Core Calculation of 1MWatt PUSPATI TRIGA Reactor (RTP) using Continuous Energy Method of Monte Carlo MVP Code System

A.K Julia[a], I. Shibata[b]

[a] Technical Support Division,
Malaysian Nuclear Agency

[b] Japan Atomic Energy Agency, Japan

Abstract. The Monte Carlo MVP code system was adopted for the Reaktor TRIGA PUSAPTI (RTP) core calculation. MVP was developed by a group of researcher of Japan Atomic Energy Agency (JAEA) first in 1994. MVP is a general multi-purpose Monte Carlo code for neutron and photon transport calculation and able to estimate an accurate simulation problems. The MVP Monte Carlo code calculation is based on the continuous energy method. This code is capable of adopting an accurate physics model, geometry description and variance reduction technique faster than conventional method. With compared to the conventional scalar method, this code could achieve higher computational speed by several factor on the vector super-computer[1]. In this calculation, RTP core was modeled as close as possible to the real core and results of keff, flux, fission densities and others obtained from MVP were outputted.

1. Introduction

RTP is a light-water moderated and pooled type research reactor with 1 MW capability. It was built in 1979 and attained the first criticality on 28 June 1982. The RTP was designed mainly for neutron activation analysis, small angle neutron scattering, neutron radiography, radioisotope production, education and training purposes. It uses standard TRIGA fuel developed by General Atomic in which the zirconium hydride moderator is homogenously combined with enriched uranium[2]. The RTP core has a cylindrical configuration surrounded with an annular graphite reflector and enclosed in the aluminium casing tank. The side and top view is presented as in Fig. 1 and 2. The specification of RTP is tabulated as in Table 1. There are 127 locations in the light-water moderator core, which can be filled either by fuel elements or other components like control rods, irradiation channels and etc. Elements are arranged in seven concentric rings: A, B, C, D, E, F, G and having 1, 6, 12, 18, 24, 30, 36 locations respectively. The distances between locations in a given ring are equal according to Safety Analysis Report of the reactor. The reactor encompassed of stainless steel-clad fuel elements with 20% enrichment for 8.5wt%, 12wt% and 20wt% uranium concentration type, graphite elements, control rods, irradiation channels and a neutron source. To ensure the integrity and capability of the core, fuel shuffling have been carried out several times. Until now, there were 13 configurations of the core with the most recent changes was carried out in July 2007.

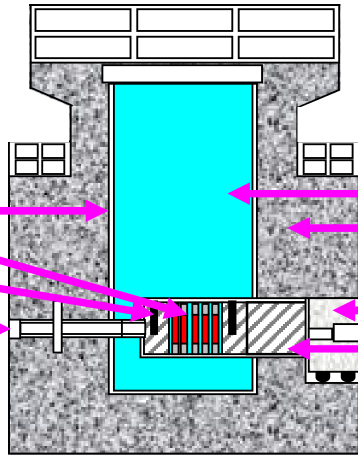

Figure 1. The side view of RTP

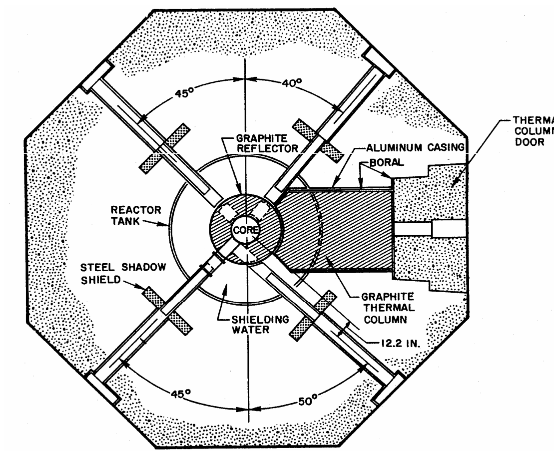

Figure 2. The top view of the reactor

Table 1. The RTP description

Items	**Description**
Type	Pool type
First Criticality	28 June 1982
Maximum Thermal Power	1MW
Average Power Density	22.8w/cm^3
Maximum Thermal Neutron Flux	1 x 10^{13}n/cm^2/s
Shape and Size of Reactor Core	Cylindrical, 110cm D x 89cm H
Coolant / Moderator	Light water
Reflector	High Purity Graphite
Fuel material	Uranium Zirconium Hydride
Enrichment of U-235	Approx. 20%

2. Neutronics Calculation Using MVP Code System

2.1. Calculation model

The RTP core configuration-11th has been calculated using MVP. In the 11th configuration, about 83 fuel elements of 8.5wt%, 15 fuel elements of 12wt% and 10 fuel elements of 20wt% were used. About three numbers of 8.5wt% fuel follower control rod were used in this configuration and it makes 111 of fuel elements were used in total. In this calculation, treatment of the nuclear reactions is based on continuous-energy method which the cross-sections was processed from the evaluated nuclear data files. In this case, neutron library of JENDL 3.3 was used. The cross section data was calculated beforehand as mention before. Table 2 below tabulated the design parameters and material composition data that used for RTP core calculation.

Table 2. Design parameters and material composition of RTP

	Fuel Element			Fuel Follower Control Rod
Geometrical data				
Radius of Zr rod (cm)	0.3175			0.3175
Radius of fuel (cm)	1.765			1.665
Air gap (cm)	0.05			0.05
Cladding (cm)	0.05			0.05
Fuel composition				
Mass of uranium				
Mass of U^{235}				
Uranium (wt%)	8.5	12	20	8.5
Enrichment (wt%)	19.9			19.9
H:Zr ratio	1.6			1.6
Absorber				B_4C
Natural Boron (%)				80%

Material (density: g/cm^3)	Atomic number density			
	Nuclide	Number density (10^{24} atom/cm^3)		
Fuel meat		8.5wt%	12wt%	20wt%
($UZrH_{1.6}$)	^{235}U	2.48433E-04	3.48056E-04	6.55951E-04
	^{238}U	9.87344E-04	1.38327E-03	2.60694E-03
	^{1}H	4.36057E-02	4.16180E-02	4.49906E-02
	^{nat}Zr	2.72535E-02	2.60113E-02	2.81192E-02
Zr rod (6.52)	^{nat}Zr		4.30407E-02	
Mo-ring (10.2)	^{nat}Mo		6.40237E-02	
Graphite (1.65)	^{12}C		8.02198E-02	
SS304 (7.889)	^{nat}Cr		1.72080E-02	
	^{55}Mn		1.50007E-03	
	^{nat}Fe		5.90284E-02	
	^{nat}Ni		7.42164E-03	
Control rod (2.465)	^{10}B		2.05170E-02	
	^{11}B		8.41080E-02	
	^{12}C		2.61562E-02	
Grid plate (anodized aluminium)	^{27}Al		6.02612E-02	
Air gap	^{16}N		3.78621E-05	
	^{16}O		1.01568E-05	
Coolant (0.99644)	^{1}H		6.66E-02	
	^{16}O		3.33E-02	

In MVP, procedure to obtain physical quantities is performed with estimators. The statistical estimation of physical quantities from random process is known as tallying. There are four types of estimators, namely; collision estimator, track length estimator, point detector estimators and surface crossing estimators [1]. In RTP case, only collision estimator and track length estimator were used. The collision estimator is a scoring method which estimated by scoring the contribution in the collisions as follows:

$$C_i = h(\Gamma_i)\, W_i\, \delta v_T(\Gamma_i)$$

Where;

C_i	=	number of the i-th events scored
Γ_i	=	the i-th collision point of a history
W_i	=	weight of the particle at the collision
$\delta_i(\Gamma_{i)}$	=	1 for $\Gamma_I \in V_T$
	=	0 for $\Gamma_I \notin V_T$

The track length estimator is used when the physical quantities could not be estimated with the collision estimator especially in vacuum region. The track length estimator could possible estimated small number of collision which is not accurately done with collision estimator. The estimated scores of i-th events as follows:

$$C_i = \sigma(L_i) W_i L_i \delta v_T(\Gamma_i)$$

Where;

L_i	=	track length of the particle passing through the rgion V_T
W_i	=	weight of the particle in the flight
$\sigma(L_i)$	=	Reaction rates are to be assumed constant in space and time along the path L_i

In order to reduce variances in the calculation, variance reduction technique was adopted to improve the statistical accuracy of Monte Carlo calculation [1]. The eigenvalue problem was treated to obtain the multiplication factor for a system containing fissile materials. The RTP core was modeled using cylindrical with central axis is parallel to the z-axis geometry. It is divided into 350 regions purposely for tallying the physical quantities, i.e. flux and fission densities data by the collision estimators and track length estimators.

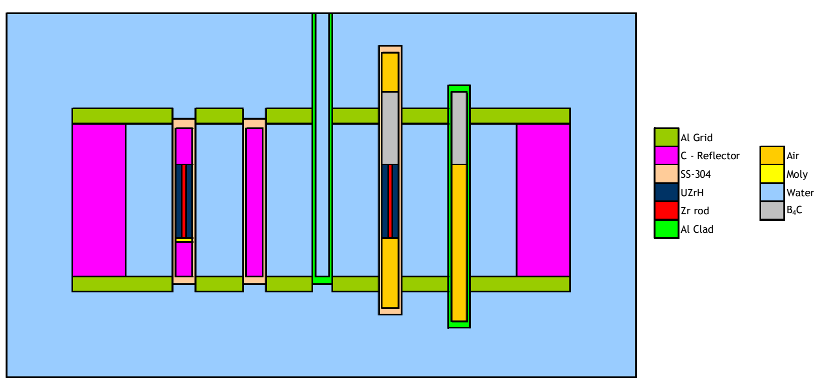

Fig.7 RTP core model

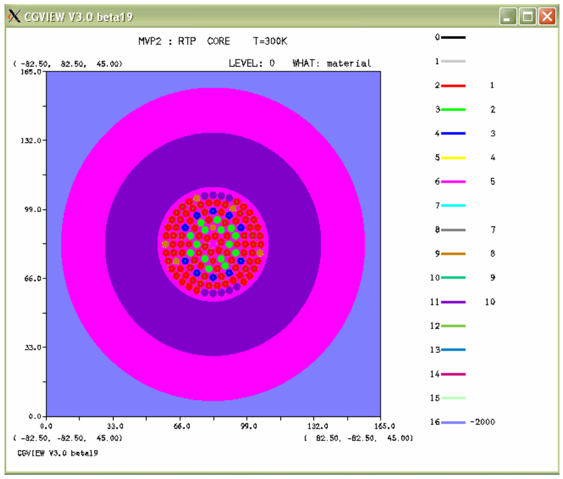

Fig.8 Geometrical representation of RTP core in MVP

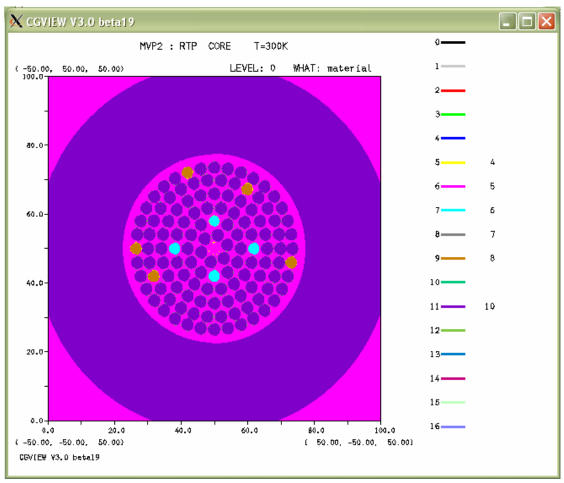

Fig. 9 Homogenized of RTP core cell

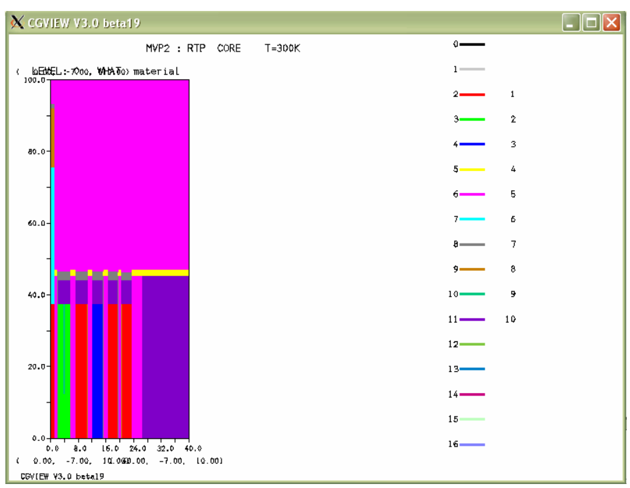

Fig.10 RTP core region in MVP

2.2. *Flux calculation*

From the MVP output, physical parameters as keff, flux, fission densities and others were outputted. Although these values were obtained, it is only outputted its relative values and therefore this values is necessarily converted into absolute value by using below equation:

$$\Phi(\text{real}) = \Phi(\text{MVP output}) * \text{standard factor}$$

The standard factor could be obtained by dividing real power with MVP power.

$$\text{standard factor } (sf) = \frac{Pr}{Pm}$$; Pr, Pm is a real power and MVP power

The flux then was calculated using as below equation:

$$\begin{aligned} \text{fission } &= VN\sigma_f \Phi_{TH} \\ &= V\frac{\rho}{M} \times 6.02 \times 10^{23}\, \sigma_f \Phi_{TH} \end{aligned}$$

where,

V	=	volume of one fuel element
N	=	number density
σ_f	=	thermal scattering cross section, 800 barns
Φ_{TH}	=	thermal flux at 800 x 10^{-24} cm^2

3. Results and Discussion

Below are the results for k-eff from MVP output.

RESULTS BY THE PRINCIPLE OF MAXIMUM LIKELIHOOD

TRACK LENGTH	KEFF= 1.07517E+00 (0.1575%)
COLLISION	KEFF= 1.07477E+00 (0.1636%)
ANALOG	KEFF= 1.07418E+00 (0.1728%)
PRODUCTION	KEFF= 1.07431E+00 (0.1764%)
NEUTRON BALANCE	KEFF= 1.07517E+00 (0.1541%)
ALL	KEFF= 1.07464E+00 (0.1504%)

RANGE IN 1 STANDARD DEVIATION

TRACK LENGTH	1.073473170 < K < 1.076859173
COLLISION	1.073012374 < K < 1.076529089
ANALOG	1.072327021 < K < 1.076039890
PRODUCTION	1.072412416 < K < 1.076202498
NEUTRON BALANCE	1.073510336 < K < 1.076823712
ALL	1.073020162 < K < 1.076251781

RANGE IN 2 STANDARD DEVIATION

TRACK LENGTH	1.071780168 < K < 1.078552175
COLLISION	1.071254017 < K < 1.078287446
ANALOG	1.070470586 < K < 1.077896325
PRODUCTION	1.070517375 < K < 1.078097539
NEUTRON BALANCE	1.071853649 < K < 1.078480400
ALL	1.071404352 < K < 1.077867591

RANGE IN 3 STANDARD DEVIATION

TRACK LENGTH	1.070087167 < K < 1.080245176
COLLISION	1.069495659 < K < 1.080045803
ANALOG	1.068614151 < K < 1.079752760
PRODUCTION	1.068622334 < K < 1.079992581
NEUTRON BALANCE	1.070196961 < K < 1.080137087
ALL	1.069788542 < K < 1.079483401

* Real variance is estimated by the method of Nuc.Sci.Eng., 125,1(1997). *

K-eff Estimator	K-eff	SigA(%)	SigR(%)	SigR/SigA
M.L.E. of NEUTRON BALANCE estimator	1.07517E+00	0.1541	0.2361	1.5325
M.L.E. of ALL estimators	1.07464E+00	0.1504	0.2409	1.6022

- Results for other estimators are reasonable.

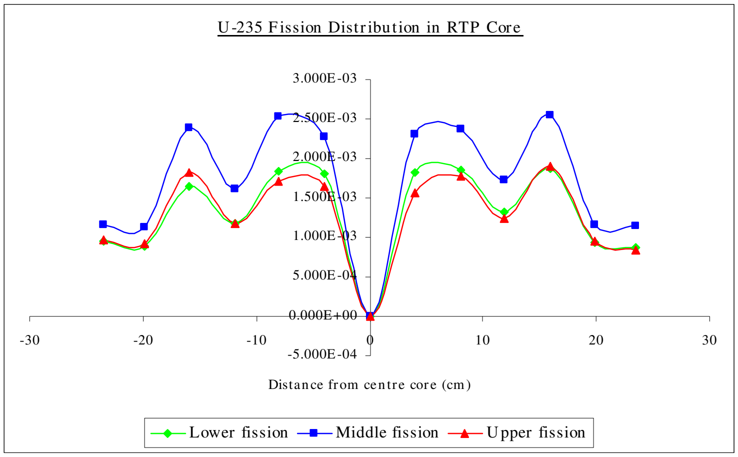

Fig. 12 Fission distribution in RTP Core

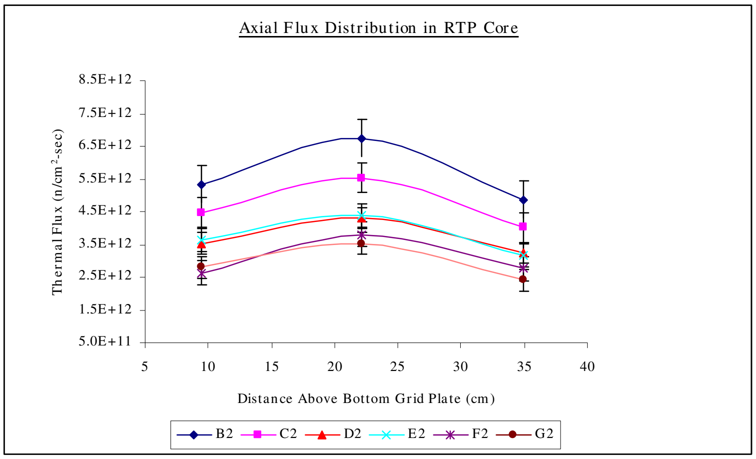

Fig. 13 Axial flux distribution in RTP Core

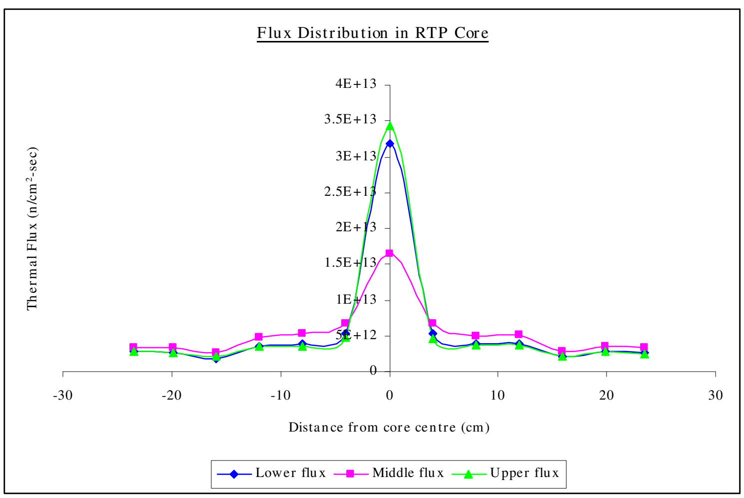

Fig. 14 Thermal flux distribution in RTP Core

4. Conclusion

The MVP code is a high-performance code which is very convenient to use and it is applicable to many other purposes, not limited to the reactor core calculation. The RTP has been modeled using this code for the first time and will continue using it in the core calculation and other appropriate application such as shielding calculation and design parameters and others. Further work will be carried out to refine the reactor core and burn up calculation also will be performed purposely for the RTP neutronic analysis. In order to achieve best estimation for DSA studies, MVP code was one of the option to be consider especially in neutronics analysis of the reactor core.

ACKNOWLEDGEMENTS

The author would like to thanks people in Japan Atomic Energy Agency for their support and sharing knowledge during this work done and also people who involved wether directly or indirectly during the preparation of this conference paper.

REFERENCES

[1] Nagaya Y. et al, 2005, MVP/GMVP II: General Purposes Monte Carlo Codes for Neutron and Photon Transport Calculations based on Continuous Energy and Multigroup Methods, Japan Atomic Energy Agency

[2] Safety Analysis Report for PUSPATI TRIGA Reactor, 2006, pp. 97-98