Fuel management methodology upgrade of Thai Research Reactor (TRR-1/M1) using SRAC computer code

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Abstract. This paper presents the effort to upgrade the fuel management methodology of Thai Research Reactor-1/Modification 1 (TRR-1/M1) which is currently under responsibility of Thailand Institute of Nuclear Technology (TINT). The more advanced SRAC computer code is being introduced to replace the TRIGAP computer code for the fuel management calculation of TRR-1/M1. With the new methodology, the hexagonal lattices of TRR-1/M1 can be modeled without approximating the lattices into cylindrical rings as performed by the TRIGAP computer code. In addition, the SRAC computer code is able to provide pin-wise results such as normalized power distribution which is unable to obtain by the TRIGAP computer code. Also, the paper compares the core excess reactivity of core loading 1 and core loading 2 calculated by SRAC computer code with the measurement data from the operation log book. The comparison shows good agreement between the calculated values and measured values. With the promising result, the SRAC computer code is expected to be employed as the usual fuel management methodology for TRR-1/M1 in the near future.

1. Introduction

Thailand Institute of Nuclear Technology (TINT) is currently responsible for the nuclear research reactor called "Thai Research Reactor-1/Modification 1" (TRR-1/M1). The existing fuel management tool for TRR-1/M1 is a computer code called TRIGAP [1] which was developed in Slovenia during the 80's. Although TRIGAP is capable of calculating reactor parameters such as core excess reactivity or neutron fluxes, this tool has several drawbacks. Since TRIGAP only models the spatial distribution of neutrons in cylindrical geometry, the TRR-1/M1 core, which is formed in hexagonal lattices, needs to be homogenized into cylindrical rings. As a result, TRIGAP is unable to provide pin-wise data such as normalized power distribution of the reactor. To overcome this, an upgrade to the existing methodology is proposed by employing a more advanced computer code SRAC [2] which has been developed in Japan since 1978. The SRAC computer code is capable of modeling the hexagonal lattice of TRR-1/M1 without the cylindrical ring approximation. In this paper, the modeling of TRR-1/M1 by the SRAC computer code is investigated to study the feasibility of adopting the SRAC computer code as the usual fuel management tool for TRR-1/M1.

2. Overview of TRR-1/M1

Historically, the Thai government built the first nuclear research reactor as an MTR type research reactor and named it as Thailand Research Reactor 1 (TRR-1). The reactor had been operated since 1972 until 1975 when it was converted to the TRIGA type research reactor. In the conversion, the high-enriched uranium fuel plate type was replaced by low-enriched uranium fuel rod type designed and marketed by General Atomics (GA). Moreover, the control system and the safety features were also replaced so that TRR-1 became essentially a TRIGA reactor. The reactor was then renamed as TRR-1/M1 to reflect this conversion and the new reactor has been operating since November 1977. As of now, TRR-1/M1 has gone through 16 core loading configurations. The current core configuration (core number 16) uses 20% enriched UZrH fuel which is loaded into two types of fuel rods, namely

8.5% wt and 20% wt uranium and both fuel rod types have SS304 cladding. Moreover, the 20% wt% fuel rod contains about 0.5% wt. Erbium as burnable poison which is intended to extend the operation lifetime of TRIGA fuel and provides significant fraction of the prompt negative temperature coefficient for reactivity feedback. The reactor cooling is provided by natural circulation of pool water, which is in turn cooled and purified in external coolant circuits. The fuel rods are positioned in a grid plate forming hexagonal configuration. The TRR-1/M1 uses five control rods, i.e., a safety rod, a regulating rod, two shim rods and a transient rod. The regulating, shim and safety rods are sealed in 304 stainless steel tubes while the transient rod has aluminum clad. The steady-state reactor power level of TRR-1/M1 can be varied up to 2 MW (thermal) and the main purposes of TRR-1/M1 are for isotope production, researches, education and training.

3. Comparison between the existing methodology and the proposed methodology

Similar to the standard methodology used for the nuclear power reactor, the existing and the proposed methodologies of TRR-1/M1 are executed in 2 steps. The first step (normally referred as cross section generation step) is performed in order to create the cross section libraries representing each unique lattice type of the reactor. For the existing methodology, the cross section libraries were already pre-generated by a 2D reactor physics code and the libraries are included as part of the TRIGAP code. On the other hand, the proposed methodology provides capabilities to generate the cross section libraries by using PIJ (utilizing 2D collision probability method) module of the SRAC code. Comparing the existing methodology to the proposed methodology for the cross section generation step, the modeling and the conditions used for the production of the pre-generated libraries may be inconsistent with the actual TRR-1/M1. While, for the proposed methodology, the modeling and the conditions of the actual TRR-1/M1 are considered in the production of the libraries.

After the cross section generation step, the second step (normally referred as reactor core calculation step) is executed to determine the reactor parameters. In the existing methodology, the TRIGAP code is unable to model the hexagonal rings of TRR-1/M1; thus, the hexagonal rings of TRR-1/M1 have to be homogenized into cylindrical rings as shown in Figure 1.

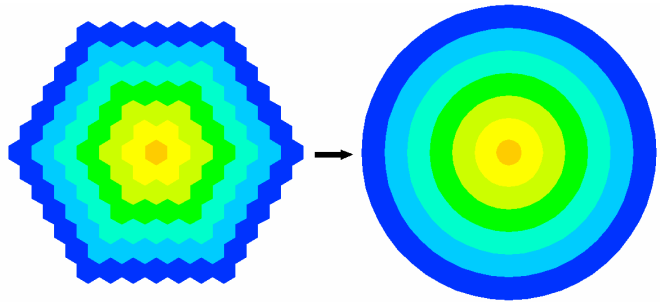

Figure 1: Fuel homogenization by the existing methodology

As a result, the TRIGAP code is unable to provide pin-wise quantities (e.g., pin-wise flux, pin-wise uranium content or pin-wise power) since the model is not pin-wise but ring-wise. Moreover, TRIGAP is only utilizing two group modeling for energy treatment which is possibly not flexible enough for neutron energy treatment of TRR-1/M1. On the contrary, the proposed methodology using the SRAC computer code is possible to model exactly hexagonal lattice configuration of TRR-1/M1 and the SRAC computer code can also utilize multi group for neutron energy treatment in the reactor core calculation step. From these comparisons, it is clear that the proposed methodology is more advantageous than the existing methodology.

4. TRR-1/M1 modeling by the proposed methodologies

TRR-1/M1 is modeled to investigate the accuracy of the proposed methodology. For group cross section generation, infinite arrays of 2D lattice models corresponding to unique lattice regions of the reactor are used. Each 2D lattice model represents an axial node which has the same material throughout axial direction. For instance, a fuel rod can be divided axially into several 2D lattice models. As for TRR-1/M1, there are three types of rods: fuel element, fuel follower control rod and air follower control rod. The axial divisions of each rod type into 2D lattice models in order to produce group cross sections are shown in Figure 2 to Figure 4.

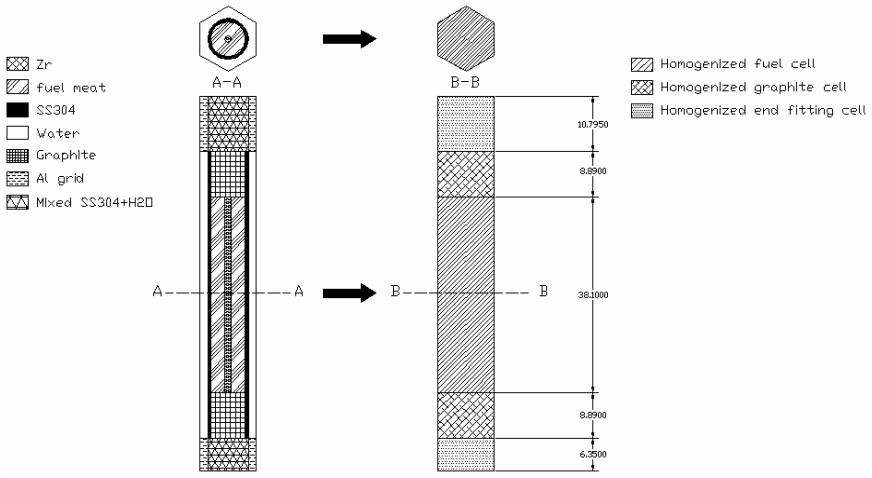

Figure 2: Axial homogenization of fuel elements

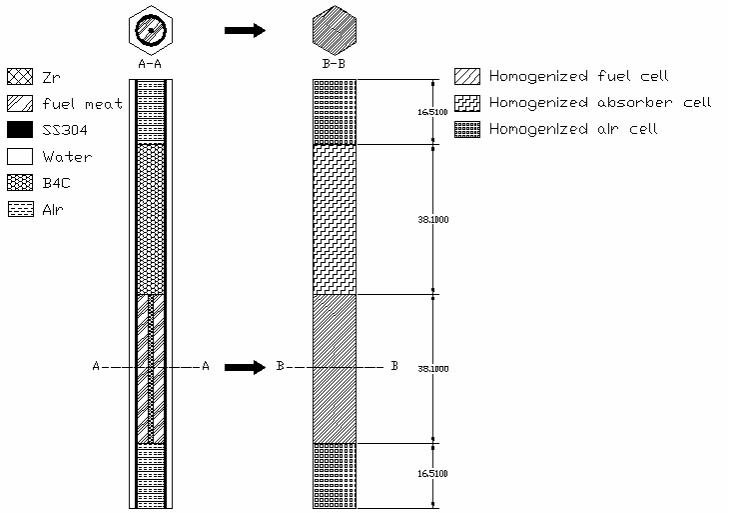

Figure 3: Axial homogenization of fuel follower control rods

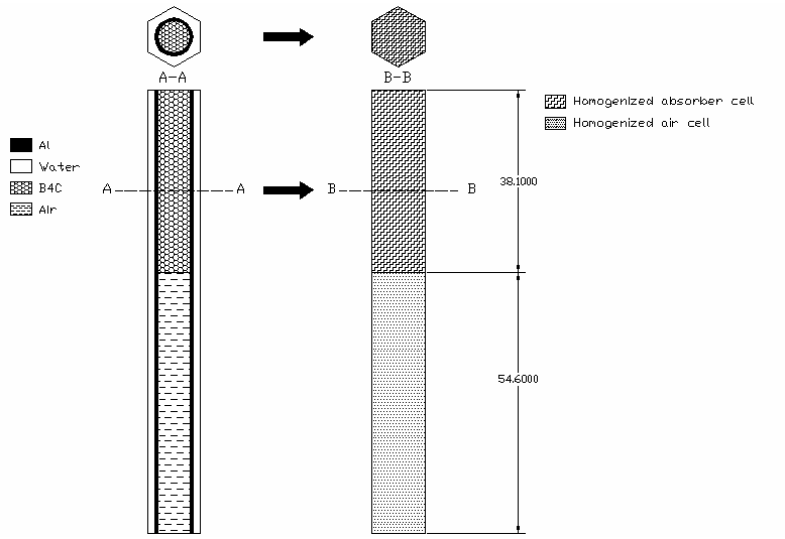

Figure 4: Axial homogenization of air follower control rods

To generate the group cross section libraries, two major modeling types are used, i.e., fuel lattice type and non-fuel lattice type. The fuel lattice type as shown in Figure 5 is employed for the group cross section production of the lattice with fuel.

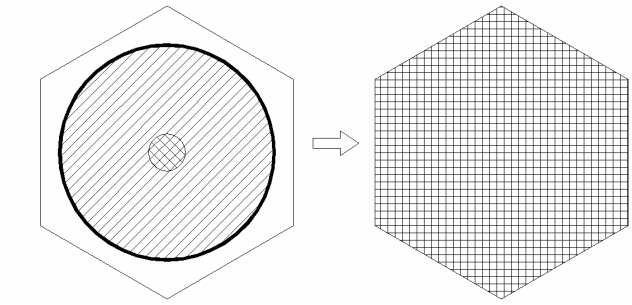

Figure 5: Homogenization of fuel type lattices

In this model, the six surfaces of the hexagonal lattice are reflective boundaries. The model consists of Zirconium rod in the center and it is surrounded by UZrH fuel. The cladding which surrounds the fuel meat is SS304 while the cladding is surrounded by outside water. The group cross section generation step produces the group cross sections using the hyperfine feature of PIJ module. This feature allows the collision probability method to utilize very fine energy group structure which is expected to be more accurate than typical fine group structure. Moreover, the group cross sections are created into 7 group structure. The group cross section libraries are generated at various burnup points from fresh to very high burnup and they are also generated at different power levels which correspond to different equilibrium temperatures. Table 1 specifies the equilibrium temperature of each material in the model as a function of the power level. It is noted that the equilibrium temperatures are obtained from the thermal-hydraulics prediction by the COOLODN code [3].

Table 1: Equilibrium temperature distribution in the 2D fuel lattice model as a function of power level

Power (kW)	T_{fuel} (C)	$T_{cladding}$ (C)	T_{water} (C)
200	173.12	120.04	39.49
400	227.74	128.24	44.51
600	277.78	133.96	47.43
800	325.54	139.09	50.51
1000	371.61	143.86	53.28
1200	416.29	148.34	55.75
1400	459.80	152.62	58.03
1600	502.30	156.75	60.48
1800	543.89	160.76	62.51
2000	584.66	164.67	64.47

On the other hand, the non-fuel type model is used for the group cross section production of the lattice without fuel as shown in Figure 6.

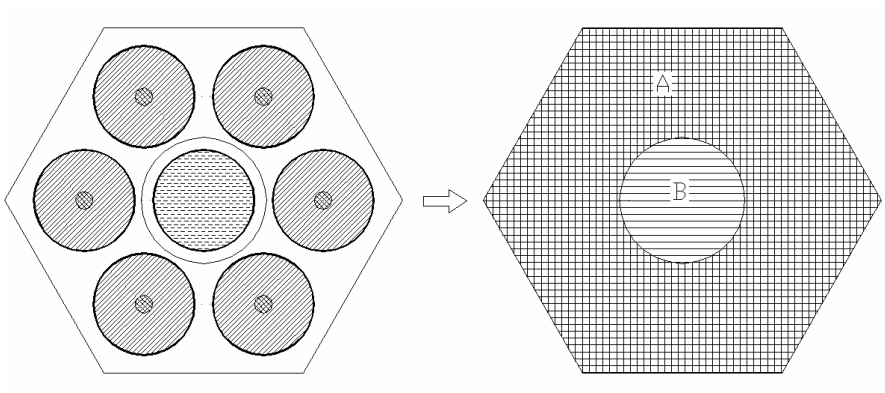

Figure 6: Homogenization of non-fuel type lattices

In this model, the hexagonal boundary of the non-fuel lattice (central circular cell) is approximated by a circular boundary (by conserving the area) and is surrounded by six typical fuel rods to provide the neutron source for the non-fuel lattice. The group cross sections of this model are created into 2 separate zones (A and B). The group cross sections of zone A are created from the regions of six fuel rods and surrounding water while those of zone B are created from the non-fuel lattice only. Therefore, only the group cross sections of zone B are used in the subsequent reactor calculation step. Like the fuel type lattice, the group cross sections of the non-fuel lattice are created into 7 group structure. Since the non-fuel lattice is considered to be non-burnable lattice, the burnup calculation of the non-fuel lattice is not performed. In addition, because the material temperature of

non-fuel lattices does not change significantly with the power level, the equilibrium temperature of each material in the non-fuel lattice is approximated to be constant at around 40C for all operating power range in the group constant generation step.

Following the group cross section generation step, the reactor calculation step is performed. The COREBN module of the SRAC system performs the reactor calculation using the group cross section libraries generated in the previous step. The reactor core is modeled by building blocks of different lattices representing different regions in both radial and axial directions. The COREBN model utilizes 90 axial layers with 60 axial layers representing the fuel section. The core loading 1 and core loading 2 of TRR-1/M1 are modeled by the SRAC code in order to validate the proposed methodology.

5. Results and Discussion

The K_{eff} of the "all-rods-out" model of the reactor core was derived and the excess reactivity was calculated by $(K_{eff} -1)/(Beta*K_{eff})$ where Beta is fraction of delayed neutrons (0.007). Table 2 presents the core excess reactivity results of TRR-1/M1 core loading 1 and core loading 2 with the comparison against operation data from the operation log book.

Table 2: Core excess reactivity results of TRR-1/M1 core loading 1 and 2

Model	K_{eff} by SRAC code	Calculated core excess reactivity by SRAC code	Operation core excess reactivity
Core #1	1.05809	7.84$	7.43$
Core #2	1.05252	7.13$	6.87$

As it can be seen from Table 2, the excess reactivity calculated by SRAC system agrees well with the operation data when considering that the operation value has inherently some amount of measurement uncertainty. There seems to be a bias of around 0.40$ between the calculated results and the operation data. In addition, the calculation by the SRAC code is able to provide normalized power distribution of the core. As an example, the normalized power distribution of core loading 1 is given as shown in Figure 7.

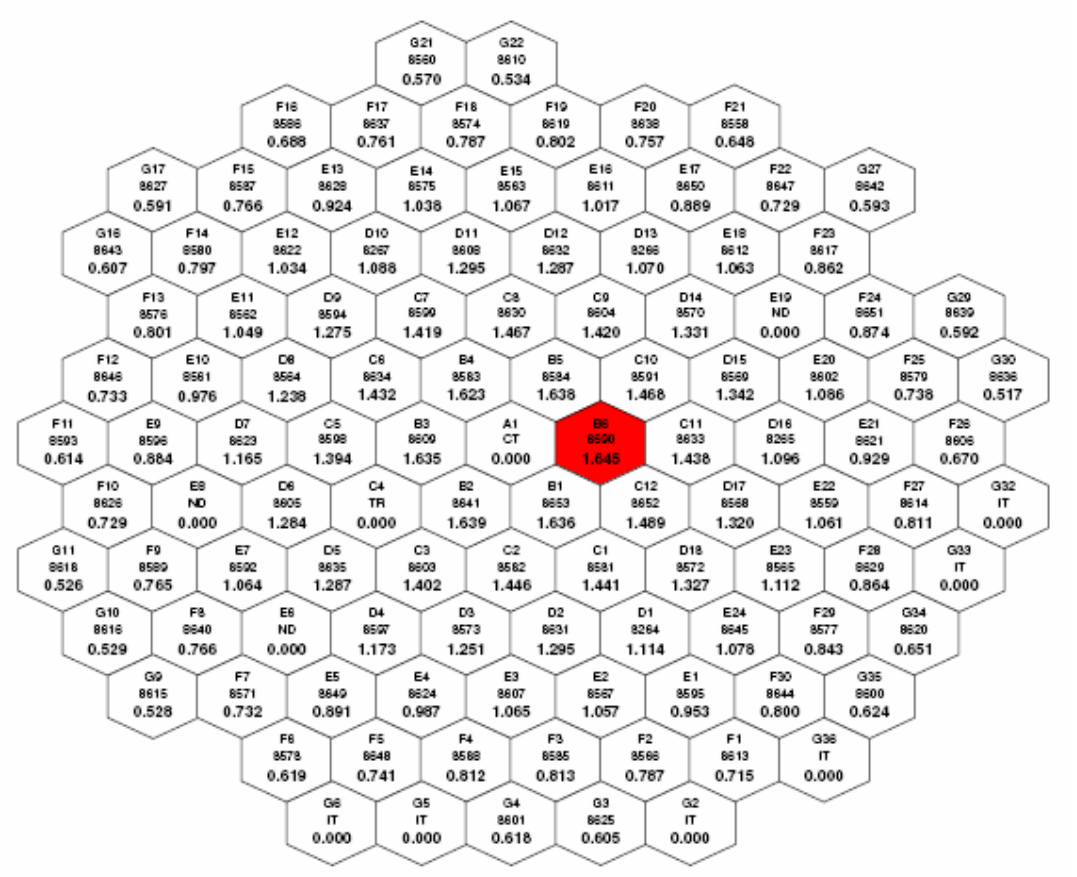

Figure 7: A plot representing normalized power distribution of TRR-1/M1 core loading 1

In addition, the change of core excess reactivity as a function of power level was calculated and is shown in Figure 8.

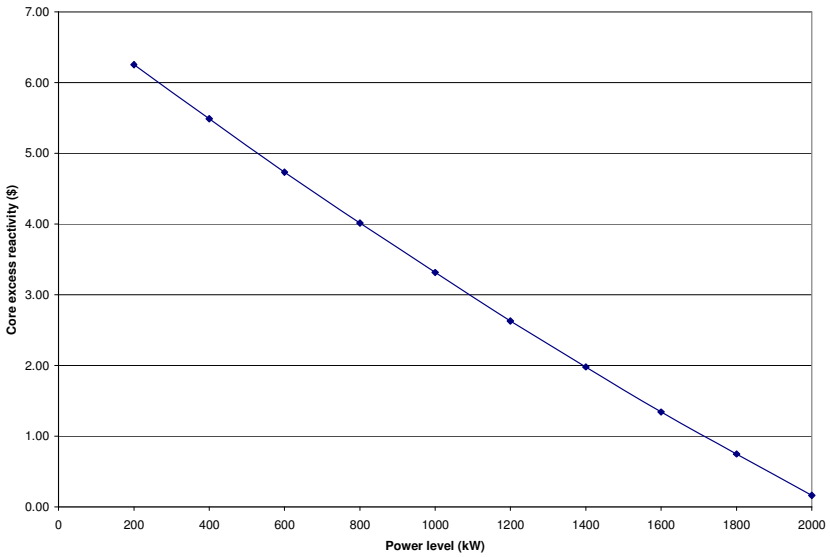

Figure 8: Core excess reactivity of TRR-1/M1 core loading 1 as a function of power level (kW)

From Figure 8, the core excess reactivity decreases quite linearly as a function of power level as excepted. This result confirms the negative reactivity feedback from power.

6. Conclusion

The upgrade of the fuel management calculation scheme for TRR-1/M1 is proposed and investigated in this paper. The proposed methodology utilizes a more advanced computer code to perform the fuel management calculation. The proposed methodology is advantageous over the existing methodology in many aspects. Most importantly, the proposed methodology is superior to the existing methodology that it is capable of providing pin-wise parameters such as pin-wise power distribution which is essential for assessing the safety of the reactor. The modeling of core loading 1 and core loading 2 of TRR-1/M1 by the proposed methodology shows good agreements with the operation data. It is expected that the proposed methodology will be employed as the standard fuel management scheme in the near future.

7. References

[1] I. Mele, M. Ravnik, 1985, TRIGAP – A computer programme for research reactor calculations

[2] K. Okumura, T. Kugo, K Kaneko and K. Tsuchihashi, 2002, SRAC (Ver. 2002); The comprehensive neutronics calculation system

[3] M. Kaminaga, 1994, COOLOD-N2: A computer code for the analyses of stead-state thermal-hydraulics in research reactors