cluster option of WIMS for geometry modeling is used for the generation of microscopic cross sections (ENDF/B-IV library) along with $(1/v)_{avg}$. The energy group structure for which the set of cross sections are obtained is provided in Table 2 [5].

Table 2: Energy structure in WIMS

Sr. No	WIMS Cross-section lib. groups	Upper limit (eV)	Mean Energy (eV)
1	1- 5	1.00000E+07	2.86531E+06
2	6-7	8.21000E+05	4.98350E+05
3	8	3.02500E+05	2.35282E+05
4	9 -11	1.83000E+05	8.64613E+04
5	12 -14	4.08500E+04	1.92995E+04
6	15-17	9.11800E+03	4.51877E+03
7	18-20	2.23945E+03	9.06898E+02
8	21-23	3.67262E+02	1.32845E+02
9	24-25	4.80520E+01	2.77001E+01
10	26-27	1.59680E+01	7.99200E+00
11	28-32	4.00000E+00	2.28035E+00
12	33-36	1.30000E+00	1.17996E+00
13	37-40	1.07100E+00	1.02030E+00
14	41-44	9.72000E-01	8.70724E-01
15	45 to 48	7.80000E-01	5.22494E-01
16	49 to 52	3.50000E-01	2.95804E-01
17	53 to56	2.50000E-01	1.58114E-01
18	57 to 60	1.00000E-01	7.07107E-02
19	61 to 64	5.00000E-02	3.53553E-02
20	65 to 69	2.50000E-02	1.58114E-04

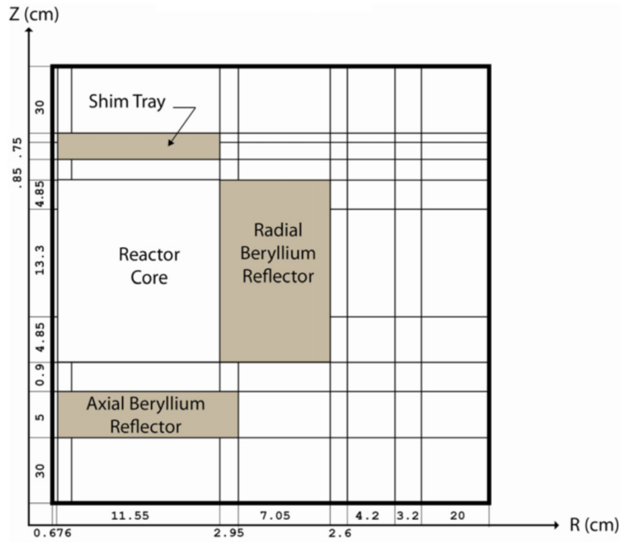

Fig. 2: The RZ-model of the MNSR system used in the CITATION calculations of HEU and the potential LEU cores [6]

From the group structure presented in Table 2, the first delayed neutron group falls within the 8th energy group, with a mean energy of 2.35282×10^2 eV. The remaining delayed neutron groups are distributed across the 6th and 7th energy groups, with a mean energy of 4.98350×10^3 eV. The kinetic parameters and multiplication factor are determined through CITATION simulations. The reactor geometry is modeled using the 2-dimensional r-z geometry option in the CITATION code, as illustrated in Fig. 2. The key steps involved in calculating the kinetic parameters are outlined in Fig. 3.

The perturbation option of CITATION code has been used which simplifies the computation of kinetic parameters.

The prompt neutron lifetime is calculated by CIATTION using Equation (1).

$$l = \frac{\sum_i \frac{V_i}{v(n)}}{\frac{1}{k}\sum_i V_i \sum_g \chi(g)\phi^*_{i,g}\sum_n \nu\Sigma_{f,n}\phi_{i,n}} \tag{1}$$

Here, i refers to mesh point location and n, g refers to energy groups. v, ϕ, ϕ^* and $\upsilon\Sigma\phi$ represents velocity, flux, adjoint flux and production rate of neutrons respectively.

CITATION also requires a set of decay constants and average delayed neutron yields. The decay constants for six groups of delayed neutron are given in Table 3.

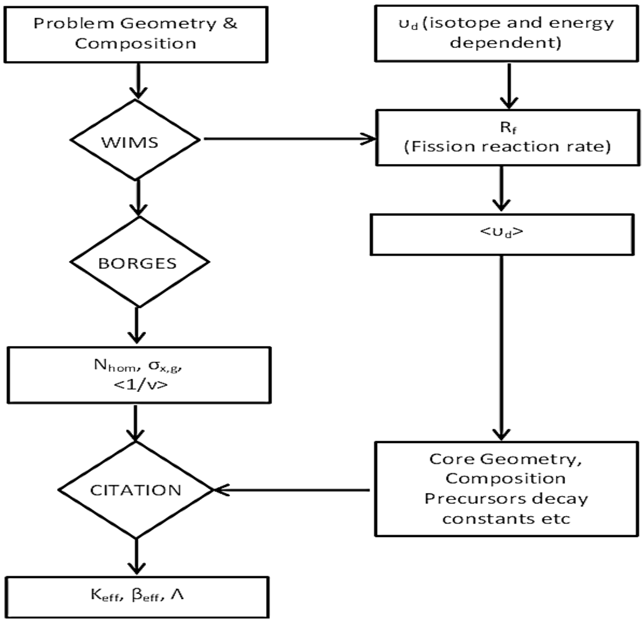

Fig. 3: Calculation flow chart

Table 3: Set of decay constant used in the calculations [13]

Delayed Group	Decay Constant (s^{-1})
1	0.0129
2	0.0311
3	0.134
4	0.331
5	1.26
6	3.21

Average delayed neutron yields are computed with the help of Equation (2) for CITATION. Where, the fast and thermal fission reaction rates are obtained from WIMSD.

$$(\overline{\nu_{dj}})^i = \frac{\int_V \int_{0.14eV}^{\infty} \nu^i_{dj}(E_f)\Sigma_f(r,E)\phi(r,\mathrm{E})\,\mathrm{dEdV} + \int_V\int_0^{0.14eV} \nu^i_{dj}(E_t)\Sigma_f(r,E)\phi(r,\mathrm{E})\,\mathrm{dEdV}}{\int_V\int_0^\infty \Sigma_f(r,E)\phi(r,\mathrm{E})\,\mathrm{dEdV}} \tag{2}$$

Where ν^i_{dj}, represents the delayed neutron yields for delayed neutron group, j and isotope, i. $\Sigma_f(r,E)$ and $\phi(r,\mathrm{E})$ are the macroscopic fission cross section and flux for energy E and position r respectively.

The nuclides ^{235}U and ^{239}Pu have significant contribution towards the delayed neutron production. Their delayed neutron yields for fast and thermal energy ranges used for averaging by equation 3 are given in Table 4.

Table 4: Delayed neutron yield for U^{235} and Pu^{239} for fast and thermal energy ranges [14]

Delayed neutron groups	Delayed neutron yield for U^{235}		Delayed neutron yield for Pu^{239}	
	Fast	Thermal	Fast	Thermal
1	0.00063	0.00052	0.00024	0.00021
2	0.00351	0.00346	0.00176	0.00182
3	0.0031	0.0031	0.00136	0.00129
4	0.00672	0.00624	0.00207	0.00199
5	0.00211	0.00182	0.00065	0.00052
6	0.00043	0.00066	0.00022	0.00027

CITATION also estimates the fraction of delayed neutron in one group using Equation (3):

$$\beta_j = \frac{\sum_i V_i\sum_g \chi'(j,g)\phi^*_{i,g}\sum_b\beta_{b,j}N_{b,i}\sum_n \nu\sigma_{f,n,b,i}\phi_{i,n}}{\sum_i V_i\sum_g\chi(g)\phi^*_{i,g}\sum_n\nu\Sigma_{f,n}\phi_{i,n}} \tag{3}$$

Here, b, j refers to delayed neutron group j and isotope b and $\chi(g)$ is the delayed neutron distribution function. $N_{b,i}$ is the number density of isotope b at mesh location i.

4. Results and discussion

4.1 HEU core model validation

The results of the HEU

MNSR core calculations are summarized in Table 5. These results are compared with available literature values for key reactor parameters, including excess reactivity, control rod worth, the worth of top Be shim plates (Fig. 4), effective delayed neutron fraction, and prompt neutron lifetime.

4.1.1 Excess Reactivity

The computed excess reactivity of the standard HEU MNSR core is 4.19 mk, which is in close agreement with the Final Safety Analysis Report (FSAR) values reported by [10], which state an excess reactivity of 4.0 mk. The deviation observed between the computed and FSAR values are minimal, indicating improved accuracy compared to other reported computed values in the literature. Furthermore, [9] conducted a more detailed reactor core modeling using the CITATION code, obtaining results that showed even closer agreement with FSAR data. This suggests that enhanced computational modeling techniques contribute to more precise estimations of excess reactivity.

Table 5: Comparison of various computed values of parameters of the standard HEU fuel.

Parameters	This Work (% error with FSAR)	[4]	[15]	[6]	FSAR (Qazi et al., 1994)
Excess reactivity (mk)	4.19 (4.8)	4.51	4.05 (1.1)	4.32 (8)	4.0
CR worth (mk)	-6.75 (0.7)	-7.55	-6.39 (-4.6)	-7.11 (6.1)	-6.7
Shut down margin (mk)	-2.56 (-5.2)	-3.03	-2.34 (13.2)	-2.79 (3.4)	-2.7
Top Be worth (mk)	20.53 (7.7)	-	-	18.98 (0.4)	19.07
Delayed neutron fraction	0.00811 (2)	-	-	-	0.00795
Prompt neutron life time (µs)	0.0494(6.7)	-	-	-	0.0463

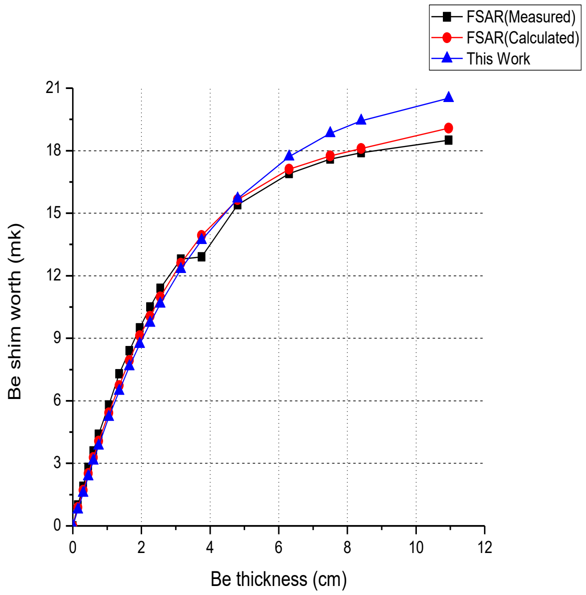

Fig 4: Top reflector worth versus thickness of top Be plates

4.1.2 Control Rod Worth

The computed control rod worth of -6.75 mk aligns exceptionally well with the FSAR-reported value of -6.7 mk [10]. Compared with other reported calculations in Table 5, the current results exhibit significant accuracy, confirming the reliability of the employed computational methods. Additionally, the results for excess reactivity and control rod worth demonstrate an improvement over those obtained in previous studies that utilized the CITATION code with macroscopic cross sections generated using the WIMS cluster option and the RZ model [6]. This improvement suggests that employing a larger number of energy groups along with average microscopic cross sections enhances the precision of diffusion theory calculations.

4.1.3 Effective Delayed Neutron Fraction and Prompt Neutron Lifetime

The computed values for the effective delayed neutron fraction and prompt neutron generation time are 0.00811 and 0.0494 µs, respectively. These results closely align with FSAR-reported values, with deviations of 2% for the effective delayed neutron fraction and 6.7% for the prompt neutron lifetime. The relatively small errors further validate the accuracy of the adopted modeling approach. The close agreement indicates that the neutron physics parameters obtained from this study are reliable for safety and operational assessments of the HEU MNSR core.

Overall, the present study's computational results for HEU MNSR core parameters strongly agree with FSAR and previously reported values in the literature. The enhanced precision, particularly in excess reactivity and control rod worth calculations, demonstrates the effectiveness of employing detailed microscopic cross sections and a higher number of energy groups in diffusion theory modeling. These findings contribute to improved predictive capabilities in reactor core physics calculations and enhance confidence in computational reactor analysis methodologies.

4.2 Potential LEU/MEU core analysis

Different LEU fuel types, i.e., UO_2 (Al, Zircaloy-4 clad), U_3Si–Al, U_3Si_2–Al, and U–9Mo–Al, have been considered for the MNSR system. The core configuration, such as the number of fuel pins, their physical and core dimensions, is the same as that of the original HEU MNSR system. This ensures the thermal-hydraulic characteristics remain unchanged. The computed results for these fuels considered for MNSR system are presented in Table 6.

The oxide fuel has relatively high density and, additionally, good thermal characteristics, making it the most attractive option among LEU fuels. However, zircaloy-4, along with aluminum, has been considered as a cladding material with oxide fuel. The higher loading of ^{235}U in oxide fuel requires relatively lower enrichment to obtain the desired value of excess reactivity, as is evident from the results in Table 6. The excess reactivity has been obtained close to the 4 mk value for all fuels with enrichment 12.5% (UO_2, Zircaloy-4 clad), 12.78% (UO_2, Al clad), 19.81% (U_3Si–Al), 22.31% (U_3Si_2–Al) and 24.29% (U–9Mo–Al) respectively.

The control rod worth and shutdown margin are computed as -5.57 mk and -1.55 mk for UO_2 fuel with Zircaloy-4 clad. The value is reasonably close with the corresponding reported values i.e. -5.748 [6], -5.437 mk [15] and -6.583 mk [4]. Similarly, the value of shutdown margin of -1.55 mk calculated in this work is also in comparable agreement with the corresponding reported values i.e. -1.768 [6] -1.43 mk [15] and -2.062 mk [4].

The enrichment values for the case of UO2 fuel with aluminum Al clad are slightly higher, i.e. 12.78%, due to the higher absorption cross-section of Al aluminum. The computed values of control rod worth as -5.61 mk and shutdown margin as -1.62 mk are in very good agreement with reported values, i.e. -5.615 mk and -1.606 mk [11], respectively.

For the case of U_3Si–Al fuel, the values of control rod worth and shutdown margin are found to be -5.84 mk and -1.79 mk which also agree well with corresponding reported values. However as mentioned above, the enrichment values obtained for potential LEU fuels including U_3Si_2–Al and U–9Mo–Al are slightly higher than 20%. The calculation results of the values for corresponding safely related parameters for both these fuels showed good agreement with the reported values.

4.3 Estimation of Kinetic parameters for potential LEU and MEU fuels

The kinetic parameters of potential LEU and MEU fuels are calculated with the procedure discussed in section 2. The values of effective delayed neutron fraction (β_{eff}) and mean neutron generation time **(**Λ) for different fuels are listed in Table 7. The β_{eff} values are not much different for these fuel types, however, the mean neutron generation time has maximum value of 4.43μs for U_3Si_2-Al fuel and minimum value of 3.70μs for U-9Mo-Al fuel.

Table 6: Comparison of various computed values of parameters for different LEU and MEU fuels.

LEU fuel	Enrichment (%)	ρ_{ex} (mk)	C.R. worth (mk)	S.D margin (mk)
UO_2 ((Zr-4 clad)				
This Work	12.5	4.02	-5.57	-1.55
[6]	12.5	4.01	-5.748	-1.736
[4]	12.6	4.52	-6.583	-2.062
[15]	12.6	4.01	-5.437	-1.43
[5]	11.2	4.33	–	–
[16]	12.45	4.73	–	–
UO_2 (Al clad)				
This work	12.78	3.99	-5.61	-1.62
[6]	12.78	4.008	-5.615	-1.606
U_3Si–Al (-38%)				
This work	19.81	4.03	-5.82	-1.79
[6]	19.81	4.04	6.041	-2.001
[4]	19.75	4.043	-6.586	-2.542
U_3Si_2–Al				
This work	22.31	4.05	-5.84	-1.79
[6]	22.31	4.028	-6.115	-2.087
[4]	19.75	4.27	-6.655	-2.365
[5]	20.7	4.3	–	–
U–9Mo–Al (42.4%)				
This work	24.29	4.16	-4	0.16
[6]	24.29	4.047	-4.529	-0.4822

Table 7: Kinetic parameters of alternate LEU/MEU fuel at BOC.

Fuel	Enrichment (%)	β_{eff}	Λ (μs)
UO_2- Zircaloy-4 clad	12.50	0.00809	4.27
UO_2- Al clad	12.78	0.00810	4.24
U_3Si-Al	19.81	0.00809	4.40
U_3Si_2-Al	22.31	0.00809	4.43
U-9Mo-Al	24.29	0.00806	3.70

Burnup dependent calculations of kinetic parameters are performed with time step of 20 days for 200 EFPD for each fuel type. This corresponds to the reactor operation of 2 hours per day; 5 days a week sustained over a period of 10 years which is equivalent to 200 EFPD.

The results of β_{eff} and Λ for different fuels are shown in Fig. 5 to Fig. 8. The parameter β_{eff} has shown decreasing trend for all fuels owing to the production of Pu isotopes in the core. However, Λ has shown an increasing trend with burnup due to decrease in the macroscopic fission cross section. A decrease in β_{eff} and an increase in Λ would negatively impact the reactor period, reducing its magnitude as the fuel undergoes burnup. However, the small magnitude of changes in kinetic parameters due to fuel depletion would not significantly impact safety or the reactivity insertion margin.

The change in kinetic parameters for conventional HEU fuel i.e. UAl_4 is given in Fig. 5. An overall decrease of $5.8x10^{-7}$ is found in $\beta_{eff,}$ which is the least among all the fuels. Whereas, an increase of 0.4μs in Λ is found at the EOC for UAl_4 fuel.

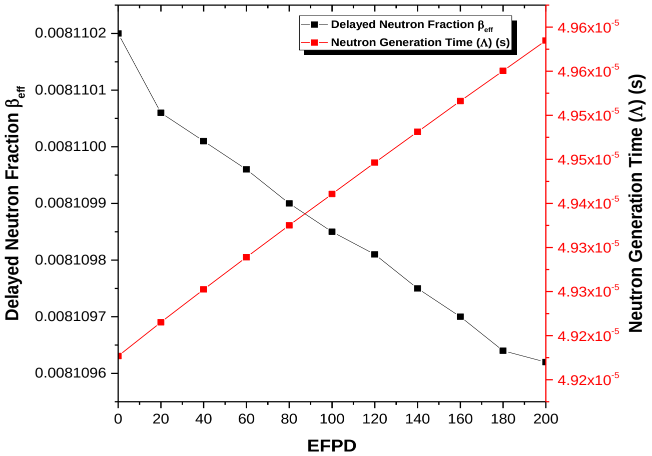

Fig. 5: Variation in kinetic parameters with burnup for standard HEU fuel.

The results for the variation in kinetic parameters are shown in Fig. 6 for ceramic fuels. The decrease in β_{eff} for ceramic fuels i.e. UO_2 with Zircaloy-4 clad and UO_2 with Al clad is $6x10^{-6}$ and $5.7x10^{-6}$ respectively. Meanwhile, the increase found in Λ are 0.3 and 0.2 μs respectively after 200 EFPDs.

The results of kinetic parameters for silicide dispersed fuels i.e. U_3Si-Al and U_3Si_2-Al are shown in Fig. 7. Both silicide fuels have shown an equal increase of 0.13μs in Λ values over the whole burnup cycle. However, the decrease in β_{eff} of U_3Si-Al and U_3Si_2-Al fuels are $4.3x10^{-6}$ and $4.0x10^{-6}$ respectively at the end of 200 EFPD.

The Fig. 8 is plotted for the calculated data for U-9Mo-Al fuel where the delayed neutron fraction has shown decrease by $3.1x10^{-6}$ and increase in neutron generation time by 0.1μs over the whole burnup cycle.

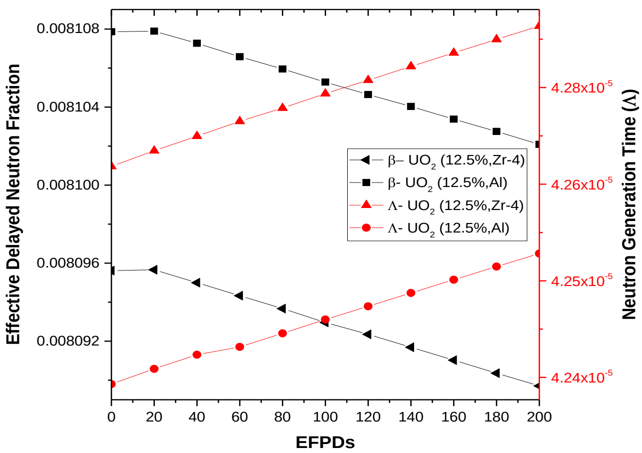

Fig. 6: Variation of kinetic parameters for UO_2 with Zircaloy-4 clad and UO_2 with Al clad with burnup.

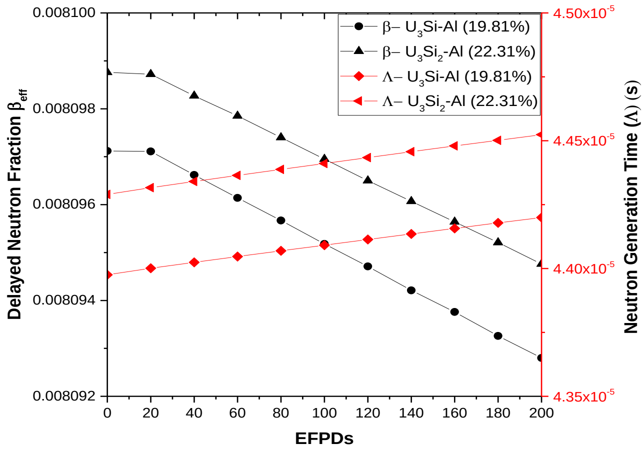

Fig. 7: Variation of kinetic parameters for U_3Si-Al and U_3Si_2-Al with burnup

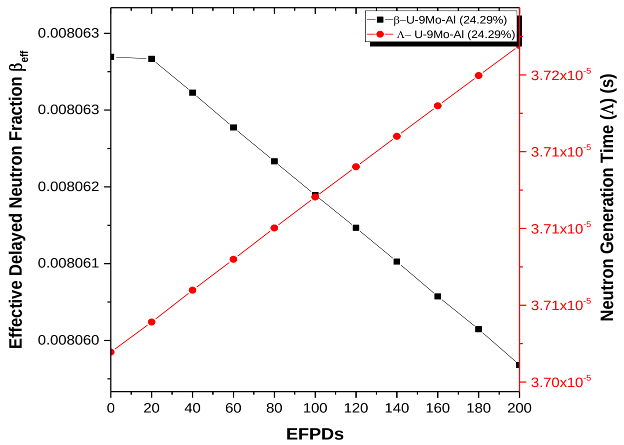

Fig. 8: The variation of kinetic parameters for U-9Mo-Al with burnup

Table 8 indicates that the net decrease in delayed neutron fraction for all fuel types is dependent on the production of Pu^{239} in the reactor core. The maximum decrease in effective delayed neutron fraction is observed in ceramic fuels as compared to other types due to relatively larger amount of Pu^{239}. A relatively hardened spectrum with Zircaloy cladding leads to an increase Pu^{239}production resulting in greater variations in the delayed neutron fraction. In contrast to Al, being lighter in mass is a more effective neutron moderator compared to zirconium alloy.

Table 8: Amount of Pu^{239} produced and net change in delayed neutron fraction.

Fuel	Change in β_{eff}	Amount of produced Pu^{239} (g)	
		This work	[5]
UO_2- Zircaloy 4 clad	$6x10^{-6}$	0.779	0.793
UO_2- Al clad	$5.7x10^{-6}$	0.773	-
U_3Si-Al	$4.3x10^{-6}$	0.528	-
U_3Si_2-Al	$4x10^{-6}$	0.472	-
U-9Mo-Al	$3.1x10^{-6}$	0.503	0.799

The neutron generation time depends on <1/v> and neutron fission cross section. The increase in Λ with the burnup can be attributed to the increase in <1/v> and decrease in fission cross section. The net increase in Λ of all considered fuel is presented in Table 8.

5. Conclusion

This work focused on determining the variation in kinetic parameters as burnup proceeds for conventional HEU and potential LEU/MEU fuels for MNSR core. The calculations were carried out using MTR-PC package. The microscopic cross sections were generated using the cluster processing option in WIMS and RZ geometry for reactor modeling in CITATION. The model validation results showed good agreement of the computed values of excess reactivity, control rod worth, shutdown margin, effective delayed neutron fraction, and prompt neutron lifetime with the corresponding values found in literature for the HEU core. Four potential fuels UO_2 (Zircaloy-4 clad & Al-clad), U_3Si–Al, U_3Si_2–Al and U9Mo–Al, are considered for the analysis of kinetic parameters as LEU fuels. The parameters β_{eff} and Λ of HEU system for the clean core are in good agreement with the safety analysis report of the reactor. The error in the delayed neutron fraction is 2%, and in the prompt neutron lifetime is less than 7%. The kinetic parameters showed variation at different burnup steps. The decrease in β_{eff} is the result of ^{239}Pu production in the core. The highest variation is found in UO_2 12.5% enriched with Zr-4 clad system, which has the highest ^{239}Pu production among all fuels. The maximum increase in Λ of 0.4μs is seen in UAl_4 90.3% enriched core. The trend of variation of the kinetic parameters is in excellent agreement with the reported values.

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