Analysis of the thermal feedback and burn up effects on kinetic parameters in TRR by the Monte Carlo method

H R Khaleghi and M Hassanzadeh

1 University of Advanced Industrial Technology Kerman Department of Electrical and Computer Engineering Department of Nuclear, Kerman, Iran
2 Nuclear Science and Technology Research Institute (NSTRI), Reactor & Nuclear Safety School, Tehran, Iran

E-mail: m_hassanzadeh1354@yahoo.com and mhasanzadeh1354@gmail.com

Keywords: kinetic parameters, LEU and HEU, MCNPX code, Tehran Research Reactor

Abstract

In the safety analysis of nuclear reactors, the knowledge about kinetic parameters is significant. Effective delayed neutron fraction ($\beta_{\text{eff}}$) and prompt neutron life time ($\ell_p$) depend on the time behaviour of the reactor power transient after reactivity insertion. Thus, in this article, the sensitivity analysis of the kinetic parameters to physical parameters such as fuel enrichment, fuel consumption (burn-up) and temperature changes in Tehran Research Reactor (TRR) has been investigated. For this purpose, this research has used MCNPX code for these parameters. According to the results, the amounts of $\ell_p$ parameter for $\frac{1}{V}$ poisoning method and Safety Analysis Report (SAR) are 47.9 and 45.0 $\mu s$, respectively. The obtained results have been compared to SAR. It was found that the relative difference is 6.4% for $\ell_p$ parameter. Moreover, the values of $\beta_{\text{eff}}$ parameter for MCNPX code and SAR are 785.64 and 777.7 pcm, respectively. When compared to the available SAR data, a relative difference of 1.1% was obtained for $\beta_{\text{eff}}$ parameter. In most cases, raising the temperature produces higher values of $\ell_p$ but reduces $\beta_{\text{eff}}$ parameters in both LEU and HEU cases. Also, any increment in burn-up no changes in $\ell_p$ parameter but $\beta_{\text{eff}}$ parameter reduces for both LEU and HEU cases.

1. Introduction

The evolution of the kinetic parameters such as the effective delayed neutron fraction ($\beta_{\text{eff}}$) and prompt neutron life time ($\ell_p$), when designing a nuclear reactor is important, because of their unstable behavior [1]. In long periods of time, the core composition is changed due to the fuel burn-up, so, the neutronic and dynamic behaviors of the reactor core might be varying from its initial condition at the Beginning of Cycle (BOC). A worthy design should be assure the safe conditions of the reactor core in the life cycle period [2].

Up to know here have been several methods to evaluate the kinetic parameters among which Monte Carlo method is the most efficient method [3]. In this method, the main components of the simulation are probability distribution function and random-number generator.

In this study, the sensitivity analysis of kinetic parameters to physical parameters such as fuel enrichment, fuel consumption and temperature changes have been done for Tehran Research Reactor (TRR) by MCNPX code [4].

Moreover, in the current study, $\frac{1}{V}$ poisoning method is applied for calculating of the kinetic parameters. This method is a simple and accurate technique way to compute the $\ell_p$ parameter in which is used of perturbation theory [5].

TRR is a 5-megawatt reactor with light water moderator and solid fuel heterogeneous. The water is used for cooling as well as for shielding. The core of reactor has been formed from Material Testing Reactor (MTR), a type of fuel assembly, which is located in a lattice. Fuel of the TRR was highly enriched uranium of 93.5% (HEU) at the start of operation, than, after the revision in its design, it has been changed to low enriched uranium of 20% (LEU). The chemical composition of the fuel is U$_3$O$_8$-AL. Table 1 shows the TRR characteristics and physical parameters [6].

© 2018 The Author(s). Published by IOP Publishing Ltd
In the current work, two cases of LEU and HEU are selected in order to calculate the neutronic and kinetic parameters for the TRR. Also, we assume the temperature changes from 20 °C to 100 °C and the period of effective fuel consumption (burn-up) has been calculated for the period of one month and in two cases of LEU and HEU [7]. Moreover, the results of this study will be validated by a comparison of the results obtained by $\frac{1}{v}$ poisoning method, Safety Analysis Report (SAR) [6, 8, 9].

2. Methodology

The MCNPX 2.6.0 code is used for calculation of kinetic parameters. This code is a three dimensional and multi-purpose code for the transport of many particles such as neutrons, photons and electrons. The neutron energy regime is from $10^{-11}$ to 150 MeV and for photon and electron is from 1 keV to 1 GeV. This code includes many capabilities such as the areas of transmutation, burn-up, and delayed particle production. Also, the ENDF/B-VI neutron libraries are used in this code.

Criticality calculations are performed by KCODE card to make the fission source converge from an initial guess distribution with an arbitrary but uniform set of points in the fuel regions. The input file for the MCNPX code is included 1000 cycles made of 50 inactive cycle and 950 active cycles with $1 \times 10^7$ histories per cycle. Tallies are in these code three basic $k_{eff}$ estimators (track-length, collision and absorption) and reports a best estimate combined $k_{eff}$ value. Criticality calculations are done within the purposes of the present study.

Equation (1) presents the main neutronic parameters $\beta_{eff}$ related with two other parameters $k_{eff}$ and $k_p$, defined as the effective multiplication factor and the prompt multiplication factor, respectively. Here, we have used Kcode card to compute $k_{eff}$ and $k_p$ parameters for the $\beta_{eff}$ calculation in equation (1) [10, 11].

$$\beta_{eff} = 1 - \frac{k_p}{k_{eff}}$$

Furthermore, the code version 2.6 is applied for the fuel burn-up calculations. The burn-up algorithm is done by the CINDER 90 code located in the MCNPX code. The CINDER. dat library file contains decay, fission yield and 63-group cross-section data. This library file must be current and accessible by this code for the burn-up calculations to use properly. KCODE and BURN cards are used to calculate burn-up and fuel depletion calculations in the MCNPX code.

To calculate and validate kinetic parameters, we have considered the following two steps calculations:

**First step:** kinetic parameters of the TRR are calculated and analyzed according to burn-up or amount of fuel consumption. At this step, we have considered the amount of burn-up for 30 days operation at 5 MW reactor power (Usually timescales TRR is in a 30-day cycle) for LEU and HEU cases [6, 8, 9].

**Second step:** kinetic parameters of the TRR have been investigated according to the effect of temperature feedbacks for two cases of LEU and HEU. At this step, we have determined the kinetic parameters for the temperature changes from 20 °C to 100 °C. However, for the change in the density of material in the core,
equation (2) has been used [12]:

\[ \rho = \frac{m}{V} = \frac{m}{V_0(1 + 3\alpha \Delta \theta)} = \frac{m/V_0}{1 + 3\alpha \Delta \theta} = \frac{\rho_0}{1 + \beta \Delta \theta} \]

where, \( \rho_0 \) refers to element density in the initial temperature (20 °C), \( \alpha \) and \( \beta \) are the linear expansion and volume expansion coefficients, respectively, \( \Delta \theta \) is the temperature changes, and \( \rho \) is the density of elements in the new temperature or desired temperature.

Here, we considered the temperatures of fuel meat and cladding on bases of ordinary and water on bases of accident conditions in a nuclear research reactor. This is a hypothetical model that can be used after the reactor shutdown or any scenarios of accidents in research reactors for analyzing neutronic and kinetic parameters.

2.1. \( \frac{1}{2} \) poisoning method

Perturbation theory method is normally used for the calculation of \( \ell_p \) parameter. For this purpose, we can add a small amount of absorbent material to the ingredients of the core, homogeneously, and compute the new \( keff \). Thus, the absorber material of boron (\( ^{10}\text{B} \)) which is very dilute (in the range of \( 10^{-8} \) to \( 10^{-9} \) atom/barn.cm) is added to each zone of reactor for creation of multi group cross section. Then, the value of \( keff \) is obtained in these conditions. Finally, we can get \( \ell_p \) by using the equation (3), [3, 5, 13].

\[ \ell_p = \lim_{N_0 \to 0} \frac{k_{eff} - k_{eff}^{B}}{k_{eff}^{B}} \times \frac{1}{N_0 \sigma_{ab} v} \]

where, \( k_{eff}^{B} \) is the effective multiplication factor by adding impurities, \( N_0 \) is boron density in terms of \( (10^{-8} \) atom/barn.cm), \( \sigma_{ab} \) is the microscopic absorption cross section of the neutron absorber (3843 barn), and \( v \) is the average thermal neutron velocity (2200 m s\(^{-1}\)). In fact, the difference between \( k_{eff} \) and \( k_{eff}^{B} \) is very small because of the concentration of \(^{10}\text{B} \) is very dilute. Thus, the sufficient number of histories in MCNPX code simulations should be tracked which requires a long computer time.

3. Results and discussion

The first core configuration of the TRR is simulated by the MCNPX code and depicted in figure 1. This core configuration of the TRR has 5 CFE (Control Fuel Element) and 14 SFE (Standard Fuel Element). According to table 1, uranium density in the meat is 2.9617 g cm\(^{-3}\) and the amounts of uranium per fuel plate and per SFE are 76 g and 290 g, respectively. Nevertheless, simulations are performed in three-dimensional detailed geometry as shown in figure 1. Finally, the kinetic parameters are calculated and analyzed for various conditions such as changing enrichment fuel, the effects of fuel consumption (burn-up) and the temperature changes in the core reactor for two cases of LEU and HEU. The obtained results are presented in two steps as follows. It is noted that the accuracy of the code input for Kcode card to calculate kinetic parameters at all steps are 1000 cycles and the number of particles history has been considered as \( 1 \times 10^7 \).

First step: After modeling the reactor core, the effects of fuel burn-up on the TRR kinetic parameters are investigated for 30 days operation at 5 MW reactor power. The kinetic parameters were calculated taking into account the effect of burn-up for two cases of LEU and HEU. Finally, the data obtained from this code are presented in figures 2–4. Figure 2 presents the reduced value of \( keff \) due to the reduction of fissile material by increasing fuel burn-up. Consequently by increasing the enrichment, this parameter has increased up to 14.8%.

The amount of fuel consumption estimated for TRR during 30 days operation at 5 MW reactor power is 29.2 MWd/kgU that is equal to 2.89% of U-235 initial number. The measured value for this parameter is \( 1.02850 \pm 0.00010 \) in the end of cycle (EOC) for case of LEU in SAR [6]. Therefore, the relative difference
between the results of this code with the measured value is about 1.05%. Also, the statistical uncertainty associated with criticality calculations in MCNPX code is approximately 10.0 pcm.

We present in figure 3 the calculated value of $\beta_{\text{eff}}$ for two cases of LEU and HEU in the TRR during 30 days operation for temperature of 25°C. As shown, the $\beta_{\text{eff}}$ values decreases in both cases as the time increasing, according to equation (1), table 2 and burn-up process. Table 2 shows summary of total and delayed neutron yield values. Usually, $\beta_{\text{eff}}$ reduces during the burn-up process due to the plutonium buildup and in the result, it causes to softening of neutron spectrum. Pu-239 has delayed neutron fraction less than U-235 (Note that U-238 has practically zero contribution to the fission reaction and its $\beta = \nu_{\text{f}}/\nu$ has no impact on $\beta_{\text{eff}}$). Because of the softening of neutron spectrum during the burn-up, neutron leakage from the core reduces and this leads to the decrease of the effect of delayed neutrons [11, 13–15].

We have calculated the value of $\beta_{\text{eff}}$ by MCNPX code, and found 785.64 pcm for case of LEU in the beginning of cycle (BOC) but, the measured value is 777.7 pcm in SAR, stated in the TRR [6]. In addition, other results found 813 pcm for the TRR [13, 15, 16]. Therefore, the relative difference between the results of our code with the measured value and the reference data in the calculations is about 1.1% and $-3.4\%$, respectively. In terms of
quantity of this parameter, in the cases of LEU and HEU and during the given time, the dropped percentages are 10.2% and 10.1%, respective.

Also, in the current study, the $\ell_p$ parameter has been calculated for two cases of LEU and HEU in the TRR during 30 days operation and the results are shown in figure 4. As can be seen, $\ell_p$ parameter is almost constant for both cases with respect to the time but the value of LEU is more than HEU. Furthermore, the obtained results for $\ell_p$ parameter for $\nu_1$ poisoning method and the SAR are 47.9 and 45.0 $\mu$s, respectively $[3, 17]$. The calculation results have been compared to the SAR. It was found that the relative difference is than 6.4%.

Second step: Finally, the effects of temperature changes on the TRR kinetic parameters are investigated by MCNPX code. At this stage, by putting TMP card in input program, kinetic parameter for temperatures of 20, 40, 60, 80 and 100 (degree centigrade) and for both cases are calculated and shown in figures 5–7. Also, the MAKXSF Code available in version 2.6 of the MCNPX code is used to produce cross-sections at different temperatures.

Figure 5 show that by increasing the temperature, the $k_{\text{eff}}$ parameter reduced for two cases of LEU and HEU in the TRR. Increment of the resonance cross-section causes the more neutrons capture in U-238. As a result, the generated neutrons in the fuel are reduced and for this reason the $k_{\text{eff}}$ parameter is reduced. On the other hand, with rising temperature and the reduction in density of moderator cause to reduce the number of moderated neutrons. Therefore, with reducing the number of thermal neutrons, the fission process reduces and consequently the $k_{\text{eff}}$ parameter reduces. Finally, the value of this parameter is decreased by temperature increasing at a rate of 2.7% and 1.8% for two cases of LEU and HEU, respectively.

### Table 2. Summary of total and delayed neutron yield values (ENDF/B-IV) $[15]$.

| Isotope | $\nu_d$ | $\nu^t$ | $\beta^t$ |
|---------|---------|---------|----------|
| U-235   | 0.0167  | 2.437   | 685      |
| U-238   | 0.0440  | 2.492   | 1766     |
| Pu-239  | 0.0042  | 2.895   | 145      |

$^a$ Thermal yield value.

$^b$ Total delayed neutron fraction.

![Figure 5. Variations of effective multiplication factor versus rising temperature.](image)

![Figure 6. Variations of effective delayed neutron fraction versus rising temperature.](image)
Also, as shown in figure 6, the $\beta_{\text{eff}}$ values decrease by raising the temperature for two cases of LEU and HEU in the TRR. The reason is that this parameter is similar to the $k_{\text{eff}}$ parameter. According to equation (1), by decreasing fission reaction the neutrons generated in the fuel are reduced as the results, the delay neutron precursors are decreased, thus, the $\beta_{\text{eff}}$ parameter is also decreased. However, the number of prompt neutrons decreases too. It is not obvious that the $\beta_{\text{eff}}$ reduces. The decreased rate found by increasing the temperature for two cases of LEU and HEU are 12.4% and 10.6%, respectively. In calculations, there is systematic error from MCNPX code for this parameter.

As shown in figure 7, the $\ell_p$ parameter increases with rising the temperature for two cases of LEU and HEU in the TRR. The value of $k_{\text{eff}}$ parameter has been decreased with increasing the temperature thus; we have found the rate of this parameter 4.9% and 5.8% for two cases of LEU and HEU, respectively.

Furthermore, in this research, the value of temperature feedback coefficients is calculated by the MCNPX code in the TRR and for two cases of LEU and HEU found to be $-12.7$ pcm K$^{-1}$ and $-11.5$ pcm K$^{-1}$, respectively. However, the value reported for this parameter in the SAR is $-14$ pcm K$^{-1}$ in the case of LEU in the core. The relative difference between the results of the MCNPX code with the value reported in the SAR for calculation of this parameter is about $-9.3%$ [6].

4. Conclusions

Reactor kinetics means the temporal behavior of the reactor without the factors that cause this behavior. In other words, the reactor kinetics is the same as the reactor dynamics for steady state. That is to say, if the core changes then these parameters will change. The kinetic parameters contain prompt neutron life time ($\ell_p$) and effective delayed neutron fraction ($\beta_{\text{eff}}$). Here, MCNPX code is used to simulate TRR kinetic parameters. The results of these simulations are validated by comparison to the results obtained by $\frac{\Delta}{\gamma}$ poisoning method and Safety Analysis Report (SAR). Then the effects of change in enrichment, burn-up and temperature are discussed below:

- The results show that with increasing of the enrichment, the $\ell_p$ parameter is reduced due to the reduction of the $k_{\text{eff}}$ and the $\beta_{\text{eff}}$ parameter remains approximately constant.
- Also, the results of this study show that the effect of burn-up has dropped up to 10.2% and 10.1% for the $\beta_{\text{eff}}$ parameter in both cases of LEU and HEU, respectively. By increasing the time, the $\ell_p$ parameter is almost constant for two cases of LEU and HEU.
- In addition, the results indicate that the $\ell_p$ parameter increases with raising temperature about 4.9% and 5.8% for both cases of LEU and HEU, respectively. The amount of $\beta_{\text{eff}}$ parameter reduces with raising temperature about 12.4% and 10.6% for two cases of LEU and HEU, respectively.
- Finally, a reader can conclude that any of the changes in enrichment, burn-up and temperature how effect on kinetic parameters which has a key role in analysing dynamics of nuclear reactors behaviour.

ORCID iDs

M Hassanzadeh © https://orcid.org/0000-0001-7655-2686
References

[1] Snoj L et al 2008 Monte carlo calculation of kinetic parameters for the TRIGA Mark II Research Reactor International Conference, Nuclear Energy for New Europe
[2] Muhammad F 2010 Kinetic parameters of low enriched uranium fuelled material test research reactor at end-of-life Ann. Nucl. Energy 37 1411–4
[3] Hassanzadeh M, Feghhi S A H and Khalafi H 2013 Calculation of kinetic parameters in an accelerator driven subcritical TRIGA reactor using MCNIC method Ann. Nucl. Energy 59 188–93
[4] Denise B P 2008 MCNPX User’s Manual Version 2.6.0, Los Alamos National Laboratory, LA-CP-07-1473
[5] Bretscher M M 1997 Evaluation of reactor kinetic parameters without the need for perturbation codes, International Meeting, CONF-87091894, Argonne National Laboratory, Argonne, Illinois USA
[6] FSAR for TRR 2009 Final Safety Analyses Report for Tehran Research Reactor Atomic Energy Organization of Iran, Tehran
[7] TRR Log Book 1993 Atomic Energy Organization of Iran (AEOI), Report Nuclear Research Center, Reactor Division, Iran, No. 24
[8] Hosseini S A and Vosoughi N 2010 Uncertainty evaluation of calculated and measured kinetics parameters of Tehran Research Reactor Nucl. Eng. Des. 240 2761–7
[9] Hosseini S A, Vosoughi N, Ghofrani M B and Gharib M 2010 Calculation, measurement and sensitivity analysis of kinetic parameters of Tehran Research Reactor Ann. Nucl. Energy 37 463–70
[10] Talamo A and Gohar Y 2010 Deterministic and monte carlo modeling and analyses of yulina-thermal subcritical assembly nuclear engineering division report, Laboratory (https://doi.org/10.2172/991100)
[11] Hassanzadeh M and Feghhi S A H 2013 Analysis of burn-up effects on kinetic parameters in an accelerator driven subcritical TRIGA reactor Ann. Nucl. Energy 62 280–3
[12] McLane V, Dunford C L and Rose P F 1976 Neutron Cross-Sections (New York: Academic)
[13] Arkani M, Hassanzadeh M and Khakshournia S 2016 Calculation of six-group importance weighted delayed neutron fractions and prompt neutron life time of TRR based on monte carlo method Prog. Nucl. Energy 88 1–12
[14] Chibinyaev A V, Alekseev P N and Teplov P S 2005 Estimation of the effect of neutron spectrum regulation on VVER-1000 fuel burnup Atomic Energy 101 680–3
[15] Lashkari A, Khalafi H and Kazeminejad H 2013 Effective delayed neutron fraction and prompt neutron lifetime of Tehran Research Reactor mixed-core Ann. Nucl. Energy 55 266–71
[16] Safaei Arshi S, Khalafi H and Mirvakili S M 2015 Preliminary thermal-hydraulic safety analysis of Tehran Research Reactor during fuel irradiation experiment Prog. Nucl. Energy 79 32–9
[17] Hassanzadeh M, Feghhi S A H and Khalafi H 2013 Calculation of the neutron importance and weighted neutron generation time using MCNIC method in accelerator driven subcritical reactors Nucl. Eng. Design 262 404–8