Validation of the Monte Carlo Code MVP on the First Criticality of Indonesian Multipurpose Reactor

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Abstract. The validation research works in BATAN are focused using Monte Carlo codes with recent nuclear data on the experimental results. In this paper, the validation results of Monte Carlo code MVP on the first criticality experimental of Indonesia Multipurpose Reactor (RSG GAS reactor) are presented. The MVP code is a continuous energy Monte Carlo code developed by Japan Atomic Energy Research Institute (JAERI). The objective this paper is to show the accuracy of the code using recent nuclear data of JEF-3.0, JENDL-3.3 and ENDF/B-VI.8. The final goal of this research is to use the code as an in-core fuel management code since the code has a module of burn-up calculation (MVP-BURN). The MVP calculations with the three libraries produced k_{eff} values with excellent agreement to experiment data since the maximum differences are less than 0.5%. For the total control rod worth, the maximum difference is 3.6%. Systematically, ENDF/B-VI.8 library gave a maximum difference compared with other libraries. Therefore, the MVP code with recent libraries can be applied for the MTR type reactor with bulky Beryllium reflector.

1. Introduction

The 30 MWth Indonesian Multipurpose reactor, Reaktor Serba Guna G.A. Siwabessy (RSG GAS) [1], was commissioned at Multipurpose Reactor Center (PRSG-BATAN) at Serpong and the first criticality of the reactor was achieved in July 29, 1987. The typical working core is achieved through some transition cores with smaller core and lower power. There are five transition cores before a full core configuration can be achieved in the sixth core.

In the equilibrium core, there are 40 standard fuel elements (FEs) and 8 control fuel elements (CEs). With the existing nominal core cycle of 20 days, the core produces energy of 600 MWD per cycle. The equilibrium core is divided into eight burn-up classes with an average burn-up step of approximately 7 % (loss of ^{235}U). The average burn-up at beginning of cycle (BOC) and the maximum discharged burn-up are about 23.8 and 56%, respectively.

The intensive research activities on the field of in-core fuel management are carried out for supporting the safe and efficient operation of the RSG GAS reactor. Those activities also support an efficient and effective core conversion program in RSG GAS reactor for using high uranium density fuel. Therefore, an in-core fuel management code of Batan-FUEL has been developed using the two-dimensional multigroup neutron diffusion method [2]. A fuel burn-up model in the reduced reactor geometry was implemented in the code to get an efficient and fast computational time [3]. The code has been used in the design of the equilibrium silicide RSG GAS core with higher uranium density [3,4]. However, a Monte Carlo method code is needed for an accurate evaluation of reactor core characteristic.

The continuous energy Monte Carlo method is well known to yield an accurate calculation because of its low number of approximations. The method has recently been enhanced with a burn-up calculation capability so that it can be used as an accurate burn-up calculation method. Several burn-up codes based on a continuous energy Monte Carlo method have been developed, e.g., MCNP-BURN [5] and MVP-BURN [6]. Since the recent computer technology declines the computational cost, the Monte Carlo method will be a main tool in the in-core fuel management analysis.

In this report, Monte Carlo calculation results for the first core of RSG GAS will be presented and compared with the experiment data. The first core was chosen because of, first, the completeness of the experiment data, and secondly, the fuel elements, absorber and beryllium reflectors were still fresh so that uncertainties increased from the fuel burn-up and absorber depletion calculations, and from the calculation of lithium poisoning in the beryllium reflectors can be eliminated.

The MVP code is a continuous energy Monte Carlo code developed by Japan Atomic Energy Research Institute (JAERI) [7]. The objective this paper is to show the accuracy of the Monte Carlo code using recent nuclear data of JEF-3.0, JENDL-3.3 [8] and ENDF/B-VI.8 [9]. This research is an initial step before using the MVP-BURN code in analyzing the in-core management of RSG-GAS reactor.

2. Core Descriptions

2.1. General description of RSG GAS reactor

The RSG-GAS reactor is a multipurpose open-pool type reactor. The reactor has nominal power of 30 MWth using 40 standard fuel elements (FE, each consisting of 21 fuel plates), 8 control fuel elements (CE, each consisting of 15 fuel plates) and 8 absorbers of AgInCd on the 10 × 10 core grid positions as shown in Fig.1. The beryllium and light water are used as the reflector and the moderator and coolant, respectively. At the nominal power, the reactor produces thermal neutron flux in order of 10^{14} n/cm^{2}s. Originally, the core used the oxide fuel (U_3O_8-Al), but, presently, the core uses the silicide fuel (U_3Si_2-Al). Reactor main data can be seen in Refs.[1,4].

Fuel elements used for RSG GAS are of the material testing reactor (MTR), i.e. plate-type fuel elements, and one fuel element consists of 21 fuel plates assembled by two side plates. The cross-sectional view of RSG GAS standard fuel element is shown in Fig. 1. One fuel plate consists of 19.75 % enriched uranium silicide meat (with uranium density of 2.96 gU/cc) embedded in aluminum matrix, and aluminum cladding. The active length of the fuel element (or the meat height) is 60 cm. The nominal 235U loading per standard fuel element is 250 g. Control fuel elements, shown in Fig. 2, with identical outer dimension consist of 15 fuel plates, that is, three fuel plates at both outer sides of the fuel elements are removed to provide space for absorber blades and therefore the nominal 235U loading for a control fuel element reduces to 178.57 g. At both sides of the control fuel element, two absorber guide plates (aluminum) are installed. A fork type control rod (0.38 cm thick Ag-In-Cd absorber meat with SS-321 cladding) can be inserted into or withdrawn out of the control fuel element.

2.2. First core of RSG GAS reactor

The full configuration of the first core of RSG GAS consists of 12 fresh standard and 6 control fuel elements while the configuration for the first criticality needs only fresh 9 standard and 6 control fuel elements. The first criticality and full core configuration of the first core are shown in Fig. 3 and Fig. 4, respectively.

As seen in Fig. 4, the core grid positions of B-6, C-7, D-6 and F-4 were used for irradiation positions. There are 15 loading steps for achieving the full core configuration. The maximum power of the first core (full configuration) was 10.17 MWth. Table 1 shows the core composition for the two core configurations.

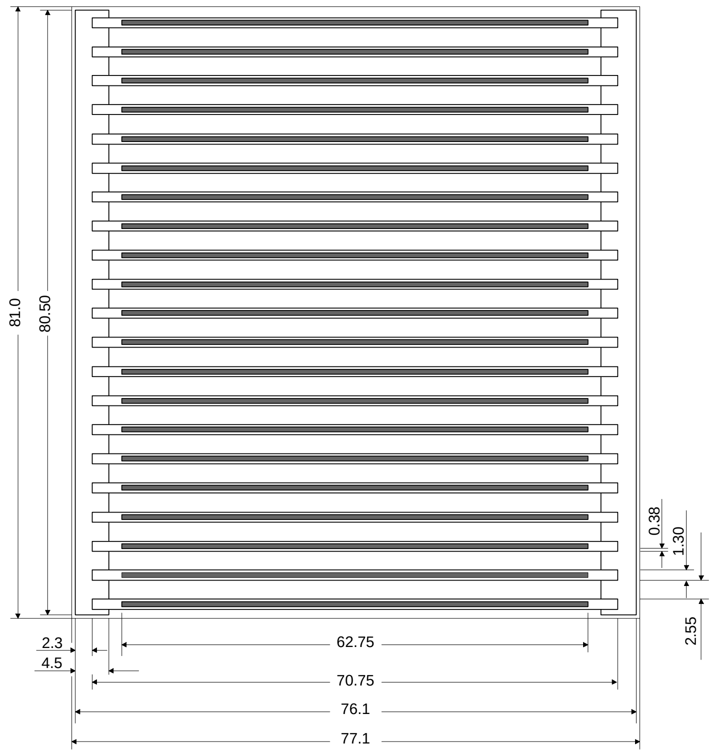


FIG. 1. RSG GAS standard fuel element (unit mm)

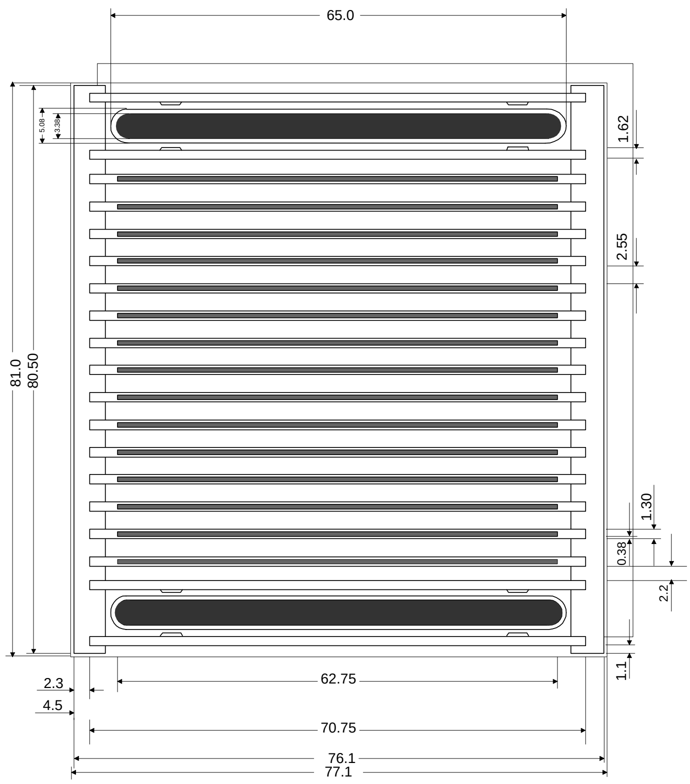
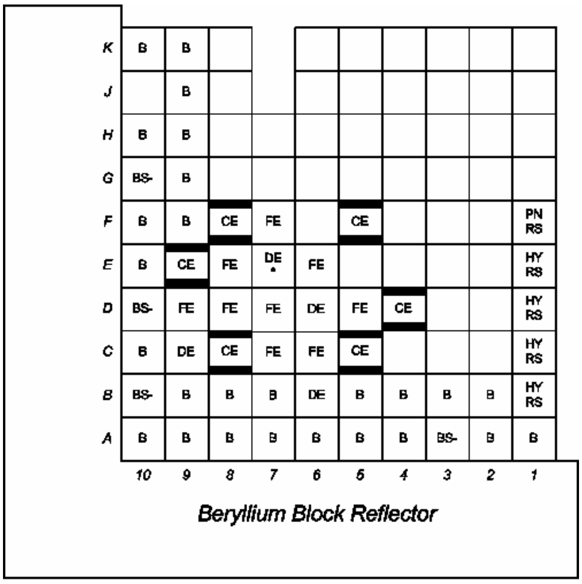
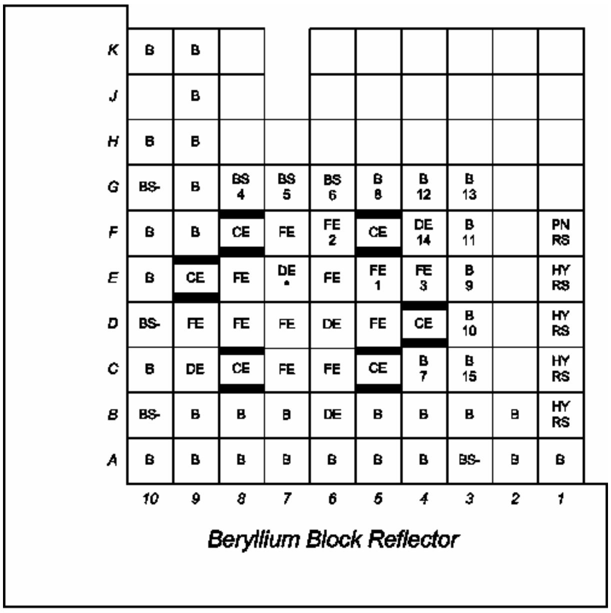


FIG. 2. RSG GAS control fuel element (unit mm)



Note : FE = Fuel Element, CE = Control Element, B = Be Reflector Element, BS = Be Reflector Element with plug, BS- = Be Reflector Element without plug, DE = Dummy Element, PNRS = Pneumatic Rabbit System, HYRS = Hydraulic Rabbit System, * = Neutron source (Cf-252)

FIG. 3. First criticality configuration of the RSG GAS first core



Note : FE = Fuel Element, CE = Control Element, B = Be Reflector Element, BS = Be Reflector Element with plug, BS- = Be Reflector Element without plug, DE = Dummy Element, PNRS = Pneumatic Rabbit System, HYRS = Hydraulic Rabbit System, * = Neutron source (Cf-252)

FIG. 4. Full configuration of the RSG GAS first core (the numbers are loading steps)

Table 1. The number of core materials for first and full core configurations

Core Material	Number of materials	
	First criticality core configuration	Full core configuration
Fuel Element	9	12
Control Fuel Element	6	6
Beryllium reflector element	31	43
Dummy element	4	5

3. Experiments and calculations

3.1. *Criticality experiments*

The evaluations are carried out for two criticality experiments. First, estimation of the number of standard fuel elements (FEs) required while there are six control fuel elements (CEs) in the core. Based on the Fig. 3, the six CEs were set at C-5, C-8, D-4, E-9, F-5 and F-8. The water coolant temperature in the primary circuit was kept low (25 °C), all neutron beam tubes were flooded (filled) by water, all fuel elements were free of xenon and samarium by operating the reactor at very low power. First criticality was achieved when the 9^{th} fuel elements was inserted (at core grid position of F-7). In this condition, the regulating rod was inserted 12.5 cm while the other five shim rods were fully withdrawn.

Second, the excess reactivity of the full core configuration is estimated. The full core was set up by loading of 3 FEs, 11 beryllium reflector elements and 1 dummy element. Based on Fig. 4, there are 15 loading steps to achieve sufficient excess reactivity for one core cycle. At each loading step, reactivity gains were measured by calibrating the difference of the regulating rod position with a reactivity meter (compensation method with the other 5 shim rods in bank configuration). The accumulated excess reactivity measured by this method should be considered as an uncorrected excess reactivity value. After correction on the measured excess reactivity was done the corrected excess reactivity of the first core was found to be 8.46 %$\Delta k/k$ (11.054 \$, β=0.00765). This excess reactivity was converted to effective multiplication factor of 1.09242.

In this paper, the total control rod worth is also evaluated. Based on the experimental results, the total control rod worth is 17.80 %$\Delta k/k$. The value was obtained by summation of single control rod worth with shim rod compensation method.

3.2. *Monte Carlo calculations*

The active part (7.71 cm × 8.1cm × 60 cm) of both standard and control fuel elements were modeled as their exact geometry and dimensions while the top and end-fitting of the elements were modeled in an approximate manner since their geometry are very complicated, that is, the structure materials were homogenized with water by volume weighting. Exact modeling approach was also taken for the active parts of the beryllium reflector elements, beryllium block elements and irradiation positions. Considering their complicated geometry, the core grid and bottom support were also treated approximately as for the top or end-fitting of fuel elements. This approximation did not deteriorate the accuracy of the Monte Carlo calculation results since it was applied in the non active parts of the core. The movable control rods (absorber blades) were modeled as their exact geometry and dimensions. Consequently, 60 cm water layer above the core had to be included in the calculation to provide enough space for the absorber blades when a control rod was fully withdrawn. Approximately 30 cm water layers were included below the core bottom support, and around the beryllium block and element reflectors. Vacuum boundary conditions were imposed on the outer boundary of the reactor system.

All Monte Carlo calculations in the present work were conducted with libraries from JEF-3.0, JENDL-3.3 and ENDF/B-VI.8 for temperature of 300 K. The measured critical effective multiplication factors were corrected when the core isothermal temperature was not identical with 300 K. All calculations were run with the number of total histories is 630.000 (NHIST) and five cycles are skipped before k_{eff} data accumulation begins.

4. Results and discussions

Table 2 shows the comparison between experiment data and MVP calculation results for first criticality and excess reactivity of RSG GAS first core. The first criticality predictions by MVP code were very close to the experiment data, especially the one with JEF-3.0 library. Among three libraries, the ENDF/B-VI.8 library produced a lowest effective multiplication factor although the difference with the experiment data was still below 0.5 %.

Table 2. Comparison of MVP calculation results with experiment data

Core Configuration		Experimental results	Calculated results		
			ENDF/B-VI.8	JEF-3.0	JENDL-3.3
First Criticality	k_{eff}	1.0	0.99554 (0.1371%)[a]	1.00132 (0.1298%)[a]	0.99763 (0.115%)[a]
	C/E[b]		0.996	1.001	0.998
Full core (all CRs are withdrawn)	k_{eff}	1.09242	1.08843 (0.1105%)[a]	1.09277 (0.1286%)[a]	1.09029 (0.1178%)[a]
	C/E[b]		1.004	1.000	1.002
Full core (all CRs are inserted)	k_{eff}	-	0.906535 (0.1412%)[a]	0.911994 (0.1733%)[a]	0.91077 (0.1325%)[a]
	C/E[b]		-	-	-
Control rod worth	$\Delta\rho$(%)	17.80	18.43	18.14	18.08
	C/E[b]		1.036	1.019	1.016

[a] Error

[b] Calculated value divided by experimental value

For the full configuration of the RSG GAS first core, the (corrected) excess reactivity data and MVP calculation results also showed a good agreement. However, compared to other libraries, ENDF/B-VI.8 library gave a highest effective multiplication factor although the difference with the experiment data was still below 0.5 %.

Table 2 shows that the ENDF/B-VI.8 and JENDL-3.3 libraries gave lower than experimental value for first criticality configuration. On the contrary, the ENDF/B-VI.8 and JENDL-3.3 libraries gave higher than experimental value for full configuration core. Fig. 5 shows that the thermal neutron fluxes of ENDF/B-VI.8 library are the highest one compared with other libraries in the beryllium reflector element. Since the full core configuration has more beryllium reflector element compared with first criticality configuration, so the calculated k_{eff} becomes higher. However, for the higher calculated k_{eff} using JENDL-3.3 is not affected by more beryllium reflector elements, but it is caused more standard fuel elements. Fig. 6 shows that the calculated neutron flux using JENDL-3.3 library is highest one compared with other libraries for all neutron energies.

The total control rod worth shown in Table 2 was obtained by a simple arithmetic summation of single control rod worth, while the single control rod worth was measured by a reactivity meter with shim rod bank compensation method. It is well known that the interference between control rods to some extent may deteriorate the accuracy of the total control rod worth obtained by the summation of single control rod worth. Combined with the uncertainty of the calculated β_{eff} value, the MVP results can be judged to be well agreed with the experiment data.

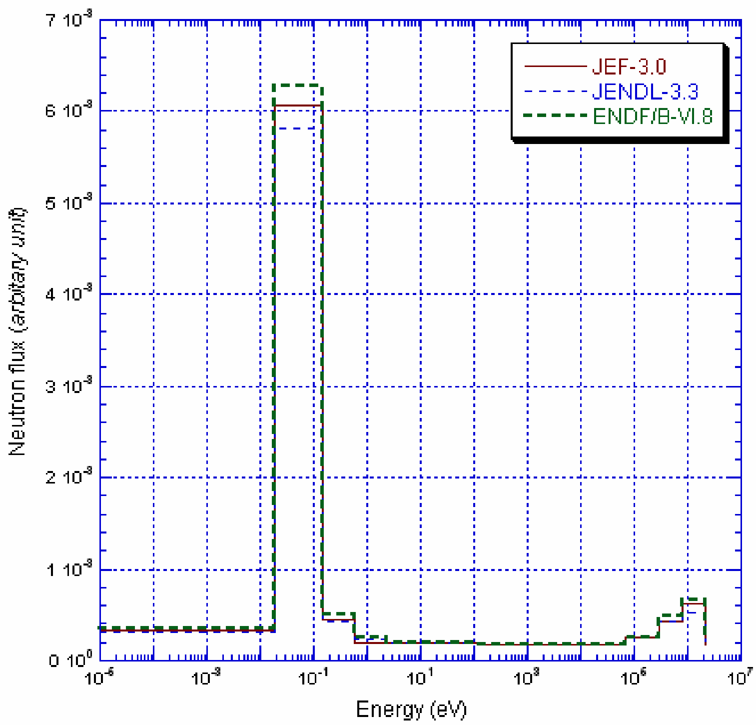


FIG. 5. Neutron flux distribution at a selected beryllium reflector element

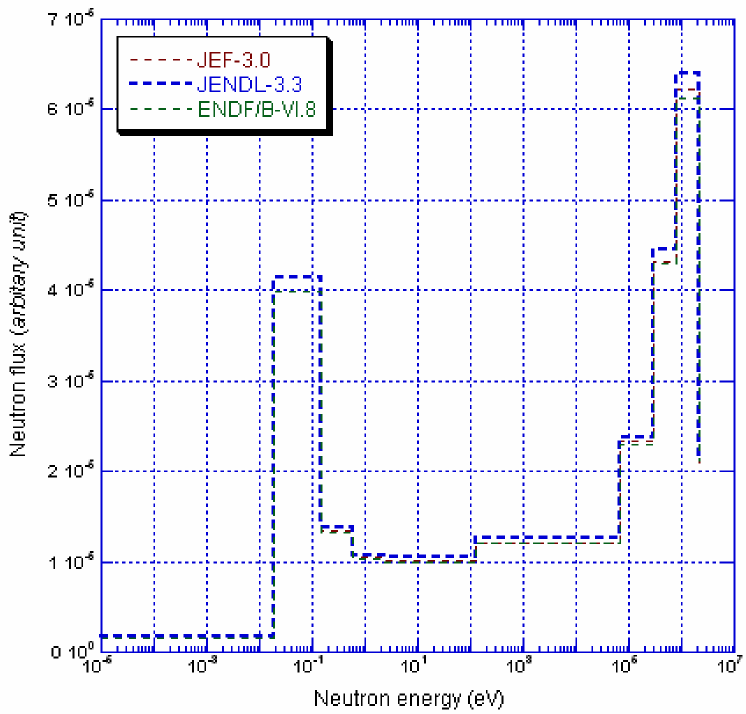


FIG. 6. Neutron flux distribution at a selected standard fuel element

5. Conclusions

Monte Carlo calculations have been conducted on the first core of the Indonesian 30 MWth Multipurpose Reactor, RSG GAS. A continuous energy Monte Carlo code MVP, developed by JAERI combined with nuclear data derived from ENDF/B-VI.8, JEF-3.0 and JENDL-3.3 libraries, was used for the whole benchmark calculations and the results were compared with experiment data. The MVP calculations with the three libraries produced k_{eff} values with excellent agreement to experiment data since the maximum differences are less than 0.5%. For the total control rod worth, the maximum difference is 3.6%. Systematically, ENDF/B-VI.8 library gave a maximum difference compared with other libraries. Therefore, the MVP code with recent libraries can be applied for the MTR type reactor with bulky Beryllium reflector.

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