The Monte Carlo Method: Versatility Unbounded In A Dynamic Computing World
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VALIDATION OF THE MCNP5 CORE MODEL OF THE PSU BREAZEALE RESEARCH REACTOR

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ABSTRACT

The Monte Carlo method is currently one of the new methodologies under development to be employed as a fuel management tool for the Pennsylvania State University Breazeale research reactor (PSBR). This paper discusses the effort to analyze the current core loading of the PSBR by the MCNP5 code. Since the initially loaded fresh fuel has been depleted due to the long operation history of the PSBR, the fuel material composition of each fuel element is needed to be determined in order to model the current PSBR core. The HELIOS code was used in this research to obtain the fuel material composition change of each fuel element. In addition, the decay calculation was introduced to the PSBR core model by the ORIGEN2.2 code to take into account of the effect of the Xe^{135} decay after reactor shutdown. The X^{135} decay was found to have a significant impact on the core excess reactivity. Subsequently, the MCNP5 core model was compared to the current core analysis methodology and it was also validated against measurements. Good agreements were observed from the code-to-code comparison and the measurement comparison.

Key Words: PSBR, Monte Carlo core calculation, Fuel depletion, Xenon decay, MCNP validation

1 INTRODUCTION

The Monte Carlo calculation has currently been employed as one of the computational methods in the fuel management area for several research reactors worldwide. This method is very attractive because of its capability to handle the very complex and detailed geometries in three-dimensions (3D). The continuous energy treatment of the Monte Carlo method has enabled it to solve the highly energy dependent problems without using the multi-group energy approximation. In the recent years, the Monte Carlo method has been introduced to model the Pennsylvania State University Breazeale Reactor (PSBR) [1]. This research has been initiated from the effort to apply new methodologies to replace the old core analysis method. In essence, the Monte Carlo method is one of methodologies under development emphasizing on performing the whole-core calculation as a reference. During almost 40 years of the PSBR operation, the PSBR core has gone through many core loadings and the current core loading is the core-loading 51 (as of time when the analysis was conducted). The long operation of the PSBR has depleted some amount of uranium fuel in the PSBR core. In addition, a number of fission products have been produced in the fuel elements as a result of the fuel depletion. The change of material composition within each fuel element has caused the changes in neutronic properties of the fuel elements. Therefore, the PSBR core needed to be analyzed for the changes of the fuel material composition. The analysis of the fuel elements is important in order to accurately predict the power distribution within the PSBR core. In this paper, the MCNP5 code [2] is used to model the full PSBR core. The excess reactivity of the MCNP5 core model is analyzed as a function of

time after reactor shutdown to observe the effect of X^{135} in the core. Finally, the MCNP5 core model is validated against experimental measurements and against the current core analysis methodology.

2 METHODOLOGY

2.1 Descriptions of the PSBR

The PSBR is a TRIGA Mark III research reactor manufactured by General Atomic. The reactor has been operated since 1965 when the core was upgraded from MTR type fuel. The PSBR is a light water cooled, pool type reactor designed for 1 MW(t) steady-state power operation (up to 2000 MW(t) when pulsing) with natural circulation cooling. The uniform lattice of the PSBR core is formed in hexagonal shape and the center of the core is the location of the central thimble which is surrounded by hexagonal rings. The rings running from the center outward are designated B, C, D, E, F and so on. The standard PSBR fuel element contains homogeneous 20% enriched U-ZrH_x mixture which can be grouped into two types according to uranium weight percentage: the 8.5% wt. fuel element and the 12% wt. fuel element. The PSBR core was initially loaded with only fresh 8.5% wt. fuel elements in 1965 and some fresh 12% wt. fuel elements were added to the core in 1972. The current PSBR core comprises of 94 fuel elements (30 of them are 12% wt. fuel elements and 64 of them are 8.5% wt. fuel elements). The PSBR uses four control rods which can be categorized in two types. These are fuel follower control rods (i.e. SA, SH and RR) and air follower control rod (i.e. TR). The current PSBR core configuration is shown in Figure 1.

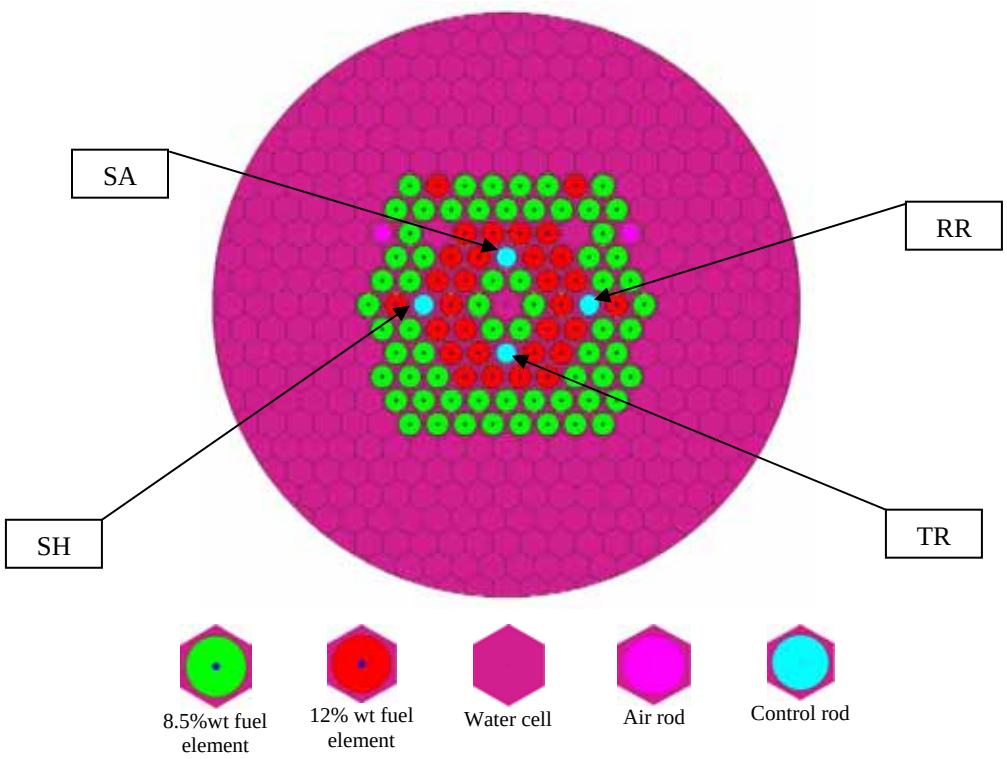

Figure 1. Current PSBR core configuration (core loading 51)

The standard PSBR fuel element consists of three axial sections. The top and the bottom sections of the fuel element are graphite bars serving as neutron reflectors. The middle section of the fuel element contains fuel-moderator mixture of U-ZrH matrix which surrounds a central Zr rod. The cladding of the fuel element is made of SS304 and the water outside the fuel element acts as coolant. The fuel follower control rod has similar geometries and dimensions to the standard fuel element except that the fuel follower control rod consists of four axial sections. These sections are the top graphite bar, the neutron absorber (B_4C), the fuel matrix and the bottom graphite bar. On the other hand, the air follower control rod consists of two axial sections: the neutron absorber at the top and the air rod at the bottom.

2.2 Depletion of fuel elements

The first task of modeling the current PSBR core is to determine the current material composition of each fuel element. Basically, the change of fuel material composition depends on the fuel burnup. The fuel material has several kinds of interactions with neutrons during operation which results in the depletion of the fuel material. The fission products which are the by-products from the fission reaction are accumulated within the fuel element during operation. The fission products are normally non-fissile materials and some of them are strong neutron absorbers. In general, the fuel element is less reactive with increasing burnup because of the accumulation of the fission products in the fuel element. In this work, the determination of the fuel material composition as a function of burnup was performed by the HELIOS1.7 code [3]. The current burnup of each fuel element was obtained by the previous core analyses [4], [5]. The two-dimensional (2D) model of the TRIGA cell was constructed by the HELIOS code as shown in Figure 2. The model comprises of the central zirconium rod, fuel-moderator mixture, cladding and the water outside fuel element in a hexagonal lattice. The six surfaces of the hexagonal lattice are reflective boundaries.

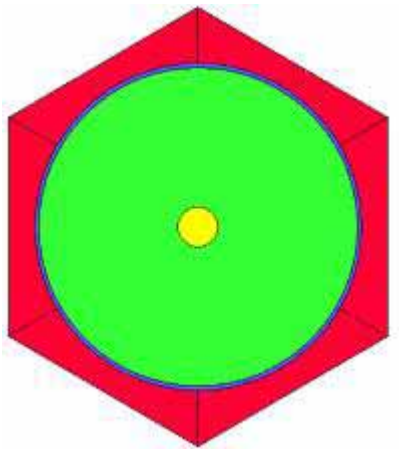

Figure 2: HELIOS model of the PSBR TRIGA cell

The burnup calculation of the TRIGA cell was performed by the HELIOS code utilizing 45 neutron cross section groups. The atomic number densities of 42 isotopes (including actinides and fission products) were obtained from the HELIOS burnup calculation. The burnup calculation was performed at the fuel and central Zr rod temperatures of 400ºC which represent

the average fuel temperature during reactor operation and the average temperature of 60ºC for the other materials (i.e. cladding and water). In addition, the burnup calculation was performed at the average fuel power of 51 W/g. of uranium. The material composition of each fuel element was determined individually at different burnup.

2.3 Full core model with xenon decay calculations

When the atomic number densities of all fuel elements in the core were obtained by the burnup calculation, the MCNP5 full core model of the PSBR was constructed by the interface code, TRIGSIM [6]. The full core model of the current PSBR core consists of 94 burned fuel elements as shown in Figure 1. The MCNP calculation was performed at room temperature and the standard continuous cross sections from the endf66 library (the newest cross section library) of the MCNP5 code were used for most isotopes. However, since the continuous cross sections of some few isotopes are not available from the endf66 library, less recent continuous cross section libraries were utilized instead. The TRIGSIM code automatically added a fission rate tally card in the MCNP input to determine the normalized power of each fuel element. The excess reactivity of the core was determined from the K_{eff} of the all-rods-out model. The reactivity worth is calculated by

$$\rho = \frac{K_{eff} - 1}{\beta K_{eff}} \tag{1}$$

where: ρ = reactivity worth in dollars ($)

β = total delayed neutron fraction (0.007 for PSBR)

It is known that the core excess reactivity is affected by the amount of Xe^{135} in the fuel elements. In fact, the amount of Xe^{135} after reactor shutdown depends on two phenomena. First, the Xe^{135} decays into Cs^{135} and, secondly, the Xe^{135} is produced by the I^{135} decay. In this analysis, the core excess reactivity is analyzed as a function of time after reactor shutdown by determining the amount of Xe^{135} and other fission products in each fuel element at different decay times using the ORIGEN2.2 code [7]. The fuel material composition at the different decay times was later used in the MCNP5 core calculation to determine the core excess reactivity as a function of time after reactor shutdown.

2.4 Comparison to current core analysis methodology

The current PSBR core analysis utilizes the same methodology used in a power reactor. In this methodology, a number of 2D transport calculations are performed to obtain cross section library as a function of several parameters including burnup. Subsequently, a few-group nodal diffusion calculation is employed to perform the core calculation. For the PSBR, the HELIOS code is used to generate the diffusion cross section library. These cross sections are used by the two-group nodal diffusion code, ADMARCH [8], to perform core calculations. The validation of the MCNP5 core model and the ADMARCH core model is performed by comparing the normalized power of each fuel element.

2.5 Validation of full core model against measurement

The experiment to measure the reactivity worth of four fuel elements was performed to verify the MCNP5 full core model. The fuel elements number 34, 203, 122 and 126 were selected for the experiment. To measure the reactivity worth of a fuel element in the experiment, the critical rod worth of the core with all fuel elements was determined. Subsequently, the measured fuel element was removed from the core and the critical rod worth of the core without the measured fuel element was determined. The difference between the critical rod worth of the core with the fuel element and the core without the fuel element is the reactivity worth of the fuel element. On the other hand, to compute the reactivity worth of a fuel element by the MCNP5 core model, two MCNP5 core calculations were performed to simulate the experiment. First, the full core model with the control rods at their critical rod positions, which were derived from the experiment, was performed to determine the reactivity worth of the core. Subsequently, the core model without the measured fuel element was simulated to determine the reactivity worth of core without the fuel element. The difference between the reactivity worth of both models is the reactivity of the fuel element which is used for the comparison with the measurement.

3 RESULTS AND DISCUSSION

3.1 Full core model with xenon decay calculations

The burnup calculation of each fuel element was performed by the HELIOS code to obtain the atomic number densities. These atomic number densities were then used in the decay calculation to determine the atomic number densities after decay. The ORIGEN2.2 code was utilized to perform the decay calculation. In the first step, the decay calculation of only one fuel element (fuel element number 128) was performed to observe the amount of Xe^{135} as a function of time after reactor shutdown. Moreover, the decay calculations were performed with the fuel element depleted at different average burnup power levels (25%-200% nominal power) to observe the behavior of the Xe^{135} decay as a function of burnup power levels. The amount of Xe^{135} of the fuel element number 128 as a function of time after reactor shutdown depleted at different burnup power levels is shown in Figure 3.

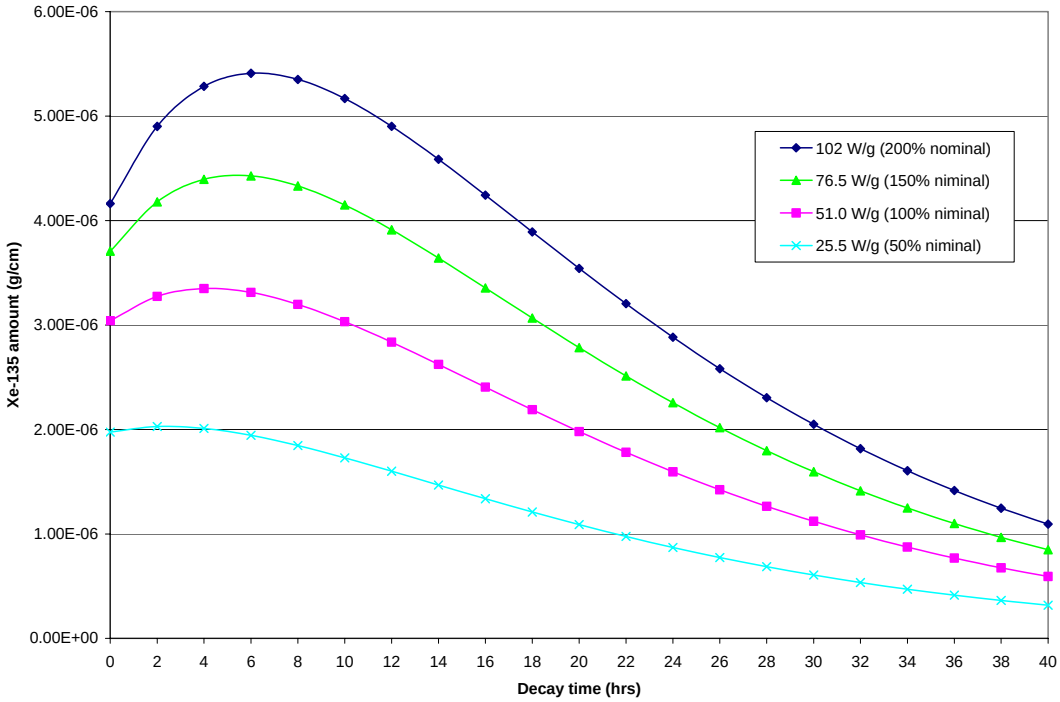

Figure 3: The amount of Xe^{135} of the fuel element number 128 as a function of time after reactor shutdown depleted at different burnup power levels

Figure 3 shows that the amount of Xe^{135} varies as a function of time. The time which the amount of Xe^{135} peaks depends on the burnup power level. For the 100% nominal power level which was used in the burnup calculation of all fuel elements, the time which the Xe^{135} peaks is approximately 5 hours. It can be further observed that the amount of Xe^{135} at the time right after the reactor shutdown is smaller than that at the time within approximately first 10 hours after reactor shutdown. Therefore, it is expected that the core excess reactivity would be smaller within first 10 hours after reactor shutdown and it would start increasing afterwards. The decay calculation of the rest of the fuel elements was later performed to include the Xe^{135} decay effect to the full core model. The atomic number densities of each fuel element at different times after reactor shutdown were input into the full PSBR core model. The PSBR core excess reactivity of the MCNP5 full core model as a function of different decay time after reactor shutdown is shown in Table 1.

Table 1: The core excess reactivity of the PSBR CL-51 model as a function of decay time after reactor shutdown

Decay time (hrs)	Reactivity worth ($)
0	$3.49 \pm 0.02(\sigma)$
12	$3.66 \pm 0.02(\sigma)$
24	$4.64 \pm 0.02(\sigma)$
40	$5.36 \pm 0.02(\sigma)$
Experiment	3.89

As it can be seen from Table 1, the decay time has a significant effect on the core excess reactivity. As the amount of Xe^{135} decreases, the core excess reactivity increases as expected. The experimental value shown in Table 1 was measured in the morning when the reactor was shutdown overnight. Therefore some amount of Xe^{135} in the PSBR core had decayed during the reactor shutdown. For the MCNP5 model, the 12 hour decay time is the best estimate of the Xe^{135} decay time after reactor shutdown since its core excess reactivity is closest to the experiment value.

3.2 Comparison to current core analysis

In this task, the comparison between the MCNP5 core model and the ADMARCH core model was performed. In the first step, the cross section generation methodology for the ADMARCH code was compared with the MCNP5 code. The cross sections used by the ADMARCH code were pre-generated by the HELIOS code which was also used to determine the atomic number densities for the MCNP5 core model. Therefore, a consistent comparison between the cross section methodology and the Monte Carlo methodology can be performed since the atomic number densities were essentially the same. In doing so, a two-dimensional model of the TRIGA cell (as in Figure 2) was constructed and computed by the MCNP5 code. It is noted that the HELIOS code utilized multi-group cross section derived from ENDF/B-V while the MCNP5 utilized the continuous energy cross section derived from ENDF/B-VI. The K_{inf} as a function of burnup of both codes are shown in Figure 4.

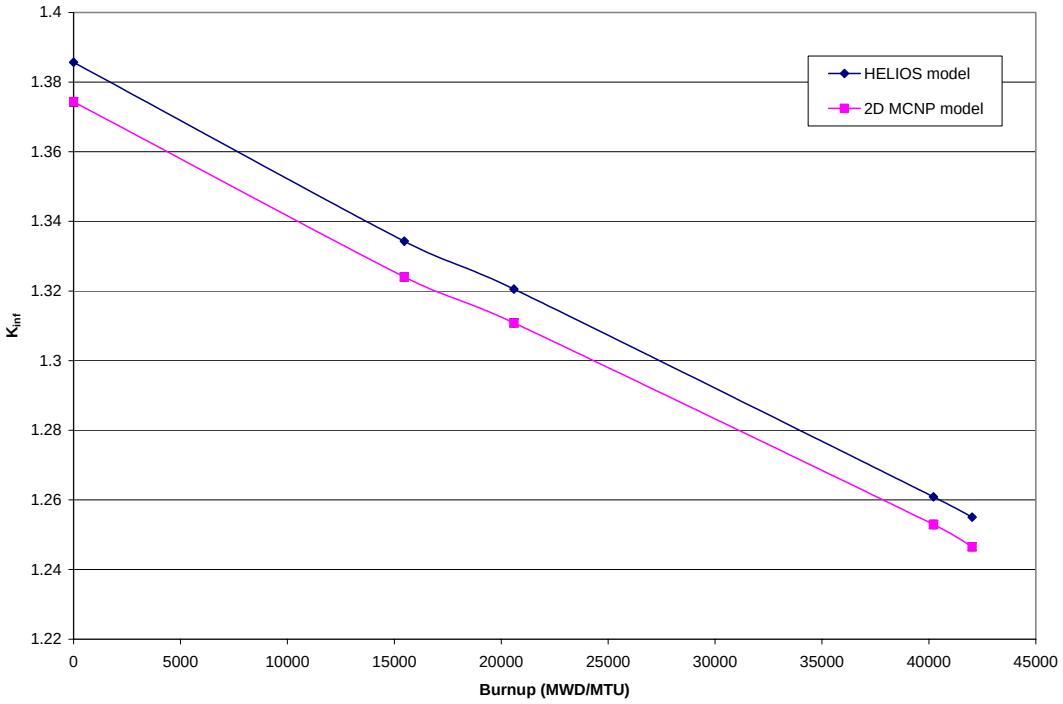

Figure 4: K_{inf} comparison as a function burnup between MCNP5 model and HELIOS model

As it can be seen from Figure 4, the HELIOS code systematically predicts higher K_{inf} than the MCNP5. Hence, it is expected that the cross sections generated by the HELIOS code would produce higher core excess reactivity when used in the core calculation.

The excess reactivity of the MCNP5 core model and that of the ADMARCH core model are compared as shown in Table 2.

Table 2: Excess reactivity comparison of the current PSBR core

Core model	Core excess reactivity ($)
ADMARCH	3.80
MCNP5	$3.66 \pm 0.02(\sigma)$
EXPERIMENT	3.89

As it can be seen from Table 2, the ADMARCH model predicts higher core excess reactivity than the MCNP5 model as expected. Furthermore, the ADMARCH core model predicts closer core excess reactivity to experiment than the MCNP5 core model. However, the ADMARCH core model does not take into account the effect of the Xe^{135} decay while the MCNP5 model utilizes 12 hours decay time. As described earlier, the experimental value was measured in the morning when the reactor was shutdown overnight and therefore some amount of Xe^{135} in the PSBR core had decayed during the reactor shutdown. In addition to the core excess reactivity, the normalized power of the ADMARCH all-rods-out (ARO) core model and the normalized power of the MCNP5 all-rods-out (ARO) core model were compared against each other. Figure 5 shows the normalized power of each fuel element comparison between the two models. The comparison shows that the normalized power distributions of both codes agree quite well. The difference of the normalized powers of the fuel elements around the interior of the core is typically less than 3% while the normalized powers of the fuel elements around the peripheries differ from each other approximately by 3%-8%.

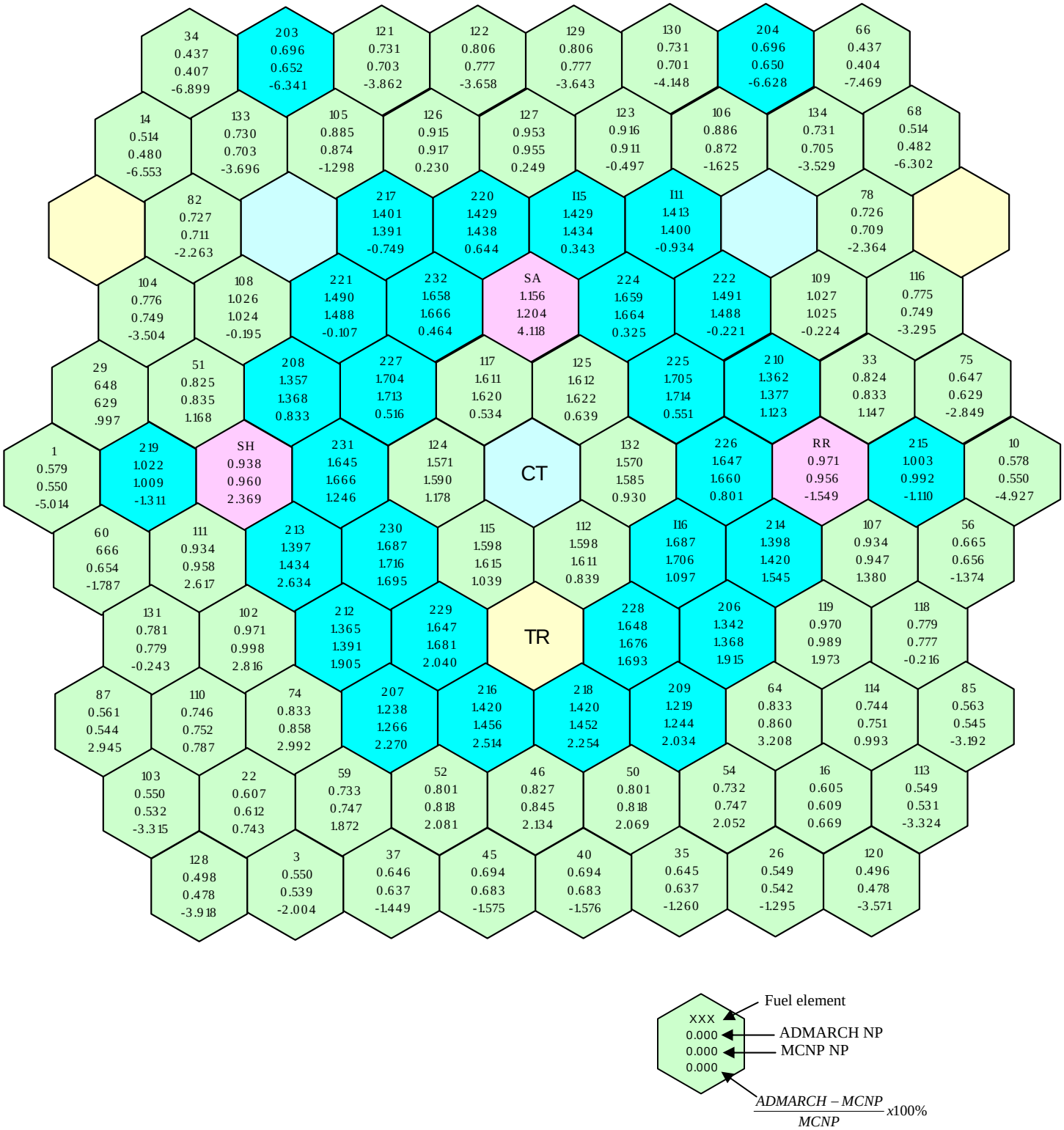

Figure 4: PSBR CL-51 ARO model

3.3 Validation against measurement

The reactivity of the selected fuel elements was calculated and compared to the measurement. The comparison between the measured data and the calculated values is shown in Table 3.

Table 3: Reactivity of the selected fuel elements

Fuel element	Reactivity (cents)	
	MCNP5	Measurement
34	$5 \pm 3(\sigma)$	11.4
126	$44 \pm 3(\sigma)$	45.0
122	$33 \pm 3(\sigma)$	30.3
203	$22 \pm 3(\sigma)$	22.5

The comparison shows good agreements between the measured data and the calculated reactivity. For all fuel elements except fuel element number 34, the reactivity calculated by the MCNP code agrees with the measured value within one standard deviation. However, for fuel element number 34, the reactivity calculated by the MCNP code agrees with the measurement within three standard deviations.

4 CONCLUSIONS

In this research, the Monte Carlo method was utilized to model the current PSBR core. The burnup calculation to determine the amount of current fuel composition was performed by the HELIOS code. The Xe^{135} decay calculation was introduced in the MCNP5 core model to take into account the effect of Xe^{135} buildup and decay after reactor shutdown. It was found that the Xe^{135} decay has a significant effect on the core excess reactivity. The MCNP5 core model with 12 hour of X^{135} decay time is chosen as the best estimate of the current PSBR core. Moreover, the MCNP5 core model was compared to the current core analysis model. The comparison shows good agreements between the normalized powers calculated by both methods. The validation of the MCNP5 core model and the measurement was performed by comparing the reactivity of some selected fuel elements. Good agreements were also observed between the MCNP5 core model and the measurement.

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