PATH FORWARD IN GEN-IV NEUTRONIC EXPERIMENTS AT LR-0 REACTOR

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ABSTRACT

Research at LR-0 reactor is focused on benchmark experiments related to reactors belonging into fourth generation. These experiments are aimed mainly at determination and benchmarking of criticality. Critical benchmarks are then submitted into international database which is operated under auspices of IRPhE (International Reactor Physics Experiment Evaluation) Project of OECD NEA databank. Experimental database broadening helps to validations of codes as well as nuclear data used for design and calculations of GEN-IV reactors. This paper summarizes the past and recent benchmark efforts applied to materials as graphite (Gas Cooled High Temperature Reactor), fluoride salts (Molten Salt Reactor), and lead (Lead Fast Reactor). Performed experiments are modelled in MCNP6.1 code with usage of different nuclear data libraries. Results of criticality measurements are then compared with calculations in form of C/E-1, where possible discrepancies become more apparent. Results show that the benchmark models accurately represent the experiments, where the most of the cases are in 1σ uncertainty interval. No apparent trend was found in results of calculations among experiments and calculations, but the models of fluorinated compounds are best represented by CENDL-3.1 data library.

1 Introduction

Reactor benchmarks at LR-0 reactor are aimed at research of neutronic parameters of advanced reactors comprising fast reactors cooled by lead (LFR), high temperature reactors (HTR), and reactors based on molten fluoride salts (MSR, FHR). Fluoride salt experiments consisted of methodological experiments with LiF-NaF (also referred to as FLINA) salt containing natural Li as well as of experiments with depleted ^{7}LiF-BeF$_2$ (also referred to as FLIBE) salt originally used in MSRE reactor operated in Oak Ridge. Measured quantities are critical parameter, fission densities and neutron spectra. Some of the experiments were peer reviewed by OECD NEA subgroup and included into IRPhEP database, which is used mainly by developers for code validation and for nuclear data library testing. Operational experience gained from repetition and standardization of experiments and the benchmark MCNP calculation model led to decision to validate the neutron spectrum of the LR-0 benchmark core with low enriched fuel as a reference, because in past, following quantities were measured or benchmarked and thus they are known: criticality [1], the neutron spectrum [2] or [3], and the fission densities [4]. When the neutron field in special irradiation cavity is established as reference, it can be then used, beside others, for spectral average cross section measurement (see **Error! Reference source not found.**), which is actual aim of the current NDS (Nuclear Data Services) project of IAEA in frame of improving of International Reactor Dosimetry Fission Fusion Files (IRDFF).

Advantage of the LR-0 reactor among other experimental facilities is the ability of precise measurement of critical parameter which is otherwise source of the large uncertainties. Uncertainties in benchmark cases, given by the engineering variations of structural materials parameters have systematic trend and are usually less than 180 pcm. Experimental values are compared with the calculations using different nuclear data libraries (ENDF/B-VII.1,

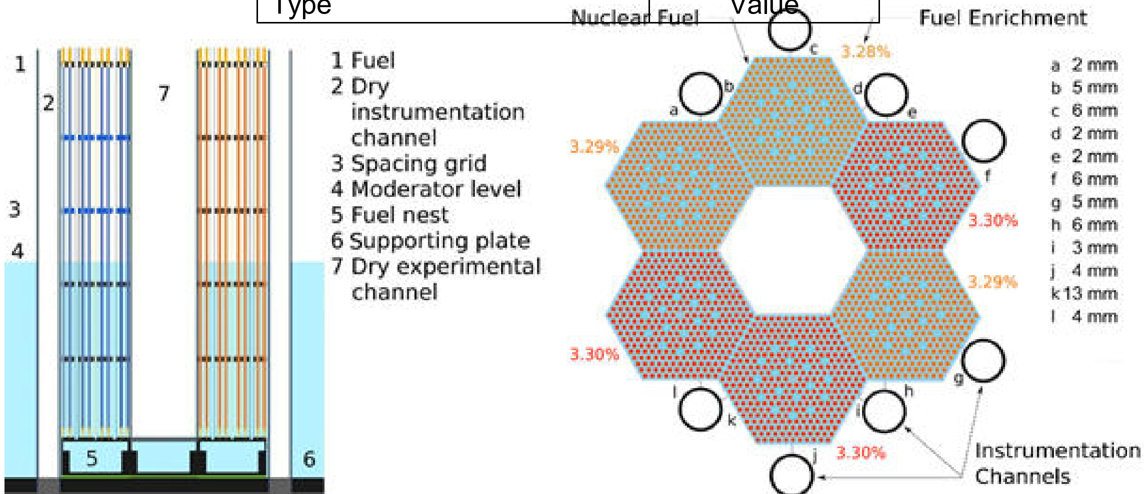
ENDF/B-VII.0, JEFF-3.2, JENDL-4, RUSFOND-2010, and CENDL-3.1). This article summarizes results of critical experiments at LR-0 reactor.

2 Materials and methods

2.1 Experiments

Discussed experiments are carried out at LR-0 reactor, which is light water pool type reactor, located in Research Centre Řež. The experimental work was carried out at room temperatures and pressures. The reactor core of benchmark experiments with triangular lattice consisted of six VVER-1000-like fuel assemblies with nominal enrichment of 3.3 % and dry experimental module located in the core centre (see Figure 1 and Figure 2). Experimental module has dimensions close to the fuel assembly pitch and thus its location in the core centre assures almost homogeneous neutron flux of fast neutrons in the module. This module can either be kept as void in the core, or can be filled by reactor structural materials to be subjected to neutronic investigation. The fission column height is 126 cm at maximum.

In the six-assembly benchmark core, the criticality as well as power changes are driven by moderator level changes. The uncertainty in critical parameter determination is equal to the uncertainty of water level measurement. Absolute uncertainty in water level determination (combines all factors influencing the measurement – calibration, mechanical tolerances) is 0.5 mm. Other sources uncertainties are given by the manufacturing tolerances and are discussed in Table 1, Sources of uncertainties attributed to benchmark experiments



Type	Value
Fuel cladding thickness	0.0016 cm
Fuel density	0.0093 g/cm^3
Fuel enrichment	0.01 % variation
^{234}U content	0.0309%
Fuel lattice pitch	0.15 cm
Moderator level	0.05 cm
Moderator contamination	2 ppm (BE)
Insertion density	1 %
Insertion outer diameter (OD)	0.05
Insertion inner diameter (ID)	0.05
^{6}Li contamination in LiF-BeF_2	100 ppm

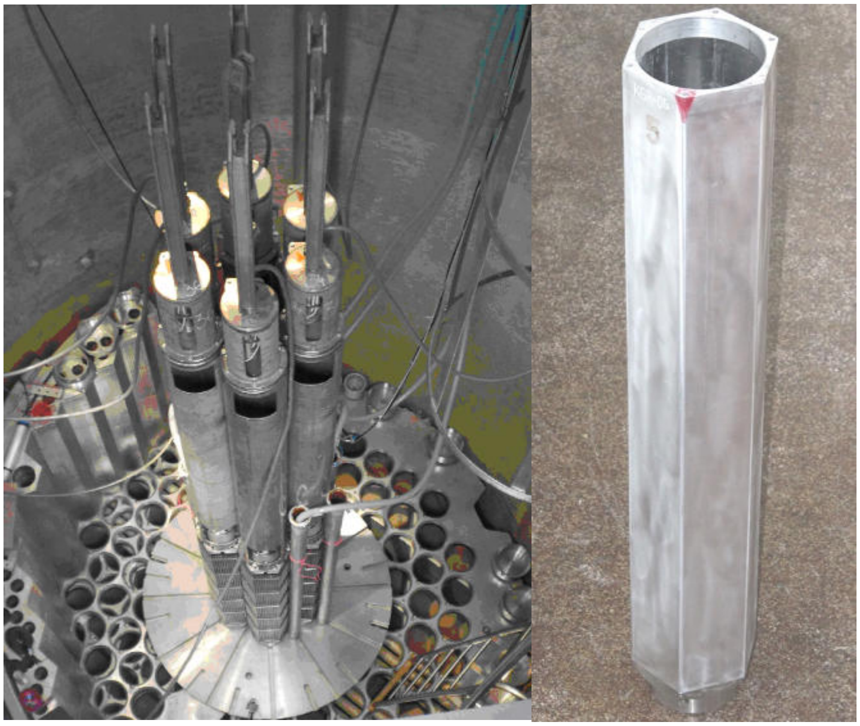


Figure 2, Sight into the reactor core (left) and photograph of the dry irradiation module (right).
Figure 1, Reactor core scheme with empty central channel (axial plane on the left, radial plane on the right)

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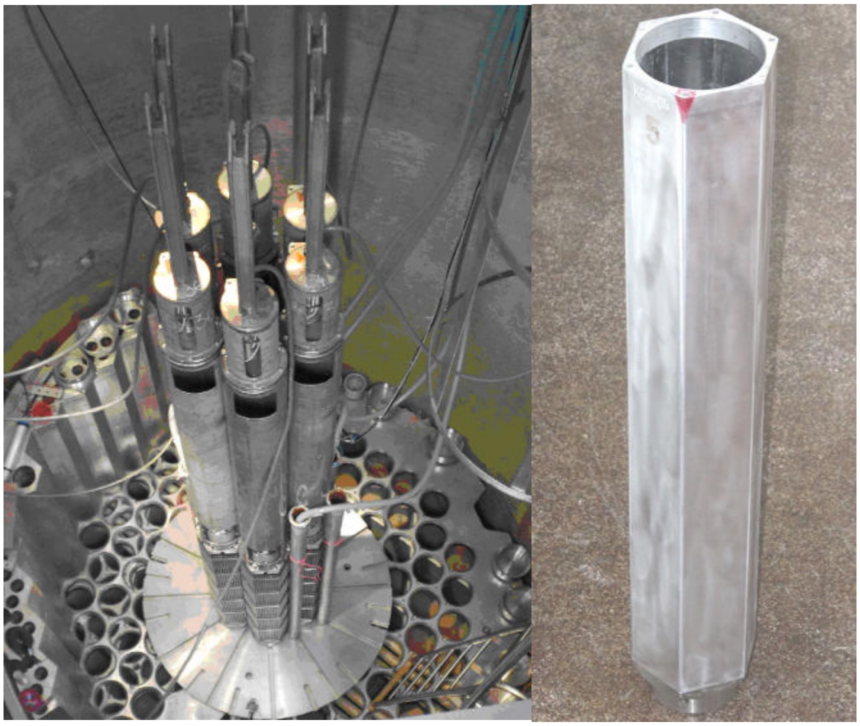

Figure 2, Sight into the reactor core (left) and photograph of the dry irradiation module (right)

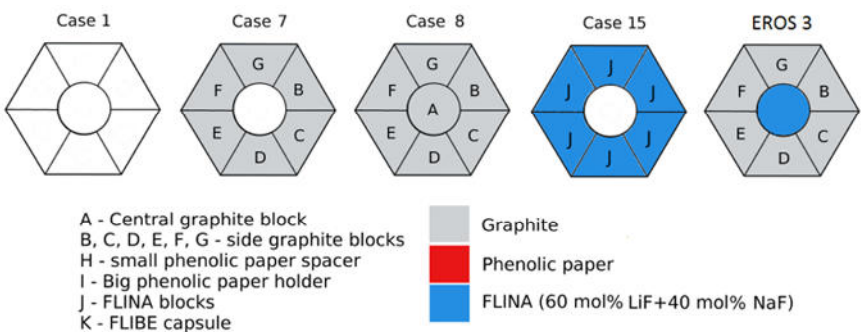

Figure 3, Scheme of used insertions – combinations of graphite and LiF-NaF modules

Figure 4, Photographs of used insertions (from left Teflon, canister with LiF-BeF2, canister with lead)

The amount of inserted materials is limited only by the dry module volume. Some of the insertions can have form of segments (see Figure 3), or can form compact cylinder or cylindrical vessels containing investigated materials (see Figure 4). Material insertion is usually positioned into the module in that way that the beginning of the fission column is equal or bellow beginning of the insertion (bottom of the module), the height of the insertion is in most cases 60 cm.

Experiments with graphite and fluoride LiF-NaF salt involved segmental insertions, to maximize the filling of the experimental module. Segments of graphite are not cladded, segments with LiF NaF salt have form of aluminium canisters filled with fluoride salt. Graphite segments and segments with LiF-NaF salt can be combined to simulate the neutronics in the salt channels of MSR. Experimental work with insertion in form of solid Teflon cylinder was carried out for investigation of potential impact of fluorine cross section in advanced reactors containing large amounts of fluoride salts, thus forming the major component of the reactor core. In this paper result of Teflon criticality is presented just for comparison with other results. Lead and LiF-BeF_2 salt are inserted in sealed cylindrical vessels (canisters) to facilitate the manipulation, as both substances are toxic. Dimensions and materials of both canisters are optimized, as both insertions have different physical and chemical properties, and thus slightly different dimensions and types of steels are used. The canister volume for this type of insertions is approximately 15 dm^3. Influence of the empty canister on neutronics was studied in separate experiments.

2.2 Calculations

Criticality calculations were carried out by MCNP6.1 [6] code with benchmark models compiled from knowledge about the physical composition of the material in the dry module and measured critical level. Calculations involved different nuclear data libraries (ENDF/B-VII.1, ENDF/B-VII.0, JEFF-3.2, JENDL-4, RUSFOND-2010, and CENDL-3.1) and were produced with statistical deviations lower than 10 pcm. Graphite insertion was simulated with utilization of S(a,b) matrix form ENDF/B-VII.0, other insertions were not subjected to special thermal scattering treatment. The same settings were used for quantification of uncertainties, which were estimated by parameter perturbations method.

Table 2, Estimation of experimental uncertainties by calculation (in pcm)

Parameter	Insertion type								
	Case 1	Case 7	Case 8	Case 15	EROS 3	Teflon	FLIBE	Lead	Canister
Fuel cladding	78	72	73	71	81	99	32	81	82
Fuel density	5	7	4	5	6	10	25	8	8
Fuel enrichment	36	36	36	38	38	44	57	39	38
^{234}U content	119	117	119	103	112	128	65	117	110
Lattice pitch	123	108	102	75	44	102	68	46	98
Moderator level	8	11	11	3	3	12	24	6	7
Moderator contamination	-	-	-	-	36	37	13	34	39
Insertion density	-	13	13	5	14	12	22	4	2
Insertion OD	-	-	-	-	-	16	-	-	-
Insertion ID	-	-	-	-	-	17	-	-	-
Insertion contamination	-	0	0	8	8	-	38	29	-
Total	**192**	**180**	**178**	**153**	**155**	**202**	**128**	**161**	**177**

All uncertainties are considered to have rectangular distribution and to be independent and not correlated. Therefore, the total uncertainty can be calculated as a square root of the sum of squares of single uncertainties. The uncertainty estimation was carried out only in ENDF/B-VII.0 library. The highest contributors to the total uncertainty are the values connected with fuel definition: lattice pitch, ^{234}U content, fuel cladding thickness, and enrichment. Total uncertainties at different experiments are almost always lower than 200 pcm, only in case of Teflon insertion, the total uncertainty reaches 202 pcm.

3 Results

Empty module (Case 1) can be considered as a reference, because the criticality of the benchmark core is not influenced by any inserted material, which can be source of other potential uncertainties. This configuration also helps to assess the insertions according to their properties; they can cause either neutron absorptions, leading to losses, or neutron in-scattering, improving the balance in the core. Thus, the experimental critical water level (see Table 3) shows the reactivity impact of given types of insertions.

Graphite insertions obviously cause substantial moderator level decrease. LiF-NaF salt insertion, containing naturally occurring isotopic mixture of Li, on the other hand, excessively increases the moderator level over 80 cm. EROS 3 experiment is the repetition of the older experiment simulating neutronics of graphite channel filled by fluoride salt. The layer of graphite is sufficiently thick to partly supress the absorbing properties of the LiF-NaF salt. Teflon insertion serves as simulator of the medium containing graphite and fluorine. Results in this case give surprisingly good agreement with calculations if assumed that the CF_2 bound molecules are considered as free gas in calculations. Empty canister, being the piece of stainless steel in the centre of the core has rather absorbing properties. In case of filling by GEN IV coolant material, lead or LiF-BeF_2 salt, the neutron balance improves slightly in comparison with reference.

Table 3, Results of critical water level measurement

Case	Critical level [cm]
Case 1	55.60
Case 7	44.38
Case 8	43.22
Case 15	80.26
EROS 3	49.36
Teflon	46.42
FLIBE	54.40
Lead	54.49
Canister	58.96

Figure 5 shows graphical comparison of calculations and experiments in pcm as C/E-1. It can be seen that results for simple materials (void, graphite, Teflon, canister) fit into the 1σ uncertainty interval (red lines). Interestingly, slightly decreasing trend can be observed for C/E-1 in case of graphite addition in case of all investigated libraries, except JENDL-4. When the LiF-NaF salt is added into the dry module either for Case 15, or EROS 3, the calculation tends to underestimate the experiment, but the results are near to the borders of 1σ uncertainties.

Larger discrepancies can be observed in case of material insertions in canisters. In both cases, LiF-BeF_2 as well as lead, the calculation results are overestimating and are above the 1σ uncertainty interval. In none of the cases thermal scattering libraries are used.
Interesting fact is also the impact of absorbing materials to overall experimental uncertainties. It is observed that the uncertainties are systematically lower for materials with increased neutron absorption in thermal spectrum.
Concerning used nuclear data libraries no apparent trend is observed among experiments and all the calculation results seem to provide satisfactory figure about the experiments. However, it is worthwhile to note that in case of experiments with fluorinated compounds, the calculations in CENDL-3.1 library generally provides the closest results. Numerical values of calculations are summarized in Table 4.

Figure 5, Results of critical moderator level measurements and corresponding calculations in different nuclear data libraries

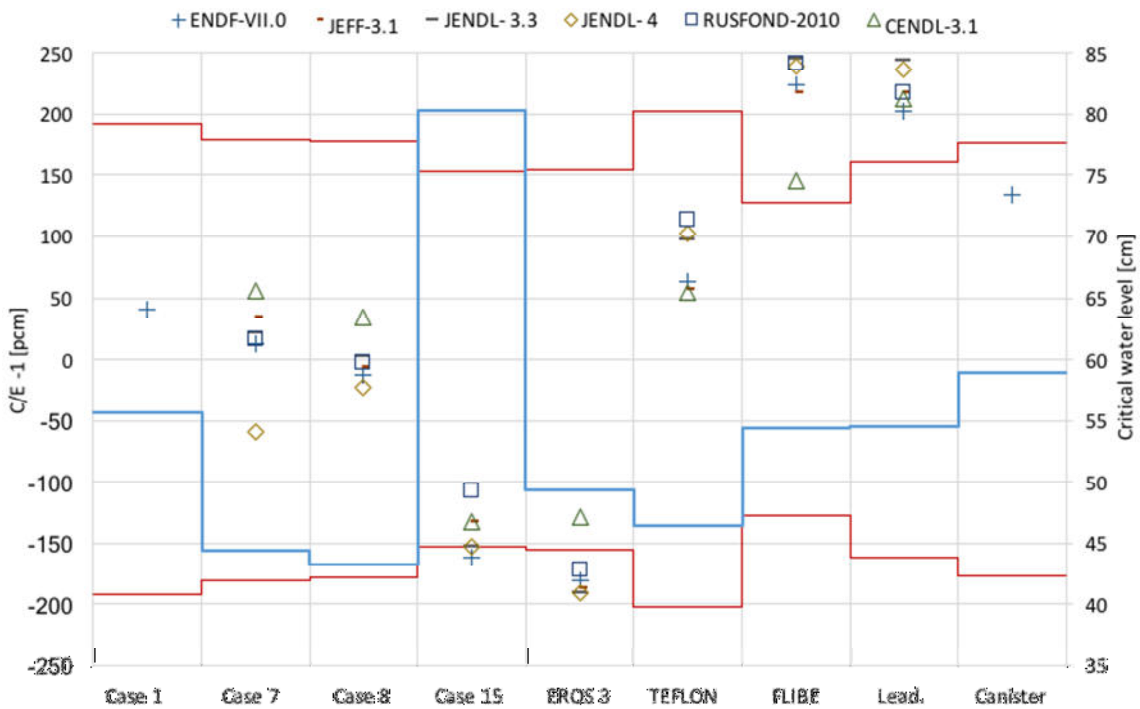

Table 4, Results of accompanying criticality calculations in different libraries

	Tot. uncert.	ENDF/B-VII.0	JEFF-3.1	JENDL-3.3	JENDL-4	RUSFOND-2010	CENDL-3.1
Case 1	192	41	-	-	-	-	-
Case 7	180	13	34	22	-59	17	56
Case 8	178	-13	-6	3	-22	-2	34
Case 15	153	-161	-133	-153	-153	-107	-133
EROS3	155	-181	-187	-191	-190	-171	-129
Teflon	202	64	57	98	103	114	55
FLIBE	128	224	217	245	239	241	146
Lead	161	203	217	244	237	217	212
Canister	177	134	-	-	-	-	-

4 Conclusions

Presented benchmark experiments are demonstrating the effort in validation of standard nuclear data libraries for usage in calculations of advanced nuclear reactors. Results show that the investigated libraries can equally be used for these calculations because no apparent "best recommendation" was found. Good agreement of experiments and calculations is supported by the fact that the majority of results are within 1σ uncertainty interval. Models indicate that the CENDL-3.1 library provides slightly better results in case of fluorinated compounds.

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6 References

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