A SRAC CALCULATION OF THE VVER 1000 CORE'S EFFECTIVE MULTIPLICATION FACTOR

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ABSTRACT: Neutronic characteristics for a VVER-1000 were investigated by using SRAC code and nuclear data library JENDL-3.3 with 107 public energy groups. The elementary lattice modules, PIJ and CITATION, have been used for modeling of the fuel rods, fuel assemblies and full core. The main Neutronic characteristics analyzed in this work include infinite multiplication factors (k_{inf}) versus burnup, the distribution of nuclide concentrations in the pin cells; the pin-wise power distribution in the assembly; the effective multiplication factors (k_{eff}), and the power distribution in the core.

Keywords: *SRAC, PIJ, CITATION, effective multiplication factor, power distribution, burnup.*

I. INTRODUCTION

Nuclear data libraries provide the data on cross sections and angular distributions of nuclear reactions with neutron from experimental research and theoretical calculations. The data are used to simulate neutronic characteristics in nuclear reactor physics. For this reason, the more accurate data are, the more accurate simulation results become.

This report is based on the OECD benchmark paper: "A VVER-1000 LEU and MOX Assembly Computational Benchmark. Nuclear Energy Agency, Organization for Economic Co-operation and Development"[1]. In this report, we present characteristics of the VVER Low Enriched Uranium (LEU) and Mixed Oxide (MOX) fuel assembly, where the calculations have been done using 3 libraries: ENDF/B 7.0, JENDL 3.2 and JENDL 3.3. The obtained results are compared to estimate the accuracy and usability of the data from the libraries with specific characteristics.

II. CALCULATION SPECIFICATIONS

In this report, we used the SRAC code with PIJ and Burnup modules to modeling LEU and MOX assemblies. The SRAC code which can be executed in UNIX and LINUX environments was developed by Japan Atomic Energy Agency (JAEA). The code consists of 107 energy groups with 74 fast groups and 48 thermal groups providing collision probability (PIJ) and resolving neutron diffusion and transport [2].

II.1. LEU and MOX assemblies

The VVER hexagonal fuel assemblies consist of 331 cylindrical rods. Fuel cladding and structural materials made by Zr-Nb composition. Fuel rods in the LEU assembly are classified into 4 types: 300 UO_2 rods with 3.7%w/t enrichment, 12 absorbed burnable rods (UO_2-Gd_2O_3-(UGD)) with 3.6%w/t UO_2 and 4.0%w/t Gd_2O_3, a water rod put in the center of the assembly and 18 guide tubes located at positions as shown in Figure 1.The fuel rods in MOX assembly are classified into 6 types: first layer consisting of 66 fuel rods with 2%w/t enrichment, second layer 96 fuel rods with 3%w/t, 138 fuel rods with 4.2%w/t in the center, 12 UGD rods, 1 water rod, 18 guide tubes in the same positions as in the LEU assembly.

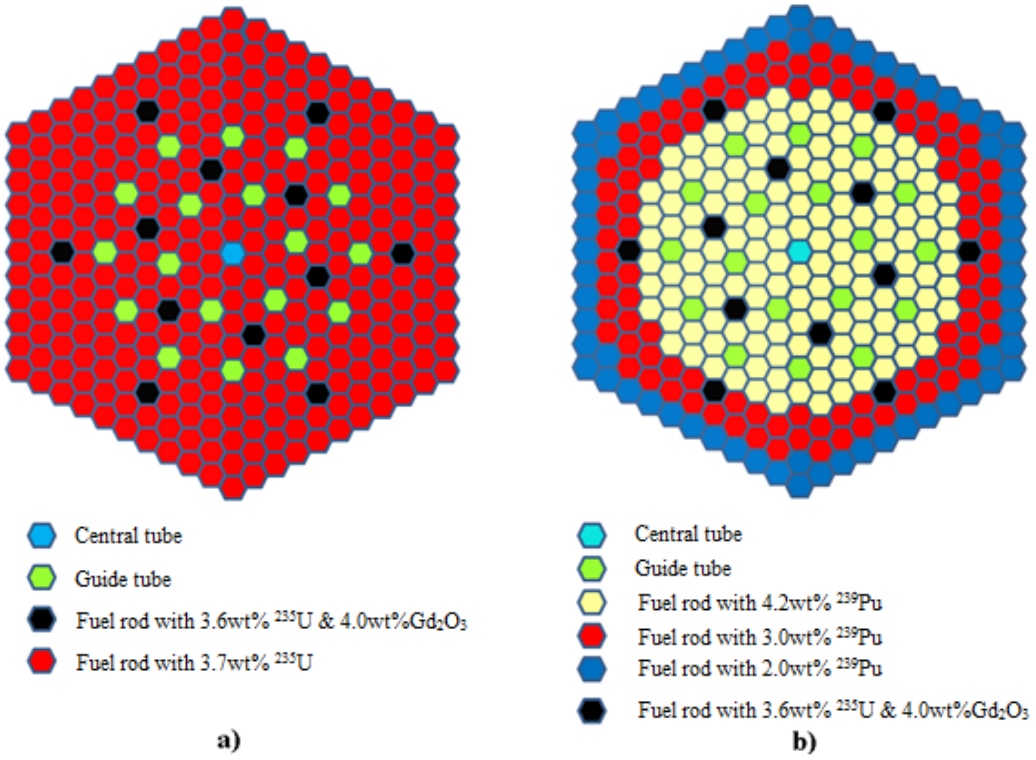

Figure1: LEU (left) and MOX (right) fuel assemblies.

II.2. State conditions

Table 1: State conditions.

State	Description	Fuel Temp. (K)	Moderator Temp. (K)	^{135}Xe and ^{149}Sm	Boron concentration in moderator (g/kg)
S1	Operating poisoned hot state	1027	575	Eq.	0.6
S2	Operating state	1027	575	0	0.6
S3	Isothermal hot state with Boron	575	575	0	0.6
S4	Isothermal hot state without Boron	575	575	0	0
S5	Cold state	300	300	0	0

State conditions are listed in Table 1.The physical characteristics of the LEU and MOX assemblies are calculated in the states: S1 is the operating poisoned hot state with ^{135}Xe and ^{149}Sm, S2 is the operating hot state, S3 is the isothermal hot state with Boron, S4 is the isothermal hot state without Boron, S5 is the cold state.

III. RESULTS AND DISCUSSIONS

III.1. Infinite multiplication factor versus burnup

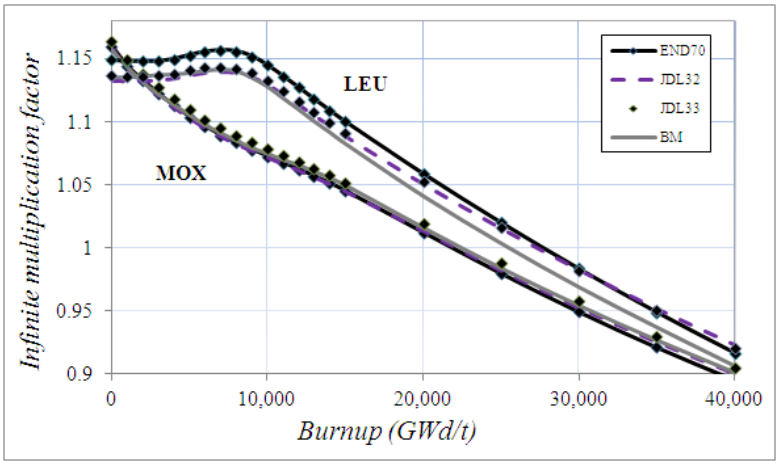

Figure 2: The infinite multiplication factor versus burnup.

Figure 2 shows the infinite multiplication factor (k_{inf}) of the LEU and MOX assemblies versus burnup evaluated from 0GWd/t to 40GWd/t. In the LEU assembly, we can see in the burnup range from 0 to 8GWd/t, k_{inf} increases lightly from 1.147 to 1.155. After that, k_{inf} decreases steadily from 1.155 to 0.9. In the MOX assembly, k_{inf} decreases from 1.159 to 0.895. The k_{inf} of MOX fuel is lower than that of the LEU. This is due to two reasons: (i) During the fuel burning ^{239}Pu in the MOX produces more neutron absorbers than ^{235}U in the LEU, and (ii) ^{238}U in the LEU fuel can be converted into fissile material ^{239}Pu. At the beginning of cycle, because MOX fuel produces more ^{135}Xe and ^{149}Sm than LEU fuel, the contribution of UGD rods in MOX assembly is not much as in LEU assembly, so gradient of the k_{inf} curve in MOX assembly is not high as in LEU assembly.

In Figure 2, the grey BM straight line is the benchmark mean value of k_{inf} and the other ones are k_{inf} obtained based on three nuclear data libraries mentioned above. For the MOX fuel the k_{inf} values are very close to the benchmark mean value. For the LEU assembly, the library ENDF 7.0 gives the k_{inf} value deviated 1.5% from the benchmark one, the JENDL 3.2 and JENDL 3.3give k_{inf} close to the benchmark in the range from 0GWd/t to 15GWd/t with a small deviation of 0.2%. However, at the k_{inf} values greater than 15GWd/t the deviation increases up to 1.5%. It can be concluded that the deviation between the k_{inf} coefficients obtained from the LEU and Benchmark is greater than that obtained from the MOX and Benchmark, especially at the high burn-up values.

III.2. Nuclear densities versus burnup

Figure 3 shows nuclear densities of ^{235}U and ^{239}Pu versus burnup, where we can see:

- The appearance of ^{239}Pu in LEU fuel is caused by fuel conversion in fuel-burning process and the appearance of ^{235}U in MOX fuel by contribution of UGD rods.

- In LEU assembly, the ^{235}U concentration at first burnup step is ~ 2.5×10^{-4}atoms/barn*cm and then decreases to ~ 5×10^{-5}atoms/barn*cm at the burnup 40GWd/t. The ^{239}Pu concentration increases dramatically at first burnup steps. This is the reason why k_{inf} curve for LEU fuel has one peak at 8GWd/t.

- In MOX fuel, the ^{235}U concentration decreases from ~ $2.07x10^{-5}$atoms/barn*cm at 0GWd/t to ~ $6.45x10^{-5}$atoms/barn*cm at 40GWd/t.

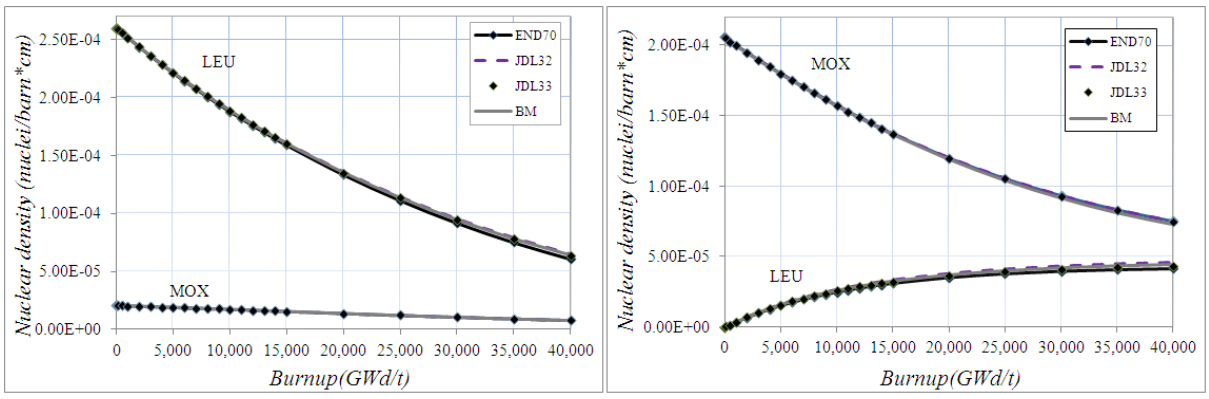

Figure 3: Nuclear densities of ^{235}U(left) and ^{239}Pu(right) versus burnup.

The ^{239}Pu concentration at beginning of the cycle is $2x10^{-4}$atoms/barn*cm and then decreases to ~ $7x10^{-5}$atoms/barn*cm at 40GWd/t. The ^{235}U and ^{239}Pu nuclear densities in LEU and MOX assemblies calculated by three libraries are very similar to the benchmark mean values.

Figure 4 shows the ^{135}Xe and ^{149}Sm concentrations versus burnup plot. As known, two these isotopes are reactor poisons, and the k_{inf} value is affected by their products. The amount of ^{135}Xe and ^{149}Sm increases rapidly at the first burnup steps. We can see, the ^{135}Xe and ^{149}Sm concentration in the MOX fuel is much higher than in the LEU fuel because the yields of ^{135}Xe and ^{149}Sm from thermal reaction of ^{239}Pu are higher than from ^{235}U.[3]

Table 2: Fission product yields (atoms per fission) from thermal fission*

Isotope	^{233}U	^{235}U	^{239}Pu
^{135}I	0.0475	0.0639	0.0604
^{135}Xe	0.0107	0.00237	0.0105
^{149}Pm	0.00795	0.01071	0.0121

*From M. E. Meek and B. F. Rider, "Compilation of Fission Product Yields," General Electric Company Report NEDO-12154, 1972.

In LEU fuel, the maximum ^{135}Xe concentration is~ $3.2x10^{-9}$atoms/barn*cm at 5GWd/t and then decreases to ~ $2.7x10^{-9}$atoms/barn*cm at 40GWd/t.

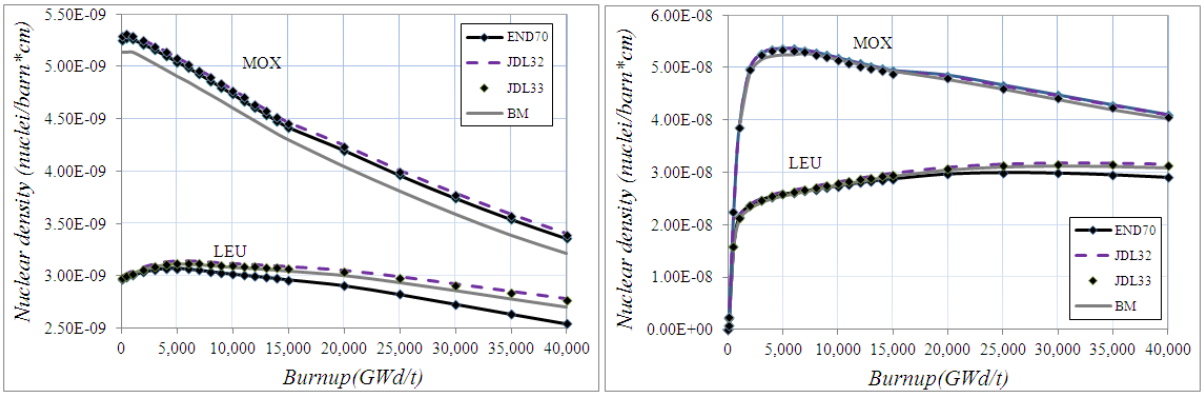

Figure 4: Poison density in fuel versus burnup for two cases: ^{135}Xe (left) and ^{149}Sm (right).

In MOX assembly, the maximum ^{135}Xe concentration is ~ 5.4x10^{-9}atoms/barn*cm at 1GWd/t and down to ~ 3.4x10^{-9}atoms/barn*cm at 40GWd/t. The accumulated amount of ^{149}Sm increases rapidly at first burnup steps and reaches a maximum value of~ 5x10^{-8} atoms/barn*cm at 5GWd/t. After that, ^{149}Sm density decreases to ~ 4x10^{-8}atoms/barn*cm at 40GWd/t.

From Figure 4 we can see that result on ^{135}Xe and ^{149}Sm concentrations with three libraries in LEU assembly are similar to the benchmark mean value at first burnup steps. However, the deviations increase at high burnup values. The calculations using the library ENDF 7.0 gives the most difference compared with the benchmark one, for example at burnup 40GWd/t the corresponding deviations for^{135}Xe and ^{149}Sm are 5.82% and 5.94%. In MOX fuel, results with 3 libraries for ^{135}Xe concentration are higher ~ 4% than benchmark mean value. The calculations using the library ENDF 7.0 gives the most difference compared with the benchmark one, for example at burnup 40GWd/t the corresponding deviations for ^{135}Xe 4.59% and for ^{149}Sm it's very close to the benchmark one.

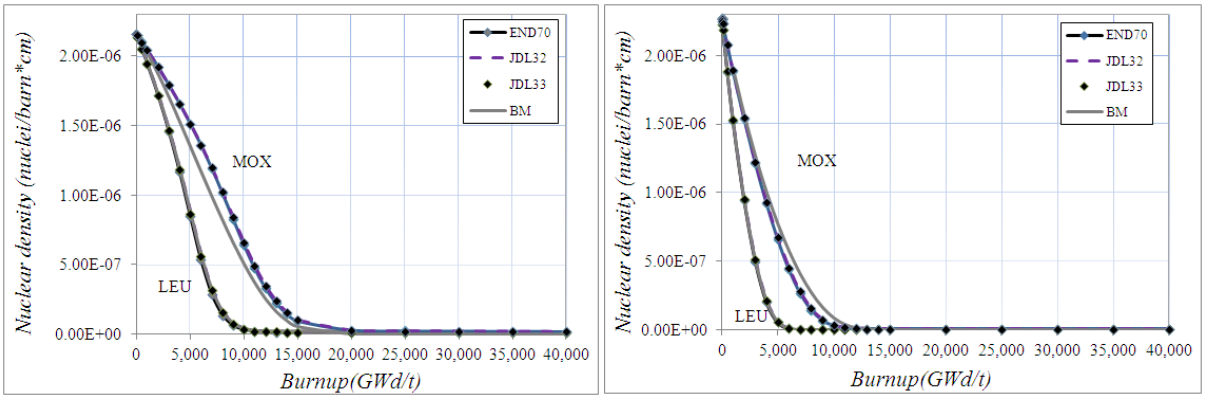

Figure 5: Nuclear density versus burnup for ^{155}Gd (left) and ^{157}Gd (right).

Figure 5 presents the ^{155}Gd and ^{157}Gd nuclear densities versus burnup plot. The role of Gd in fuel assemblies is to equalize reactivity at beginning of the cycle. In general, the Gd concentration decreases dramatically to 0 when the burnup increases from 0GWd/t to 10GWd/t. For LEU assembly the results for ^{155}Gd and ^{157}Gd concentrations with three libraries are similar to the benchmark one. For MOX assembly, the corresponding results for ^{155}Gd concentration are higher than benchmark one, where the ENDF 7.0 gives the greatest difference. For example, at 6GWd/t, the corresponding deviation is ~ 15.89%. The results for ^{157}Gd concentration are similar to benchmark one in the burnup range from 0GWd/t to 6GWd/t. At high burnup steps^{157}Gd concentrations are lower than benchmark, where the ENDF 7.0 gives the greatest difference, ~ 4.76% at 8GWd/t.

III.3. Reactivity coefficients

a. Effect on Boron in moderator

The absorption effect by Boron is obtained in comparison between their activities at Isothermal hot state with Boron (S3) and without Boron (S4). The $\Delta\rho$ value can be calculated by theformular: $\Delta\rho = \rho(S4) - \rho(S3) = \frac{1000x\,[k(S4) - k(S3)]}{k(S4)x\,k(S3)}$ (mk).The specific $\Delta\rho$ values in LEU and MOX assemblies at 0, 20, 40GWd/t are presented in Table 3.

Table 3: Effect of Boron in moderator.

Fuel Assembly	Burnup (GWd/t)	BM	EDF 7.0		JDL 3.2		JDL 3.3	
		Δρ	*Δρ*	*Deviation*	*Δρ*	*Deviation*	*Δρ*	*Deviation*
LEU	0	40.23	40.42	0.47%	40.31	0.21%	40.25	0.06%
	20	43.44	43.69	0.56%	43.34	0.23%	43.13	0.72%
	40	51.41	51.56	0.30%	51.00	0.80%	50.73	1.33%
MOX	0	23.20	23.10	0.42%	23.07	0.59%	22.79	1.77%
	20	32.54	32.13	1.25%	31.94	1.83%	31.60	2.88%
	40	42.64	41.45	2.79%	41.04	3.75%	40.71	4.54%

The BM in Table 2 is benchmark mean value, while the columns ENF 7.0, JDL 3.2 and JDL 3.3 show the results using libraries ENDF 7.0, JENDL 3.2 and JENDL 3.3.We can see in Isothermal hot state without Boron, $\Delta\rho$ values are higher than in Isothermal hot state with Boron. At high burnup steps, change in the reactivity coefficients is relatively big. This is because the Boron concentration in moderator is kept at 0.6g/kg while the fuel reactivity decreases.

The Δρ values for LEU assembly are more much higher than in MOX assembly. This is due to a fact that in the burning process the MOX fuel produces more neutron absorbers than LEU fuel. For this reason, the absorption effect by Boron in MOX assembly is not strong as in LEU assembly.

Compared with the benchmark mean value, the results given by ENDF 7.0 are in good consistence, where a minimum deviation is 0.3% and the maximum is 2.79%.The corresponding deviations are higher in cases of the JENDL 3.2 and JENDL 3.3, the maximum deviations are 3.75% and 4.54% for JENDL 3.2 and JENDL 3.3, respectively.

b. Effect on fuel temperature

Effect on fuel temperature is obtained by the reactivity comparison between the Operating state(S2) and Isothermal hot state(S3). The$\Delta\rho$ values can be calculated by theformular: $\Delta\rho = \rho(S2) - \rho(S3) = \frac{1000x\,[k(S2) - k(S3)]}{k(S2)x\,k(S3)}$ (mk).The specific $\Delta\rho$ values in LEU and MOX assemblies at 0, 20, 40GWd/t are presented in Table 4.

Table 4: Effect of fuel temperature.

Fuel Assembly	Burnup (GWd/t)	BM	EDF 7.0		JDL 3.2		JDL 3.3	
		Δρ	*Δρ*	*Deviation*	*Δρ*	*Deviation*	*Δρ*	*Deviation*
LEU	0	-9.86	-10.52	6.71%	-10.42	5.68%	-10.46	6.06%
	20	-12.87	-12.74	1.01%	-12.46	3.16%	-12.68	1.46%
	40	-15.63	-14.97	4.24%	-14.64	6.36%	-14.85	5.00%
MOX	0	-12.18	-11.80	3.10%	-11.68	4.11%	-11.74	3.58%

	20	-13.84	-13.76	0.58%	-13.65	1.38%	-13.68	1.18%
	40	-15.98	-15.37	3.81%	-15.16	5.11%	-15.19	4.93%

In Table 4, when fuel temperature increases from 575K to 1027K, $\Delta\rho$ decreases from 9.86mk at 0GWd/t to 15.63mk at 40GWd/t in LEU assembly and from 12.18mk at 0GWd/t to 15.98mk at 40GWd/t in MOX assembly. Compared with the benchmark the library ENDF 7.0 gives a greater deviation than two the other libraries. For the ENDF 7.0 the maximum deviation is 6.71% and minimum is 0.58%, while for the JENDL 3.2 and JENDL 3.3 the maximum deviations are 6.36 and 6.06%, and the minimum ones are 1.38 and 1.18%, respectively.

c. Effect on Isothermal hot state.

Effect on Isothermal hot state is obtained by the reactivity comparison between the Isothermal hot state without Boron (S4) and Cold State(S5). The $\Delta\rho$ values can be calculated by the formular: : $\Delta\rho = \rho(S4) - \rho(S5) = \frac{1000x\,[k(S4) - k(S5)]}{k(S4)x\,k(S5)}$ (mk). The specific $\Delta\rho$ in LEU and MOX assemblies at 0, 20, 40GWd/t are presented in Table 5.

When the fuel and moderator temperatures increase the negative reactivity is quite high. In LEU assembly the Δρ values increase from 41.73mk at 0GWd/t to 50.3mk at 40GWd/t. In MOX assembly the Δρ increase from 47.96mk at 0GWd/t to 52.73mk at 40GWd/t.

Table 5: Effect on Isothermal hot state.

Fuel Assembly	**Burnup (GWd/t)**	**BM**	**EDF 7.0**		**JDL 3.2**		**JDL 3.3**	
		Δρ	***Δρ***	***Deviation***	***Δρ***	***Deviation***	***Δρ***	***Deviation***
LEU	0	-41.73	-43.22	3.58%	-42.01	0.68%	-42.72	2.37%
	20	-47.92	-48.78	1.80%	-47.79	0.27%	-47.43	1.03%
	40	-50.3	-50.99	1.38%	-50.00	0.60%	-48.80	2.98%
MOX	0	-47.96	-48.51	1.14%	-48.62	1.38%	-47.37	1.23%
	20	-54.28	-54.44	0.30%	-54.00	0.51%	-52.34	3.58%
	40	-52.73	-54.49	3.33%	-53.45	1.36%	-51.40	2.54%

Library JENDL 3.2 gives the results closest to the benchmark mean value, its maximum and minimum deviation is, respectively, 1.38% and 0.27%. The ENDF 7.0 and JENDL 3.3 give the results with higher deviations, typically maximum deviation of 3.58%.

IV. CONCLUSION

In this report, we presented the calculation results on the infinite multiplication factor (k_{inf}), nuclear densities of ^{235}U, ^{239}Pu, ^{135}Xe, ^{149}Sm, ^{155}Gd, ^{157}Gd; reactivity coefficients ($\Delta\rho$) versus burnup for LEU and MOX assemblies using three nuclear data libraries ENDF 7.0, JENDL 3.2 and JENDL 3. All the obtained results were compared with the benchmark mean values.

The results on infinite multiplication factor, nuclear densities obtained from two libraries JENDL 3.2 and JENDL 3.3 are closer to the benchmark than library ENDF 7.0. However, the results on the reactivity coefficients by ENDF 7.0 is better than two the others libraries. Thus, we can conclude on the advantage and disadvantage of each data library used in analyzing the neutronic characteristics of VVER. As a result, the data selection depends on both physics and calculation code.

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