Nuclear data sensitivity and uncertainty analyses on the first core criticality of the RSG GAS multipurpose research reactor

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Abstract. The results of criticality, sensitivity and uncertainty (S\U) analyses on the first core criticality of the Indonesian 30 MWth Multipurpose Reactor RSG GAS (MPR-30) using the recent nuclear data libraries (ENDF/B-VII.1 and JENDL-4.0) and analytical tools available at present (WHISPER-1.1) are presented. Two groups of criticality benchmark cases were carefully selected from the experiments conducted during the first criticality approach and control rod calibrations. The C/E values of effective neutron multiplication factor (k) for the worst case was found around 1.005. Large negative sensitivities were found in (n,γ) reaction of H-1, U-235, Al-27, U-238 and Be-9 while large positive sensitivities were found in U-235 (total nu and fission), H-1 (elastic), Be-9 (free gas, elastic) and H-1 S(α, β) (lwtr.20t, inelastic). The S\U analysis results concluded that the uncertainties of k originated from the nuclear data were found around 0.6% which covered well the [C/E-1] values. Differences in the sensitivities amongst the two nuclear data libraries were also identified, and recommendation for improving the nuclear data library was given.

1 Introduction

With the growing computing speed, the Monte Carlo method has been widely adopted in the reactor physics simulation including the neutronics benchmark calculation of Indonesian Multipurpose Research Reactor RSG GAS in order to evaluate the effective neutron multiplication factor k [1], control rod worth, power distribution, and kinetics parameters [2]. Furthermore, many Monte Carlo codes such as McCARD [3], MCNP6.2 [4] and Serpent2 [5] have the capability to calculate the sensitivity coefficient (S) of k to each isotope for each neutron reaction at each energy group. Later, using S and the covariance data from the nuclear data library, the uncertainty (U) of k to the nuclear data library can be calculated.

In this paper, the sensitivity and uncertainty (S\U) analyses on the first core criticality of the Indonesian Multipurpose Research Reactor RSG GAS are evaluated using the modern nuclear data libraries such as ENDF/B-VII.1 [6] and JENDL-4.0 [7] using Whisper-1.1 [8] within MCNP6.2 software distribution.

2 Description of RSG GAS

The Indonesian multipurpose research reactor, Reaktor Serba Guna G.A. Siwabessy (RSG GAS) is a beryllium-reflected, light-water-moderated and -cooled, 30 MWth (max.) pool-type reactor. The first criticality of the reactor was achieved in 1987. Reactor main data is shown in Table 1. Fuel elements (FEs) are plate-type, and each consists of 21 fuel plates assembled by two side plates. One fuel plate contains 19.75% enriched uranium oxide meat dispersed in aluminum matrix, and aluminum cladding on both sides of the meat. Control fuel elements (CEs) 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. A fork type control rod can be inserted into or withdrawn out of the control fuel element.

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. At each loading step, reactivity gains were measured by calibrating the difference of the regulating rod (RR) position with a reactivity meter and it had been corrected by a method described elsewhere [1]. The first criticality and full core configurations of the first core are shown in Figs. 1 and 2, respectively. Meanwhile, Table 2 lists the calculation cases of the first core of RSG GAS.

3 Results and Discussion

As mentioned previously, the sensitivity coefficients in this study have been calculated by MCNP6.2 using 10,000 neutron histories and 10,000 neutron generations with 100 inactive ones. These calculation parameters provide standard deviation of 8 pcm for the k value. Then, Whisper-1.1 has been used to compute the uncertainty using the

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Table 1. Design Parameters of RSG GAS [1]

General	
Reactor type	Pool type
Fuel element type	LEU oxide MTR
Cooling system	Forced convection Down Flow
Moderator/coolant	H_2O
Reflector	Be and H_2O
Maximum power (MWth)	30
Core Characteristics (full core configuration)	
No. of fuel elements	40
No. of control elements	8
No. of fork type absorbers (pairs)	8
Nominal cycle length (fpd)	25
Avg. burn-up at BOC (% loss of U-235)	23.3
Avg. burn-up at EOC (% loss of U-235)	31.3
Avg. discharge burn-up at EOC (% loss of U-235)	53.7
Fuel/control elements	
Fuel/control element dimension (mm)	77.1×81×60
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)	0.54×62.75×600
Fuel meat material	U_3O_8Al
U-235 enrichment (w/o)	19.75
Uranium density in meat (g/cc)	2.96
U-235 loading per FE (g)	250
U-235 loading per CE (g)	178.6
Absorber meat material	Ag–In–Cd
Absorber thickness (mm)	3.38
Absorber clad material	SS-321
Absorber clad thickness (mm)	0.85

Table 2. Calculation cases of RSG GAS first core

First Group	
First criticality and excess reactivity loading	*k* value
First criticality (9 FEs, 6 CEs, RR=475 mm)	1.0
Full core (12 FEs, 6 CEs, CRs all up)	1.0886
Full core (12 FEs, 6 CEs, CRs all down)	N/A
Second Group	
Calibrated rod/grid position	*k* value
JDA06/C-8 (600 mm/290 mm)	1.00008
JDA01/E-9 (600 mm/284 mm)	1.00008
JDA03/F-8 (600 mm/293 mm)	1.00008
JDA05/C-5 (600 mm/288 mm)	1.00008
JDA04/F-5 (600 mm/290 mm)	1.00008
JDA07/D-4 (600 mm/282 mm)	1.00008

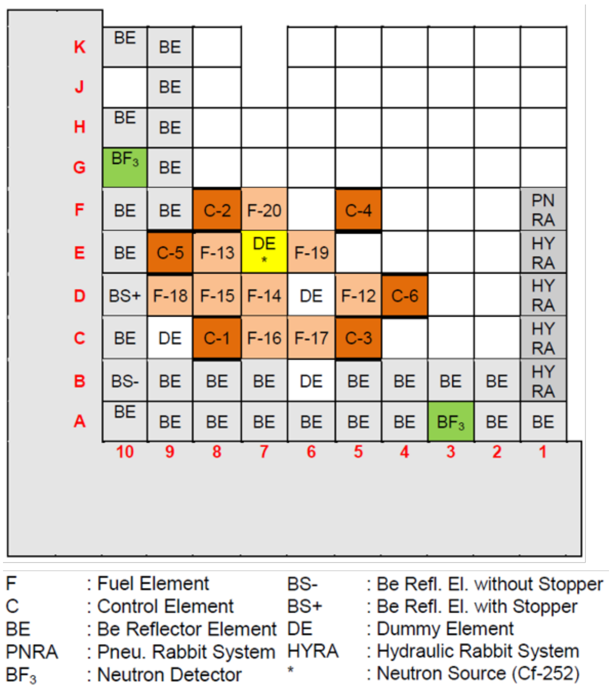

Figure 1. First criticality core configuration of RSG GAS first core [1]

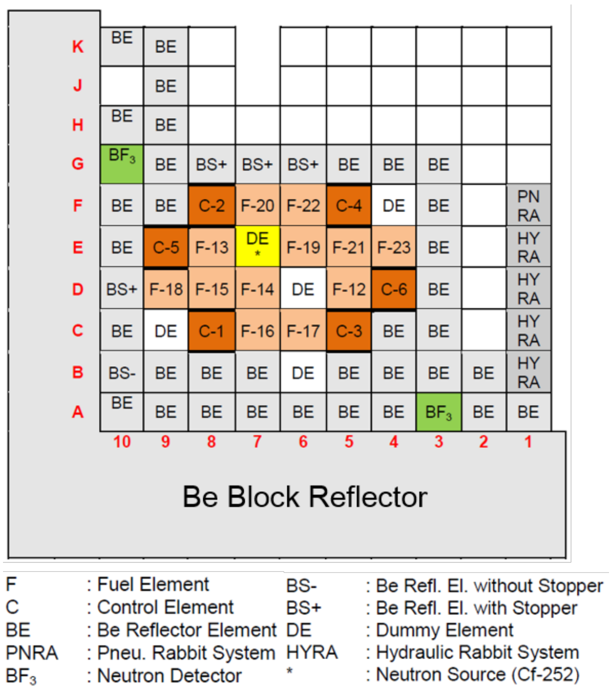

Figure 2. Full core configuration of RSG GAS first core [1]

44-group covariance data available in the nuclear data library. The sensitivity and uncertainty analyses have been performed for all cases in the first- and second-group of first core criticality of RSG GAS. However, the sensitivity coefficient is discussed only for configuration JDA06/C-8 since other core configurations produce similar values. To further confirm the results, the sensitivity coefficients gen-

Table 3. Sensitivities of k in decreasing direction

Nuclei/ Element	Neutron Reaction	Sensitivity Coefficient by MCNP6.2 ENDF/B-VII.1	JENDL-4.0	JENDL/ENDF	MCNP6.2/Serpent2 Coefficient ENDF/B-VII.1	JENDL-4.0
H-1 (free gas)	(n,γ)	-1.69E-01	-1.70E-01	1.01	1.00	1.00
U-235	(n,γ)	-1.22E-01	-1.22E-01	1.00	1.00	1.00
Al-27	(n,γ)	-5.04E-02	-4.93E-02	0.98	0.99	0.98
U-238	(n,γ)	-3.79E-02	-3.82E-02	1.01	1.00	1.00
Be-9 (free gas)	(n,γ)	-5.41E-03	-4.76E-03	0.88	0.64	0.61
Mn-55	(n,γ)	-4.39E-03	-4.41E-03	1.01	1.00	0.99
Ag-109	(n,γ)	-3.65E-03	-3.62E-03	0.99	1.07	1.04
Be-9 (free gas)	(n,α)	-3.09E-03	-3.00E-03	0.97	N/A	N/A
In	(n,γ)	-2.71E-03	-2.66E-03	0.98	1.08	1.05
Ag-107	(n,γ)	-1.88E-03	-1.92E-03	1.02	1.04	1.03
Cd	(n,γ)	-1.63E-03	-1.58E-03	0.97	1.07	1.10
Fe	(n,γ)	-1.34E-03	-1.34E-03	1.00	0.99	1.01
O-16	(n,α)	-1.24E-03	-1.21E-03	0.98	N/A	N/A

Table 4. Sensitivities of k in increasing direction

Nuclei/ Element	Neutron Reaction	Sensitivity Coefficient by MCNP6.2 ENDF/B-VII.1	JENDL-4.0	JENDL/ENDF	MCNP6.2/Serpent2 Coefficient ENDF/B-VII.1	JENDL-4.0
U-235	ν_{tot}	9.95E-01	9.95E-01	1.00	1.00	1.00
U-235	(n,f)	3.51E-01	3.50E-01	1.00	1.00	1.00
H-1	(n,el)	2.70E-01	2.90E-01	1.08	0.92	0.99
Be-9	(n,el)	6.03E-02	6.04E-02	1.00	0.99	1.01
lwtr.20t	(n,inl)	5.11E-02	3.32E-02	0.65	N/A	N/A
O-16	(n,el)	4.44E-02	4.38E-02	0.99	1.01	1.02
Al-27	(n,el)	2.64E-02	2.72E-02	1.03	0.95	0.94
Be-9	(n,2n)	1.13E-02	1.15E-02	1.01	1.00	1.00
Al-27	(n,inl)	1.13E-02	9.95E-03	0.88	1.02	0.99
U-238	(n,el)	5.21E-03	4.71E-03	0.90	1.14	0.98
U-238	ν_{tot}	5.01E-03	5.05E-03	1.01	1.01	1.00
be.20t	(n,inl)	4.97E-03	4.73E-03	0.95	N/A	N/A
U-238	(n,f)	3.44E-03	3.48E-03	1.01	1.01	1.02
be.20t	(n,el)	3.43E-03	2.88E-03	0.84	N/A	N/A
U-238	(n,inl)	2.30E-03	2.09E-03	0.91	1.08	0.91
Fe	(n,el)	2.07E-03	1.82E-03	0.88	1.13	0.96

erated by Serpent2 are also reported and compared to the ones from MCNP6.2.

First, the sensitivity is discussed. The sensitivity coefficients of k on each nuclide and each nuclear reaction are sorted according to their signs which are negative coefficients which reduce the k, and positive coefficients which increase k. The negative sensitivity coefficients are listed in Table 3 and they are dominated by (n,γ) reactions. The top five contributors of negative sensitivity coefficients are (n,γ) reactions of H-1, U-235, Al-27, U-238, and Be-9. Meanwhile, the positive sensitivity coefficients as summarized in Table 4 are contributed by several nuclear reactions of different isotopes. It is noticed that there are also contribution from S(α,β) libraries. The top five contributors of positive sensitivity coefficients are ν_{tot} and (n,f) of U-235, (n,inl) of H-1, (n,el) of Be-9, and (n,inl) of H-1 S(α,β). Furthermore, Tables 3 and 4 also compare the value of the sensitivity coefficient between MCNP6.2 and Serpent2. It is shown that generally the coefficients agree very well between the two codes except for the (n,γ) of Be-9 (about -36%). A further analysis is required to understand the cause of this large difference.

The sensitivity coefficients of the two nuclear data libraries are also shown in Tables 3 and 4. In Table 3, (n,γ) of Be-9 has large discrepancy between the 2 libraries, about -12%. Meanwhile, in Table 4, (n,inl) of H-1 S(α,β) between the 2 libraries differs significantly, about -35%. There are also large differences due to the (n,el) of Be-9 S(α,β) (about -16%), (n,inl) of Al-27 (about -12%), and (n,el) of Fe (about -12%). This shows that the particular neutron reaction cross sections of those aforementioned isotopes differ significantly between the two libraries.

Next, the uncertainty is presented. The uncertainty analysis has been calculated for all core configurations using only ENDF/B-VII.1 nuclear data library, and it is shown in Table 5. The calculated k value by MCNP6, and the ratio of calculation to experiment are also listed in the table. It is observed that the uncertainties of k originating from the nuclear data library are positive and around 6%. It shows more contribution from the positive sensi-

Table 5. Uncertainty analysis result (ENDF/B-VII.1)

First Group			
First criticality and excess reactivity loading	k value	[C/E-1]	Uncertainty
First criticality (9 FEs, 6 CEs, RR=475 mm)	1.00482	0.05	0.00609
Full core (12 FEs, 6 CEs, CRs all up)	1.09889	0.06	0.00564
Full core (12 FEs, 6 CEs, CRs all down)	0.91940	N/A	0.00595
Second Group			
Calibrated rod/grid position	k value		
JDA06/C-8 (600 mm/290 mm)	1.00269	0.003	0.00583
JDA01/E-9 (600 mm/284 mm)	1.00260	0.003	0.00584
JDA03/F-8 (600 mm/293 mm)	1.00366	0.004	0.00583
JDA05/C-5 (600 mm/288 mm)	1.00375	0.004	0.00583
JDA04/F-5 (600 mm/290 mm)	1.00402	0.004	0.00583
JDA07/D-4 (600 mm/282 mm)	1.00375	0.004	0.00583

tivity coefficients listed in Table 4. It is worth to mention that the calculated uncertainties are comparable to (C/E-1) of k. It should be noted that the evaluated uncertainties in this study only consider the ones originating from the nuclear data library. The experimental measurements also have uncertainty as well as the modeling of the core using MCNP. All of these uncertainties will impact the uncertainty of k.

4 Conclusions

The sensitivity and uncertainty analyses on the first core criticality of RSG GAS using two modern nuclear data libraries (ENDF/B-VII.1 and JENDL-4.0) have been performed using Whisper-1.1. Serpent2 has also been used to verify the generated sensitivity coefficients by MCNP6.2. In the sensitivity analysis, it was found that (n,γ) reactions of H-1, U-235, Al-27, U-238, and Be-9 provide large negative values, while ν_{tot} and (n,f) of U-235, (n,inl) of H-1, (n,el) of Be-9, and (n,inl) of H-1 S(α,β) give large positive values. The discrepancy of the sensitivity coefficients of the two nuclear data library has also been identified. On the other hand, the uncertainty analysis shows that the uncertainties of k to ENDF/B-VII.1 nuclear data library are about 0.6% which are comparable to the [C/E-1] values of k. For the future study, the uncertainty analysis should be continued for JENDL-4.0 when its covariance data is available for Whisper-1.1. Moreover, this study can be also performed using the latest nuclear data library, ENDF-B/VIII.0.

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