Development of reference benchmark neutron field in LR-0 reactor

HABILITATION Thesis

(Collection of articles)

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March 2022

Rez, Czech Republic

Abstract

A neutron reference field is defined as a permanent reproducible neutron field with well defined neutron fluence rate and neutron energy spectra. The standard reference field has neutron spectra characterized employing neutron spectra measurements by means of time-of-flight measurements. Historically there was only one standard neutron reference field, ^{252}Cf(s.f.). In 2017 into neutron standards, there was also included ^{235}U(n_th, fiss), which becomes secondary neutron standard.

Reference neutron benchmark fields are defined as permanent and reproducible neutron fields, less well characterized as a standard field, but still acceptable as a measurement reference by a community of users. This work deals with the development of a reference neutron benchmark field in a special core placed in an LR-0 reactor. In this reference neutron benchmark field, there were precisely measured neutron and gamma spectra using stilbene spectrometry. This is very important in the case of this reactor benchmark field because it was observed that neutron spectra above 6 MeV are nearly identical with ^{235}U PFNS, and also it was confirmed that the gamma has a negligible effect on measured neutronic quantities. Namely, the impact of the photo-nuclear reactions (γ,n) competing with (n,2n) in the production of the same residual nucleus was shown to be negligible. To confirm the spatial distribution of the field, there was also validated the fission density distribution across the driver core.

This precise characterization of the core is important and well usable also in other issues. It's worth noting it is an excellent tool for validation of neutronic properties of materials inserted in the center of the special core, or surrounding it, for example, sand, whose neutronic parameters are important not only in space programs but also in used fuel storage management, because major material of earth crust is SiO_2. The most recent also interesting application was validation of neutronic description of stainless steel, which is important in criticality safety issues as stainless steel is the major component of water moderated reactors internals.

The essential results, namely criticality, flux distribution, neutron spectra and also some of the measured spectral averaged cross section evaluated as SACS averaged in ^{235}U PFNS were benchmarked in the IRPhEP database. This database is the most reliable one, which is used in improvements of nuclear cross sections. The field became one of the IRDFF-II reference benchmark neutron fields, and the measured data were used for improving the newly developed IRDFF-II neutron dosimetry library.

Keywords: Cross section, Neutron dosimetry, Neutron spectrometry, Validation of cross sections

Abstrakt

Referenční pole neutronů je definováno jako permanentní reprodukovatelné pole neutronů s dobře definovanou distribucí hustoty toku neutronů a jejich energetickým spektrem. Standardní referenční pole předpokládá nezávislé měření neutronového spektra prostřednictvím metody TOF. Historicky bylo definováno pouze jedno standardní neutronové referenční pole, konkrétně pole ^{252}Cf(s.f.), přičemž v roce 2017 bylo zahrnuto také ^{235}U(n_th, fiss), které se takto stalo sekundárním neutronovým standardem. Referenční benchmarková neutronová pole jsou definována jako stálá a reprodukovatelná neutronová pole, méně dobře charakterizovaná jako standardní pole, ale stále ještě přijatelná pro referenční měření komunitou expertních uživatelů a hodnotitelů jaderných dat. Tato práce se zabývá vývojem a sestavením referenčního pole neutronů ve speciální aktivní zóně umístěné v reaktoru LR-0. V tomto referenčním poli neutronů byla precizně charakterizována neutronová a gama spektra pomocí stilbenové spektrometrie. Tento studovaný případ referenčního pole je výjimečný mezi reaktorovými poli, protože v případě tohoto benchmarkového pole, bylo demonstrováno, že neutronová spektra nad 6 MeV jsou téměř totožná s ^{235}U PFNS. Zároveň se potvrdilo, že gama má v tomto referenčním poli zanedbatelný vliv na měřené neutronické veličiny. Konkrétně vliv fotojaderných reakcí (γ,n) konkurujících (n,2n) při produkci stejného produkovaného jádra se ukázal jako zanedbatelný. Pro další ověření bylo rovněž provedeno detailní mapování hustoty štěpení ve speciálním aktivní zóně. Tato přesná charakteristika aktivní zóny je velmi dobře využitelná i v dalších oblastech neutronové fyziky. Je to vskutku vynikající nástroj pro validaci neutronických vlastností materiálů vložených buď do středu speciální aktivní zóny nebo ji obklopující. Toto bylo využito například v případě oxidu křemičitého, jehož neutronové parametry jsou důležité nejen ve vesmírném programu, ale také při řízení skladování používaného paliva, protože hlavní materiál zemské kůry je právě SiO2. Nejnověji byla tato sestava použita pro ověření neutronového popisu nerezové oceli, kterýžto je zcela zásadní v otázkách „criticallity safety“, protože nerezová ocel je hlavní součástí vodou moderovaných reaktorů.

Zásadní výsledky, jmenovitě kritičnost, rozložení neutronového toku, neutronová spektra a také některé naměřené spektrem vážené účinné průřezy, vyhodnocené SACS vážené v ^{235}U PFNS, byly rovněž publikovány v databázi IRPhEP. Lze jen dodat, že tato databáze je jednou z nejvěrohodnějších databází reaktorových experimentů používaných při vylepšování účinných průřezů. Sestavené referenční pole bylo zahrnuto mezi benchmarková referenční pole, a data na něm změřená se stala součástí nové neutronově dozimetrické knihovny jaderných dat IRDFF-II.

Acknowledgments

I would like to thank my wife Petra and the whole family, especially all four daughters, for their support and understanding.

I would also like to thank all collaborators covering all involved colleagues from Research Center Rez, and as the research work is wide, also people from other collaborating institutions without which contribution such wide research cannot be realized. It is especially colleagues from Masaryk University Brno, University of Defence, UJV, Czech Metrology Institute, Czech technical university, KIT, IAEA, Nuclear Physics Institute.

Among all, I would express my thank to Zdenek Matej without who supported the NGA device which is now an extremely well-defined neutron spectrometer, wouldn’t reach such a scientific level.

Thank you.

Michal Kostal

Content

1 Introduction

The cross section averaged over ^{235}U fission spectrum is a fundamental quantity that can be used in the evaluation of nuclear data. Many experiments focused on the determination of Spectral Averaged Cross Sections weighted in ^{235}U Prompt Fission Neutron Spectrum were performed in light water reactors using enriched uranium fuel. In these reactors, there is already some amount of water moderator between the uranium fuel and the irradiated sample. Due to decreasing character of hydrogen cross-section, the high energy tail of the reactor spectrum in cores with water moderators may be harder than the pure prompt fission neutron spectrum at some specific conditions. This is, for the example case of VR-1 reactor (see Figure 1) [1].

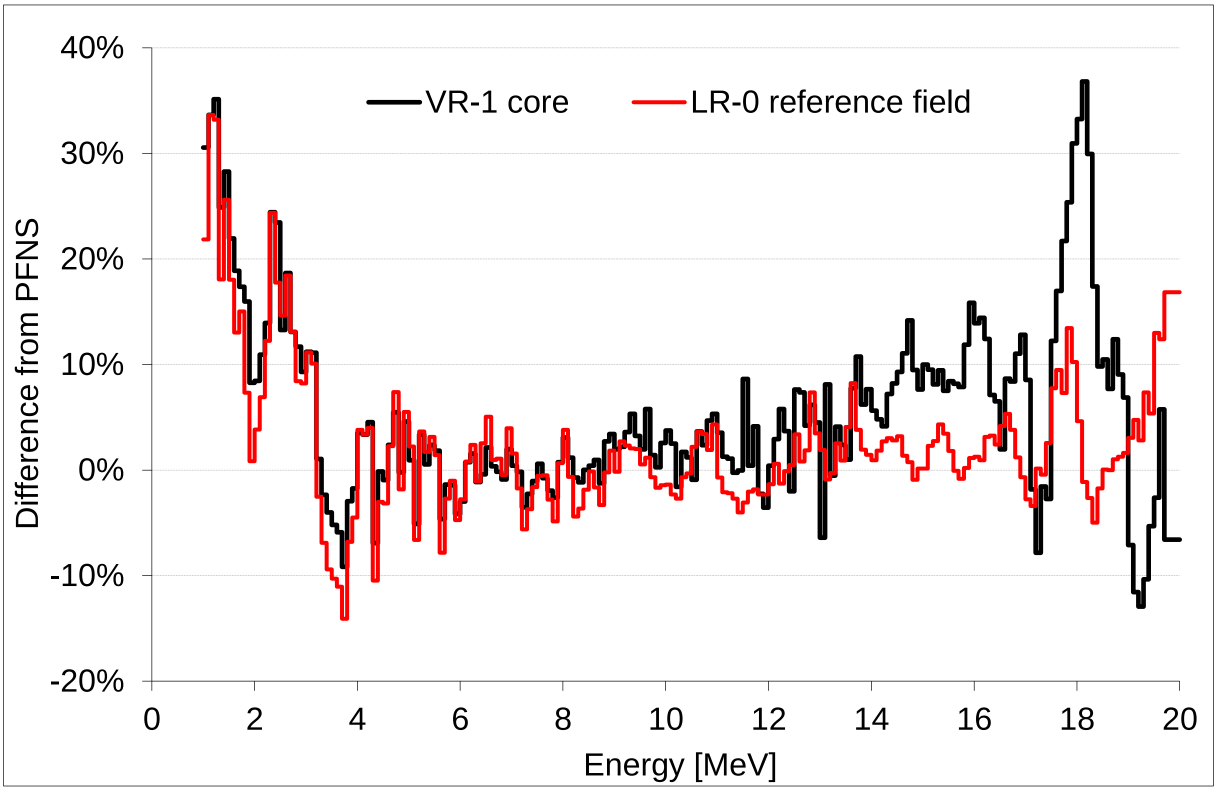

Figure 1: Comparison of VR-1 and LR-0 reactor spectra with 235U PFNS.

The figure demonstrates the fact, that whereas LR-0 spectra are close to 235U fission spectra, the VR-1 shows some differences in the region above 14 MeV. These differences in the high-energy tail of the fission spectra in the LR-0 and VR-1 reactors compared to the pure 235U prompt fission neutron spectrum can be explained by the differences in the macroscopic cross sections of the homogenized cores (see Figure 2). The total cross section of the homogenized LR-0 core is higher compared to VR-1. Also, the VR-1 homogenized cross section has decreasing character in the region above 15 MeV. The combination of both facts causes a non-negligible decrease in interaction rates with increasing energy in the VR-1. The situation in LR-0 is different because the cross section is nearly constant, and the oscillations from the average cross section are relatively lower (in percentage terms); thus, the interaction rate is nearly constant. This fact is propagated in nearly constant energy-dependent attenuation of high energy neutrons (> 10 MeV) in the LR-0, while in the VR-1, it decreases. Reflection of this fact is a harder tail of spectrum above 15 MeV in the VR-1 reactor.

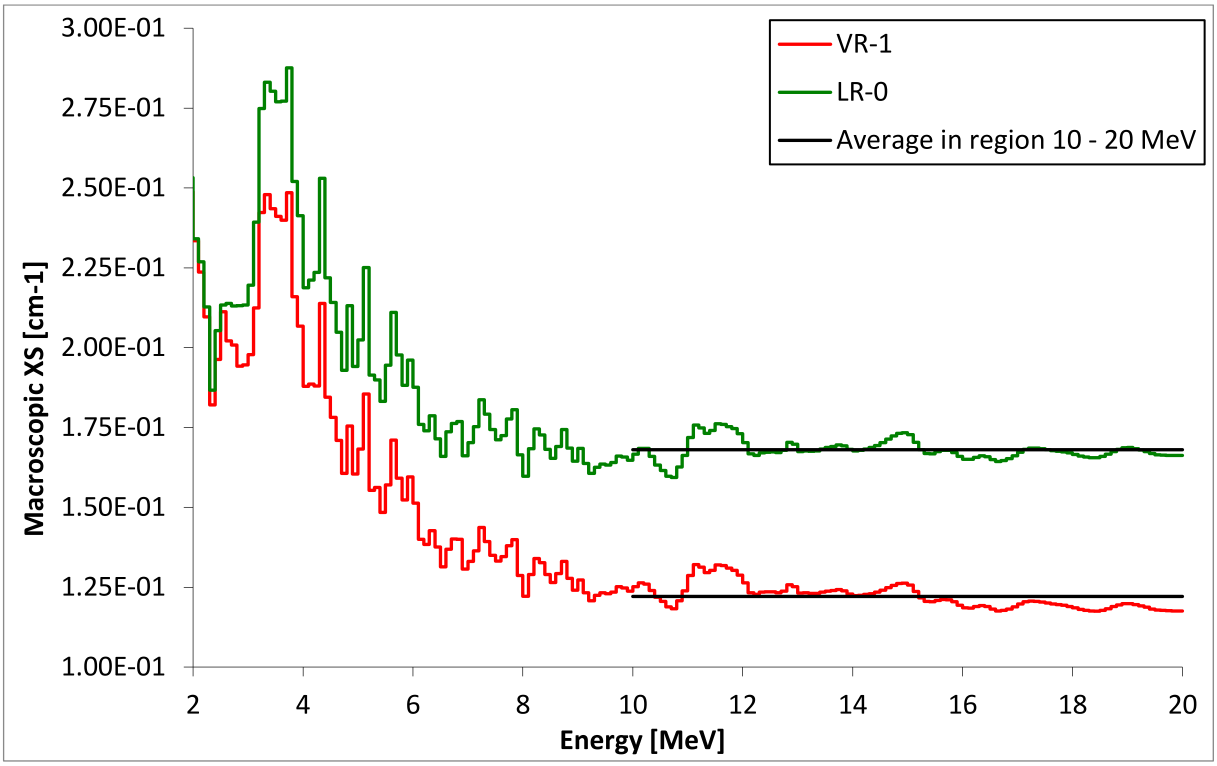

Figure 2: Macroscopic cross sections of VR-1 and LR-0 cores (ENDF/B-VIII data [2])

The above presented shows the importance of the International Nuclear Data Committee conclusion that purity of high energy tail of reactor spectra must be ensured prior to evaluation as 235U PFNS [3]. The same report also states that spectrum averaged cross sections in fast spectra in reactors where reference spectrum is not defined are more suitable for data verification/validation rather than for the adjustment. Their usefulness depends on the accuracy with which the spectrum is known. Ideally, it should be supplied with the full covariance matrix, or at least the uncertainties.

It is the reason why integral cross section data measured LR-0 team in VR-1 reactor [1] or LVR-15 reactor [4] were used for validation and verification of neutron dosimetry cross sections.

1.1 Focus of thesis

This work focuses on summarizing my results from the area of development of reference neutron benchmark field and its use for measurement of spectral averaged cross sections (SACS).

The measurements using 235U(n_th, fiss) are sometimes realized in a reactor environment. In this case, the purity of the spectrum must be checked by (Monte Carlo) simulations, ideally confirmed experimentally. For such purpose, it is necessary to develop a precise mathematical model of the core covering the volume where the samples were activated. The model has to be validated for as most parameters as possible. In the presented case, there were validated criticality, neutron flux spatial distribution, neutron spectra, gamma spectra, fission density profile. This model was used for the determination of corrections which allows the evaluation of measured reaction rates as 235U prompt fission averaged cross sections in high energy threshold reactions. In the case of capture reactions, the reaction rates are evaluated as LR-0 averaged cross sections, and despite the fact they cannot be used in the evaluation of nuclear cross sections they are usable for validations of existing evaluations.

The results were published in enclosed publications. All of them were renowned physical journals. The published data were used as the base for validation of the IRDFF-II nuclear data library, which is the best proof of the quality of results obtained in the LR-0 reference neutron benchmark field. This work is the continuation of my Ph.D. thesis in meaning that while the Ph.D. thesis deals with validation and development of suitable models for neutron transport in construction parts of VVER-1000 reactor, the habilitation thesis deals with the improvement of cross sections of monitors used for estimations of neutron flux in energetic reactor pressure vessels.

1.2 Structure of thesis

The text is divided into major milestones in the development of the reference neutron benchmark field. The first part, chapter 2, presents the LR-0 reactor where the neutron reference field was set. Chapter 3 deals with the characterization of the critical parameter. It is the important part because the uncertainty of calculations reflects the discrepancy between measured and calculated critical parameter keff. Chapter 4 contains the description of methods and results obtained in the characterization of neutron flux profiles. It was realized using activation detectors and precise HPGe semiconductor spectrometry. Chapter 5 is focused on the determination of neutron and gamma spectra in the location where the reference field was identified. It was realized using stilbene spectrometry. The calibration was tested in a special Si filtered field with quasi monoenergetic lines, which was developed by the author of this thesis.

Chapter 6 deals with validation of fission density profile and is complementary to profile of neutron flux inside reference field. It was realized using precise methodology developed by the author of this thesis during his Ph.D. Chapter 7 summarizes obtained results and discusses possibilities of use of developed field and its use in future work. Finally, chapter 8 summarized the obtained results. The collection of publications is attached to the end of this work.

2 LR-0 reactor

The neutron reference field was found in a specifically designed core assembled in the LR-0 reactor. The LR-0 is a zero power light water pool type reactor operated by the Research Centre Řež (Czech Republic). Continuous nominal power is 1 kW with a thermal neutron flux of about 10^9 cm^{-2}·$s^{-1}$and a fast neutron flux (above 1 MeV) of 2×10^8 cm^{-2}·s^{-1}. An illustration of the LR-0 reactor and a scheme of the core configuration for this experiment are shown in Fig. 1.

The main reactor feature that allows these experiments to be performed is flexible rearrangements of the core. The experiments are realized at atmospheric pressure and room temperature. The change of the moderator level or control-cluster position is used to control reactor power. Continuous maximal nominal power is 1 kW with neutron thermal-flux density ≈ 10^{13} n m-2 s-1. The LR-0 reactor is located in Řež near Prague (Czech Republic). The first criticality was reached in December 1982.

The LR-0 fuel elements are radially identical with those used in VVER NPPs, axially they are shortened to 125cm.

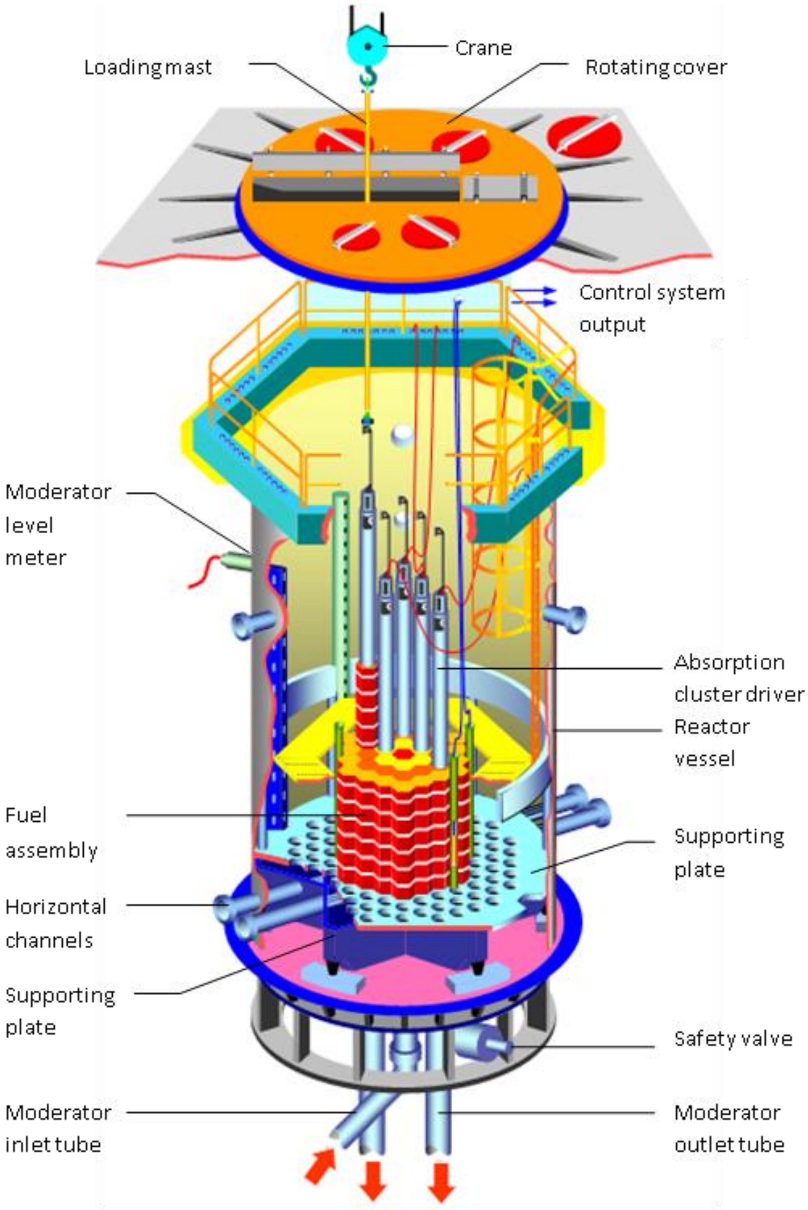

Figure 3: Scheme of LR-0 reactor [5]

The main objective of the projects on the LR-0 reactor is to create the experimental databases, which could be utilized for verification of codes and libraries used for neutronic calculations in safety analyses of the criticality of cores, storage, and transport-cask lattices for VVER type reactors [6]. In combination with VVER-1000 Mock-Up with full scale simulators of internals, downcomer, reactor pressure vessel, and even biological shielding. It is also used as a validation tool in criticality issues [6]., reactor dosimetry of internals [7], reactor pressure vessel [8], and also validation tool for mathematical models focused on concrete biological shielding [9].

The reference benchmark neutron field was identified in a dry channel surrounded by six fuel assemblies enriched to 3.3 wt. %. This geometry is simple and very simple to repeat. This simplicity also allows to use it in different validations of transport cross sections, where the studied material is inside of the core, or surrounds it. The radial cut of used geometry is plotted in Figure 4, the axial plot in Figure 5.

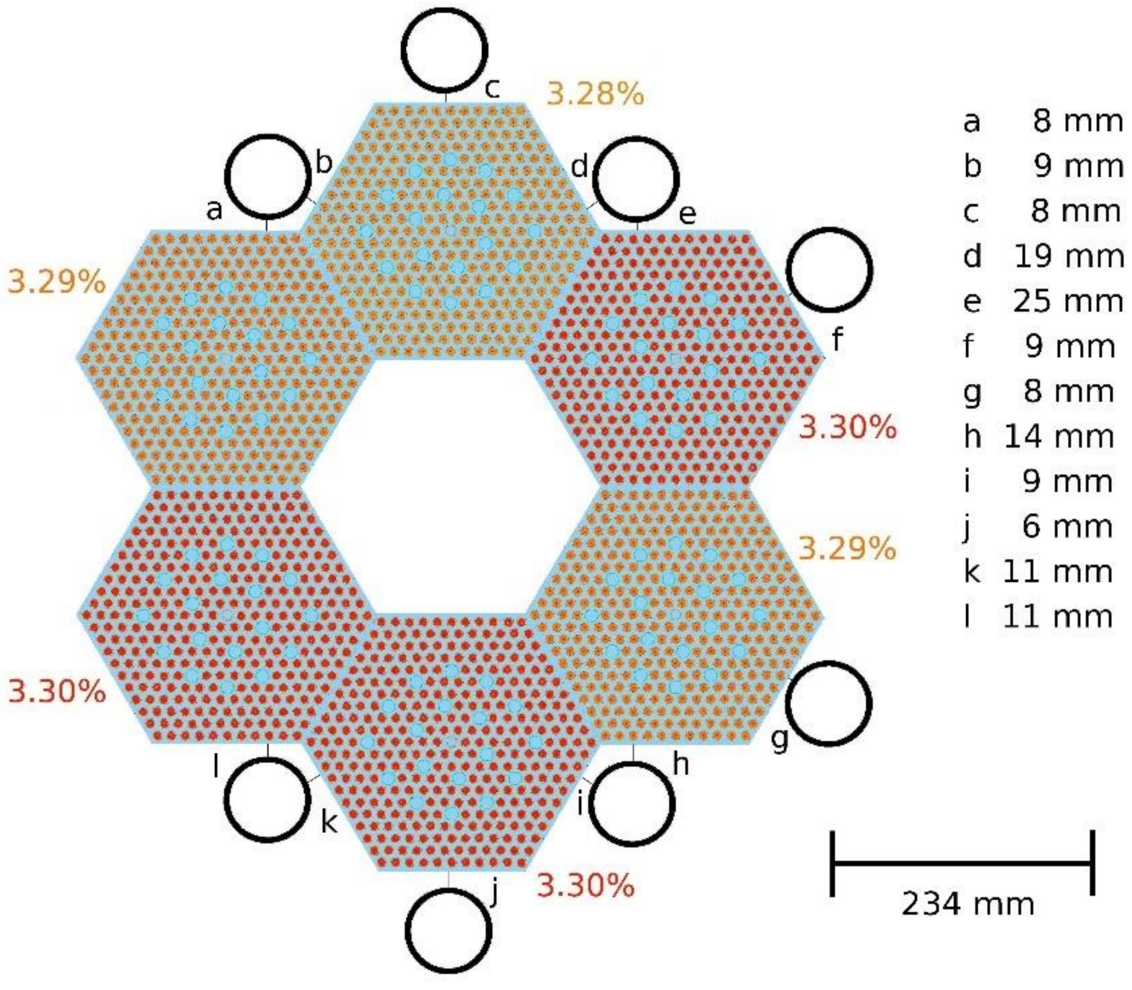

Figure 4: Radial cross-section of the core with specified enrichment, the distances are upright from pin [5]

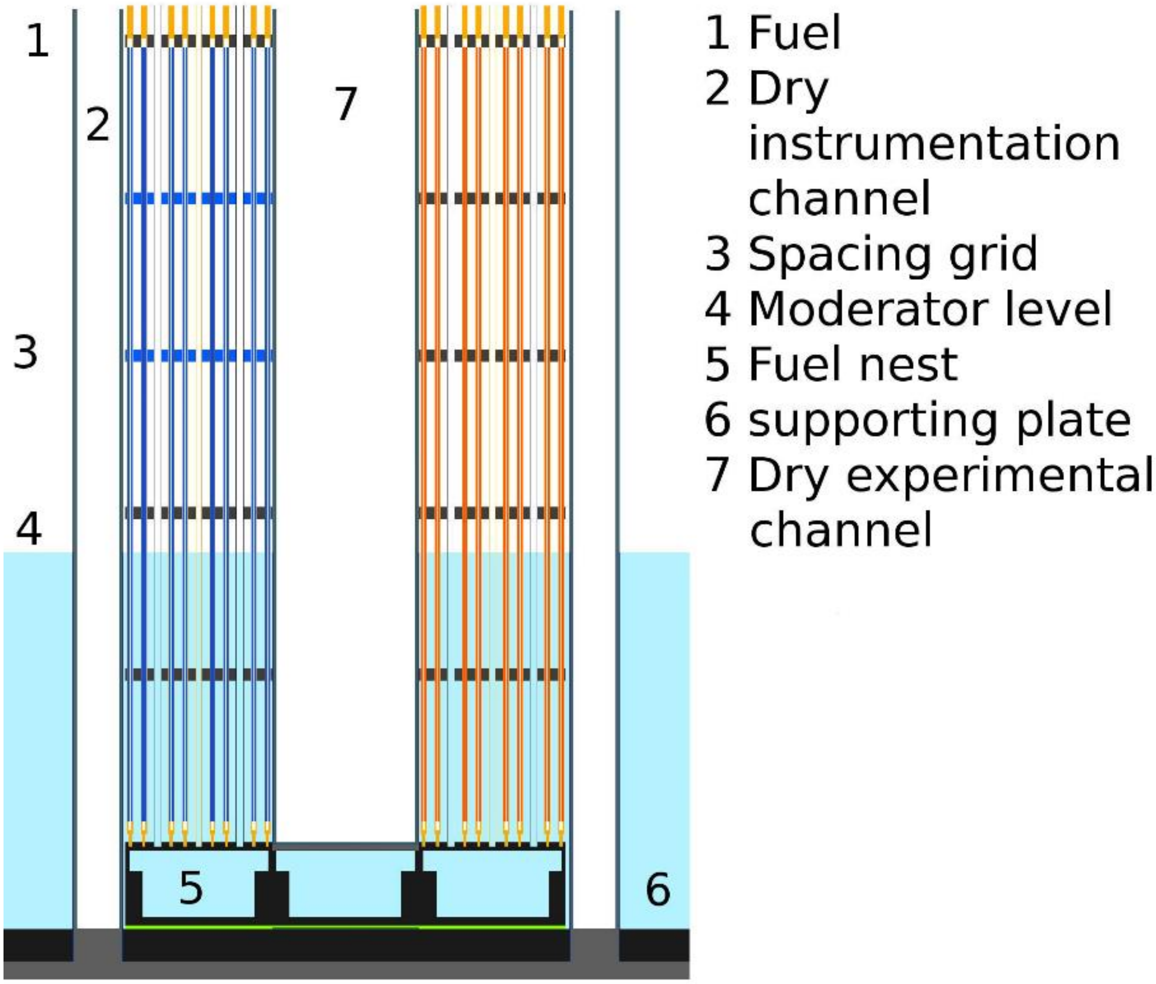

Figure 5: Axial section of core [5]

3 Characterization of critical parameter

The reactivity in the LR-0 reactor can be driven by water level as well as by control clusters with rods filled by B_4C. These control clusters are essential in VVER-1000 Mock-Up, where the fuel is fully flooded. This fully flooded core is important in deep penetration experiments, where the upper water reflector decreases leakage flux, thus minimizing room effect, which could distort deep penetration experiments [10].

In smaller cores, the reactivity is driven exclusively by the water moderator level.

In that case, the moderator level (H_{cr}) acts as a critical parameter. If the moderator level slightly exceeds the H_{cr}, the one-group asymptotic approximation [5] may be used to express the reactivity

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 ref. [12]).

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

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 [13].

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

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

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 the height of 10.0 cm with the 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 [14]). This value is taken as the standard uncertainty for critical water-level height.

The measured critical height was used as a parameter in the mathematical model describing the critical core. As the core was just critical, experimental k_{eff} is 1. As k_{eff} obtained using of the mathematical model is very close to 1, it can be assumed the developed mathematical description of the core is valid in the meaning of criticality.

3.1 Articles in collection

[15] Michal Košťál, Vojtěch Rypar, Ján Milčák, Vlastimil Juříček, Evžen Losa, Benoit Forget, Sterling Harper, Study of graphite reactivity worth on well-defined cores assembled on LR-0 reactor, Ann. of Nucl. En., Vol. 87, 2016, pp. 601-611, ISSN 0306-4549

4 Characterization of neutron flux profile

The neutron flux density spatial distribution is an important parameter describing neutron field [16], [17]. Especially when irradiated samples have non-point character, the field is strongly connected with neutron flux gradient in the sample [18].

Neutron flux can be monitored by both active and passive methods. Monitoring by passive measures is highly important in applications where the active measurements are infeasible. These applications include, for example, absolute measurements for reactor dosimetry (e.g., behind RPV in NPP) or fusion applications. Therefore it is necessary to use especially materials with known properties (namely activation cross section) in a given energy spectrum [19], [20]. In the case of the benchmark reference field, the profile has been characterized by means of the MCNP model [21].

The reaction rates were evaluated using gamma activity determined by means of semiconductor gamma spectrometry. The efficiency curve has been determined by calculation using a validated mathematical model [18]. The model has been compiled using geometrical parameters obtained from radiography (see Figure 7) and experimentally determined insensitive layer [22].

The activation foils with well defined dosimetric reactions (^{58}Ni(n,p), ^{197}Au(n,g), ^{181}Ta(n,g)) were placed in geometrically well defined positions Figure 6. The reactions were selected to cover the thermal, epithermal, and fast part of neutron spectra.

The experimental data were compared with the developed mathematical model. Due to satisfactory agreement between both, it can be concluded the developed model is valid in the meaning of flux distribution. The details can be found in [18].

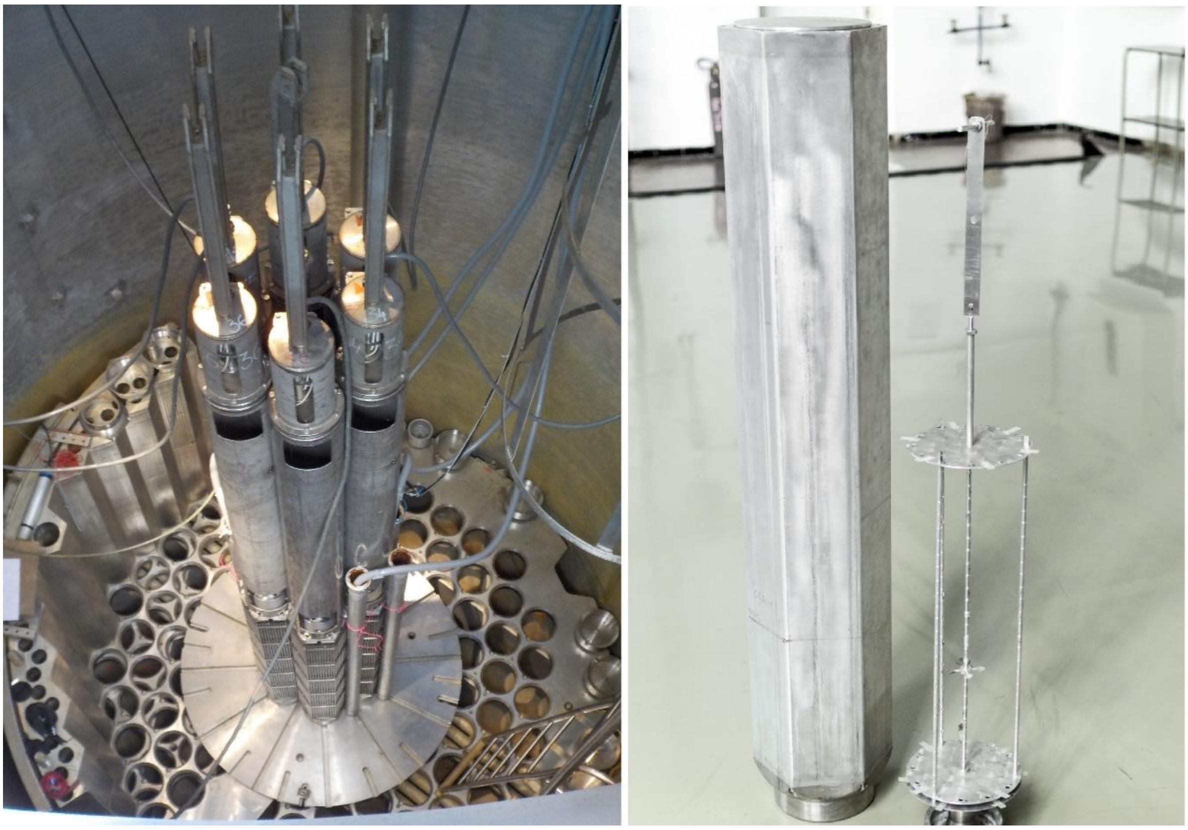

Figure 6: Overhead view inside the LR-0 reactor with a special core without a moderator (left) and a dry experimental channel with activation foil holder (right) [18]

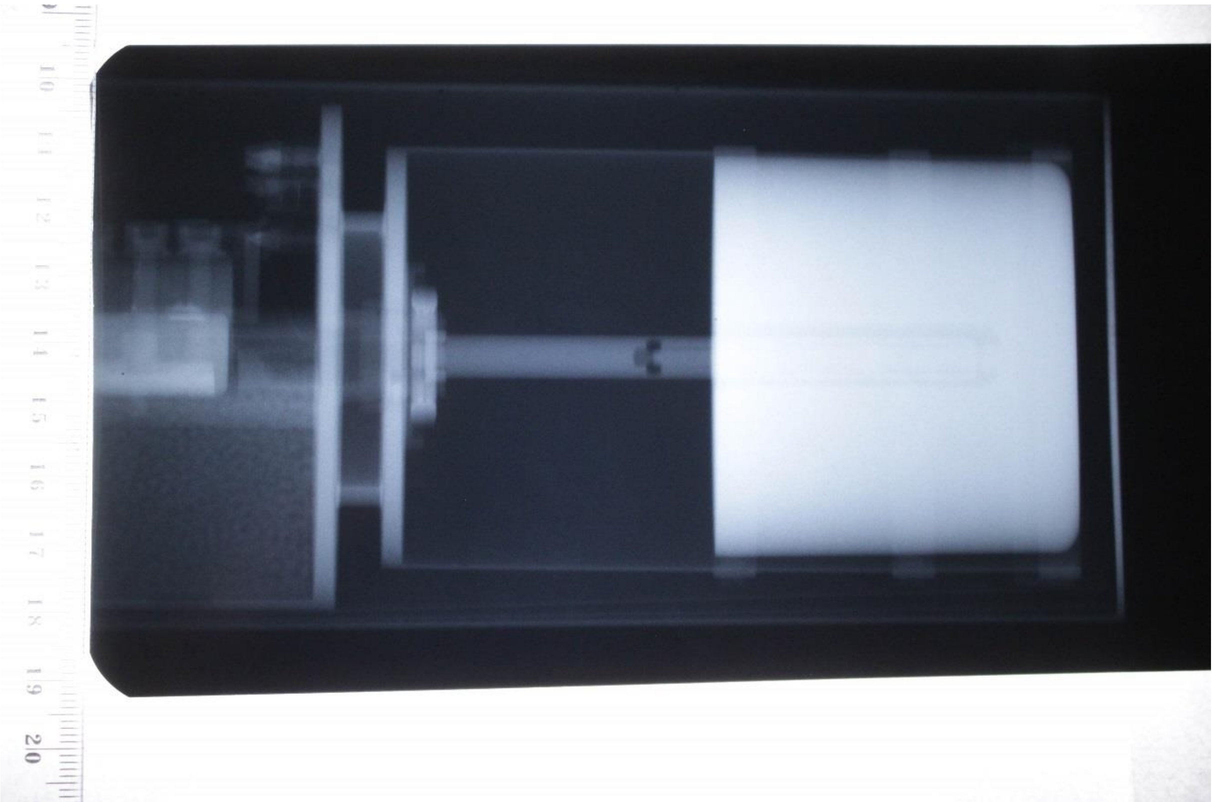

Figure 7: HPGe GEM35 detector radiogram used for determination of mathematical model

4.1 Articles in collection

[18] Košťál, M., Schulc, M., Šimon, J., Burianová N., Harutyunyan D., Losa, E., Rypar, V., Measurement of various monitors reaction rate in a special core at LR-0 reactor, Annals of Nuclear Energy, 2018, 112, pp. 759–768

5 Characterization of neutron and gamma spectra

The knowledge of neutron spectrum is crucial for describing neutron field because neutronic cross section strongly depends on neutron energy. Often the spectrum is evaluated based on deconvolution from the set of reaction rates [23]. In the case of the LR-0 reference field, the neutron spectrum was measured using stilbene scintillation spectrometry [24]. As the light outputs of the scintillator coming from gamma and neutron reactions are different, their precise characteristics are important in the determination of the stilbene response function and calibration [25]. The neutrons and gammas in mixed fields can then be distinguished using PSD [26]. The specialists from University of Defence (Brno, Czech Republic) tested the developed stilbene spectrometer in a Physikalisch-Technische Bundesanstalt (PTB, Braunschweig, Germany) reference neutron field. An accelerator, producing mono-energetic neutron fields (1.2 MeV, 2.5 MeV, 5 MeV, 14.6 MeV, and 19 MeV), has been used for absolute sensitivity and detector response function studies [27].

The gamma spectrum is not primary focused, but its shape is important, because in reactor experiments, there is often a question on the gamma effect on the purity of the experiment. Namely, the problem might be the contribution from (g,n) reactions whose product is the same as those originating from (n,2n) reactions.

The resulting electrons from gamma interactions are standardly used for organic scintillator calibration. This procedure is applicable under the assumption that the detection system, including crystal, photomultiplier tube, and associated electronics, is linear. The neutron calibration curve is determined using the gamma calibration curve (Figure 8) from the measurement using the gamma standards' Compton edge energy. An example of the calibration for ^{60}Co peaks is depicted in Figure 9. The linearity of the system is confirmed when the channels of the peaks and their corresponding energies have a linear dependence. For this measurement, the linearity is plotted in Figure 10.

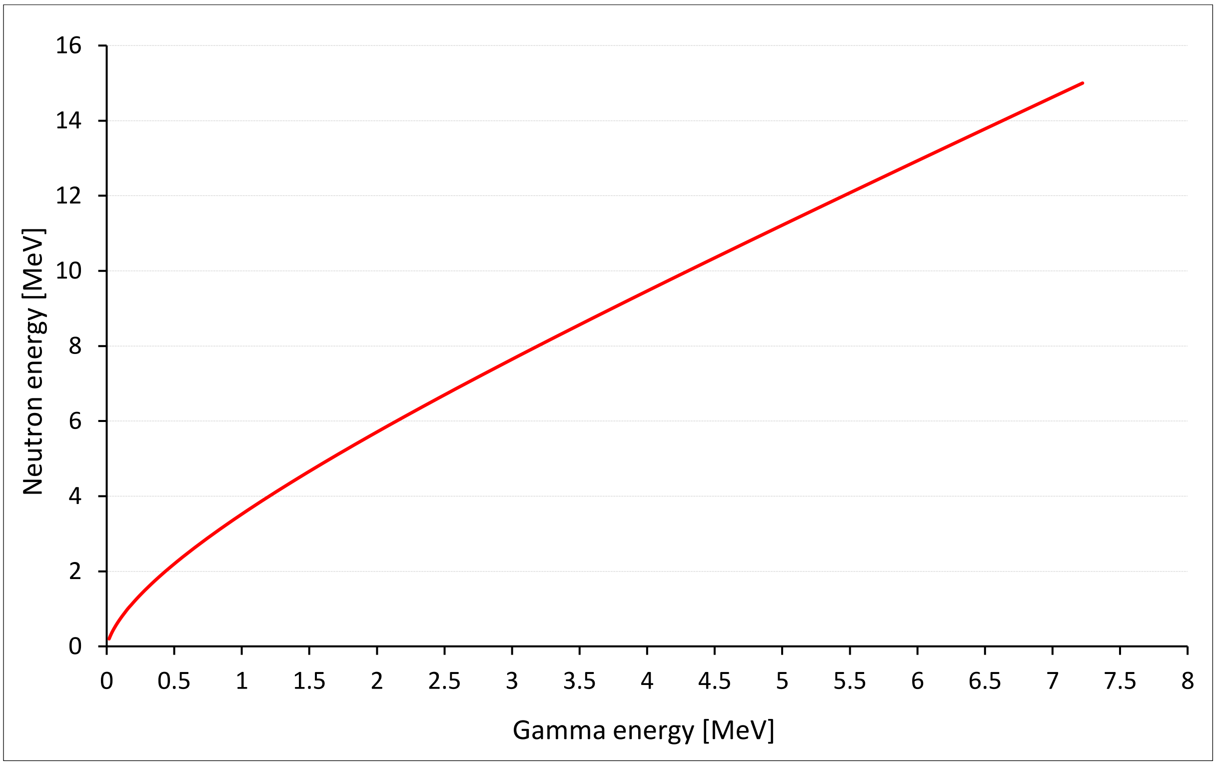

Figure 8: Relation between Neutron and Gamma Energy in Stilbene Detector.

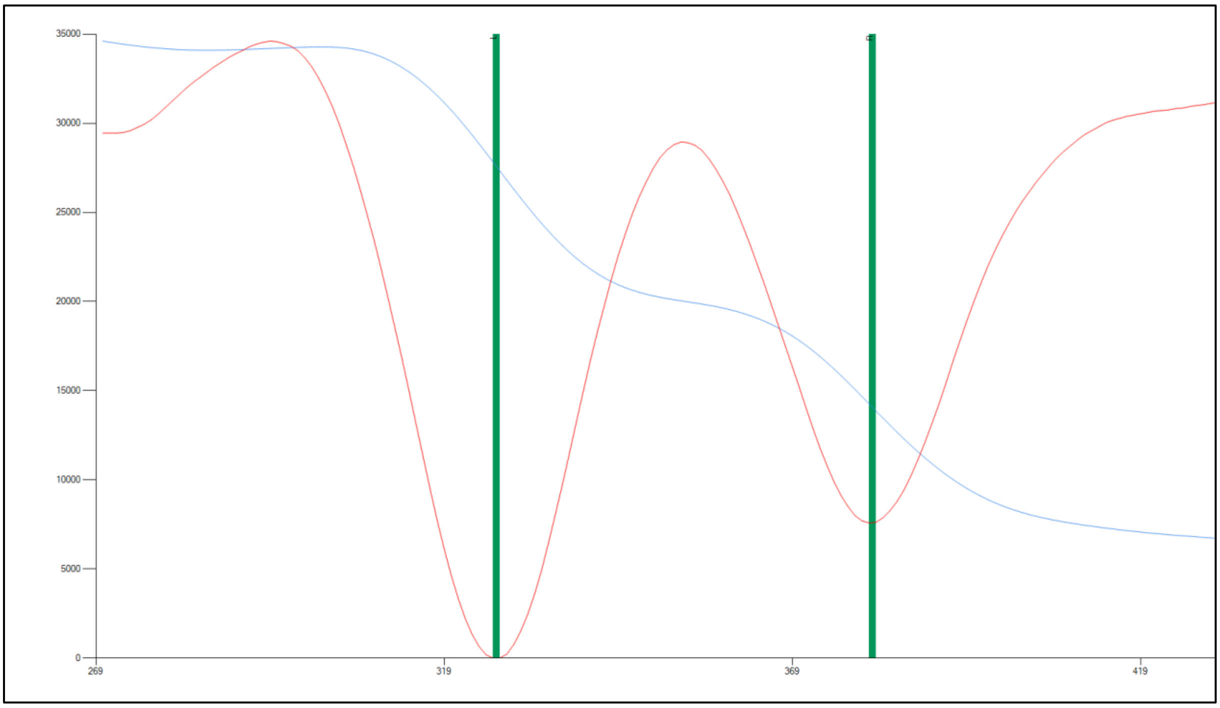

Figure 9: Calibration of the Apparatus Spectrum using ^{60}Co Peak.

The electron spectrum is the blue line, the differentiation of the spectrum with visible inflections is the red line. Green lines show channel number settings for energy scale calibration.

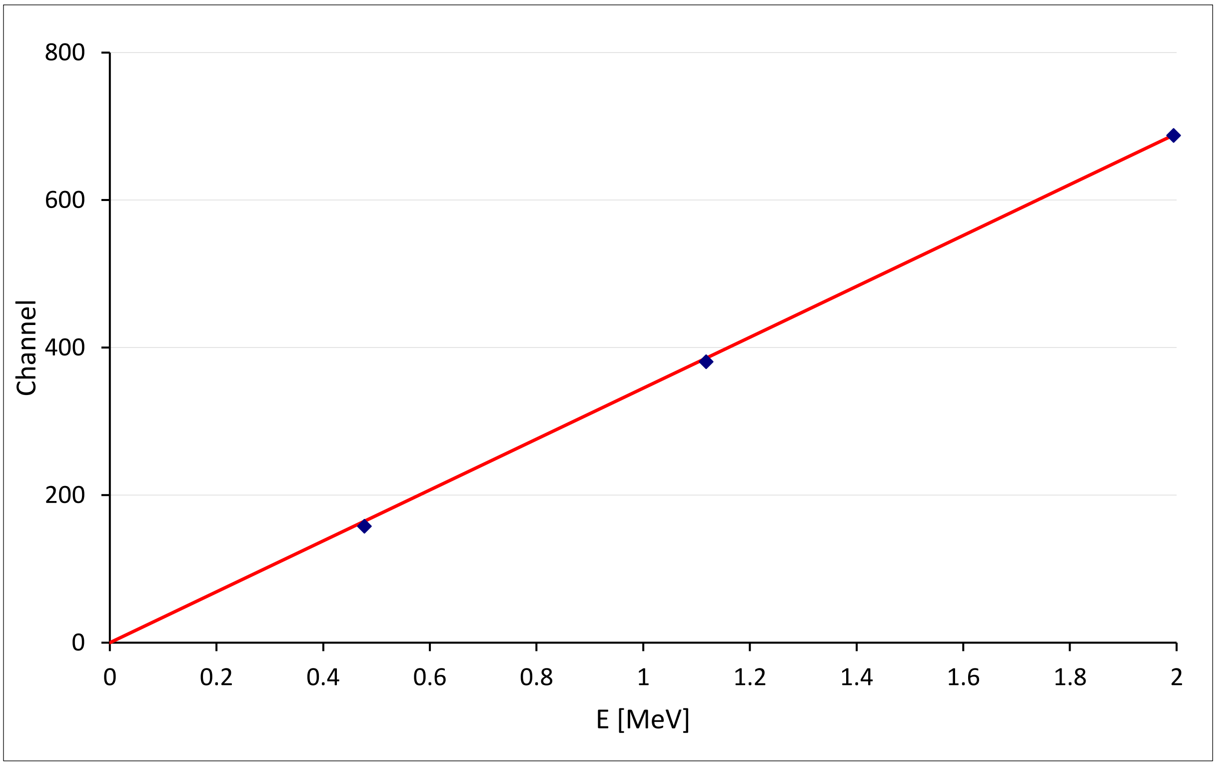

Figure 10: Linearity of Calibration (Cs line, Co line, 1H(n,γ)) UPMT=1400 V.

As the calibration is indirect, there is often a question on the reliability of obtained results. Due to this fact the methodology has been tested in many various fields. It was tested in VR-1 reactor radial channel [28], in Mock-Up of VVER-1000 in LR-0 reactor [29], in DT generator neutron field [30] and also in radioisotope field [31]. Due to indirect evaluation, the uncertainties of the methodology are very complicated to determine. Often, they are determined with use of some standard and comparison between the standard values and values obtained by the tested methodology. The bias between both can be understood as the uncertainty of the method itself. In this case it by means of comparison between as bias between tabulated spectra [19], [20] and stilbene scintillation spectrometry determined ^{252}Cf(s.f.) spectra (Figure 11). Based on such agreement, in fission like smooth spectra, which also cover neutron spectra in reference benchmark neutron field the systematic uncertainty coming from problems in deconvolution are not bigger than 5%.

For testing of calibration there was also developed silicon filtered beam in the LVR-15 research reactor. It is formed by the LVR-15 fission neutrons passing by one meter of single-crystalline silicon [32]. Due to the character of Si cross section, the field contains several significant peaks in the fast neutron energy range (see Figure 12). Testing in such neutron fields is very valuable because it can reveal specific problems in the deconvolution matrix of the detection system, which may stay hidden in fields with a smooth structure and can provide a tool for proper energy calibration.

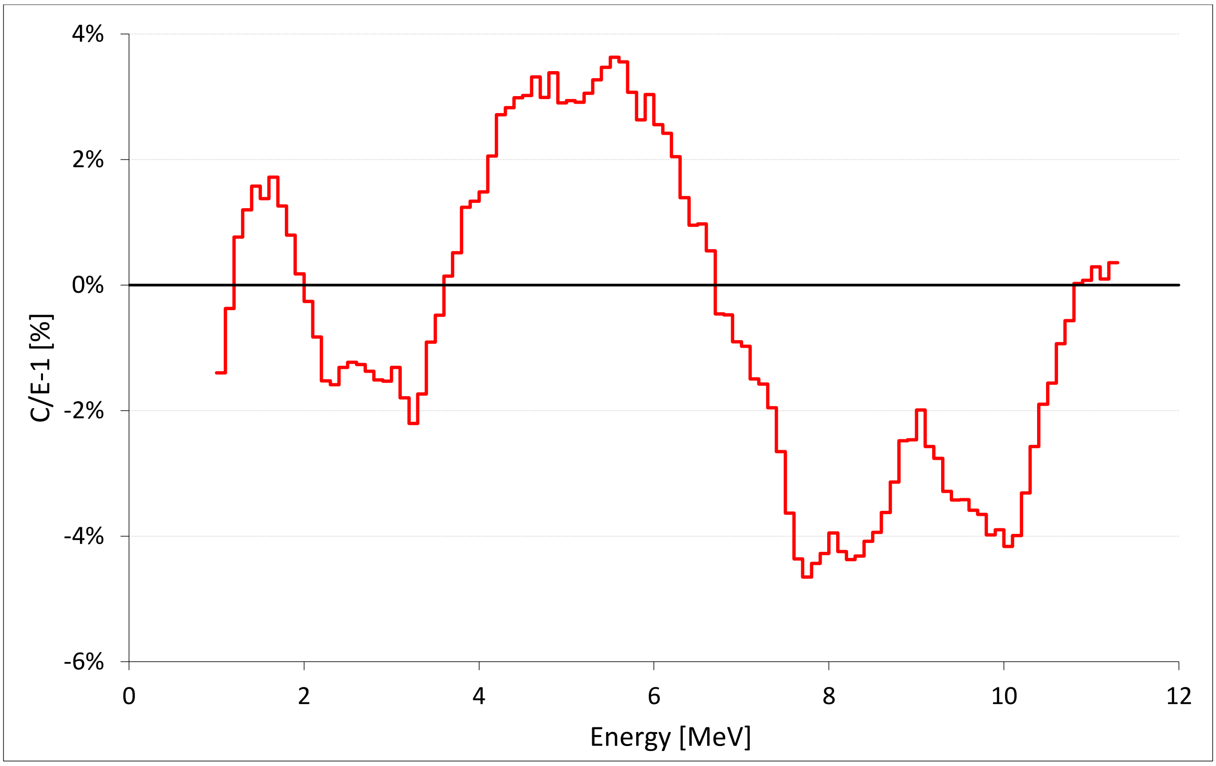

Figure 11: Eval/E-1 of 252Cf(s.f.) for flux 1 m from point source measured with NGA-01 and ø 10 mm × 10 mm crystal used in experiments

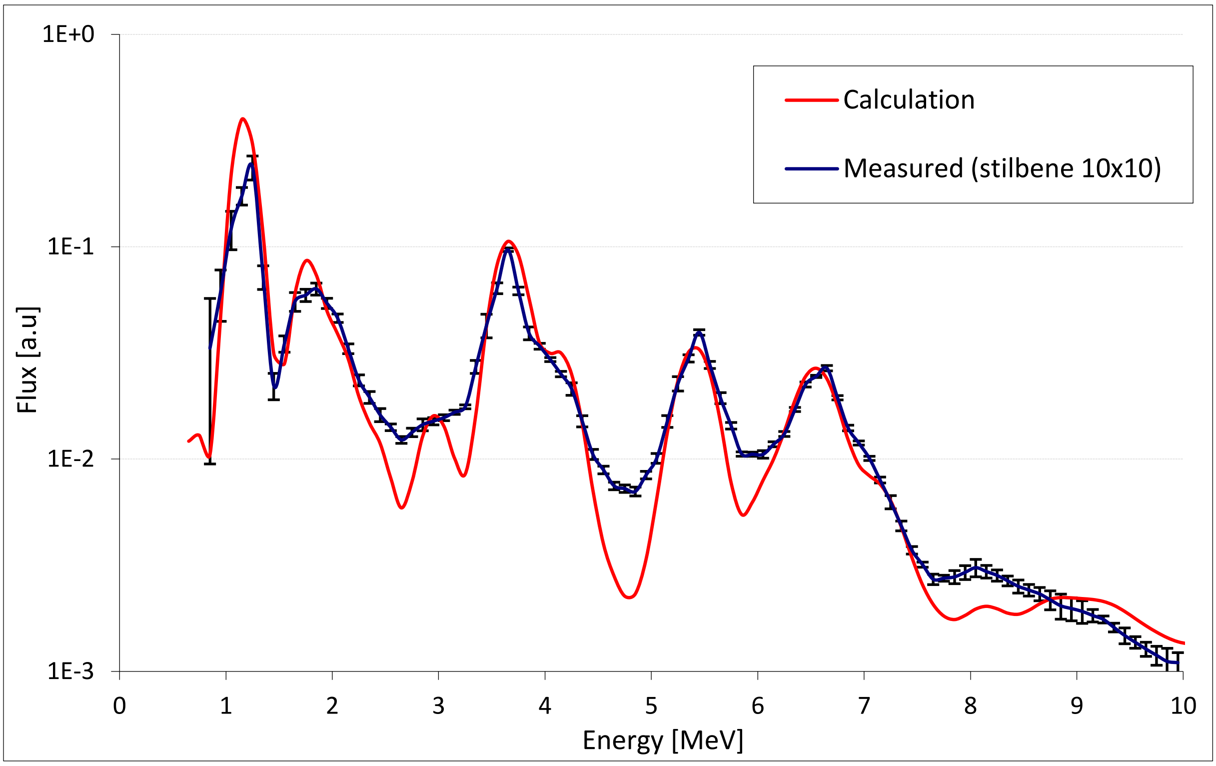

Figure 12: Comparison of calculated and measured Si filtered fluxes.

Using the above described and tested methodology the neutron spectra in reference position and region 1 – 14 MeV were determined, in region 0.1 – 1 MeV proportional counter was used [33]. The comparison between calculated and measured spectra is plotted in Figure 13. There is reported good agreement between calculation and experiment. Due to such agreement, it can be concluded the developed mathematical model for the LR-0 reference benchmark neutron field is valid in terms of neutron spectra in the reference position.

Moreover, it was demonstrated the neutron spectra above 6 MeV are undistinguishable from ^{235}U PFNS (see Figure 14). It means the neutrons with energy above 6 MeV have nearly the same energy distribution as ^{235}U fission neutrons. It also means using proper normalization, the integral cross sections of reactions induced by neutrons with energy above 6 MeV can be evaluated as integral cross section weighted by ^{235}U PFNS.

This is a valuable result because it allows using LR-0 date also for testing of ^{235}U prompt fission neutron spectra when the cross section of dosimetric reactions is well known [24].

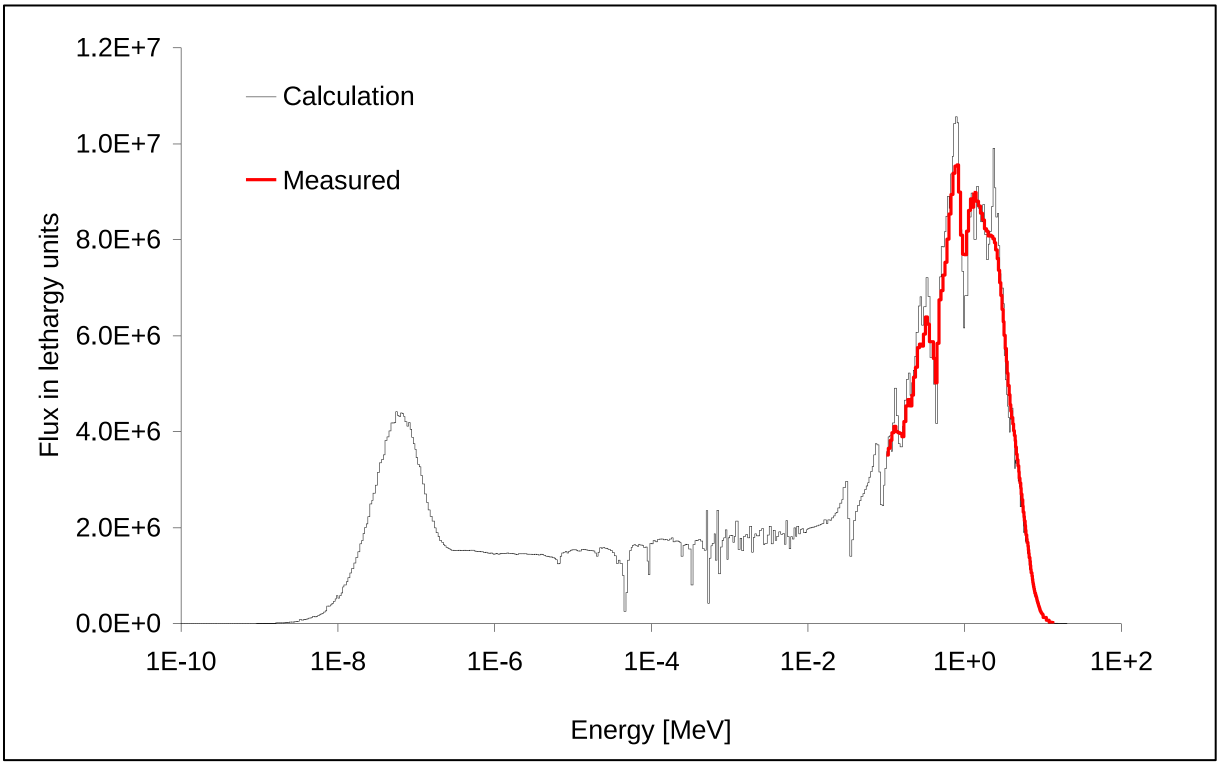

Figure 13: Comparison of Calculated, Measured neutron flux in the same core arrangement [34]

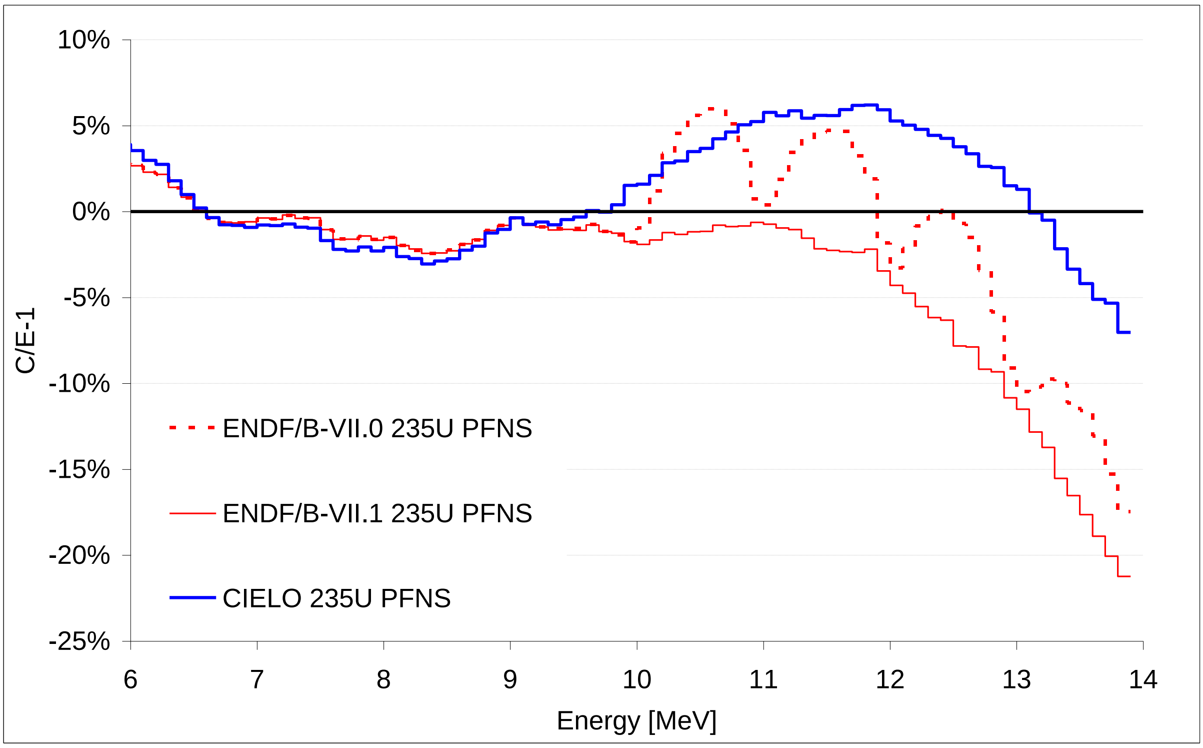

Figure 14: Identity between LR-0 and ^{235}U PFNS in ENDF/B-VII.0, ENDF/B-VII.1 and CIELO [24]. The CIELO evaluation is the newest one.

The gamma spectrum was evaluated as well. Both calculation and agreement show that gamma spectra go up to 10 MeV (see Figure 15). This result is important especially in the evaluation of (n,2n) reaction cross section, because of the impact of the photo-nuclear reaction which leads to the additional production of the same residual nuclei. The threshold of most of (n,2n) reactions is above 10 MeV. Thus they are not affected by gammas [35]. Only exception is ^{197}Au(n,2n), where the threshold of $^{197}Au(\gamma,n)^{196}Au$ is about 8.1 MeV [36]. It means that some ^{196}Au can are of (g,n) origin instead of proposed (n,2n). However, in studied volume this effect can be neglected because it is not higher than 0.1%.

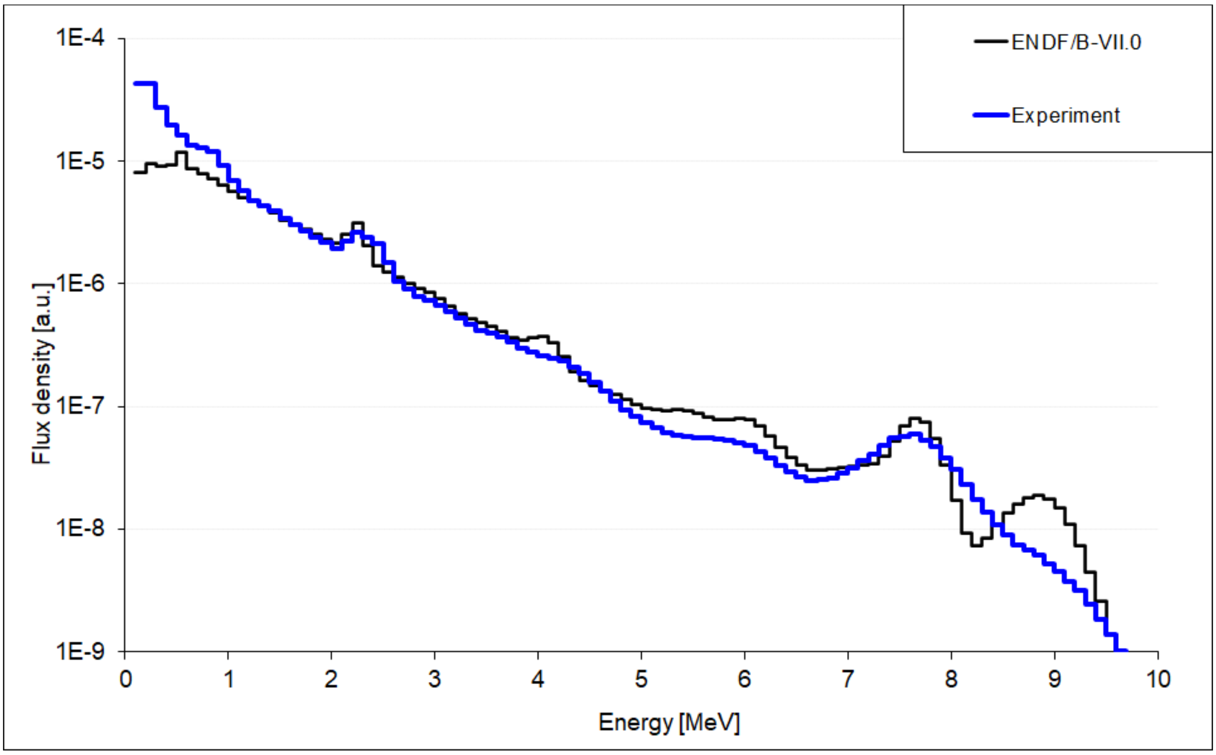

Figure 15: Comparison of measured and calculated gamma spectra in reference neutron field [37]

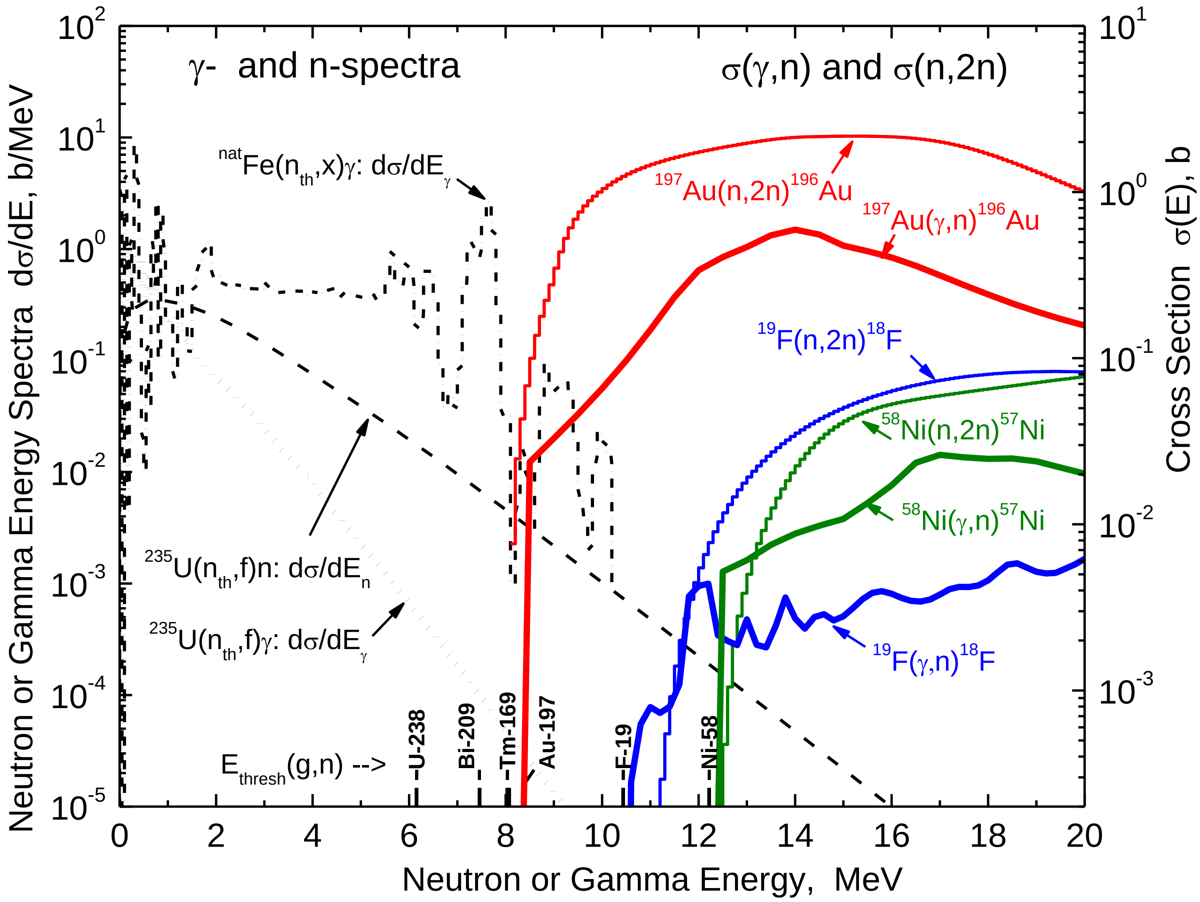

Figure 16: Cross section of selected (g,n) reactions with gamma spectra induced by neutron capture on iron.

5.1 Articles in collection

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6 Characterization of the fission density profile

The neutron field in the developed reference benchmark neutron field is formed by neutrons emitted by the driver core. Due to this fact, the validation of the source term, namely spatial distribution of neutron emissivity is the valuable cross check of the previously validated neutron field spatial distribution in reference volume because the neutron emission from fission is isotropic.

The neutron spectra in the core differ only slightly; thus, the concentration of fission products is nearly proportional to fission density in a defined point in fuel [38]. Due to such proportionality can be simply derived from gamma activity of fission products after defined irradiation (3). The details of the methodology can be found in [39].

This, developed by me during my Ph.D., the approach using single peak and fission products with well defined fission yields was also reported in [40], [41]. There is also possible to measure total gamma activity induced during irradiation [42]. In this approach, the relative fission density profile is derived by means of comparison with a reference fuel pin whose decay is monitored together with the measured one. The decay correction is determined semi-empirically because due to the fact that the wide integral is formed by many peaks with various decay parameters, the analytical solution is unrealistic.

The details of the decay corrections used in the evaluation of fission density in a single peak can be found in equations 5 – 8. The time evolution is plotted in Figure 17.

$$F_j^i = \frac{NPA_j^i(t)}{\eta^i \cdot \varepsilon^i \cdot \lambda^i} \cdot \frac{1}{N^i(t)} = NPA_j^i(0) \cdot \frac{1}{\eta^i \cdot \varepsilon^i \cdot \lambda^i \cdot N^i(0)} \tag{3}$$

$$NPA_j^i(0) = NPA_j^i(t) \cdot \frac{1}{e^{-\lambda^i \cdot t}} \cdot \frac{\lambda^i \cdot \Delta T}{1 - e^{-\lambda^i \cdot \Delta T}} \tag{4}$$

Where:

F_j^i fission rate determined via the i-th nuclei and j-th pin; $N^i(t)$ the calculated number of observed nuclei in fuel pin when 1 fission/s occurs, in time t after irradiation end; $NPA_j^i(t)$ measured Net Peak Area j-th pin of the observed nuclei i and selected peak, λ^i decay constant of a selected nuclide; η^i efficiency of HPGe for the selected gamma line of the i-th nuclide; ε^i gamma branching ratio of the selected peak from observed nuclei i; t start j-th pin measurement; ΔT length of j-th pin HPGe measurement

During the measurement, the detector response depends on actual activity and apparatus sensitivity to the measured gamma photons as $Response = \int A(t) \times \varepsilon \times \eta . dt$. The isotope activity decays by decay law as $A = A(0) \times e^{-\lambda . T}$. By integration can be obtained:

$$\int A(t).\,dt = \int_0^{Tmeasurement} A_{Startup} \times e^{-\lambda . t}\,dt = A_{Startup} \times \frac{1 - e^{-\lambda . Tmeas.}}{\lambda.} \tag{5}$$

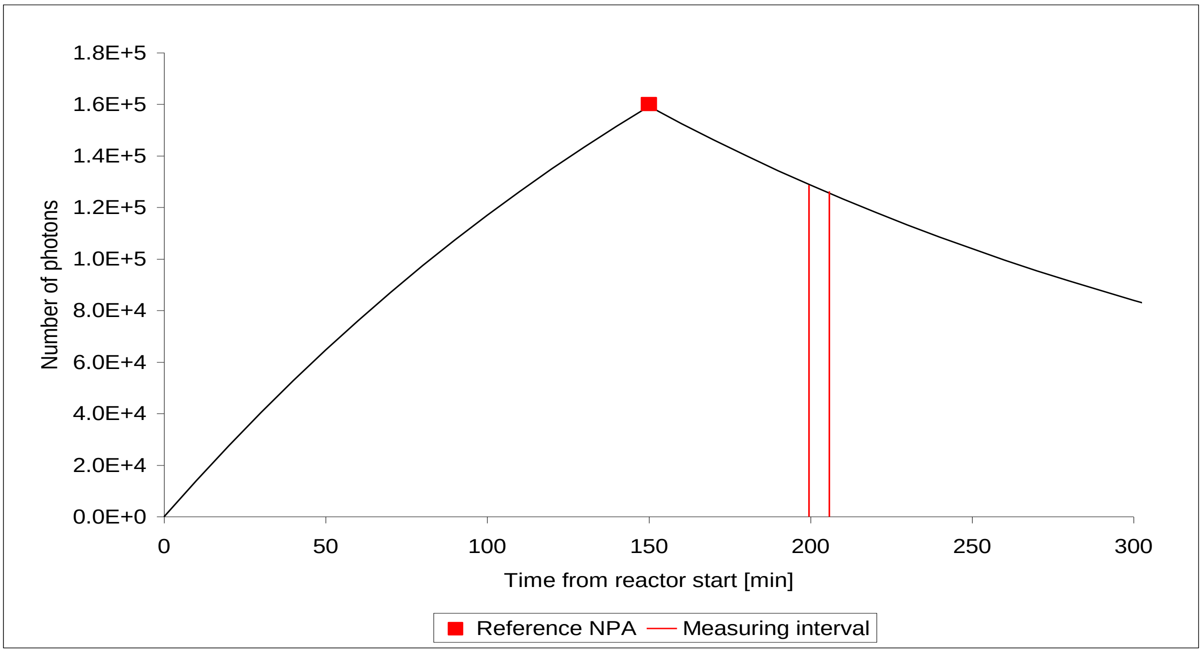

Figure 17: Time scheme of the measurement.

The correction factor, taking into account the decay compared with the non-decay case, is then obtained just by division by $A_{startup}.T_{meas.}$ Thus the correction factor is:

$$\frac{\int A(t).dt}{A_{Startup} \times T_{measurement}} = \frac{1 - e^{-\lambda.T_{meas.}}}{\lambda \times T_{meas.}} \tag{6}$$

The second task is a correction to decay between the start of measurement and the reference point. This correction is very simple, using the decay formula $N = N(0) \times e^{-\lambda.T}$, and it is reflected in equation (11). Similarly, as in the previous case, when half-life is very long, no correction is necessary, as the $-\lambda \times \Delta T$ approaches 0.

$$\frac{N^{REF}}{N_{startup}} = \frac{1}{e^{-\lambda.\Delta T}} \tag{7}$$

$$N_{ref} = N_{measured}.\frac{1}{e^{-\lambda.\Delta T}} \times \frac{\lambda.T_{meas.}}{1 - e^{-\lambda.Tmeas.}} \tag{8}$$

For correct determination of fission density it is the correct essential determination of gamma activity. Due to this request the characterization of HPGe [22] used in the fuel experiment was realized as well. The developed mathematical model of used HPGe seems to be suitable, as related uncertainty is estimated from bias between standard and evaluated activity (see Figure 18, Figure 19).

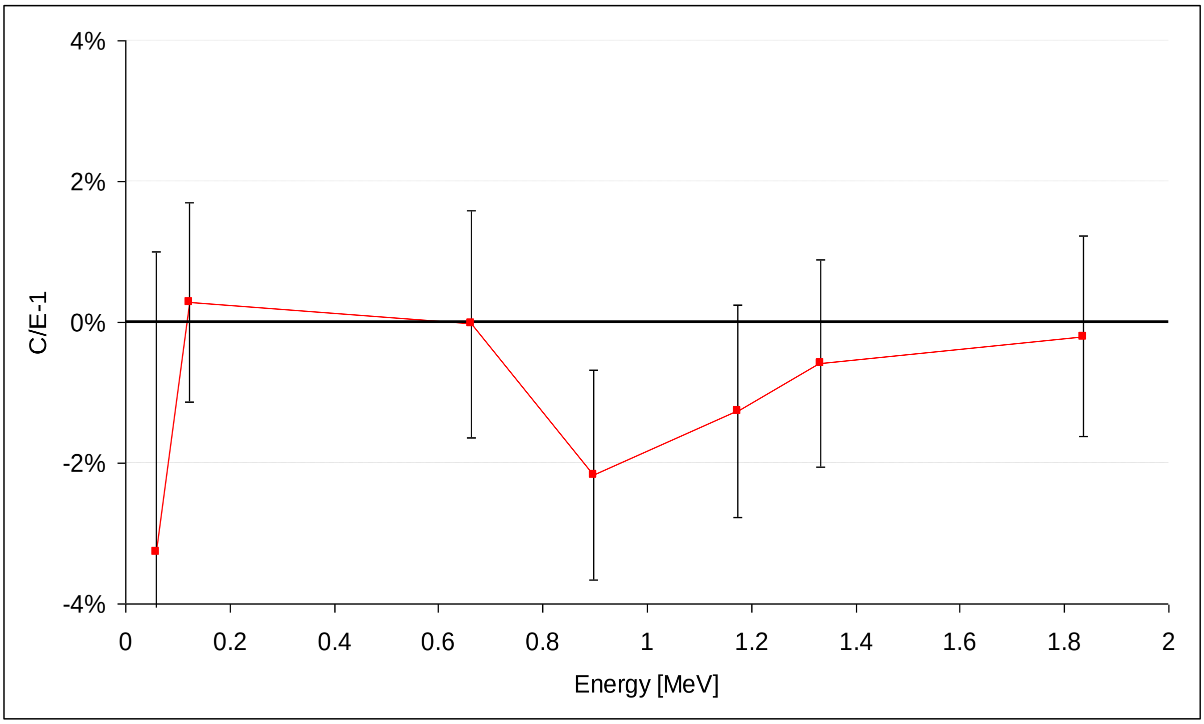

Figure 18: Validation of HPGe detector with etalon with a set of gamma emitters (^{252}Am, ^{57}Co, ^{60}Co, ^{137}Cs, ^{88}Y), etalon source in the middle of collimator slit, 11.5 cm from detector cap.

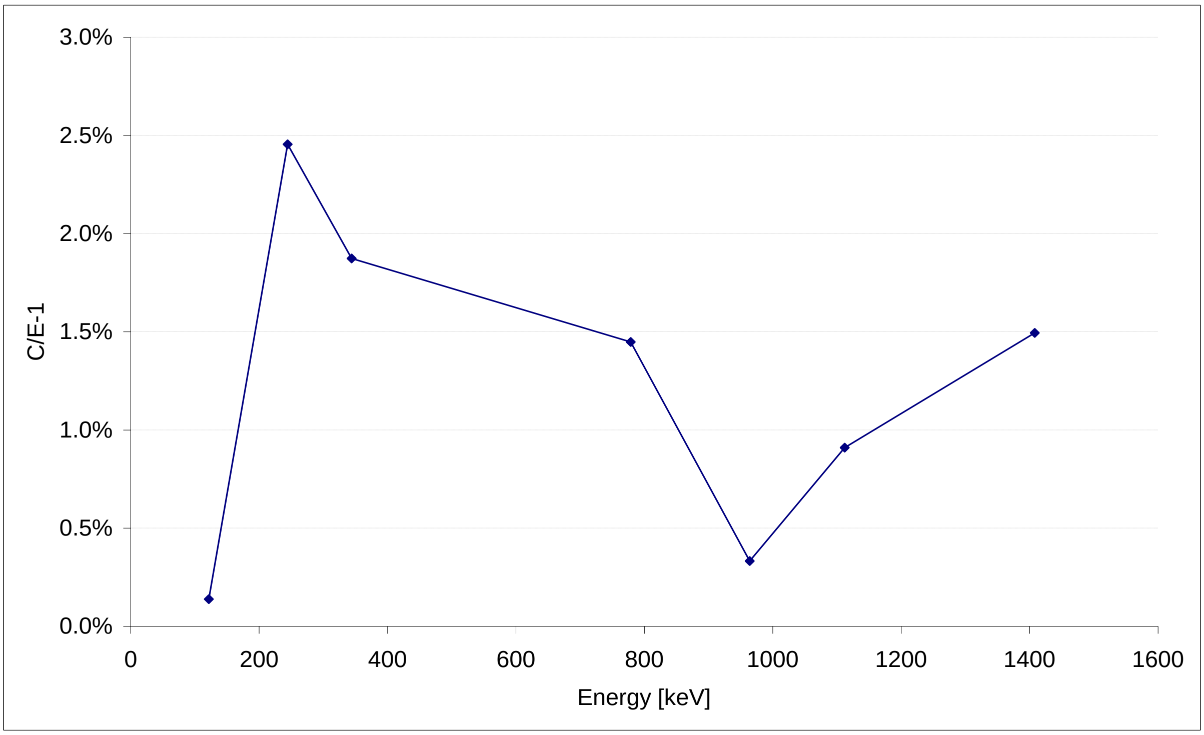

Figure 19: Validation of HPGe model with ^{252}Eu source in geometry where HPGe is out of shielding and point source is 11.5cm from detector cap (the same distance 11.5 cm is between fuel pin and HPGe End Cap).

The methodology described above was applied for the studied case of driver core surrounding the reference volume [43], [38]. The obtained results are in very good agreement with the theoretical prediction. Based on such agreement it can be concluded the fission density in the mathematical model of driver core is in good agreement with reality.

This result is important because the spatial distribution of fission density confirms the spatial distribution of neutron flux in the central cavity because neutrons in reference volume are mostly formed by transmitted or scattered fission neutrons. This validity of the mathematical model of driver core also allows using the whole core in validation issues. This is valuable in the testing of various evaluations of interesting materials. It was for example stainless steel which is important construction material [43], or silicon oxide which is important in space reactor technology or storage casks [44], because SiO_2 is the major component of earth crust and occurs in planetary rocks as well.

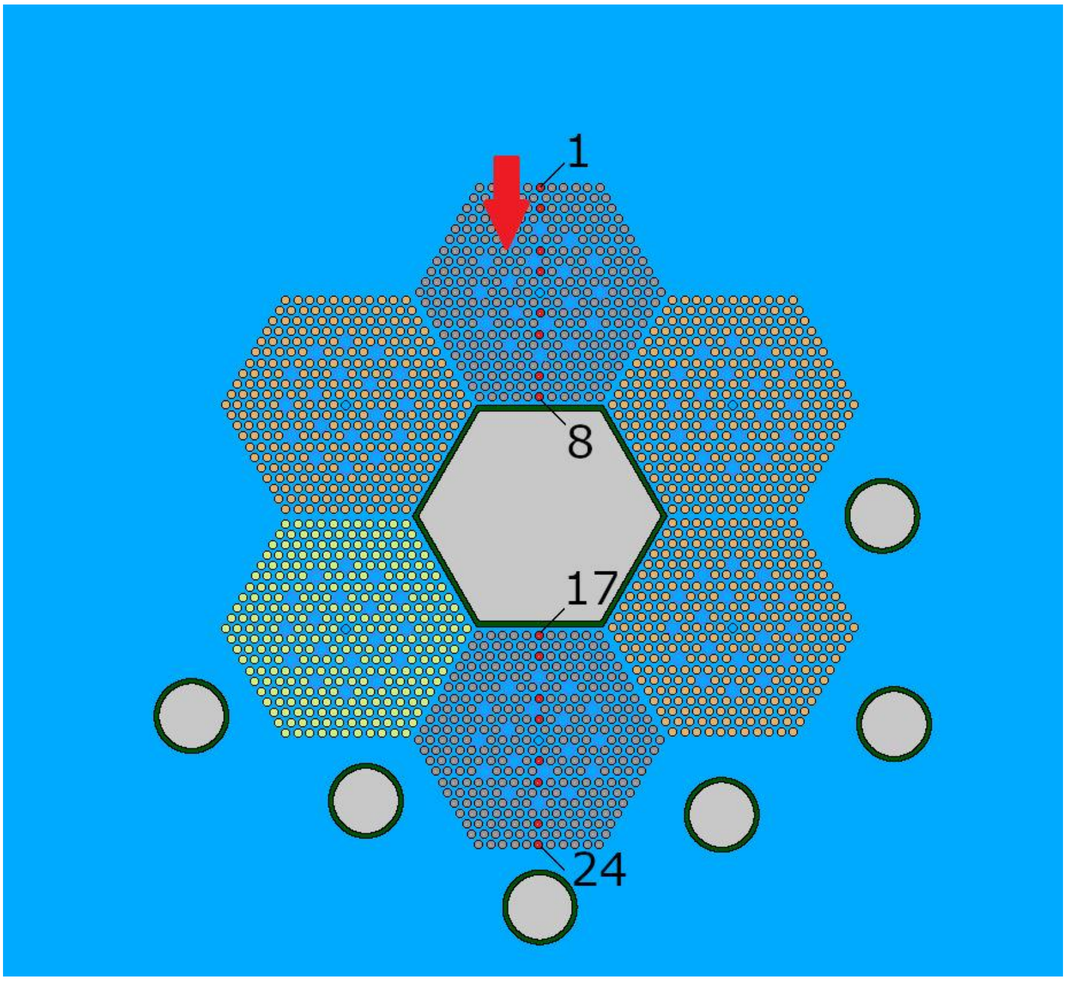

Figure 20: selection of pins where pin power was determined [43]

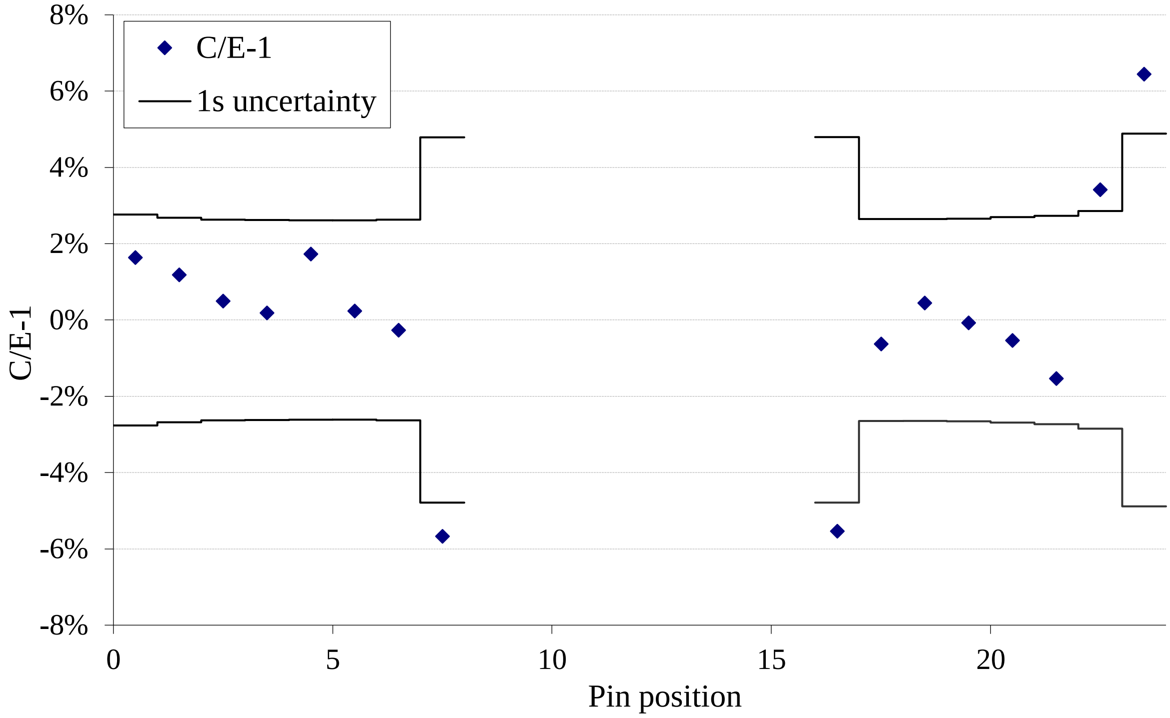

Figure 21: C/E-1 of fission density determined for selected pins [43]

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7 Reference neutron benchmark field for dosimetrical cross section measurements

Due to an extremely well defined neutron field with defined critical parameters, neutron flux spatial distribution and neutron spectra in reference benchmark neutron field and also the spatial distribution of fission density of driver core the reference field was defined [45]. Confirmation of negligible effect of (γ,n) reactions additionally to the conventional path allows to neglect this effect. Non zero production from measured reactions is only in the case of ^{197}Au(n,2n). In this case contribution of ^{197}Au(g,n) to conventional ^{197}Au(n,2n) into total production of ^{196}Au is smaller than 0.1%.

Because of the undistinguishable shape of the LR-0 reference field and ^{235}U(n_th, fiss) PFNS [16], [24] in the region above 6 MeV the set of reaction rate was evaluated as spectral averaged cross section averaged in ^{235}U PFNS. The normalization is relatively simple, its principle is taking into account the share of identical tails in both spectra. Namely, in the evaluated ENDF/B-VIII.0 ^{235}U(n_{th},fiss) PFNS, about 2.566% of total emitted neutrons have energy above 6 MeV. In the LR-0 spectrum only 0.714% of neutrons have energy above 6 MeV. Thus, for determination of the SACS, the flux which is used for normalization of calculated RR must be divided by a factor 3.594 which reflects that a large amount of thermal and epithermal neutrons is added to the part of "true PFNS".

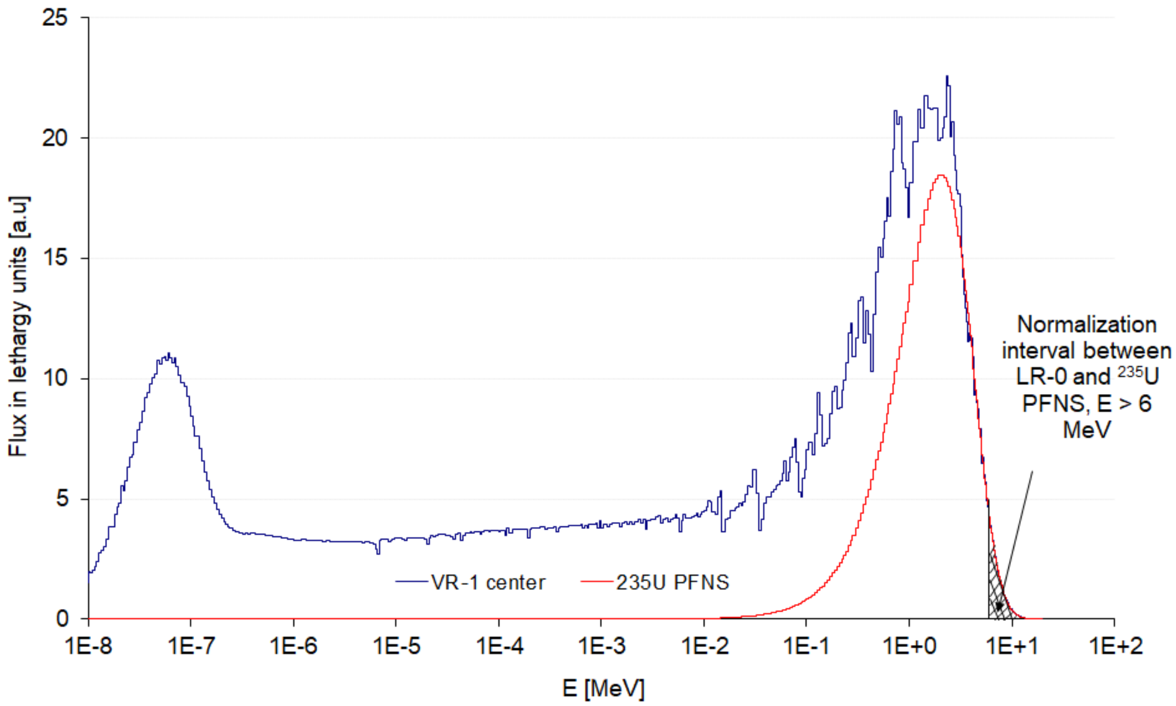

Figure 22.: Graphical interpretation of similarity in VR-1 spectra and ^{235}U(n_{th},f) PFNS and following flux normalization.

Using this approach large set of reactions were measured, and using above mentioned approach evaluated as spectral averaged cross sections averaged in ^{235}U PFNS [34], [45],[35], [46], [47], [48]. Most of this data was adopted by a part of the IRDFF-II neutron dosimetry library [49]. They are used for improving cross sections because such data are usable in nuclear data evaluation [3].

The newly developed reference field was benchmarked [5], [35] and now is part of the IRPhEP database, thus the whole experimental set is available for improving and testing new nuclear data libraries. The reference field itself becomes part of the prestigious world database of reference benchmark neutron fields under the IRDFF-II database.

The methodology developed in LR-0 was applied in the VR-1 Czech Technical University reactor. The set of data was used for testing of new neutron metrology library IRDFF-II [50]. The good agreement confirms a good correspondence of IRDFF-II with the experiment.

There was recommended a new dosimetric reaction, ^{58}Ni(n,x)^{57}Co which is valuable especially in fields with higher energy neutrons. Results obtained in VR-1 [50] and also in ^{252}Cf(s.f.) [51] validate the ENDF/B-VIII.0 evaluation for the ^{58}Ni(n,x)^{57}Co reaction. These current results are in previously reported measurements [52][53][54]. The suitable properties of this reaction make it a good candidate for the next update of the IRDFF library. This reaction will be beneficial for the reactor dosimetry field due to its long half-life and relatively high threshold makes it a good candidate for characterization of the accelerator high-energy neutron fields [55].

The reference field was also tested in combination with special filters [56]. This data will be benchmarked in the future.

7.1 Articles in collection

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8 Conclusions

Based on my experiments the reference benchmark neutron field was defined. It becomes part of the prestigious world database of reference benchmark neutron fields and now is part of the IRDFF-II database. It is a reflection of the extremely well-defined neutron field with defined critical parameter, neutron flux spatial distribution and neutron spectra in reference benchmark neutron field and also the spatial distribution of fission density of driver core. These parameters also allow applying the driver core of reference neutron field in validation of neutron transport of materials put into the core center or into reflector position. This was successfully done in the case of stainless steel and silicon oxide. These materials are essential in reactor technology thus realized experiments will help to improve their transport cross sections.

There was measured a large set of reaction rates of selected dosimetric reactions in the reference field. The reaction rates of reactions with thresholds above 6 MeV were evaluated as spectral averaged cross section averaged in ^{235}U PFNS. These new experiments realized in the newly developed reference benchmark neutron field helped to improve the dosimetric cross sections, and those data becomes a part of the IRDFF-II neutron dosimetry library. The reaction rates of reactions with a threshold below 6 MeV were evaluated as spectrum averaged cross sections averaged in LR-0 spectra and this data helps to improve the current evaluations.

The newly developed evaluation methodology was applied in the VR-1 Czech Technical University reactor. The large set of experimental data was published in the EXFOR database and expanded database for testing and improving dosimetric cross sections.

There was recommended a new dosimetric reaction, ^{58}Ni(n,x)^{57}Co which is valuable especially in fields with higher energy neutrons. Comprehensive validation in the standard field and testing in the VR-1 field makes it a good candidate for the next update of the IRDFF library. This reaction will be beneficial especially for neutron dosimetry of the accelerator formed high-energy neutron fields.

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