Study of graphite reactivity worth on well-defined cores assembled on LR-0 Reactor

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Abstract

Graphite is an often-used moderating material on the basis of its good moderating power and very low absorption cross section. This small absorption cross section permits the use of natural or low-enriched uranium in graphite moderated reactors. Graphite is now being considered as the moderator for Fluoride-salt-cooled High temperature Reactors (FHR). The critical moderator level was measured for various graphite block configurations in an experimental dry assembly of the LR-0 reactor. Comparisons with experiments were performed between Monte Carlo simulation tools for which satisfactory agreement was obtained with the exception of some systematic discrepancies. The larger discrepancies were observed when using the ENDF/B-VII.0 library. To decrease the uncertainties, based on conservative assumptions, relative comparisons were done. The results provided by the different nuclear data libraries are within 3 sigma interval of experimental uncertainties. It has been determined that differences between the results of calculations are caused by variations in the (n,n), (n,n'), (n,g) reactions and also by various angular distributions, while the (n,g) cross section variations play only a minor role for these configurations.

1 Introduction

The favorable moderating properties and chemical stability of graphite make it a strong candidate for generation IV reactor designs such as the High Temperature Graphite Reactors (HTGR) and the newly developed Fluoride-salt-cooled High temperature Reactors (FHR). However, the neutron interaction evaluations of graphite have recently been questioned and re-assessed as part of the Japanese HTGR program ([JENDL-4](#), [Goto et al 2011](#)). Consequently, there is strong interest in experiments which study nuclear systems involving graphite.

The intent of this paper is to present criticality measurements from the LR-0 reactor with various graphite configurations and compare these measurements with high-fidelity Monte Carlo simulations using various nuclear libraries. The LR-0 reactor is a light water reactor that achieves criticality by increasing the water moderator level. In this experiment, a dry channel containing graphite blocks is placed in the center portion of the core prior to critical level measurement.

These experiments are performed in the frame of The Memorandum of Understanding on nuclear power between Czech Ministry of Trade and Industry and U.S. DOE ([DOE 2013](#)). One of the studied geometries was previously performed within the scope of the SPHINX and EROS projects [Losa et al 2014](#), [Hron and Mikisek 2008](#).

2 Reactor arrangement

The measurements were performed in a special core assembled in the experimental zero-power reactor LR-0 (Research Centre Řež Ltd.). Reactor LR-0 is an experimental light-water-moderated zero-power reactor originally designed for research of VVER-1000 and VVER-440 type reactor cores, spent-fuel storage lattices, and benchmark experiments.

Power control in this core arrangement is achieved by changing the moderator level. The LR-0 fuel elements are radially identical to VVER-1000 fuel (with lattice pitch 12.75 mm), except axially shortened. The continuous maximal operating power is 1 kW with thermal neutron flux level of approximately $10^9 \text{ n cm}^{-2} \text{ s}^{-1}$. The LR-0 reactor is located in Řež near Prague (Czech Republic). Criticality was first reached in December 1982.

The reactor's main feature is that it enables experiments using a flexible support plate allowing arbitrary composition of assemblies or experimental dry assemblies into which experimental configurations can be inserted. For the experiments presented in this paper, the core contained six fuel assemblies of 3.3% enrichment (see Figure 1 and Figure 2). Slight enrichment variations of the as-fabricated assemblies are observed and were modeled using precise uranium inventory data.

Figure 1.: Radial cross-section of the core with specified enrichment, the distances are upright from pin

The various graphite configurations are placed into the centrally located experimental dry assembly. The channel is made of aluminum and the graphite blocks are positioned 2.9 cm below the bottom level of the fissile column. The reactivity worth effect of various graphite arrangements (see Figure 3) was tested by means of critical moderator water level measurement.

Figure 2.: Axial section of core with graphite filled experimental dry assembly (Case 8)

3 Experimental and calculation methods

3.1 Experimental arrangement

Figure 3 presents the various graphite configurations tested in the core. Cases 1-8 show the effect of an increase of graphite content which would allow to identify potential problems with the absorption cross section. Additional cases were modeled to verify symmetry and composition of the graphite blocks.

The graphite used in this experiment has a density of 1.72 g/cm^3 and impurities concentration below 0.2 ppm of boron equivalent. This meets the nuclear grade graphite limits [Bolewski et al 2005](#). For comparison, the graphite used in RBMK-1000 reactors has a density of 1.675 g/cm^3 and 6.8 ppm of boron equivalent [Pliukene et al 2011](#).

Figure 3.: Various configurations of the graphite insertion

The geometries with partial graphite loading required the use of spacers in order to assure accurate knowledge of the core geometry. The upper and lower spacers are made of phenolic paper locked in defined axial positions by an aluminum pin. Two types of spacer were used (see Figure 4). They differ by the bulk volume of phenolic paper. The bigger one with a measured positive reactivity worth of 87 ± 10 pcm is intended for cases 2 and 3 while the smaller one, intended for cases 4, 5, 6, 10, 11, and 12, has a small effect of 28 ± 9 pcm. To account for the spacer, the criticality eigenvalues from simulations were reduced by a correction factor that assumes a linear shielding effect. Nevertheless, this effect is small compared with other measurement uncertainties, especially with the small holder where the effect is about 15 times smaller than the related calculation-to-experiment uncertainties of about 200 pcm.

Figure 4.: Smaller spacer (left) with 3 graphite blocks and bigger spacer (right) with 2 graphite blocks

3.2 Critical height adjustment

Criticality is obtained by adjusting the moderator level (H_{cr}) for a given core map and enrichment of the fuel assemblies. When the moderator level is slightly above H_{cr} , reactivity may be expressed via a one-group (asymptotic) approximation as (1) (see [Szatmary et al 1985](#)).

$$\rho(H) = \frac{C}{(H_{cr} + \lambda_z)^2} \left[\frac{1}{\left(1 + \frac{H - H_{cr}}{H_{cr} + \lambda_z}\right)^2} - 1 \right]; \rho = \frac{k_{eff} - 1}{k_{eff}}; k_{eff} = \frac{k_{\infty}}{1 + M^2 B^2}; \quad (1)$$

where C is constant; $C = \frac{M^2 \pi^2}{k_{\infty}}$; B^2 geometry buckling, M^2 migration area and λ_z is the axial extrapolation length.

If reactivity is less than 25 ¢, a Taylor expansion of relation (2) around H_{cr} may be used. Reactivity for various moderator heights above the critical level was measured using the inverse kinetics method with time-dependent neutron counts. The digital reactimeter and data acquisition were implemented using an independent EWS computer system described in [Juricek 2009](#).

$$\rho(H) = f(H, a_1, a_2) = a_1 \cdot (H - a_2) \cdot \left(1 - \frac{3}{2} \frac{H - a_2}{a_2 + \lambda_z} \right) \quad (2)$$

where $a_1 = \frac{\delta \rho}{\delta H}$; $a_2 = H_{cr}$

The critical moderator level reactivity coefficient and associated standard deviations are determined by non-linear regression analysis of relation (2) or from its linear approximation, which does not include axial relaxation length.

Critical measurements of the various cores are presented in the next section. More details regarding the application of such a method can be found in [Kyncl et al 2008](#).

The tolerance of the measured H_{cr} for the LR-0 reactor, based on level meter manufacturer technical data, is 0.003 cm. The total uncertainty of H_{cr} at the 1σ level is determined from the tolerance of the level meter and its calibration. The uncertainty of the level meter calibration is determined by the precision at which the electrical needle is positioned. This is an electric contact, used for repeated level meter tests, which is fixed on the vessel wall at a height of 10.0 cm with uncertainty of 0.05 cm. The combined uncertainty of the critical water level value H_{cr} , which is the uncertainty of the needle level combined with uncertainty from statistical regression analysis is approximately 0.058 cm (see [Kyncl et al 2008](#)). This value is taken as the standard uncertainty for critical water-level height.

3.3 Calculation methods

The calculations were performed via the Monte Carlo method using the experimentally determined moderator height, using primarily the MCNP5 code [LANL 2003](#) and selected nuclear data libraries: [ENDF/B-VII](#), [JEFF-3.1](#), [JENDL-4](#), [JENDL-3.3](#), [ROSFOND-2010](#) and [CENDL-3.1](#).

The MCNP5 results were also compared with those performed in MCNPX 2.6.0 [Pellowitz 2007](#) and OpenMC [Romano and Forget 2013](#).

The estimated k_{eff} from moderator level measurements is based on the tolerances of the measured critical height H_{cr} for LR-0, on the moderator level meter's technical specifications (0.003 cm) and on electric contact uncertainty (0.05 cm) of the electric needle used for level meter calibration. The tolerances in other parameters were 0.5 mm in lattice pitch uncertainty, 0.0016cm in cladding thickness uncertainty, 0.016 g/cm³ in graphite density, 0.01 % in fuel enrichment and 0.0044 g/cm³ in fuel density uncertainty. The presence of impurities in the graphite will have only a minor impact (less than 10 pcm) on the outcome of the calculations due to their low concentration (0.2 ppm evaluated in boron equivalent) and the experimental graphite configuration inserted in the core. Consequently, the impurities present in the graphite are not simulated, but the uncertainty interval is broadened by the estimated effect.

These uncertainties were estimated using Monte Carlo simulations. All MCNP calculations employed the same number of histories per generation (10,000) for 5,000 active generations, skipping 50 generations for source convergence. Statistical uncertainties associated with the Monte Carlo calculations are approximately 10 pcm for both the perturbed and reference cases. The k_{eff} uncertainty for graphite density uncertainty has been evaluated by means of the perturbation card option, while all others has been evaluated using the parameter perturbation method and corresponds to a one-sided tolerance or uncertainty. A 3-sigma parameter perturbation was used in these calculations and scaling to one sigma uncertainty was applied (3).

$$U_i = \frac{|k_{\text{eff}}(x_i + \delta x_i) - k_{\text{eff}}(x_i)|}{\delta x_i} \times u_i \quad \delta x_i = 3.u_i \quad (3)$$

U_i standard k_{eff} uncertainty of the parameter,

u_i uncertainty in the parameter corresponding to one standard deviation,

δx parameter change used in the calculation of k_{eff} ,

The largest portions of the total uncertainty, which is about 200 pcm, are the uncertainty in cladding thickness (~130 pcm), lattice pitch (~130 pcm), and fuel density (~70 pcm). The uncertainty in fuel enrichment was suppressed by using more precise inventory data to about 70 pcm, about half the value of the previous evaluation (see [Kostal et al 2013](#)). This value, although notably smaller than previously, does not significantly decrease the total uncertainty because of other dominating effects. Assuming that all the uncertainties are non-correlated, the total uncertainty is the square root of the sum of the squares of all combined uncertainties. Details of the uncertainty estimation are listed in Table 1.

Table 1. Sources of uncertainty in pcm, calculated in ENDF/B VII.0.

Case	total uncertainty	lattice pitch	Water level uncertainty	Cladding thickness	Graphite impurities	Graphite density	Fuel enrichment	Fuel density
1	208	137	14	139			68	14
2	214	132	11	153	1	2	69	16
3	217	139	14	151	2	4	69	14
4	206	126	15	144	6	6	72	14
5	201	115	19	146	3	8	71	16
6	211	144	20	136	2	10	68	19
7	192	93	21	149	8	12	72	18
8	186	90	24	139	6	13	77	19
9	212	130	14	151	4	2	69	14
10	208	119	14	154	2	6	69	17
11	188	79	12	151	10	6	74	18
12	205	129	13	142	2	5	68	10

4 Results

Experimentally determined critical configurations, namely the moderator level, of the assembled core with various graphite configurations in the dry experimental assembly are listed in Table 2. It is obvious that with increasing the amount of graphite (cases 1-8), the moderator level decreases. Graphite decreases the moderator level by replacing air by a denser low absorption moderator that reflects neutrons back to the fuel assemblies, thus reducing leakage.

With 3 graphite blocks in the core (Cases 4, 10, 11, and 12), it can be observed that geometries with radial symmetry (Cases 10 and 11) produce measured critical heights that are lower than those in asymmetric geometries (Cases 4 and 12). This effect can be explained by better filling of the channel which results in lower axial leakage in the symmetric cases.

Table 2.: Measured critical height past the beginning of the fissile column

Case	H_{cr} [cm]	Uncertainty ²	$\delta\rho/\delta H$ [pcm/cm]	Standard deviation ¹	Graphite position
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1	55.65	0.05	257.36	8.08	-
2	53.34	0.05	294.85	5.30	G
3	51.08	0.05	254.99	9.68	F G
4	49.13	0.05	263.29	3.16	E F G
5	47.45	0.05	360.46	8.63	D E F G
6	45.84	0.05	362.20	7.50	C D E F G
7	44.43	0.05	368.65	11.72	B C D E F G
8	43.29	0.05	438.26	4.24	A B C D E F G
9	54.07	0.05	271.91	8.55	A
10	48.90	0.05	216.09	5.92	B D F
11	48.89	0.05	242.51	3.22	C E G
12	48.85	0.05	247.89	8.52	B C D

¹ Standard deviation from the reactivity non-linear regression analysis.

² The total uncertainty of critical moderator level at the 1σ level. It is determined by combining the tolerance of the level meter, the uncertainty of the fixed needle position and standard deviation from the reactivity non-linear regression analysis.

4.1 Comparison of experiments with MCNP5 simulations

The simulation results, with spacer correction, are listed in Table 3 and presented graphically in Figure 5. It is worth noting that all results are within the uncertainty range. One interesting observation is that ENDF/B-VII.0 overestimates while JENDL-3.3 underestimates all results. The mean average deviation in both cases is about 150 pcm. Note that the differences between each library are relatively independent of the graphite geometry. This suggests that the differences between libraries are primarily due to the description of the driver. The best agreement was obtained using JEFF-3.1 data with the mean average deviation of about 30 pcm.

It is worth noting that similar rates of underestimation with regard to ENDF/B-VII.0 data were reported in previous works. In figure 6, the first case [Kostal et al 2014](#) is a special triangular core moderated by water with average enrichment of 2.07%; the second case is a VVER-1000 mock up [Kostal et al 2013](#) which is moderated by water with diluted boric acid (2.68 g/kg) and an average enrichment of 2.27%.

Table 3.: Difference between calculation from MCNP5 using various libraries and experiment (expressed in pcm)

Case	ENDF/B-VII.0	JEFF-3.1	JENDL-3.3	JENDL-4	ROSFOND-2009	CENDL-3.1
1	183	50	-92	138	77	53
2	173	37	-126	106	58	20
3	158	4	-116	72	32	6
4	134	2	-122	90	21	-18
5	138	13	-168	71	41	6
6	111	-5	-163	66	35	2
7	168	35	-129	81	31	42
8	134	-10	-141	56	22	-2
9	200	58	-67	151	63	64
10	99	-32	-179	19	-2	-30
11	120	-29	-197	59	-11	-56
12	112	-16	-172	35	-18	-11

Figure 5.: Comparison of rate of agreement for various libraries. Experimental uncertainty is depicted by the gray envelope

Figure 6. Selected library underestimation with regard to ENDF/B-VII.0 for various cores

4.2 Comparison of experiments with various Monte Carlo codes

The calculations were carried out with various nuclear data libraries, but also with various codes, namely MCNP5, MCNPX 2.6.0 ([Pellowitz 2007](#)) and OpenMC ([Romano and Forget 2013](#)).

MCNP5 and MCNPX 2.6.0 were compared using every library previously mentioned. In most cases, MCNP5 overvalues the MCNPX 2.6.0 results. However, the overestimation rate is small with a mean average deviation of about 30 pcm with uncertainty of about 15 pcm. A graphical comparison is plotted in Figure 7.

MCNP5 and OpenMC were compared using only the ENDF/B-VII.0 library. In this case, a systematic underestimation or overestimation cannot be observed. The mean average deviation is about 20 pcm. The graphical comparison is plotted in Figure 8.

Figure 7.: Comparison between MCNP5 and MCNPX results. Statistical uncertainty is depicted by the gray bars

Figure 8.: Comparison between MCNP5 and OpenMC. Statistical uncertainty is depicted by the gray bars

5 Comparison of graphite evaluations for unique fuel library

The increasing amount of graphite in the cores translates to an increase in graphite scattering rate (see Figure 9). This increase is compensated by a decrease in hydrogen scatter rate (see Figure 10).

The increase in graphite scattering rate is predicted similarly by all libraries. As for the decrease in hydrogen scattering rate, slight deviations are observed from the JEFF-3.1 library and the CENDL-3.1 library.

Variations can be observed in the graphite capture rates (see Figure 11). JENDL-4 gives the highest values, and CENDL-3.1 the lowest. It is worth noting that the amount of graphite is relatively low, thus the total capture in graphite is small.

Figure 9.: Scatter rate of graphite in various data libraries

Figure 10.: Scatter rate of hydrogen in moderator and reflector in various data libraries

Figure 11.: Capture rate of graphite in various data libraries

The results presented in the previous sections have relatively high uncertainties and therefore they are not suitable to assess the accuracy of the graphite absorption cross section. The experimental uncertainties presented in Table 1 are very conservative, and most of the effects are systematic. One way to reduce the uncertainties is to evaluate the cases relative to the void channel as a reference case. Assuming an axially constant contribution from the cladding thickness, fuel enrichment, and fuel density, the systematic effect of the mentioned uncertainties are canceled. Note that this relative evaluation does not suppress the effect of the lattice pitch uncertainty. The pitch uncertainty is quite difficult to measure, and the value given in Table 1 comes from a very conservative analysis. In order to find a more reliable measure of the lattice pitch uncertainty, experiments were performed by reloading the cores independently five times each for three different cases. The uncertainty is found from the variance of the experimental results. This experimentally determined uncertainty in core geometry will cover both effects of lattice pitch uncertainty and water level uncertainty. This experiment was realized for cases 1, 7, and 8 because they do not need a spacer which can also introduce some discrepancy. The results are listed in Table 4. It is obvious from the presented data that the experimental uncertainty is notably smaller than the theoretically estimated one. After using the relative evaluations and the experimentally measured geometric uncertainty, the only values from Table 1 that remain to be taken into account are the uncertainties in graphite density and graphite purity. However, this uncertainty is quite low and does not exceed 14 pcm.

Table 4.: Average critical levels from 5 independent experiments per case

Case	Mean value [cm]	Standard deviation [pcm]	Maximal deviation [pcm]
1	55.599	8	20
7	44.381	16	41

The experimentally determined moderator levels from the repeated experiments (see Table 4) were used for numerical determination of k_{eff} . As mentioned previously, the results are relative to the dry void experimental channel (Case 1). In the presented results there are discrepancies between ROSFOND-2010 and ENDF/B-VII (see Table 5) even though the graphite cross sections are the same (see Table 8). So again, it seems that the differences in calculated reactivity are due to differences in the driver fuel cross sections. In order to suppress the effect of fuel description variations, models were made using one constant library for the fuel while varying the library used for the graphite. The JEFF-3.1 library was used for the fuel as it shows the best agreement in the reference case with a void dry experimental channel (Case 1). The results of this calculation are listed in Table 6.

Table 5.: Bias between selected cases with graphite and void channel (Case 1)

Case	ENDF/B-VII.0	JEFF-3.1	JENDL-3.3	JENDL-4	ROSFOND-2010	CENDL-3.1
7	-38	-33	-42	-69	-39	-25
8	-58	-53	-72	-93	-77	-61
Uncertainties						
7	26	24	25	25	25	24
8	32	30	31	31	31	31

Table 6.: Bias between selected graphite cases and void channel (Case 1) using JEFF-3.1 for driver core description

Case	JEFF-3.1	ENDF/B-VII.0	JENDL-3.3	JENDL-4	ROSFOND-2010	CENDL-3.1	UNC
7	-33	-27	-41	-58	-35	-11	24
8	-53	-53	-52	-88	-52	-25	30

Table 7.: Sensitivity to variation in selected reaction cross section calculated in ENDF/B-VII.0 library in [pcm]

Case	10% perturbation		2% perturbation				
	$^{6}\text{C} (n,n')$	$^{6}\text{C} (n,g)$	$^{235}\text{U} (n,f)$	$^{235}\text{U} (n,g)$	$^{238}\text{U} (n,f)$	$^{238}\text{U} (n,g)$	$^1\text{H} (n,g)$
1	-	-	655	-244	156	-413	-100
2	19	-1	653	-244	156	-412	-152
3	41	-1	654	-244	156	-412	-99
4	65	-2	654	-244	155	-412	-99
5	85	-2	654	-245	155	-412	-100
6	108	-3	655	-245	155	-412	-100
7	127	-4	655	-245	155	-412	-100
8	145	-5	654	-244	155	-412	-154
9	18	0	655	-244	156	-413	-100
10	66	-2	654	-244	155	-412	-100
11	64	-2	654	-245	155	-412	-100
12	61	-2	654	-244	155	-412	-99

The results show that all data libraries underestimate the criticality for cases with graphite with regard to the reference geometry without graphite (case 1). The rate of underestimation between full graphite geometry with regard to the void case is about 53 pcm for most libraries, but in the case of JENDL-4 the rate of underestimation is 88 pcm, and in the case of CENDL-3.1 the rate of underestimation is 25 pcm. It is worth noting that these deviations near the related uncertainties of about 30 pcm for this case. This JENDL-4 to ENDF/B-VII.0 underestimation might be caused by both variations in (n,g), (n,n), (n,n') reaction cross sections and also by a slight change in the scattered neutron angular distribution probability (see Figure 12). ENDF/B-VII.0 has a slightly higher elastic cross section and also a more isotropic distribution than the JENDL-4. The forward preference, however slight, causes neutrons to leave the graphite faster. Therefore, the epithermal neutron flux in graphite in JENDL-4 is lower on account of the driver core flux. When the driver core epithermal neutron flux is higher, the neutron capture rate in hydrogen is higher while the fuel capture rate is smaller. This reflects in lower fission neutron production and finally lower k_{eff} .

CENDL-3.1 has the same angular distribution as ENDF/B-VII.0, but a more dramatic change in the scatter cross section is present (see Figure 13). This higher cross section increases the flux in graphite and also its moderating properties. This slightly reduces capture in hydrogen followed by higher fuel absorption. The overall effect is a higher fission rate and finally higher k_{eff} .

The effects of (n,n), (n,n') on k_{eff} can only be observed in this type of core geometry where the diffusion length is larger than the dimensions of the central graphite block.

In contrast to the presented results, calculations (Díez et al 2013) for the BR-1 reactor show a much more notable effect of (n,g) on the k_{eff} . The BR-1 reactor located at the Belgian Nuclear Research Centre SCK•CEN in Mol (Belgium) (see Wagemans et al. 2009) is an air-cooled reactor with graphite moderator. Since this reactor uses graphite as both reflector and moderator, the graphite / fuel weight ratio is very high. It is obvious that in this case the sensitivity to the capture in carbon plays a bigger role than in the presented experiments where the insertion acts as additional moderator and central reflector. Also, since the diffusion length is larger than the dimensions of the central graphite block, the absorption probability of the entering neutrons is relatively small. The modifications in angular distribution, observable in the LR-0 experiments, are not observable in a graphite moderated and reflected core because the angular effects are suppressed by the large graphite dimensions.

The similar comparison between JENDL-4 and ENDF/B-VII.0 results can be found in (Goto et al 2011). The presented results of the HTTR calculation show also a trend of underestimation, however higher, of JENDL-4 results with regard to ENDF/B-VII.0 or JEFF-3.1. The reason is a more notable effect of (n,g) on the k_{eff} as reported in (Díez et al 2013).

Table 8.: Comparison of weighted graphite cross sections in case 8

	(n,n) [b]	(n,n') [mb]	(n,g) [mb]
ENDF/B-VII.0	4.1528	1.645	0.699
JEFF-3.1	4.1528	1.645	0.699
JENDL-3.3	4.1527	1.707	0.737
JENDL-4	4.1527	1.707	0.805
ROSFOND-2010	4.1528	1.645	0.699
CENDL-3.1	4.1531	1.628	0.699

Figure 12.: Comparison of scattered neutron emission probability

Figure 13.: Comparison of CENDL-3.1 and ENDF/B-VII.0 graphite cross sections

6 Conclusions

The critical moderator level was measured for various graphite configurations in an experimental dry assembly in the center position of the LR-0 reactor. Generally, the agreement between calculation and experiment is satisfactory as all of the results are within one standard deviation. It is worth noting in this core arrangement that the ENDF/B-VII.0 results overestimate by about 100-180 pcm while the JENDL-3.3 results underestimate by about 70 – 170 pcm. Similar underestimations and overestimations are also observed in other reported experiments with comparable fuel enrichment.

It can be observed that increasing the amount of graphite in the central experimental channel decreases the critical moderator level. This is due to a decreased leakage rate and also increased moderation in the central experimental channel over the void case.

The evaluation of the full graphite channel relative to the void channel shows underestimation of reactivity by every library. In order to suppress the effect of fuel description variations, calculations were made using only JEFF-3.1 to describe the fuel and varying libraries to describe the graphite. These calculations found that the CENDL-3.1 results underestimate by about 25 pcm, ENDF/B-VII.0 by 53 pcm, and JENDL-4 by 88 pcm while related uncertainties are 30 pcm. The differences between the results are primarily due to the variations between the libraries in the (n,n) and (n,n') reactions while the (n,g) cross section variations play only a minor role for these configurations. In the case of the JENDL-4, angular distribution also differs and impacts the results.

The studied core geometries, especially those with a dry void central assembly, have relatively low moderator level reactivity coefficients, although higher than in the lower enriched core reported in [Kostal et al 2014](#), thus enabling measurement of effects with the low reactivity worth of ~20-30 pcm.

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