การวิเคราะห์เครื่องปฏิกรณ์ปรมาณูวิจัยด้วยรหัสคอมพิวเตอร์เอ็มวีพี

นทีกูล เกรียงชัยพร

ฝ่ายจัดการเครื่องปฏิกรณ์ สถาบันเทคโนโลยีนิวเคลียร์แห่งชาติ โทรศัพท์ 02-579-5230

Email: Nateekool@hotmail.com

บทคัดย่อ

รหัสคอมพิวเตอร์ทริกาพี (TRIGAP) ได้ถูกนำมาใช้ในการวิเคราะห์การจัดแกน และเชื้อเพลิงสำหรับเครื่องปฏิกรณ์ ปวว-1/1 ในช่วงปี พ.ศ. 2533 ซึ่งรหัสคอมพิวเตอร์ทริกาพีนี้ได้ถูกพัฒนาขึ้นมาสำหรับการวิเคราะห์แกนทริกาโดยเฉพาะ แต่เนื่องจากรหัสคอมพิวเตอร์นี้ได้ถูกพัฒนามาเป็นเวลานานแล้ว วิธีการที่นำมาใช้ในการคำนวณนั้นจึงมีการประมาณการและสมมติฐานค่อนข้างมาก ดังนั้นจึงมีการคิดที่จะนำรหัสคอมพิวเตอร์อื่นที่สามารถคำนวณได้ถูกต้องและแม่นยำมากกว่ามาใช้เป็นทางเลือก โดยในปัจจุบันมีการนำวิธีการคำนวณแบบมอนติคาร์โล (Monte Carlo) มาใช้ในการวิเคราะห์เครื่องปฏิกรณ์อย่างแพร่หลาย รหัสคอมพิวเตอร์เอ็มวีพี (MVP) เป็นรหัสคอมพิวเตอร์หนึ่งที่ใช้วิธีการคำนวณแบบ มอนติคาร์โล ที่สามารถวิเคราะห์ปัญหาได้ในระบบ 3 มิติ และใช้พลังงานต่อเนื่อง รวมทั้งยังมีโมดูลเอ็มวีพีเบิร์น (MVP-BURN) ที่ใช้ในการคำนวณค่าการเผาผลาญเชื้อเพลิงด้วย รายงานฉบับนี้จะแสดงการเปรียบเทียบค่าวิกฤติที่ได้จากการคำนวณด้วยเอ็มวีพีและค่าที่ได้จากผลการทดลอง ซึ่งให้ผลการเปรียบเทียบเป็นที่น่าพอใจ

คำสำคัญ: มอนติคาร์โล เครื่องปฏิกรณ์วิจัยของไทย เอ็มวีพี ปปว.-1/1

TRR-1/M1 Core Analysis with MVP

Nateekool Kriangchaiporn

Reactor Management Section, Thailand Institute of Nuclear Technology Tel. 02-579-5230

Email: Nateekool@hotmail.com

Abstract

Since early 1990s, the in-core fuel management of TRR-1/M1 has been performed by TRIGAP. This code was specifically developed for reactor physics calculations of the TRIGA-type reactor. However, because of its limitations in geometrical and cross sectional options, the attempt of using other techniques/codes are provoked. Nowadays, the choice of using the Monte Carlo method to perform core analysis becomes more satisfaction with acceptable computational time. The MVP is one of the codes that utilize the Monte Carlo method with continuous-energy library. It is able to explicitly model the problem in 3-D geometry. It also has a burn-up calculation feature called MVP-BURN. The aim of the current work is to apply the MVP code for TRR-1/M1 core analysis. In this paper, the MVP code was verified with the experiment results for the fresh core and some burn-up cores. The calculated-eigenvalue results agree well with the experimental data within an acceptable range of statistical error.

Keyword: Monte Carlo, Thai-Research Reactor, MVP, TRR-1/M1

Introduction

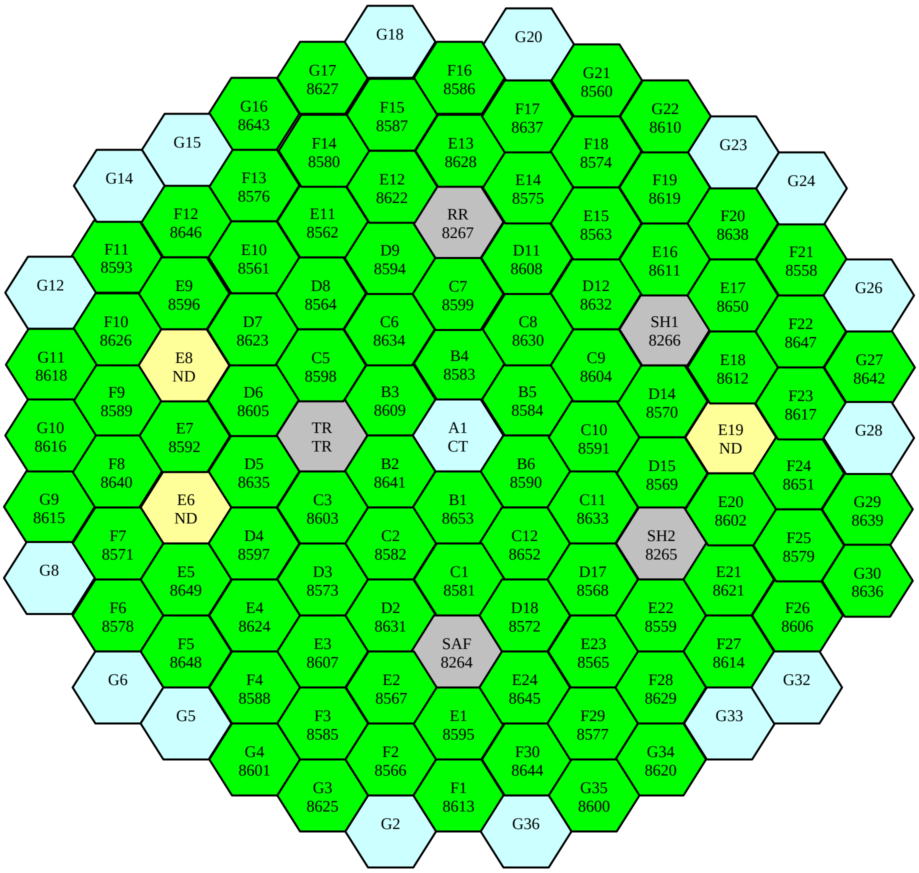
In mid 1980s, the TRIGAP code was introduced for Thailand Research Reactor-1/Modification1 (TRR-1/M1) fuel management and core analysis. The TRIGAP was purposely developed for TRIGA Mark II reactor, which has the annular array. This code has some disadvantages because of the simplifications and approximations applied to the code. TRIGAP is based on two-group diffusion equation (group boundary at 1 eV) in one dimensional cylindrical geometry. It is solved in the finite differences approximation by fission density iteration method. The physical model of TRIGAP is not appropriate for the problems involving strong spectral and spatial variations of neutron flux distribution due to the two-group approximation and ring homogenization. It is appropriate for simple compact uniform loading patterns with only one type of fuel elements in the same ring. It fails to predict correct burn-up for mixed rings or for regions near control rods and in-core irradiation channels. As a consequence, the attempts of using other techniques/codes are provoked.

Recently, the Monte Carlo technique is widely used for reactor criticality and core analysis calculations. Even though this method needs more computational time than other methods, it can handle complicated problems with heterogeneous model in three-dimensional (3D) geometry and continuous energy. For this reasons, in order to overcome the disadvantages of TRIGAP, the Monte Carlo technique is chosen to perform the TRR-1/M1 core calculations. The MVP code, which is available at TINT, is one of the nuclear codes that using the Monte Carlo technique. This paper presents the verification of the MVP code against experimental data of TRR-1/M1 for control rod worth, core excess reactivity, and criticality calculations.

Methodology

General Description of TRR-1/M1

The TRR-1/M1 is a TRIGA Mark III-type reactor, in which the core was converted from MTR-type core in 1975. It is arranged in the hexagonal shape. The core is composed of the rings moving radially outward labeled as the B-, C-, D-, E-, F-, and G-ring, respectively, from the center position (CT). The TRIGA reactor uses uranium-zirconium-hydride (UZrH) fuel, which has a prompt negative temperature coefficient. In the reactor, neutron is moderated by light water and ZrH, and is reflected by water and graphite. The reactor has a steady-state thermal power rating of 2 MW originally loaded only with 8.5 wt-% uranium fuel elements as shown in Fig.1. Later, in 1978, the 20-wt% uranium fuel element was introduced in the core in order to extend the useful lifetime of the fuel element. Since then, the burned 8.5-wt% uranium fuels were gradually replaced with the fresh 20-wt% uranium fuels core-by-core. Fig. 2 shows the configuration of TRR-1/M1 core of the second core. At present, for core16, the core consists of 105 standard fuel rods (56 of 8.5 wt % and 49 of 20wt%), 4 control rods (fuel-follower), a transient rod (air-follower), 3 neutron detectors, 1 Am-Be neutron source and an in-core IR production facilities.



CT : Central Thimble (A1)

: Neutron Detectors (E6, E8, E19)

: 8.5 wt% Fuel

: Water

TR: Transient Rod (C4)

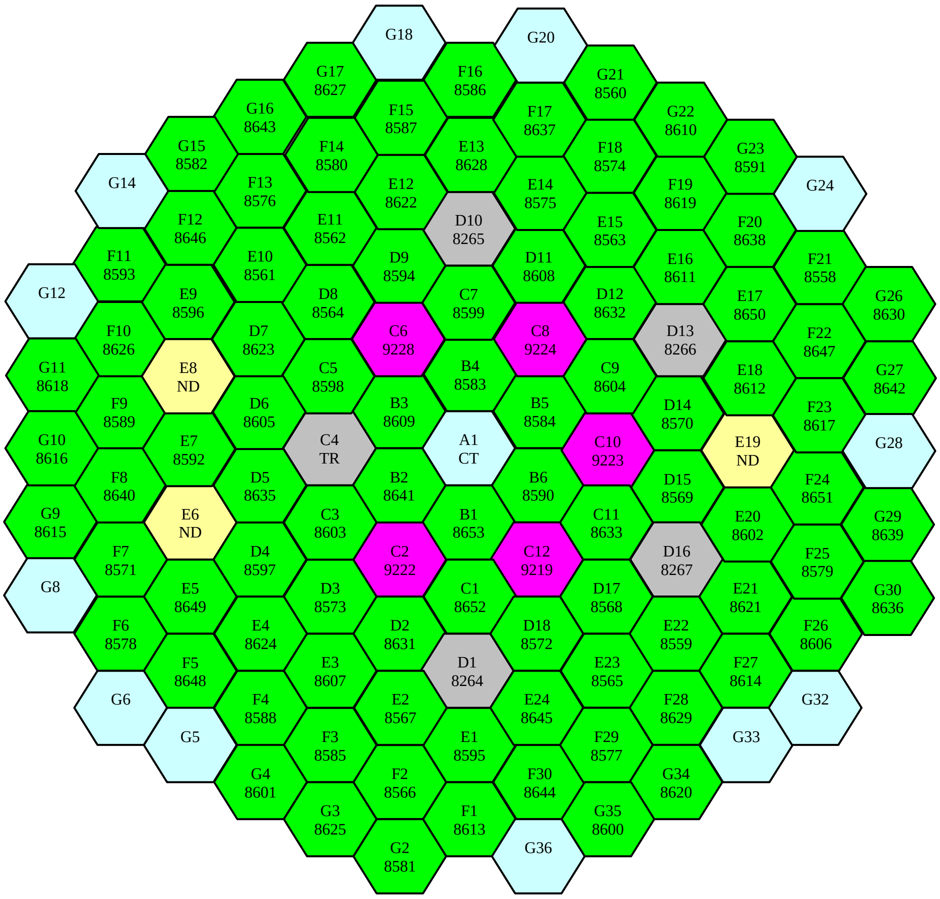
RR: Regulating Rod (D10)

SH1: Shim Rod (D13)

SH2: Shim Rod (D16)

SAF: Safety Rod (D1)

Fig. 1 Configuration of the TRR-1/M1 Core I



CT : Central Thimble (A1)

: Neutron Detectors (E6, E8, E19)

: 8.5 wt% Fuel

: 20 wt% Fuel

: Water ND: Neutron Detectors (E6, E8, E19)

TR: Transient Rod (C4)

RR: Regulating Rod (D10)

SH1: Shim Rod (D13)

SH2: Shim Rod (D16)

SAF: Safety Rod (D1)

Fig. 2: Configuration of the TRR-1/M1 Core II

MVP Description

In this paper, the MVP code was used to perform the TRR-1/M1 core analysis. MVP is a continuous energy Monte-Carlo code, it can handle complicated structures with minimum geometrical approximations. Specific cross section libraries are generated from the evaluated nuclear data (JENDL-3.3, ENDF/B-VI, JEF-3.0 etc.) by using the LICEM code. The neutron cross sections in the unresolved resonance region are described by the probability table method. The neutron cross sections at arbitrary temperatures are available for MVP by just specifying the temperatures in the input data. In addition, MVP has a coupling code called MVP-BURN and a burn-up calculation module BURN which solves a depletion equation analytically based on the modified Bateman's method with microscopic capture, fission and (n,2n) reaction rates obtained with MVP. The MVP-BURN is well validated by several burn-up benchmark calculations and analyses of post irradiation experiments.

Criticality and Burn-up Calculations

The core criticality and burn-up calculations were performed with the MVP and MVP-BURN codes using ENDF/B-VI pointwise cross-section library for all nuclides. The burn-up calculations were performed in three-dimensional geometry modeled for the whole core of each core cycle using the standard chain model (u4cm6fp50bp16T). The Predictor-Corrector (PC) method was applied for all time steps. In the PC-method, MVP calculations are done twice in each time step (beginning of step and end of step) to get averaged microscopic reaction rates during a burn-up time step interval. The number density of each pin were obtained and kept track at the end of each burn-up core calculation. For the criticality calculation, 20 inactive cycles were run, followed by 80 active cycles, each of 5000 histories. Table 1 shows the core burn-up parameters using in this study.

Table1: Core burn-up parameters

ITEM	VALUE
Power (MW)	1
Fuel Temp ($^{\circ}$C)	355
Moderator Temp($^{\circ}$C)	39
Cladding Temp($^{\circ}$C)	327
Burn-up period (MWD)	
Core1	61.23
Core2	76.00
Core3	86.72
Core4	47.29
Core5	119.64
Core6	62.65

Results and Discussions

Control rod worth

The control rod worth of Core1 at the beginning-of-cycle (BOC) was calculated under cold and clean condition. For each control rod, two eigenvalue calculations were performed, one with fully inserted position and another with fully withdrawn position, while the other control rods are at the middle positions. The individual control rod worth was calculated by Eq.1.

$$CR\,Worth = \frac{k_1 - k_2}{k_1 k_2} \Big/ \beta_{eff} \qquad \text{Eq.1}$$

Where: k_1 is the eigenvalue of CR fully withdrawn case

k_2 is the eigenvalue of CR fully inserted case

β_{eff} is the effective delayed neutron fraction for TRIGA, 0.007

The measured and calculated control rod worths are presented in Table2. The results show that the difference of the predicted rod worth and experimental data are less than 50 cents with ±10 cents deviation for each rod. The total rod worth of Core 1 is overestimated by 1.37 dollars.

Table2: Control Rod Worth for Core1@BOC

	Control rod Worth ($)					
	Transient	**Shim1**	**Shim2**	**Safety**	**RR**	**total**
Experiment	3.25	2.72	2.79	3.11	3.14	15.01
MVP	3.66±0.10	3.01±0.10	3.13±0.10	3.27±0.10	3.31±0.10	16.38±0.50
Diff (MVP-Exp.)	0.41±0.10	0.29±0.10	0.34±0.10	0.16±0.10	0.17±0.10	1.37±0.50

Power Distribution

The individual element power was calculated under unrodded conditions. Fig.2 shows the normalized power distribution, calculated using Eq.2, at BOC of Core 1. The obtained results are within 1% uncertainty. It is found that the power peaking factor is 1.72 at B5 located in ring B, which is in the expected ring.

$$NP_i = P_i / \overline{P} \qquad \text{Eq.2}$$

Where: NP_i is the normalized power of the fuel element in the *i* position

P_i is the power produced by the fuel element in the *i* position

$\overline{P}$ is the average power produced by a fuel element in the core

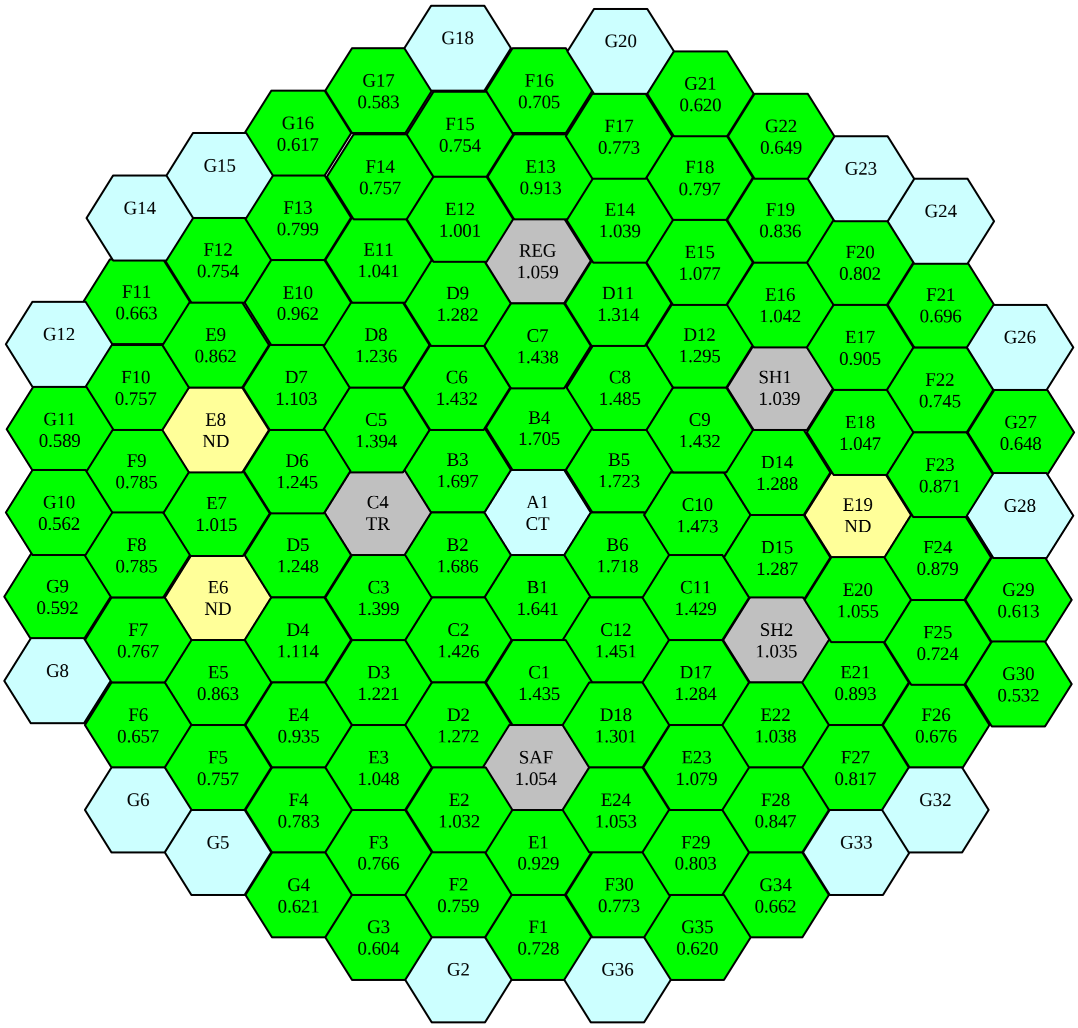


Fig. 3: **NP distribution of the TRR-1/M1 Core 1 for ARO**

Excess reactivity

The excess reactivities of critical cores 1 through 7 were determined when all control rods at their completely withdrawn positions as shown in Table3. The calculations were performed at the beginning of each core cycle under cold and clean condition. For the experimental data, the core excess reactivity was obtained by the sum of each control rod reactivity at its position under critical and cold condition. The results illustrated that the calculated core excess reactivities are overestimated in the range of 50 to 85 cents within 10 cents deviation.

Table3: Core Excess Reactivity

CORE	EXPERIMENTAL (DOLLARS,$)	CALCULATED (DOLLARS,$)	DIFF= CAL-EXP (DOLLARS,$)
1	7.43	8.04±0.10	0.61±0.10
2	6.87	7.67±0.10	0.80±0.10
3	6.06	6.79±0.10	0.73±0.10
4	6.00	6.82±0.10	0.82±0.10
5	7.01	7.83±0.10	0.82±0.10
6	5.92	6.45±0.10	0.53±0.10
7	5.96	6.45±0.10	0.49±0.10

Conclusion

Several techniques were used for neutronics calculations. As known, the Monte Carlo method provides good reference data among all the methods. In this paper, the MVP code was chosen to perform the TRR-1/M1 core analysis. The calculated results were verified with the experimental data. The comparisons show good agreement within standard deviation for control rod worth of core1 and core excess reactivity of cores 1 to 7.

Acknowledgement

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