การใช้รหัสคอมพิวเตอร์ SRAC เพื่อการจัดการเชื้อเพลิงนิวเคลียร์ สำหรับเครื่องปฏิกรณ์ปรมาณูวิจัยของไทย ปปว-1/1

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ฝ่ายจัดการเครื่องปฏิกรณ์ฯ สถาบันเทคโนโลยีนิวเคลียร์แห่งชาติ

บทคัดย่อ

งานวิจัยนี้เป็นการปรับปรุงวิธีการจัดการเชื้อเพลิงนิวเคลียร์สำหรับเครื่อง ปปว-1/1 โดยใช้รหัสคอมพิวเตอร์ SRAC ที่มีรูปแบบการจัดเรียงเชื้อเพลิงแบบหกเหลี่ยม ทำให้สามารถคำนวณค่าภาคตัดขวาง (cross section) และค่าการกระจายตัวของกำลังต่อแท่งเชื้อเพลิงในรูปแบบ 3 มิติ (3-D calculation) เป็นต้น ซึ่งการคำนวณจะแบ่งเป็น 2 ขั้นตอน ได้แก่การคำนวณหาภาคตัดขวาง (cross sections) ที่มีโครงสร้างกลุ่มตั้งต้นแบบละเอียดมากและการคำนวณแกนจากค่าภาคตัดขวางที่ได้ในขั้นแรกในแบบ 4 กลุ่มพลังงาน ในงานวิจัยนี้รหัสคอมพิวเตอร์ SRAC ถูกนำมาใช้ในการจำลองแกนปฏิกรณ์หมายเลข 1 เพื่อเป็นการทดสอบความสามารถในการจำลอง ผลของค่าเอกซ์เซสรีแอคติวิตีของแกนปฏิกรณ์หมายเลข 1 สอดคล้องกับผลการทดลองดี ซึ่งจะนำรหัสคอมพิวเตอร์นี้มาใช้แทนการจัดการเชื้อเพลิงนิวเคลียร์ สำหรับเครื่อง ปปว -1/1 ในอนาคต

คำสำคัญ: รหัสคอมพิวเตอร์ SRAC การจัดการแกนเชื้อเพลิงนิวเคลียร์ เครื่องปฏิกรณ์ปรมาณูวิจัยของไทย

Utilization of SRAC Computer Code as a Nuclear Fuel Management Tool for Thai Research Reactor (TRR-1/M1)

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Abstract

An upgrade of fuel management methodology for TRR-1/M1 is presented in this paper. The advanced code system called “SRAC” is applicable to hexagonal arrangement reactor core thus it is applicable for TRR-1/M1 as a fuel management tool. The methodology is executed in 2 steps, namely hyperfine group structure cross section generation step and reactor core calculation step by applied SRAC’s capabilities to create group cross section in to four group structure. For verification purpose, the TRR-1/M1 core configuration number 1 is modeled by the new methodology in this paper. The result of the core excess reactivity of this core shows good agreement with the operation data from operation log-book. This methodology is expected to be employed as a fuel management tool for TRR-1/M1 in the near future.

Keywords: SRAC computer code, Fuel management, Thai Research Reactor, TRR-1/M1

1. Introduction

Fuel management is one of the essential aspects for the nuclear reactor operation. Basically, fuel management involves the planning of nuclear fuel use in the most economical manner without violating safety constraints. Since there are several types of neutron interactions within the nuclear reactor, the tool used for fuel management must be able to properly account for the underlying physics. For instance, the birth of neutron through fission, the loss of neutron through neutron capture and leakage and the change of neutron energy levels are, in nature, multi-dimensional and they must be modeled correctly by the fuel management tool. Routinely, computer programs are used as the fuel management tool of the nuclear reactor. These computer programs are embedded with models capable of (to some extents) describing the physical phenomena and also solving for the quantities of interest within the nuclear reactor. It is important to employ the fuel management tool because the behaviors and safety parameters of the reactor can be forecast.

For TRR-1/M1, which is presently the only Thailand's nuclear research reactor, the existing fuel management tool is a computer program called TRIGAP [1]. 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, a new methodology to replace the existing methodology is proposed. This paper describes the proposed methodology, which is actually derived from the methodology used for fuel management in typical commercial nuclear power reactors. In addition, this paper presents the results of the reactor core calculation by the proposed methodology in comparison with the available operation data.

2. Overview of TRR-1/M1

Thailand Institute of Nuclear Technology (TINT) is currently responsible for the nuclear research reactor called "TRR-1/M1" which is located in Bangkok. Historically, the reactor was built as an MTR research reactor type and it was named as Thailand Research Reactor 1 (TRR-1). The reactor had been operated since 1972 until 1975 when it was converted to the TRIGA research reactor type. 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. Description of the existing and proposed methodologies

This section provides a brief description of both existing and proposed methodologies used for the fuel management calculation of TRR-1/M1.

- **Existing methodology**

The existing methodology for fuel management of TRR-1/M1 utilizes the computer program called TRIGAP which was developed in Slovenia during the 80's. This methodology divides the reactor into several regions and each region represents a unique zone within the reactor core. The execution of the existing methodology has 2 steps, that is, group cross section generation step and reactor calculation step. The group cross section generation step is performed by a 2D reactor physics code to produce two-group cross sections representing different neutronic properties of each region. For TRR-1/M1, the group cross sections were earlier generated and stored in a library file. The generated two-group cross sections are subsequently employed in the reactor calculation step. The reactor calculation of TRIGAP is based on the neutron diffusion equation in cylindrical geometry and it is capable of performing burnup calculation. However, the cylindrical geometry modeling capability of TRIGAP is not most suitable for TRR-1/M1 since the core of TRR-1/M1 is arranged in hexagonal lattices. In order to model TRR-1/M1 by TRIGAP, the hexagonal rings of TRR-1/M1 have to be homogenized into cylindrical rings. Consequently, the fuel material gradient among fuel rods within each hexagonal ring is smeared. Figure 2 shows the typical modeling of TRR-1/M1 core by TRIGAP.

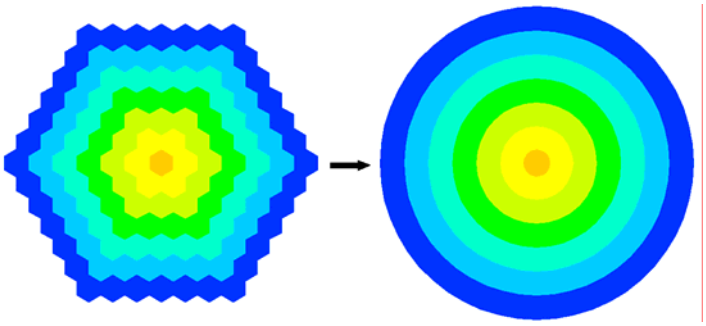

Figure 2: Fuel homogenization by the existing methodology

As a result, this methodology 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 instead. Moreover, TRIGAP is only utilizing two group modeling for energy treatment which is probably appropriate for light water reactors. However, the neutron energy distribution of TRIGA reactors is rather different than that of light water reactors. The two-group modeling of TRIGAP is possibly not flexible enough for neutron energy treatment of TRR-1/M1.

- **Proposed methodology**

The proposed methodology is derived from the standard methodology for commercial nuclear power reactors and it is applied for TRR-1/M1 as presented in this paper. Like the existing methodology, the proposed methodology divides the reactor core into several regions and is executed in 2 steps. However, both steps are performed by more advanced computer programs collectively packaged into one system called “SRAC” [2] which has been developed in Japan since 1978. The group cross section generation is performed by the PIJ module of SRAC system which utilizes 2D collision probability method. Typically, the group cross section generation is performed using infinite arrays of 2D lattice models corresponding to unique lattice regions of the reactor. In essence, 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 depending on its geometry. 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 3 to Figure 5.

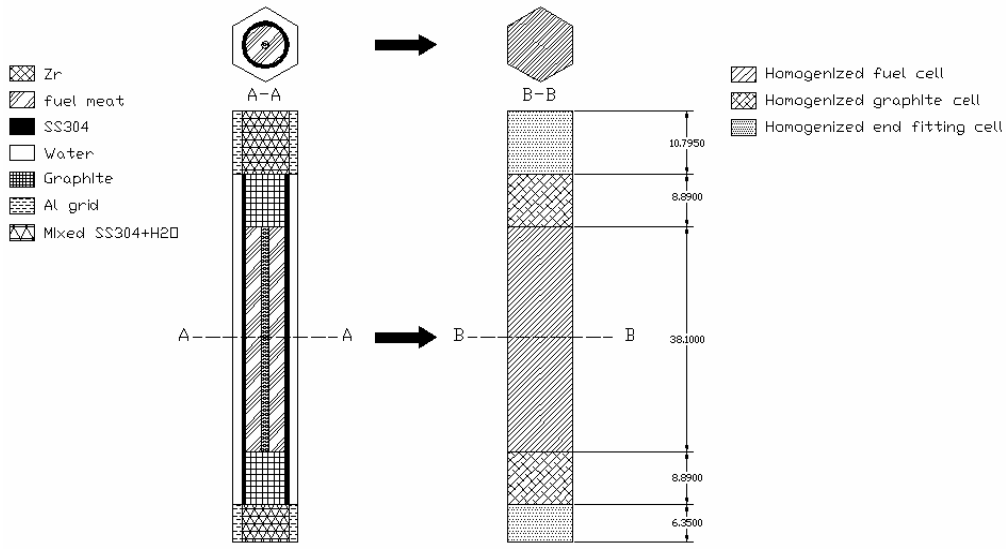

Figure 3: Axial homogenization of fuel elements

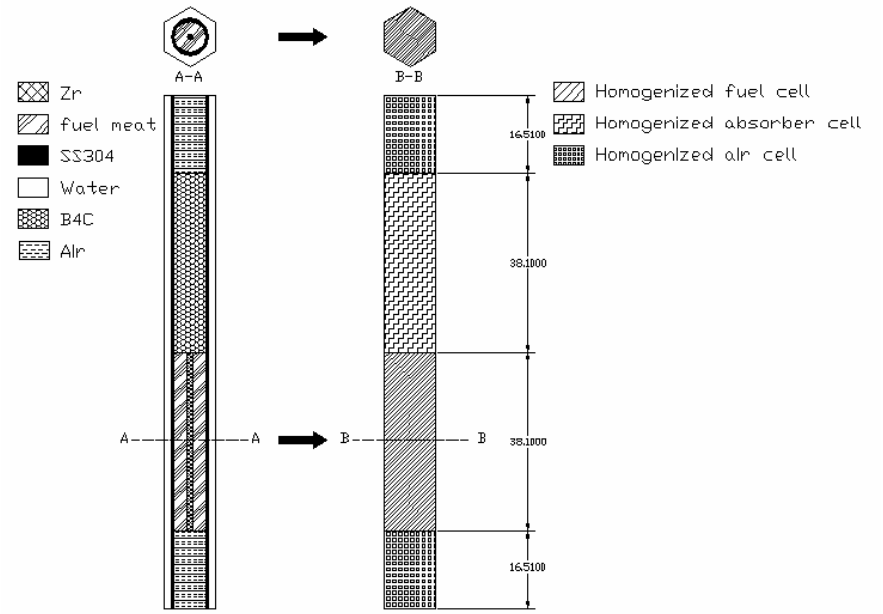

Figure 4: Axial homogenization of fuel follower control rods

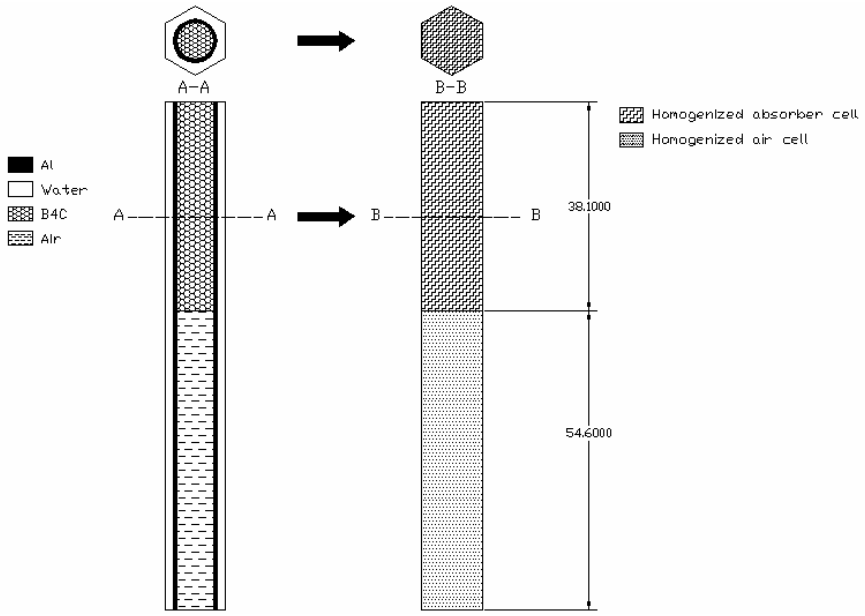

Figure 5: Axial homogenization of air follower control rods

Furthermore, to generate the group cross sections, the 2D lattices of TRR-1/M1 can be classified into two major models, i.e., fuel type lattice and non-fuel type lattice. The fuel type lattice as shown in Figure 6 is employed for the group cross section production of the lattice with fuel.

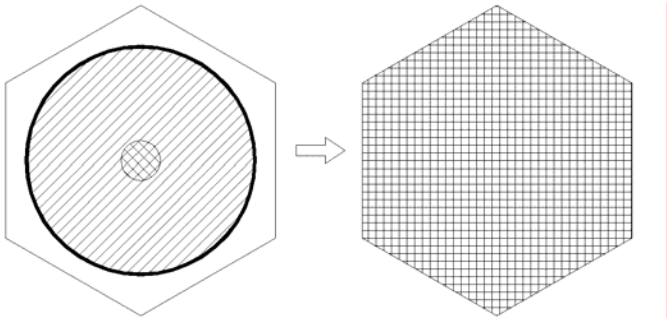

Figure 6: Homogenization of fuel type cells

In this model, the six surfaces of the hexagonal lattice are reflective boundaries. The fuel model consists of Zirconium rod in the center of the rod and it is surrounded by UZrH fuel. The fuel cladding which contains the fuel meat is SS304 while the fuel 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 4 group structure in the proposed methodology while they are created into two group structure in the existing methodology. Moreover, the group cross sections of the proposed methodology 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 7.

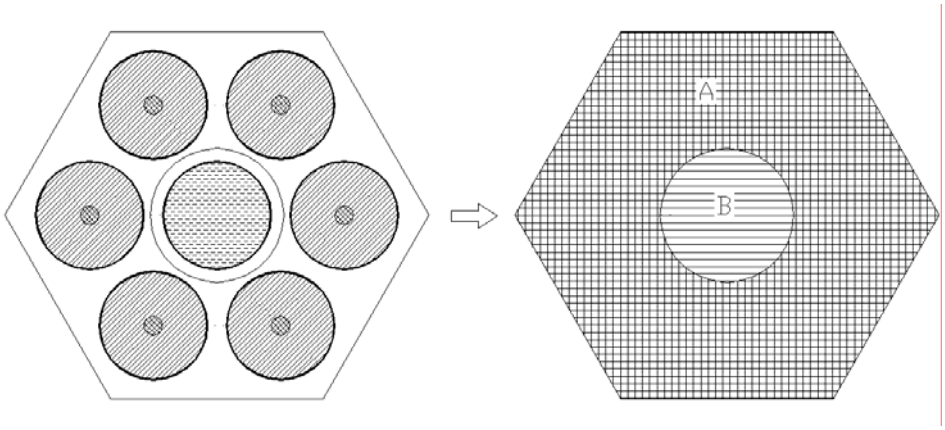

Figure 7: Homogenization of non-fuel type cells

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. This methodology creates the group cross sections of this model 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 4 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 proposed methodology uses the COREBN module of the SRAC system to perform the reactor calculation and the group cross sections generated in the previous step are used in this calculation. As mentioned earlier, the reactor core is modeled by building blocks of different lattices representing different regions in both radial and axial directions. The COREBN module has a capacity of modeling hexagonal lattice of TRR-1/M1 core. To verify the proposed methodology, the core configuration number 1 as shown in Figure 8 is modeled.

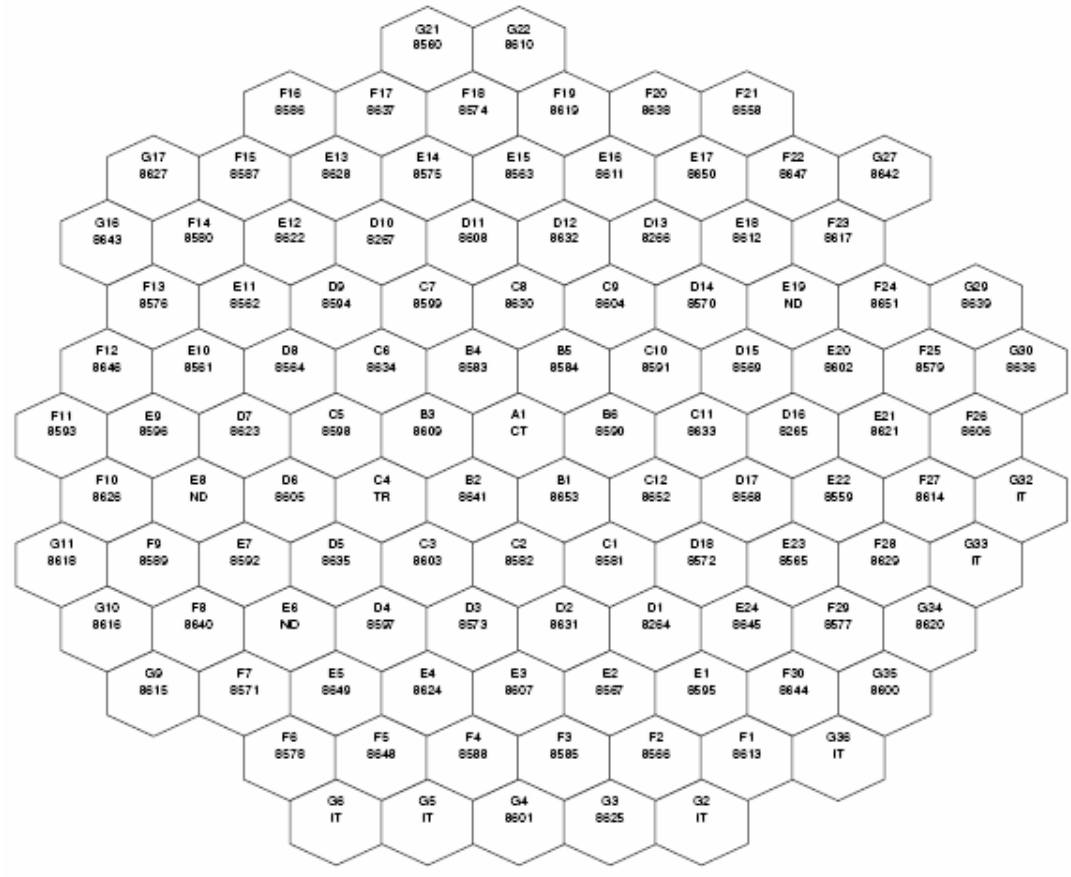

Figure 8: Core configuration number 1

The label in each lattice represents the fuel and non-fuel components of the reactor core. The label in number is the representation of a fuel rod lattice. It is noted that the core configuration number 1 uses only 8.5% wt fuel rod type (denoting by label beginning with “8”). Essentially, the largest portion of the

"IT", "ND" and "CT" lattices is water and the other materials in these lattices have small effects to neutronic properties of the lattice. Thus, these lattices are approximated by water lattices.

4. Results and Discussion

The group cross sections of different lattices were generated by the PIJ module of the SRAC system and the reactor core calculation of TRR-1/M1 core number 1 was performed afterwards. 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 number 1 with the comparison against operation data from the operation log book.

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

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$

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. In addition, the calculation by the SRAC system is able to provide normalized power distribution of the core as shown in Figure 9.

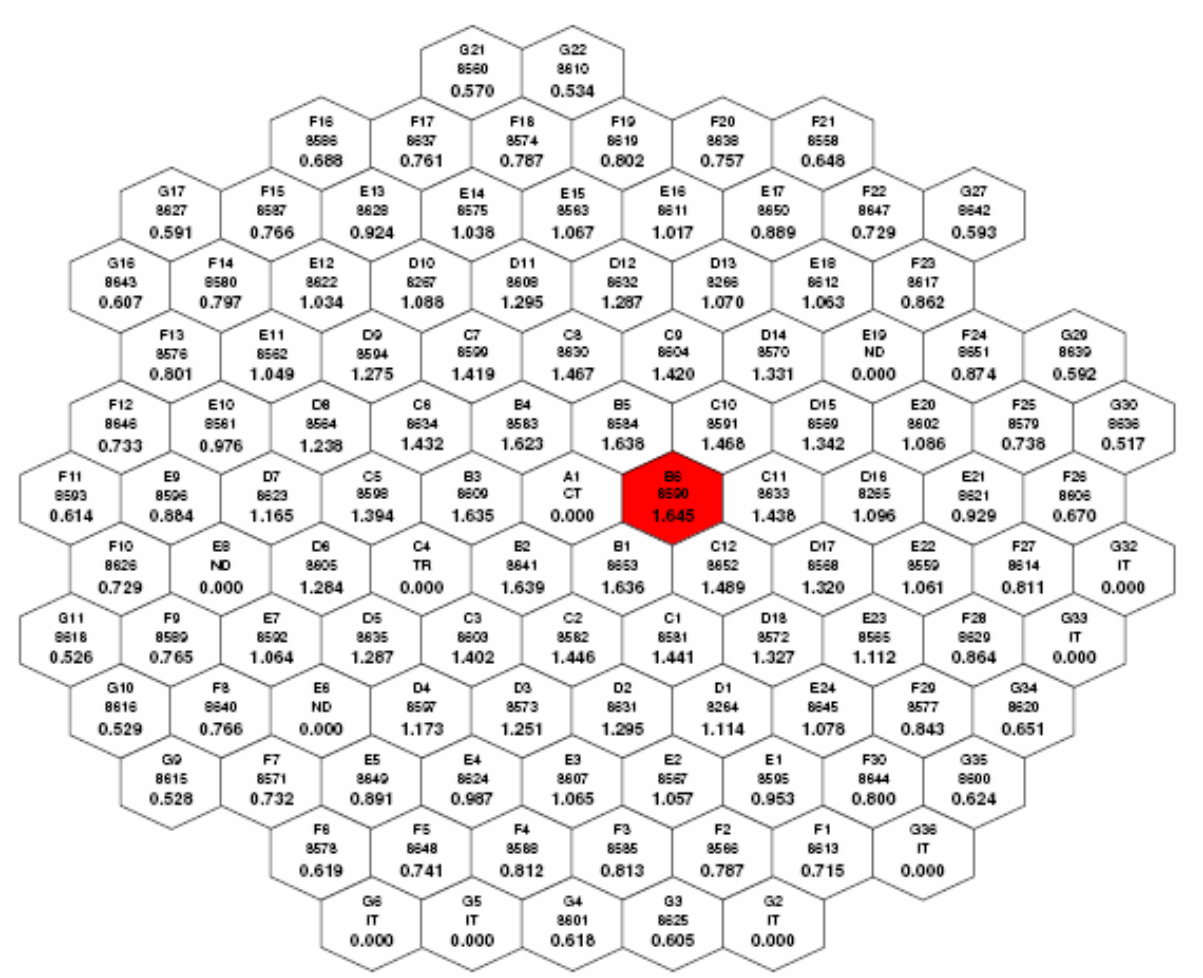

Figure 9: A plot representing normalized power distribution of TRR-1/M1 core #1

Also, the change of core excess reactivity as a function of power level is obtained from series of reactor core calculations at various equilibrium temperatures. Figure 10 presents the changes of core excess reactivity as a function of power level of TRR-1/M1 core #1.

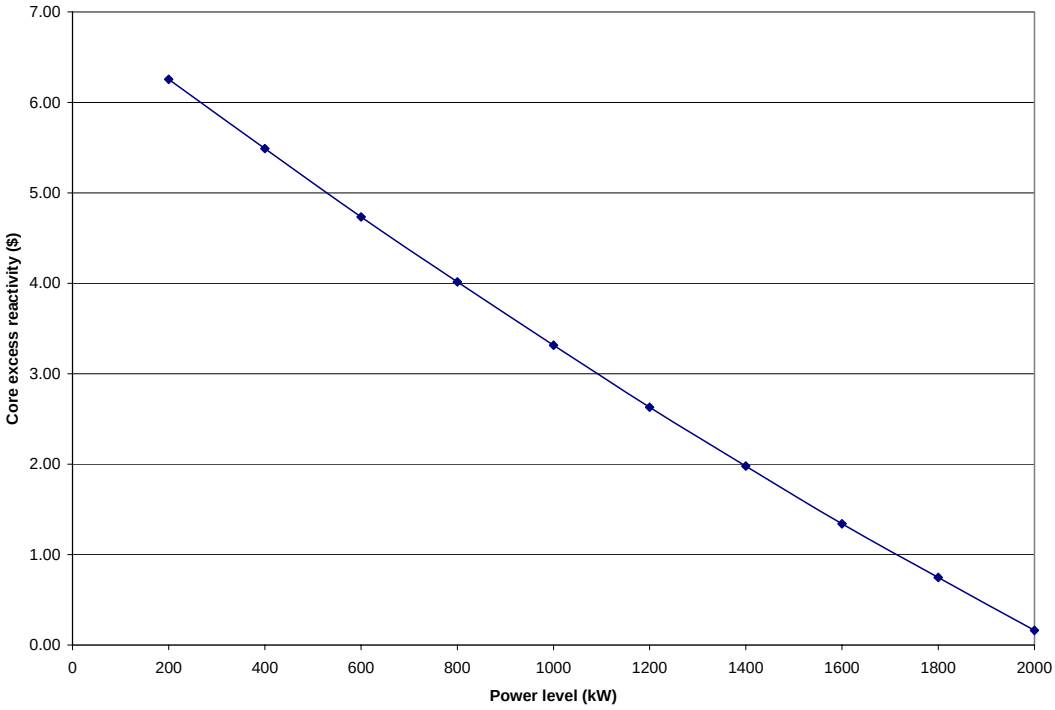

Figure 10: Core excess reactivity of TRR-1/M1 core #1 as a function of power level (kW)

From Figure 10, the core excess reactivity decreases quite linearly as a function of power level as excepted. This result confirms the negative temperature feedback of the TRR-1/M1.

5. Conclusion

A new methodology for fuel management calculation of TRR-1/M1 is proposed to replace the existing methodology. The proposed methodology is derived from the methodology typically used in the commercial nuclear power reactors. The more advanced SRAC code system is utilized in this proposed methodology. The modeling of the TRR-1/M1 core number 1 by the proposed methodology demonstrates good agreement with operation data from the log book in term of core excess reactivity. Moreover, the proposed methodology is able to predict the change of core excess reactivity as a function reactor power. Also, the pin-wise normalized power distribution of the reactor core can be obtained by the proposed methodology while it is not possible by the existing methodology. It is evident that the proposed methodology is functioning well and it can be absolutely adopted as a new fuel management tool for TRR-1/M1 in the near future.

6. References

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