The First Core Criticality Analysis of the RSG GAS Multipurpose Research Reactor using the Newly Released JENDL-5 Nuclear Data Library

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ABSTRACT

In the present work, the criticality experiments of the clean first core of the G. A. Siwabessy Multipurpose Reactor (RSG GAS) were analyzed by using the newly released Japanese Nuclear Data Library version 5 (JENDL-5) to contribute to the validation effort of the JENDL-5, especially for the application on the beryllium reflected, light-water moderated, low-enriched uranium (LEU, 19.75 % enrichment) fueled material testing reactors (MTRs). Compared to the other types of reactors, the number of testing/research reactors are very limited, and it is expected that the results would be beneficial not only to the research reactor communities but also to the criticality safety communities. In the past, we have already conducted similar criticality analyses using the then-available nuclear data libraries such as JENDL-3.2, 3.3, 4.0, ENDF/B-VII.1, VIII.0, and JEFF. In the present work, a continuous energy Monte Carlo MVP3 code is used in conjunction with the newly released JENDL-5 nuclear data library. The JENDL-5 keff C/E values are around 1.005 which shows the high fidelity of the library. Sensitivity analysis results reveal relatively large reactivity changes (-50 to -60 pcm) when U-235, U-238, or water JENDL-5 library is changed by the one of JENDL-4.0. Furthermore, the H-1 and O-16 contributions to the water reactivity change are found to be -162 pcm and +96 pcm, which are considerably large. The JENDL-5 average values of effective delayed neutron fraction (βeff), generation time (Λ), and Rossi-Alpha parameter (α) are found to be 738 pcm, 73 μsec, and 101 sec^{-1}, respectively. These average values are very close to the ones of JENDL-4.0, 738 pcm, 74 μsec, and 100 sec^{-1}, respectively.

KEYWORDS

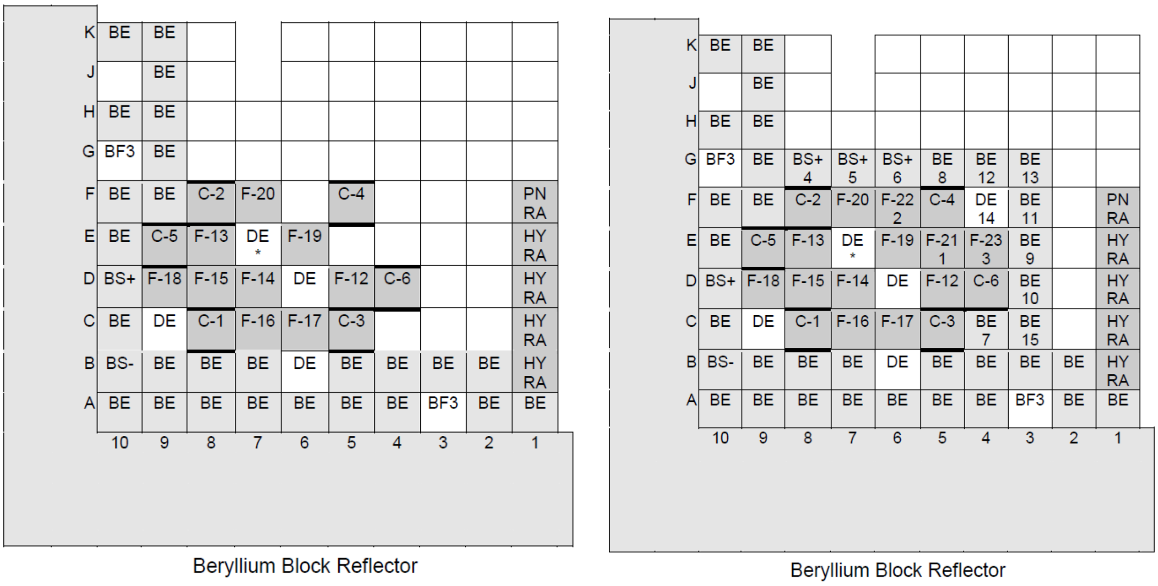
JENDL-5, Criticality benchmark, RSG GAS first core criticality, LEU oxide fuel, Be reflector

1. INTRODUCTION

Almost twelve years after the Japanese Evaluated Nuclear Data version 4 (JENDL-4.0) was released (2012) [1], the new version, i.e., the JENDL-5, was released in December 2021 [2]. In our previous work [3, 4, 5, 6], the criticality experiments of the clean first core of the G. A. Siwabessy Multipurpose Reactor (RSG GAS) [7] were used to validate the JENDL and other worldwide used nuclear data libraries (ENDF and JEFF). In the present work, the same criticality experiments were analyzed by using the new JENDL-5 to contribute to the validation effort of the JENDL-5, especially for the application on the beryllium reflected, light-water moderated, low-enriched uranium (LEU, 19.75 % enrichment) fueled material testing reactors (MTRs). The remainder of this paper is organized as follows. Section 2 presents a brief description of the RSG GAS reactor. Section 3 explained the methodology used in the criticality calculations. In Section 4, the results of the criticality analysis, control rod worth, core excess reactivity, and kinetic parameters are discussed. The conclusions and the future work are given in Section 5.

2. BRIEF DESCRIPTION OF RSG GAS

The RSG GAS is a light-water moderated and cooled, pool-type MTR that uses beryllium (Be) as its neutron reflector. It is designed with a maximum thermal power of 30 MW. The first criticality of this reactor was achieved in July 1987. The first core used an MTR plate-type oxide fuel (U_3O_8-Al) with low enrichment, 19.75 wt % U-235 (at present the core has been converted to silicide fuel). Fig. 1 shows the first criticality and full core configurations of the first core of RSG GAS. The main reactor data for the first core is shown in Table I. The standard fuel element (FE) and the control fuel element (CE) are shown in Fig. 2. Other main specifications of the reactors, as well as the detail of the criticality experiments, can be found elsewhere [5, 7].



	10	9	8	7	6	5	4	3	2	1
K	BE	BE								
J		BE								
H	BE	BE								
G	BF3	BE								
F	BE	BE	C-2	F-20		C-4				PN RA
E	BE	C-5	F-13	DE *	F-19					HY RA
D	BS+	F-18	F-15	F-14	DE	F-12	C-6			HY RA
C	BE	DE	C-1	F-16	F-17	C-3				HY RA
B	BS-	BE	BE	BE	DE	BE	BE	BE	BE	HY RA
A	BE	BE	BE	BE	BE	BE	BE	BF3	BE	BE

	10	9	8	7	6	5	4	3	2	1
K	BE	BE								
J		BE								
H	BE	BE								
G	BF3	BE	BS+ 4	BS+ 5	BS+ 6	BE 8	BE 12	BE 13		
F	BE	BE	C-2	F-20	F-22 2	C-4	DE 14	BE 11		PN RA
E	BE	C-5	F-13	DE *	F-19	F-21 1	F-23 3	BE 9		HY RA
D	BS+	F-18	F-15	F-14	DE	F-12	C-6	BE 10		HY RA
C	BE	DE	C-1	F-16	F-17	C-3	BE 7	BE 15		HY RA
B	BS-	BE	BE	BE	DE	BE	BE	BE	BE	HY RA
A	BE	BE	BE	BE	BE	BE	BE	BF3	BE	BE

F	: Fuel Element	BS-	: Be Refl. El. without Stopper
C	: Control Element	BS+	: Be Refl. El. with Stopper
BE	: Be Reflector Element	DE	: Dummy Element
PNRA	: Pneu. Rabbit System	HYRA	: Hydraulic Rabbit System
BF_3	: BF_3 Neutron Detector	*	: Neutron Source (Cf-252)

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BF_3	: Neutron Detector	*	: Neutron Source (Cf-252)

Figure 1. First criticality (left) and full core (right) configurations of the RSG GAS first core (plain numbers in the core show loading step of fuel, Be reflector and dummy elements after the first criticality; empty lattices contain water)

3. METHODOLOGY

3.1. Selection of Criticality Benchmark Cases

There were two sequences of criticality experiments conducted for the first core of RSG GAS, i.e. (1) the criticality approach experiment for achieving the first criticality, and (2) the criticality experiment for core excess reactivity loading followed by control rod worth calibrations. At the end of the first sequence of the criticality experiment, the first criticality was achieved at the 9-th loading of standard fuel element [5], i.e. the regulating rod (RR) was pulled out slowly (reactor periods were greater than

50 s) while the other 5 shim rods were fully withdrawn. The insertion position of the RR at the first criticality was 475 mm. This configuration, shown on the left side of Fig. 1, was chosen as one of the most appropriate for this study.

In the second sequence of the criticality experiment, the loading of additional fuel elements and reflector elements was conducted to achieve a full core configuration with sufficient excess reactivity for one core cycle. The reactivity gains were measured at each loading step 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 measurement of the accumulated excess reactivity by this method should be considered as an uncorrected value. The excess reactivity value was then corrected by a method described elsewhere [5]. After that, the control rod calibrations were conducted following the excess reactivity loading for the six control rods with various methods. Many combinations of control rod positions can be found during the calibrations which gave a critical core condition. In this work, the combinations of control rod positions that occurred during control rod calibration with the bank compensation method were selected for the benchmark calculations.

Table I. Main reactor data of RSG GAS first core

Reactor Type	Pool Type
Fuel Element Type	LEU Oxide MTR
Moderator/Coolant	H_2O
Reflector	Be & H_2O
Nominal Power (MWth)	10.7
No. of Fuel Elements	12
No. of Control Elements	6
No. of Fork Type Absorber (pairs)	6
Fuel/Control Element Dimension (mm)	77.1 x 81 x 600
Fuel Plate Thickness (mm)	1.3
Coolant Channel Width (mm)	2.55
No. of Plate per Fuel Element	21
No. of Plate per Control Element	15
Fuel Plate Clad Material	$AlMg_2$
Fuel Plate Clad Thickness (mm)	0.38
Fuel Meat Dimension (mm)	0.54 x 62.75 x 600
Fuel Meat Material	U_3O_8-Al
U-235 Enrichment (wt. %)	19.75
Uranium Density in Meat (g/cm^3)	2.96
U-235 Loading per Fuel Element (g)	250
U-235 Loading per Control Element (g)	178.57
Absorber Meat Material	Ag-In-Cd
Absorber Thickness (mm)	3.38
Absorber Clad Material	SUS-321
Absorber Clad Thickness	0.85

3.2. Criticality Calculation Model and Conditions

The active part (7.71 x 8.1 x 60 cm^3) of both FE and CE 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 is very complicated, that is, the structure materials were homogenized with water by volume weighting. An exact modeling approach was also taken for the active parts of the Be reflector elements, Be 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 MVP3 [8] 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, a 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.

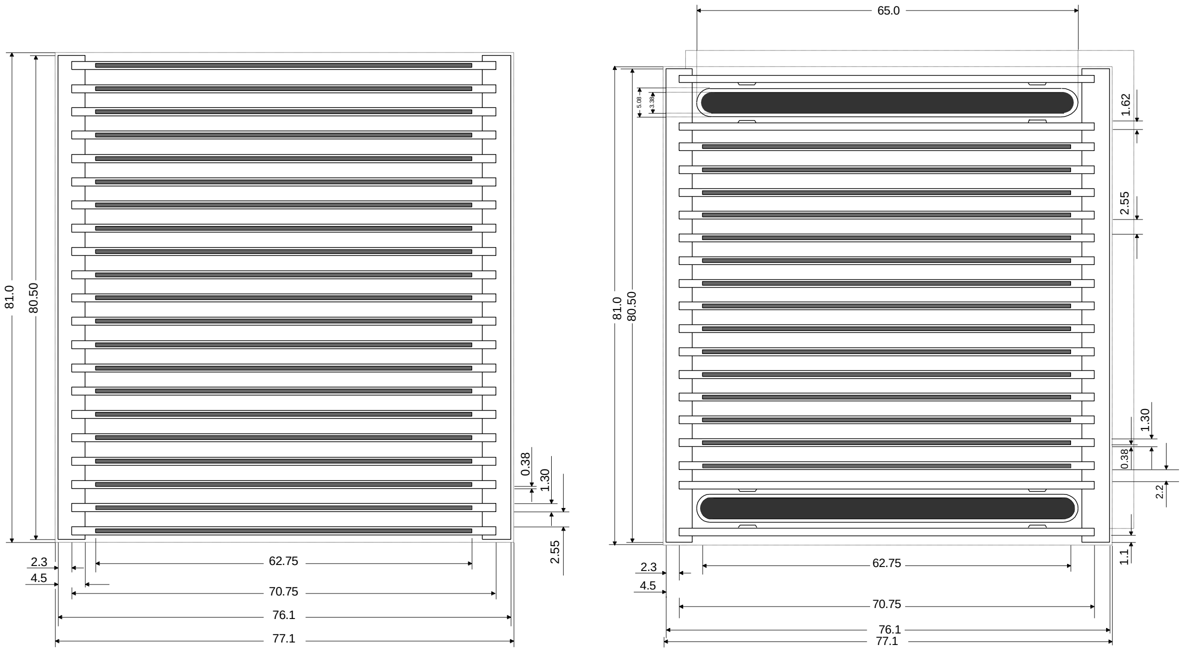

Figure 2. Standard fuel element (left) and control fuel element (right) of the RSG GAS (unit mm)

All MVP3 calculations in the present benchmark were conducted with the JENDL-5 library for a room temperature of 300 K. The measured critical effective multiplication factors were corrected when the core isothermal temperature was not identical to 300 K. The total number of batches (generations) was 10,000 where each batch consists of 10,000 histories, i.e. the total number of effective histories was 100 million. Additional initial 100 batches were skipped to guarantee the fundamental mode had been achieved before statistical evaluation of keff and other tallies were conducted. Under these calculation conditions, the fractional standard deviation (FSD) for keff was less than 0.01 % for all cases. The main benchmark results are the keff values which can be directly compared with the experimental/measured ones. The inter-library comparison results also provide valuable information, differences on the critical keff, excess reactivity, control rod worth, and kinetics parameters amongst libraries will also be presented and discussed. As for the calculations of the kinetics parameters, the MVP3 differential operator sampling with perturbed source effect and default number of propagation batches (10) was adopted as in our previous work [9].

4. ANALYSIS RESULTS

The analysis results are shown in Tables II and III for criticality-related values and kinetic parameters, respectively. In the tables, the results for the JENDL-4.0 of our previous work are also shown for inter-library comparison. As mentioned in Section 3, since the benchmark cases are divided into two groups according to the two sequences of critical experiments, the tables are also divided accordingly.

In general, for the first group of Table II, one can observe that although the keff C/E values of JENDL-5 are very close to 1.0, the values are slightly larger than the ones of JENDL-4.0. The maximum difference between the two libraries is found for the full core configuration with all control rods inserted, i.e. around 262 pcm. In the same table, the experiment data (keff or the excess reactivity) for the full core is included but needs several remarks since it is not purely experiment data. The measurement detail of core excess reactivity and the correction method is given in [5]. It should be noted here that the measured excess reactivity must be converted from $ unit to absolute value to get the (absolute unit) keff, and therefore, the accuracy depended on the accuracy of the calculated effective delayed neutron fraction, βeff (=765 pcm). No measured data were then available for βeff so further assessment of the accuracy of the calculated βeff was not possible. This factor also contributed to the difference between the experiment data and Monte Carlo results. The sub-criticality of the full core when all control rods were inserted was not measured and only calculation results are available. Combined with the calculated excess reactivity values, the calculated total worth of all control rods can be estimated. The experimental data for the total control rod worth was obtained by a simple arithmetic summation of the single control rod worth, while the single control rod worth was measured by a reactivity meter with the 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 the single control rod worth [10]. Combined with the uncertainty of the calculated βeff value, the MVP3 results for JENDL-5 and JENDL-4.0 can be judged to be well agree with the experiment data.

The second group of Table II shows the comparison between MVP3 results with several critical conditions of the RSG GAS first core that occurred during control rod calibrations. These critical conditions were achieved when a calibrated rod was fully inserted and the other rods were in a certain bank position. The measured critical conditions shown in the table were already corrected for the Joule effect (which turned out to be negligible) of the primary cooling pumps. Unfortunately, the measurements for the critical conditions were not repeated so no reliable measurement error prediction can be deducted. Similar trends of keff C/E values can be observed for the two JENDL libraries.

A sensitivity study was conducted to investigate the reactivity change when a single nuclide or material library of JENDL-5 is replaced by one of the JENDL-.40, and the result is summarized in Fig. 3. The main nuclides composing the fuel region are U-235, U-238, O-16 and Al-27 (meat and clad) and H-1 and O-16 (water moderator/coolant), while the reflector region mainly consists of Be-9. In other reactor structural regions, Al-27, water (H-1 and O-16) are dominant. It should be noted that one can not separate the bounded hydrogen $S(\alpha,\beta)$ thermal library from the free gas (fast) library of MVP libraries, therefore, the reactivity change from replacing the library is the summation of both $S(\alpha,\beta)$ thermal and fast library of hydrogen. In the study, the reactivity change of water is further broken down into H-1 and O-16 contributions. Relatively large reactivity changes are observed for U-235, U-238, and water, namely around -50 to -60 pcm. Furthermore, the contribution of H-1 is -162 pcm while the one of O-16 is +96 pcm. Al-27 and Be-9 show negligible reactivity changes. Relatively strong absorber materials, i.e. Ag, In, and Cd, also do not show large reactivity changes.

Table III shows the kinetic parameters calculated by using the JENDL-5 library, and the results are also compared to the ones using JENDL-4.0 [9]. The JENDL-5 average values of βeff, generation time (Λ), and Rossi-Alpha parameter (α) for the second group benchmark cases are found to be 738 pcm, 73 μsec, and 101 sec^{-1}, respectively. These average values are very close to the ones of JENDL-4.0, 738 pcm, 74 μsec, and 100 sec^{-1}, respectively. It is assumed that there is no significant revision of delayed neutron data of U-235 and U-238 from JENDL-4.0 to JENDL-5. The calculated βeff values are slightly smaller than the βeff value (765 pcm) provided by the reactor vendor, which was used during the commissioning of the reactor. The non-critical core kinetic parameters (from the first group of Table III) are listed for reference only.

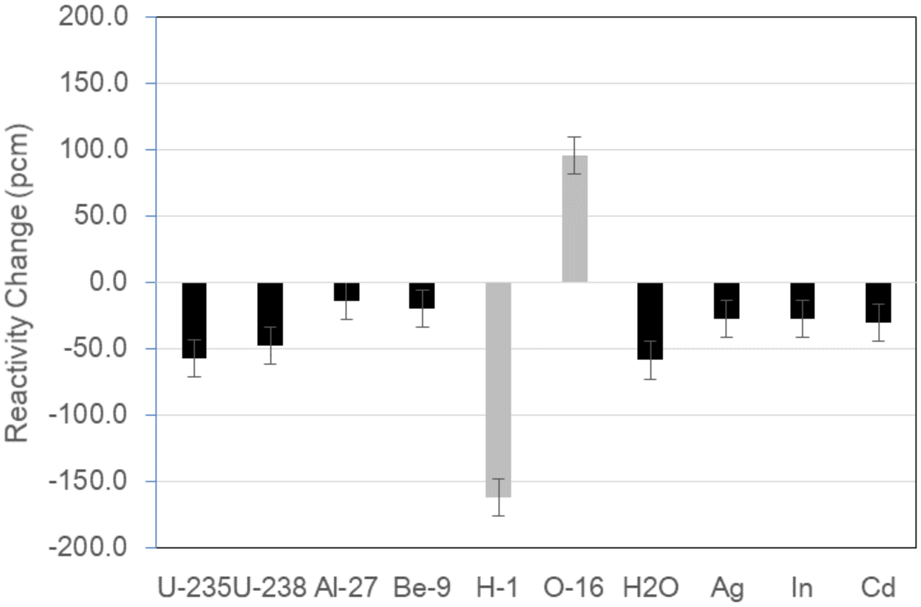

Figure 1. Sensitivity analysis results by changing a single JENDL-5 library with JENDL-4.0

5. CONCLUSIONS AND FUTURE WORK

Criticality analysis of the newly released JENDL-5 nuclear data library against the criticality experiments of the first core of RSG GAS has been conducted using the continuous energy Monte Carlo MVP3 code. In general, the JENDL-5 keff C/E values are around 1.005 which shows the high fidelity of the library. Sensitivity analysis results reveal relatively large reactivity changes (-50 to -60 pcm) when U-235, U-238, or water JENDL-5 library is changed by the one of JENDL-4.0. Furthermore, the H-1 and O-16 contributions to the water reactivity change are found to be -162 pcm and +96 pcm, which are considerably large. The reactivity changes obtained in this sensitivity analysis show the same order and trend as reported in Table 4 of Ref. [11] (The dk/k values of uranium-fueled light-water moderated system when each nuclide is changed from JENDL-5). As for future work, we plan to conduct sensitivity and uncertainty of keff of JENDL-5 similar to our previous work for JENDL-4.0 and ENDF/B-VII.1 [12].

Table II. Criticality analysis results for JENDL-5 (present work) and JENDL-4.0 [5]

Core configuration		Experiment data	JENDL-4.0	JENDL-5	Δρ(J5-J4) (pcm) d)
First criticality (9 FEs, 6 CEs, RR=475 mm)	keff	1.0	1.00243	1.00496	251
	C/E		1.002	1.005	-
Full core (9 FEs, 6 CEs, CRs all up)	keff	1.09242 a)	1.09854	1.10054	165
	C/E		1.006	1.007	-
Full core (9 FEs, 6 CEs, CRs all down)	keff	n.c. b)	0.91880	0.92102	262
	C/E		-	-	-
Control rods worth	Δρ(%)	17.80 c)	17.81	17.71	-97
	C/E		1.000	0.995	-

Calibrated rod / grid position (calibrated rod position / other rod bank position)		Experiment data	JENDL-4.0	JENDL-5	Δρ(J5-J4) (pcm) d)
JDA05 / C-5 (600 mm / 288 mm)	keff	1.00008 a)	1.00342	1.00536	192
	C/E		1.003	1.005	-
JDA06 / C-8 (600 mm / 290 mm)	keff	1.00008	1.0026	1.00470	208
	C/E		1.003	1.005	-
JDA07 / D-4 (600 mm / 282 mm)	keff	1.00008	1.00277	1.00516	237
	C/E		1.003	1.005	-
JDA01 / E-9 (600 mm / 284 mm)	keff	1.00008	1.00231	1.00455	222
	C/E		1.002	1.004	-
JDA04 / F-5 (600 mm / 290 mm)	keff	1.00008	1.003	1.00519	217
	C/E		1.003	1.005	-
JDA03 / F-8 (600 mm / 293 mm)	keff	1.00008	1.00281	1.00479	197
	C/E		1.003	1.005	-

a) β=0.00765, excess reactivity was already corrected (see text)

b) n.c. = not conducted

c) Shim rod bank compensation method, measured by a reactivity meter, summation of single control rod worth

d) FSDs for all keff are less than 0.01% (J4: JENDL-4.0, J5: JENDL-5)

Table III. Kinetic parameters analysis results for JENDL-5 (present work) and JENDL-4.0 [9]

First criticality and excess reactivity loading	Kinetics Parameters	JENDL-4.0	JENDL-5	JENDL-4.0/ JENDL-5
First criticality (9 FEs, 6 CEs, RR=475 mm)	β_{eff} (pcm)	751.4 ± 3.4	749.5 ± 3.4	1.003
	Λ (μsec)	82.79 ± 0.08	81.82 ± 0.07	1.012
	α (1/sec)	90.8 ± 0.4	91.6 ± 0.4	0.991
Full core (12 FEs, 6 CEs, CRs all up)	β_{eff} (pcm)	727.7 ± 3.2	728.9 ± 3.1	0.998
	Λ (μsec)	76.51 ± 0.07	75.56 ± 0.06	1.013
	α (1/sec)	95.1 ± 0.4	96.5 ± 0.4	0.986
Full core (12 FEs, 6 CEs, CRs all down)	β_{eff} (pcm)	742.0 ± 3.5	747.2 ± 3.5	0.993
	Λ (μsec)	69.17 ± 0.06	68.20 ± 0.06	1.014
	α (1/sec)	107.3 ± 0.5	109.6 ± 0.5	0.979

Calibrated rod / grid position (calibrated rod position / other rod bank position)	Kinetics Parameters	JENDL-4.0	JENDL-5	JENDL-4.0/ JENDL-5
JDA06 / C-8 (600 mm / 290 mm)	β_{eff} (pcm)	738.6 ± 3.3	737.8 ± 3.3	1.001
	Λ (μsec)	73.60 ± 0.06	72.72 ± 0.06	1.012
	α (1/sec)	100.4 ± 0.5	101.5 ± 0.5	0.989
JDA01 / E-9 (600 mm / 284 mm)	β_{eff} (pcm)	734.6 ± 3.3	743.3 ± 3.3	0.988
	Λ (μsec)	75.32 ± 0.06	71.90 ± 0.06	1.048
	α (1/sec)	100.9 ± 0.5	103.4 ± 0.5	0.976
JDA03 / F-8 (600 mm / 293 mm)	β_{eff} (pcm)	742.3 ± 3.4	733.0 ± 3.4	1.013
	Λ (μsec)	73.42 ± 0.06	72.31 ± 0.06	1.015
	α (1/sec)	101.1 ± 0.4	101.4 ± 0.5	0.998
JDA05 / C-5 (600 mm / 288 mm)	β_{eff} (pcm)	741.9 ± 3.4	735.2 ± 3.3	1.009
	Λ (μsec)	73.67 ± 0.07	72.73 ± 0.06	1.013
	α (1/sec)	100.7 ± 0.5	101.1 ± 0.5	0.996
JDA04 / F-5 (600 mm / 290 mm)	β_{eff} (pcm)	737.1 ± 3.4	739.2 ± 3.3	0.997
	Λ (μsec)	75.32 ± 0.07	74.30 ± 0.07	1.014
	α (1/sec)	97.9 ± 0.5	99.5 ± 0.5	0.984
JDA07 / D-4 (600 mm / 282 mm)	β_{eff} (pcm)	735.0 ± 3.4	741.8 ± 3.3	0.991
	Λ (μsec)	73.93 ± 0.07	73.09 ± 0.06	1.012
	α (1/sec)	99.4 ± 0.5	101.5 ± 0.5	0.980

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REFERENCES

[1] Shibata, K. et al. JENDL-4.0: A New Library for Nuclear Science and Engineering. J. of Nucl. Sci. Technol. 2011, vol. 48(1), pp. 1-30.
[2] Iwamoto, N. et. al. Japanese Evaluated Nuclear Data Library Version 5: JENDL-5. J. Nucl. Sci. Technol. 2023, vol. 60(1), pp. 1-60.
[3] Liem, P. H. Science and Technology Agency Internal Report. 1995, JAERI, Japan.
[4] Liem, P. H. Monte Carlo Calculations on the First Criticality of the Multipurpose Reactor G. A. Siwabessy. Atom Indonesia. 1998, vol. 24(2).
[5] Liem, P. H. and Sembiring, T. M. Benchmarking the New JENDL-4.0 Library on Criticality Experiments of a Research Reactor with Oxide LEU (20 w/o) Fuel, Light Water Moderator and Beryllium Reflectors. Ann. Nucl. Energy. 2012, vol. 44, pp. 58-64.
[6] Hartanto, D. and Liem, P. H. Analysis of the First Core of the Indonesian Multipurpose Research Reactor RSG-GAS using the Monte Carlo Serpent Code and the ENDF/B-VIII.0 Nuclear Data Library. Nucl. Eng. Tech. 2020, vol. 52(12), pp. 2725-2732.
[7] BATAN. Multipurpose Reactor G. A. Siwabessy Safety Analysis Report - Rev. 7. 1980. Jakarta.
[8] Nagaya, Y. et al. MVP/GMVP Version 3: General Purpose Monte Carlo Codes for Neutron and Photon Transport Calculations Based on Continuous Energy and Multigroup Methods. JAEA-Data/Code 2016-019. 2016.
[9] Liem, P. H. et al. Kinetics Parameters Evaluation on the First Core of RSG GAS (MPR-30) using Continuous Energy Monte Carlo Method. Prog. Nucl. Energy. 2018, vol. 109, pp. 196-203.
[10] Liem, P. H. et al. Study on the Control Rod Interaction Effect in RSG GAS Multipurpose Reactor (MPR-30). Ann. Nucl. Energy. 2002, vol 29(6), pp. 701-716.
[11] Tada, K. et al. JENDL-5 Benchmarking for Fission Reactor Applications. J. of Nucl. Sci. Technol. 2023. DOI:10.1080/00223131.2023.2197439.
[12] Liem, P. H. et al. Sensitivity and Uncertainty Analysis on the First Core Criticality of the RSG GAS Multipurpose Research Reactor. Prog. in Nucl. Energy. 2019, vol. 114, pp. 46-60.