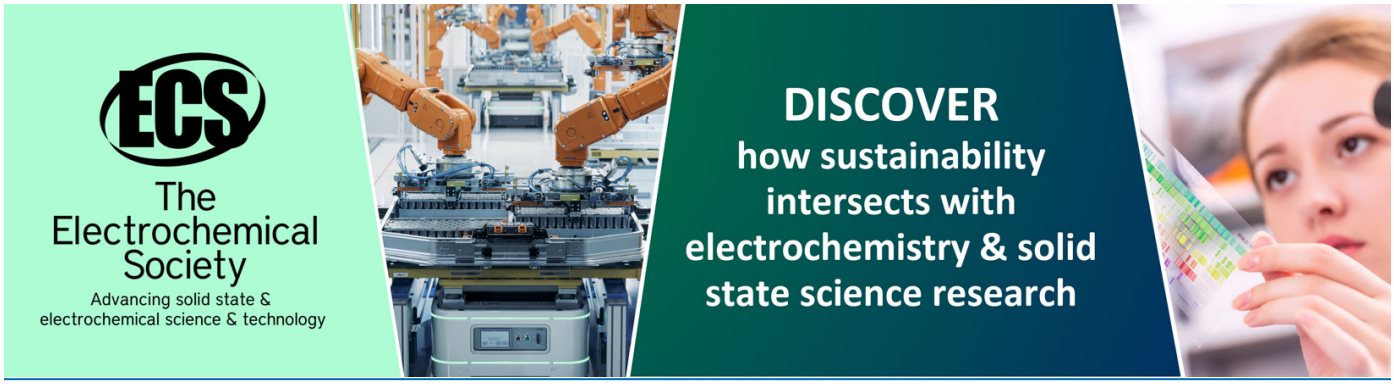
An overview of the development of a multigroup Monte Carlo code for TRIGA reactors



IOP Conf. Series: Materials Science and Engineering **555** (2019) 012007 doi:10.1088/1757-899X/555/1/012007

An overview of the development of a multigroup Monte Carlo code for TRIGA reactors

M R Omar[1], J A Karim[2] and T L Yoon[1, a)]

[1]School of Physics, Universiti Sains Malaysia, 11800 USM, Penang, Malaysia.
[2]Reactor Technology Center, Technical Support Division, Malaysian Nuclear Agency, 43000 Bangi, Selangor, Malaysia.

[a]tlyoon@usm.my

Abstract. This paper discusses the overview of the development of TRIMON, a next-generation core management code for TRIGA reactors. The newly developed code employs Monte Carlo method which uses diffusion theory type homogenized neutron cross section data, thus, speeding up the stochastic simulation via simplification of a complex core geometry. It has the capability of running three-dimensional core simulation up to 16 neutron energy groups. In general, the group neutron cross sections are pre-processed and homogenized via the Effective Diffusion Homogenization (EDH) method. The important features of TRIMON include core eigenvalue calculation, fuel burnup management, three-dimensional power and neutron flux distribution prediction. In this paper, the numerical results obtained using TRIMON were compared with the actual measured operational data. The performance comparison was also made between TRIMON and MCNP.

1. Introduction

In this work, the development of a new neutronics calculation code coupled with direct fuel burnup calculation for TRIGA reactors is reported. TRIMON is a stochastic code written in Fortran which integrates homogenized cross section data generated using diffusion theory with multi-group Monte Carlo method.

Majority state-of-the-art Monte Carlo codes are still inferior in core design and direct fuel burnup calculation despite their extremely powerful computational capabilities [1]. In this context, transferring the fuel burnup information of the previous core calculation to correct the neutron cross-section data for the use of current core calculation can be tedious in most general multi-purpose Monte Carlo codes. As a result, various strategies have been published by researchers on linking Monte Carlo codes to fuel depletion codes such as ORIGEN and CINDER [2]. However, these strategies are poorly outlined and require an intermediate linking code that is not accessible to all nuclear engineers.

Unfortunately, authors in [3] demonstrated that thousands of MCNP eigenvalue calculation cycles were required to converge at a good estimation of core effective multiplication factors, k_{eff}. Essentially, TRIMON addressed this issue by introducing diffusion-theory type homogenized neutron cross sections in the Monte Carlo method. Apparently, there is a lack of research on such initiative. Previous work [5] involves modifying scripts to feed in homogenized cross section data into MCNP, a

continuous energy Monte Carlo code. Yet, the validity and the theoretical aspect of such trial are poorly understood.

As such, we have attempted to outline a systematic procedure to computationally utilize homogenized group cross section data in Monte Carlo method. The new code is also capable of modelling three-dimensional reactor core, thus enabling nuclear engineers to study the axial power peaking characteristics of the reactor core. In order to account the effects of fuel depletion on reactor behaviour, it is also integrated with core direct fuel burnup calculation.

TRIGA is a commercial research reactor manufactured by General Atomics, USA. To date, the reactor has been installed in 24 different countries. The reactor has been used for various applications such as radioisotopes production, non-destructive testing, and research on the properties of matter and for education and training. The reactor is a pool-type water reactor and the reactor core is loaded with hydride fuel-moderator element, specifically UZrH.

Reaktor TRIGA PUSPATI (RTP) is a 1 MWth TRIGA installed at Malaysian Nuclear Agency, Bangi Malaysia. RTP core is a cylindrical-shaped core containing 127 designated core locations to accommodate fuel elements and other non-fuel elements such as control rods and irradiation facilities. The reflector is made up of graphite and the reactor assembly is equipped with four boron carbide control rods. RTP is fuelled with 8.5%wt, 12%wt and 20%wt UZrH fuel with the enriched to 20% U-235. The fuel is clad by a 0.05cm thick SUS-304 can.

2. Theory

2.1 Unit cell homogenization

At the most fundamental level, a reactor core can be constructed using its basic lattice structures which are known as unit cells. The geometry of a unit cells is defined such that the entire reactor core can be build using repetitions of the unit cell. For instance, the unit cells forming a TRIGA core are illustrated in Fig. 1.

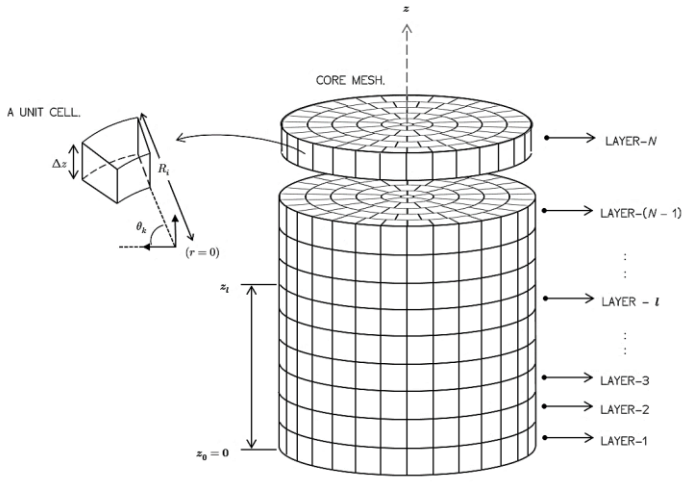


Figure 1. Geometry of a unit cell and TRIGA core mesh.

In TRIMON, each unit cell represents a homogenized reactor sub-region with constant homogenized group neutron cross sections, Σ. Ordinarily, when tracing the random walks of a neutron history, the current unit cell which contains the neutron is mapped so that the value of Σ for the cell can be retrieved and used for simulating the subsequent nuclear events.

2.2 Fuel burnup effect

In TRIMON, fuel burnup accumulation due to previous core operations is not neglected during cross section data pre-processing. The burnup, b, for each fuel cell in percent U^{235} is given by the general power series correlation,

$$b = \sum_{j=1}^{N} \beta_j p_{\text{cell}}^{j} (\Delta t)^j \tag{1}$$

where N is the order of the series, β_j are the coefficients of the series and Δt is the burnup increment in days. During neutron cross section data pre-processing, the fuel density and enrichment are corrected according to the prescribed value of b so that the number density of the fuel region within the unit cell can re-calculated.

2.3 Fuel temperature effect

Inevitably, fuel temperature gives a significant impact on homogenization calculation due to Doppler broadening effect of the neutron cross section at the resonance neutron energy region. For a typical TRIGA reactor, the temperature of a fuel meat in a unit cell is derived based on the empirical formula suggested by [1],

$$T_{\text{cell}} = T_{\text{cool}} + \sum_{n=1}^{N} a_n \alpha_r \alpha_z \left\{ \frac{m}{M} \left(1 - \frac{b}{100} \right) P \right\}^n \tag{2}$$

where T_{cool} is the temperature of coolant, N is the power series order, a_n is the coefficient of the power series, α_r is the radial power form factor at the fuel cell location, α_z is the axial power form factor at the fuel cell location, m is the mass of Uranium in the fuel rod, M is the total mass of uranium within the reactor core, b is the fuel rod burnup in percent, and P is the nominal core power.

2.4 Monte Carlo random walk overview

To begin with, neutrons are assumed to be moving in a straight line in the direction $\mathbf{\Omega}$ from an original position, $\mathbf{r}$. In fact, all neutrons are only produced by fission reactions. According to [6], other types of neutron production can be neglected, and their effects are imperceptible.

In general, a random walk step of a neutron is initiated at position $\mathbf{r}$ and ends at the position of the next collision event, $\mathbf{r}_c$. Next, the current unit cell, Λ, is mapped where the position $\mathbf{r}$ is located inside Λ. Thus, the distance to the next collision, d_c, can be predicted,

$$d_c = -\frac{\ln \xi}{\Sigma_t} \tag{3}$$

where Σ_t is the total homogenized neutron cross section of the unit cell Λ and ξ is a random number where $\xi \in [0,1)$.

Next, the distance to the nearest boundary of the current unit cell, d_b, along the neutron trajectory direction, $\mathbf{\Omega}$, is determined. If the distance to the next collision event is greater than the distance to the nearest cell boundary, i.e. $d_c > d_b$ then the neutron is transported to the boundary instead. Otherwise, the neutron is transported to the next collision point.

At the collision site, the neutron will proceed for a neutron–nuclear interaction. All interactions such as neutron scattering, absorption and fission reaction are processed according to their corresponding algorithms outlined by [7]. On the other hand, if the neutron is transported to the cell boundary, the distance to the next collision will be calculated using the total neutron cross section of the neighbouring unit cell. This process reiterates until the neutron is absorbed by a nucleus or escapes out of the reactor core. When a neutron is absorbed, the simulation stops and the next neutron from the fission bank are simulated.

In general, eigenvalue calculation is a transport simulation of neutrons studying the ability of a system, i.e. a nuclear reactor core, to sustain fission chain reactions. In reactor theory, the eigenvalue, k_{eff}, is interpreted as the ratio between the number of neutrons in successive generations, with the fission reaction regarded as the neutron birth event that distinguishes the generations of neutrons.

In general, TRIMON employs the method of successive generations (MSG) which was first introduced by [8]. Interestingly, MSG necessitates history tracking of a finite number of neutrons from one generation to another. Initially, an initial guess of k_{eff} and the fission site distribution, $\Psi(\mathbf{r})$, must be done. Here, $\Psi(\mathbf{r})$ contains the information where fission reaction is possible. As the eigenvalue calculation progresses, k_{eff} and $\Psi(\mathbf{r})$ will stochastically converge and equilibrate. It is important to ensure that the value of k_{eff} and $\Psi(\mathbf{r})$, converge and this can be ensured by skipping a few fission generation cycles before tally accumulation is made. In fact, $\Psi(\mathbf{r})$ converges slower than that of k_{eff} [9].

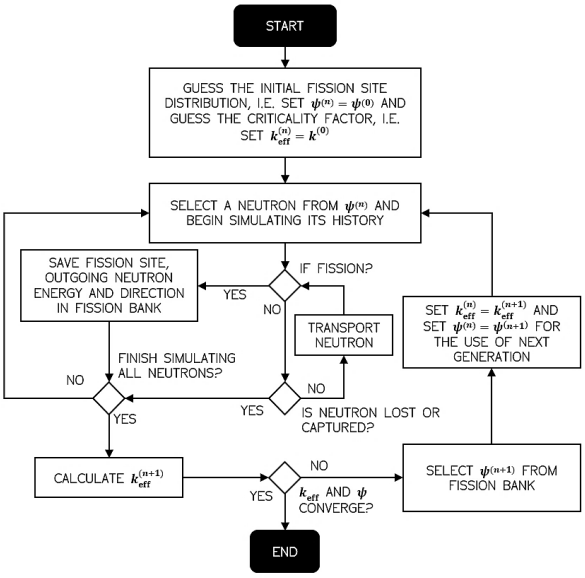


Figure 2. Method of successive generations (MSG) implemented in TRIMON.

3. Results and discussions

Throughout all eigenvalue calculation done in this work, the temperature of the coolant was set at 293K, the reactor core was xenon-free, and the power of the reactor is at negligible thermal power (<0.01kW). A total of 50000 neutron histories were simulated per fission cycle.

3.1 TRIMON validation with RTP operational data

The details of the operation histories of RTP for the first three operational cores are given in Table 1. The summary of the results of the comparison is given in Table 2. In the operational core benchmark problem, the mixed core configurations of Puspati TRIGA Reactor (RTP) which consists of 8.5%wt and 12%wt $UZrH_{1.6}$ fuels were simulated using TRIMON. The results obtained using TRIMON were compared with the actual operational data. Each core configuration is assigned to a specific core identification, i.e. Core-*N* for the N^{th} operational core. When a specific core configuration has reached its end of cycle and reconfigured, the new core configuration is assigned to a new identification, i.e. Core-(*N*+1).

Table 3 indicates a reduction of k_{eff} at the end of each operational core cycle (EOC). Then, the value of k_{eff} was restored every time the core is reconfigured at the beginning of the cycle (BOC).

During fuel reconfiguration, the existing high burnup fuels may be replaced with new fresh fuels, and some fuels loaded in the core may be reshuffled with the existing lower burnup fuels.

Table 1. First three operational core histories.

Configuration	Average Power (kW)	Total Burnup Given (MWd)	Accumulated Burnup (MWd)	Operating Hours (days)
Core-1	455.015	43.4	43.4	95.3
Core-2	600.580	45.8	89.2	76.4
Core-3	674.223	42.6	131.8	63.1

Table 2. Summary of eigenvalue calculations using TRIMON for Core-1 to Core-3. Measured k_{eff} obtained from the RTP operational log book is also given. Difference between calculated and measured, Δk_{eff}, is also given.

Core	k_{eff} TRIMON	k_{eff} Measured	Δk_{eff}
Core-1 BOC	1.05190 ± 0.00028	1.05312 ± 0.00038	0.00122 ± 0.00047
Core-1 EOC	1.03310 ± 0.00030	1.03095 ± 0.00037	0.00215 ± 0.00048
Core-2 BOC	1.05701 ± 0.00043	1.05377 ± 0.00039	0.00324 ± 0.00058
Core-2 EOC	1.03702 ± 0.00038	1.04004 ± 0.00038	0.00298 ± 0.00054
Core-3 BOC	1.05432 ± 0.00026	1.05323 ± 0.00039	0.00109 ± 0.00047
Core-3 EOC	1.04258 ± 0.00039	1.04193 ± 0.00038	0.00065 ± 0.00055

BOC: Beginning of operational core cycle.
EOC: End of operational core cycle.

3.2 TRIMON performance benchmark with MCNP

In this benchmark problem, the eigenvalue calculations of RTP first operational core (Core-1) was performed using TRIMON and MCNP. This section reports the evaluation of the gain brought by TRIMON compared to the local information lost due to the use of homogenized neutron cross section data. Figure 3 shows TRIMON imposes a slightly greater relative error of k_{eff} compared to MCNP. However, TRIMON exhibits better simulation time where the computational rate is at 15 fission cycles per minute compared to MCNP which is at 5 fission cycles per min.

In this work, the initiative to integrate homogenized group data in the Monte Carlo method is merely to reduce the complexity of the core model and to increase the performance of the Monte Carlo simulation. Notably, this approach reduces the spatial variation of neutron cross sections within the reactor core model. However, this attempt may introduce local information lost due to spatial averaging, thus, increasing the relative error of k_{eff}. Interestingly, the presented result shows that the performance gain brought by TRIMON is greater than the local information lost. Thus, the efficiency of TRIMON and MCNP is expressed using the figure of merit (FOM) as given in Table 3. Note also that a higher FOM value indicates better computational efficiency.

Table 3. TRIMON and MCNP performance result for Core-1. Calculation was done using 50000 neutrons per fission cycle with 200 fission cycles.

Parameters	TRIMON	MCNP
Simulation Time, T (min)	13.7	45.6
Converged k_{eff}	1.05154	1.05971
Δk_{eff}	0.00817 ± 0.00057	
Relative Error, R	0.00044	0.00032
Figure of Merit, $1/R^2T$	517744	154784

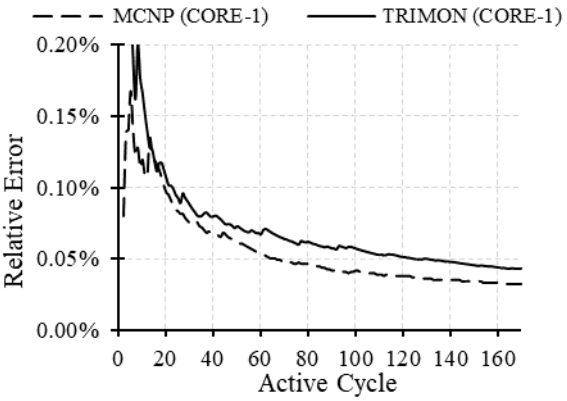


Figure 3. Relative error evolution for Core-1.

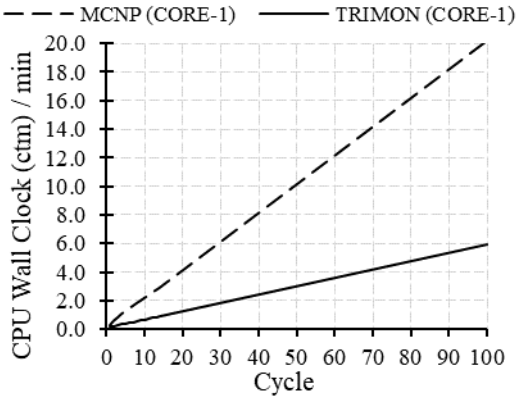


Figure 4. CPU wall clock of the first 100 cycles using TRIMON and MCNP.

4. Conclusions

In summary, the development of a Monte Carlo core calculation code for TRIGA reactors, TRIMON, is presented. TRIMON was first tested and applied to RTP. Essentially, the new code integrates local fuel burnup effect in Monte Carlo simulation, supplanting the existing state-of-the-art Monte Carlo codes which are still less optimal in terms of direct fuel burnup calculation and simulation performance.

The simulation results obtained from the first test run of the code indicates a good agreement with the experimental results obtained from the actual reactor operational data. Perhaps, the integration of diffusion-type homogenized neutron cross section with Monte Carlo method proposed in this paper can serve as a guideline for future researchers to explore the feasibility of using the similar technique in other nuclear reactor types.

5. References

[1] Peršič A, Žagar T, Ravnik M, Slavič S, Žefran B, Ćalić D, Trkov A, Žerovnik G, Jazbec A and Snoj L 2017 "TRIGLAV: A program package for TRIGA reactor calculations," *Nucl. Eng. Des.*, **318** p 24–34.

[2] Goorley T, James M, Booth T, Brown F, Bull J, Cox L J, Durkee J, Elson J, Fensin M, Forster R A, Hendricks J, Hughes H G, Johns R, Kiedrowski B, Martz R, Mashnik S, Mckinney, Pelowitz D, Prael R, Sweezy J, Waters L, Wilcox T and Zukaitis T 2016 Features G of MCNP6 *Ann. Nucl. Energy*, vol. 87, p 772–783.

[3] Rabir M H, Md Zin M R and Usang M D 2016 Neutron flux and power in RTP core-15 *AIP Conf. Proc.* **1704(1)** p 50018.

[4] Wang K, Li Z, She D, Liang J, Xu Q, Qiu Y, Yu J, Sun J, Fan X and Yu G 2015 RMC - A Monte Carlo code for reactor core analysis *Ann. Nucl. Energy* **82** p 121–129.

[5] Kuijper J C, Van Der Marck S C and Hogenbirk A 2007 Using Homogenized macroscopic group cross sections in continuous-energy Monte Carlo neutron transport calculations with MCNP *Jt. Int. Top. Meet. Math. Comput. Supercomput. Nucl. Appl.*
[6] Persic A, Slavic S, Ravnik M and Zagar T 1998 *TRIGLAV - a program package for research reactor calculations*.
[7] Romano P K and Forget B 2013 The OpenMC Monte Carlo particle transport code *Ann. Nucl. Energy* **51** p 274–281.
[8] Lieberoth J 1968 Monte Carlo Technique to solve the static eigenvalue problem of the boltzmann transport equation *Nukl. 11* **213(19)**.
[9] Cho N Z and Chang J 2009 Some outstanding problems in neutron transport computation *Nucl. Eng. Technol.* **41(4)** p 381–390.