Book of Abstracts

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DECAY DATA AND DELAYED NEUTRON

Summation calculations for reactor antineutrino spectra, decay heat and delayed neutron fractions,involving new TAGS data and evaluated databases

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Abstract: Three observables of interest for present and future reactors depend on the beta decay properties of the fission products: the reactor decay heat (critical for safety and economy), antineutrinos from reactors (critical for non-proliferation and fundamental neutrino physics) and delayed neutron emission (critical for the operation and control of reactors).

We will present new results from summation calculations of the three quantities quoted above, performed either with evolved independent yields or cumulative yields coupled with fission product decay data, from various nuclear data bases or models. The global impact of published Total Absorption Gamma Spectroscopy (TAGS) measurements will be shown, in order to understand the progresses done in the last decade in the predictions of these observables. In addition, new TAGS results from the latest experiment of the TAGS collaboration at the JYFL facility of Jyvaskyla will be displayed as well as their impact on the antineutrino spectra and the decay heat associated to fission pulses of the main actinides.

Recent improvements of 93mNb and 103mRh activity measurement methodology and associated nuclear data for reactor dosimetry applications

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Abstract: Reactor dosimetry is based on the analysis of the activity of irradiated dosimeters, such as ^{93m}Nb and ^{103m}Rh, to derivate in particular the fast neutron flux in nuclear reactors. The activity measurement of these dosimeters is performed by X-ray spectrometry, but the low-energy of emitted photons makes it difficult to derive reliable results with low uncertainties.

Results of recent work conducted by the CEA to improve these characterisations are presented in this paper. Several approaches have been pursued in parallel, a) improvement of the efficiency calibration of HPGe detectors using both experiments and Monte Carlo simulations for a better estimation of the corrective factors for the geometry (efficiency transfer, self-attenuation, coincidence summing) and of fluorescence and self-fluorescence effects. b) Improvement of the knowledge of the ^{93m}Nb and ^{103m}Rh decay schemes. The latter required performing specific experiments: irradiation of $RhCl_3$ in the ISIS reactor followed by a complete process of activity measurements of the ^{103m}Rh solution by liquid scintillation and measurement of the absolute photon emission intensities by X-ray spectrometry. Result analysis showed the need of a complementary experiment using ^{103}Pd (under analysis) in order to try to balance the decay scheme of ^{103m}Rh, the multipolarity of the gamma transition being a major issue.

A synthesis of the main results of the work conducted since 2015 is then shown as well as a first feedback on the reactor dosimetry results in terms of uncertainty and C/E bias upgrading. Finally, outcomes are given concerning the future work to be performed for the complete analysis of the available experimental results in the aim of the evaluation of the derived nuclear decay data and of their possible inclusion in the reference nuclear data libraries.

Energy Dependence of Delayed Neutron Data

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Abstract: In order to safely work, a reactor has to be in a critical state, meaning that the productions of neutrons by fission should be compensated by the loss of neutrons due to absorption and leakage. The reactivity measures how far a reactor is from criticality, and is estimated from the measurements of the reactor period. The two quantities are linked by the Nordheim equation, which requires the knowledge of delayed neutron data. The quantities of interest are: the average delayed-neutron yield ($\boldsymbol{\nu_d}$), the kinetic parameters ($\boldsymbol{a_i}, \boldsymbol{\lambda_i}$) and the spectra ($\boldsymbol{\chi_{d,i}}$) and can be computed by summation calculations. In particular,the $\boldsymbol{\nu_d}$ and the $\boldsymbol{\chi_{d,i}}$ are needed for the estimation of the $\boldsymbol{\beta_{eff}}$, the effective fraction of delayed-neutron, the real quantity determining the margin of control of a reactor. This safety parameter is particularly important in fast reactors, where the margin to prompt criticality is smaller. The energy dependence of the $\boldsymbol{\nu_d}$ depends on the library under consideration. Mastering it is essential, especially in fast reactors where the fission rate goes from some keV to some MeV. Since the 3.1 version of JEFF library, the energy dependence of $\boldsymbol{\nu_d}$ for the most important actinides is described by the semi-empirical Lendel model.

In this work, we will investigate the improvements of such model using GEF (GEneral description of Fission observables), a code describing the observables for spontaneous fission, neutron-induced fission and, more generally, for fission of a compound nucleus form any other enrance channel, with given excitation energy and angular momentum [1]. In this work, the GEF code will be used to study the energy dependence of fission yields, which can be used, in return, to compute delayed neutron quantities by summation calculations. The energy dependence of delayed neutron data, obtained using JEFF-3.1.1's thermal fission yields and ENDF/B-VIII.0's radioactive decay data, will then be compared with experiments. Some more physical considerations (fission modes) might be taken into account.

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EVALUATION OF NUCLEAR DATA

Hybrid R-Matrix Evaluation of Neutron-Induced Reaction Cross Sections of O-16

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Abstract: The evaluation of neutron-induced reaction cross section of light nuclei is still a challenge in nuclear data evaluation. The main difficulty is the extended energy range of resonances due to the reduced level density. Thus standard evaluation techniques based on statistical model calculations can only be applied at relatively high incident neutron energies, while there is no adequate nuclear model providing a quantitative description of the resonance regime with predictive power. In this contribution we present an evaluation of neutron-induced reactions of O-16 based on the hybrid R-matrix technique. This recently developed R-matrix description is characterized by a smooth transition to the statistical model calculations which can only reasonably applied at incident neutron-energies beyond 13 MeV. Starting with an optimized statistical model description of available experimental data higher than 13 MeV, an R-matrix description of corresponding low-energy data is performed which matches perfectly in the transition region. The most important aspect of the hybrid R-matrix technique is the use of a unique matching radius in all open channels which is of physically reasonable size, i.e. larger than the range of the nuclear interaction. The latter is an important feature which represents a necessary first step towards a deeper understanding of the resonance regime.

The work was supported by grants of the Eurofusion Consortium (Materials) and the Austrian Academy of Sciences. The views and opinions expressed herein do not necessarily reflect those of the European Commission.

Testing Compound Nucleus Hypothesis Across the ^{17}O Nucleus Excitation

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Abstract: The excited compound nucleus ^{17}O* has been studied over (n,α) and (α,n) cross sections modeling, respectively for ^{16}O and ^{13}C targets in their ground states. The modeling is fulfilled within the Reich-Moore formalism. Two separate ways were used to obtain the (α,n) cross sections: a direct kinematic calculation on one side and on the other side, an inversion of the (n,α) cross section. Resonance parameters according the resolved resonance range (0 to 6 MeV) are borrowed from JEFF-3.3 file. Firstly, the modeling is carried out in the “neutron laboratory system” (using the classic frame of the projectile kinetic energy). The parameters corresponding to the resonance levels, which are provided in the neutron laboratory system in the JEFF-3.3 file, are converted in the “alpha laboratory system” for reactions where alpha particle is the projectile. In the second time, the implementation is designed in the center of mass system of the excited compound nucleus. The resonance parameters are thus converted in that reference framework. The comparison between reciprocal cross section and the one obtained through direct method enables the investigation of compound nucleus hypothesis over the studied energy domain.

Unified approach for multiband optical model in soft deformed even-even and odd-A nuclides

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Abstract: Multi-band coupling and nuclear softness were recently shown to affect substantially the calculated compound-nucleus cross sections for even-even actinides. using coupled channels method. The lack of experimental data for many nuclides motivates building a regional optical potential for all actinides allowing the optical data predictions for poorly investigated nuclides. This goal requires a unified approach for both even-even and odd-A nuclei.

A description of vibro-rotational excited bands in odd-A nuclei is proposed, assuming that the collectivity in odd nuclei is given by the collective excitations of the even-even core.

A regional optical potential is obtained using nucleon scattering experimental data for 238U and 232Th. This potential is used to calculate 233U data by fitting the nuclear deformations. 233U is chosen as an example of odd-A actinide nucleus since it has clearly identified collective rotational bands built on quadrupole and octupole vibrations of the core. The changes in predicted compound nucleus formation cross-sections due to several nuclear softness effects are analysed and compared for 233U and 238U: multiband coupling, stretching of the rotating nucleus, and volume conservation in vibrating nucleus. It is shown that these changes may reach 5-15% for incident neutron energies below 100 keV. Direct level excitation cross-sections for levels from quadrupole and octupole bands are comparable with direct cross-sections on second and third excited ground-state band levels.

Undertaken calculations demonstrate the impact of nuclear softness on calculated optical model cross sections both on even-even and odd actinides.

Study of the Photon Strength Function and Level Density in the Gamma Decay of n+234U

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Abstract: In nuclear energy applications, it is important to know the gamma flux and intensity from the sum of all nuclear reactions. This is only possible if databases are complemented by the results of nuclear reaction models. In the framework of the statistical model, the gamma de-excitation of a nucleus depends on the spin-parity, level density, and the so-called Photon Strength Function (PSF). The photon strength function of ^{235}U was studied by measuring the gamma de-excitation cascade in radiative capture reactions on ^{234}U with the Total Absorption Calorimeter (TAC) at n_TOF at CERN. The TAC is a gamma 4π calorimeter, which detects almost all the gamma rays emitted from the nucleus. This experiment provides information on gamma multiplicity and on the gamma spectra that can be compared with numerical simulations. The specialized code DICEBOXC was used to simulate the gamma cascade while GEANT4 was used for the simulation of the interaction of these gammas with the TAC materials. Simulation models and parameters can be improved and validated thanks to the data measured with the TAC. The results of this study will be presented and discussed.

Improvement of the CIELO evaluation of neutron induced reactions on natural iron

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Abstract: The validation work carried out on the CIELO/ENDF/B-VIII.0 evaluations [1,2] identified a significant problem with iron evaluation: an underestimation by up to 30% of the transmitted fast neutron flux with energies from 0.85 up to around 10 MeV. The underestimation in the outgoing neutron range from 0.85 to 4 MeV was observed both in neutron leakage measurements of 252Cf(sf) and D+T neutron sources placed inside thick iron shells (e.g., see Fig.32 in Ref.[3]). Modifications of the ENDF/B-VIII.0/CIELO iron evaluations carried out within the INDEN project are described, in particular changes to the inelastic and elastic cross sections from 0.85 up to 10 MeV of neutron incident energy. Additional modifications of the neutron capture on 54Fe and inelastic neutron scattering cross section on 57Fe in the resonance regions are discussed.

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High-resolution evaluation of the ^{235}U(n,f) cross section from 3 keV to 30 keV

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Abstract: ^{235}U neutron-induced fission cross-section is commonly used as reference for determining other isotope fission cross-section. However, below 150 keV this cross section is only included as Standard at the thermal point and recently its integral value between 7.8 eV and 11 eV [1]. The resolved resonance region, spanning up to 2.25 keV, has been re-evaluated with high resolution in the last ENDF/B-VIII release [2] and a SAMMY resonance analysis was done by L. Leal et al. [3] including the work of Paradela el al. presented at WONDER-2015 [4] in this energy range, and taken into account the IAEA Reference file. Above 2.25 keV the last ENDF/B-VIII shows a coarse energy resolution even though its fine agreement with the point-wise dataset in the IAEA Reference file, whilst JENDL-4.0 is giving only an smooth line.

The aim of this work is to combine the ^{235}U(n,f) low-background and high-resolution experimental data obtained at the CERN-nTOF facility with the ones of Weston & Todd taken at ORNL-ORELA [5], in order to produce a very fine grid (2000 bin/decade) datafile with normalisation to the IAEA Reference file. The extremely-high energy calibration required to reproduce the resonance sharp profiles is based on the outstanding nTOF Data Acquisition System with a resolution below 0.1% with reference to the 8.78 eV resonance and to the sharp Al(n,g) capture deep at 5.904 keV (20 eV FWHM).

The comparison of the so-evaluated profile with the experimental data and with the evaluated ones will be discussed.

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Study of the (n,γf) process on ^{239}Pu

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Abstract: In the neutron resonance range, fission cross section of 239Pu can be seen as a sum of the "direct" fission and (n,γf) reactions. In that case, five channel widths should be considered for a proper evaluation, those are: two opened fission channels for Jπ=0+, one opened fission channel for Jπ=1+ and two J-dependent for the (n,γf) reaction. The evaluated file of the fission cross section considers the Jπ=1+ resonances as being produced only through direct (n,f) reactions, reflecting the sizeable contribution of the (n,γf) process. This should have an impact on the determination of the capture and fission widths involved in the Reich-Moore approximation of the R-matrix theory. The present work aims to investigate this impact by using the CONRAD code and the Γγf from Bouland, in comparison with the work from Trochon. Prompt neutron multiplicity (νp) has been also reproduced including the contribution of the (n,γf) process.

Statistical Multistep Analysis of Neutron Activation Cross Sections of Aluminum used as Cladding in Miniature Neutron Source Reactors

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Abstract: Theoretical calculations has been carried out for the stable isotope of Al to predict the (n,p), (n,2n) and (n,α) activation cross sections completely from threshold up to 20 MeV using EXIFON 2.0 which is a global and standard minimum model description without explicit account for higher-order effects. The results provide adequate data for application within the reactor energy spectrum, compares fairly with recent measurements and evaluations. The observed differences give a measure for deviation from an average nuclear behavior for the isotope within the energy range studied with the possibility of further improvement if higher-order effects are accounted for.

MICROSCOPIC AND INTEGRAL MEASUREMENTS

Trends on major actinides from an integral data assimilation

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Abstract: Nuclear data uncertainties on major actinides as they stand in current nuclear data libraries still do not meet Gen IV core requirements. Integral data assimilation can contribute to their improvement if attention is being given to ways that minimize the possibility of creating compensating errors. Parametric studies have been done so as not be too dependent to the nuclear data covariance data that are still perfectible or even missing. Marginalization technique has been used for light and structural isotopes for which approximations in the integral data assimilation technique are rather high.
Integral measurements with reliable experimental techniques have been selected. They are ICSBEP, IRPhE and MASURCA critical masses, PROFIL irradiation experiments and the FCA-IX experimental programme (critical masses and spectral indices). Highly reliable calculation analyses are possible with the use of as-built geometries calculated with the TRIPOLI4 Monte Carlo code. The C/E values have uncertainties (experimental and modelling ones) are much smaller than current nuclear data uncertainties calculated using appropriate nuclear data covariance matrices and sensitivities. They can hence be used in an integral data assimilation with highly profit with the CONRAD code solving the Bayes equation.
The trends on the JEFF3.1.1 ^{235}U capture cross section are quite consistent with recent differential measurements. The information comes from the simultaneous use of PROFIL sample irradiation analyses and various ^{235}U enriched critical masses.
Assimilation results suggest also a 2.5% decrease for ^{238}U capture from 3 keV to 60 keV, and a 4-5% decrease for ^{238}U inelastic in the plateau region. For this energy range, uncertainties are respectively reduced from 3-4 to 1-2% and from 6-9% to 2-2.5% for ^{238}U capture and ^{238}U inelastic. The simultaneous use of GODIVA and FLATTOP-^{235}U and JEZEBEL ^{239}Pu and FLATTOP-^{239}Pu is relevant to reassess ^{238}U inelastic cross sections, as their critical mass C/E are very sensitive to this cross section as the reflecting properties of the depleted Uranium blankets of the FLATTOP cores.
The increase trend on ^{239}Pu capture cross section of around 3% in the [2 keV-100 keV] energy range is driven by an underestimation of the PROFIL ^{240}Pu/^{239}Pu ratio C/E.
For ^{240}Pu capture cross section, the increase is of around 4% in the [3 keV-100 keV] energy range and is suggested by PROFIL C/E. This trend goes in the same direction as the recent ENDF/B.VIII evaluation though at a much lower level.

Total Monte Carlo acceleration for the PETALE experimental programme in the CROCUS reactor

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Abstract: The PETALE experimental programme intends to improve neutron cross sections in the MeV energy range for heavy reflector elements of PWRs. It mainly consists in a transmission experiment through stainless steel and elemental metal plates interleaved with thin dosimeter foils in the reflector of the CROCUS reactor. In this frame, uncertainty propagation and data assimilation are required in order to optimise the experimental setup. This article will present the methodology developed using the Total Monte Carlo (TMC) approach and a modified version of the Serpent2 Monte Carlo code. The choice of the TMC approach has been done to study the applicability of a Bayesian Monte Carlo approach with the TENDL random cross sections library for the data assimilation foreseen on this experimental configuration.

Due to the low neutron flux in the dosimeters, the two main difficulties encountered concern the convergence of the reaction rate estimation in the dosimeters, and the large number of independent calculations required by the TMC. Specific developments have been implemented in Serpent2 using variance reduction and Correlated Sampling (CS) technics. This latter allows to propagate simultaneously different random neutron cross section files. This TMC-CS methodology will be detailed, and compared to reference TMC and sensitivity-based uncertainty propagations. Results obtained in the case of the uncertainty propagation of iron to the reaction rates in the dosimeters of the experimental setup will be discussed. For comparison purposes, an application to the hmi-001 benchmark will also be discussed. It consists of a highly enriched uranium/iron metal core reflected by a stainless-steel reflector. This system is characterized by a high dependence of the iron uncertainty on the k_{eff}, highlighting the discrepancies between the different approaches and their respective advantages.

An Experimental Programme optimized with Uncertainty Propagation: PETALE in the CROCUS Reactor

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Abstract: The PETALE experimental programme in CROCUS intends to contribute to the improvement of neutron cross sections in the MeV energy range for stainless steel, particularly in the prospect of heavy reflector elements of PWRs. It mainly consists in several transmission experiments, first through nuclear-grade stainless steel, and separately through its elemental components of interest – iron, nickel and chromium. The metal reflector is composed of successive metal plates set in a box in air, and interleaved with activation foils. In addition, reflectors' reactivity worth estimates, in-core spectra characterisations, and plate activation measurements, when possible, will be performed.

Because of the opposition between the low neutron flux in the dosimeters in the range of interest, requiring irradiations of maximal power and duration, and the operational constraints of the teaching activities at the EPFL CROCUS facility, an optimization of the experimental programme was carried out. Reaction rates of interest were maximised, thus sensitivity to target nuclear data, while integrated power was minimized. It called also for an estimation and a reduction of technological uncertainties in the setup design. This work was performed by propagating uncertainties using the Total Monte Carlo (TMC) approach with random ACE files from the TENDL 2017 library and the Serpent2 Monte Carlo code. In this article, the final design of the experimental programme optimized with this methodology is presented, starting from in-core devices design, to dosimeters materials and target uncertainties selection, and finally operation prospects.

Integral experiments for bismuth nuclear data validation at the fast VENUS-F reactor

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Abstract: VENUS-F is a fast zero-power reactor that serves as a mock-up of the MYRRHA fast reactor/accelerator driven system, which will use lead-bismuth eutectic as coolant and spallation target. VENUS-F uses solid lead and/or bismuth as coolant simulator and reflector. This paper describes an experimental campaign dedicated to bismuth data validation carried out in a core configuration where bismuth was used as a coolant simulator in all the fuel assemblies (in total 625 kg of high-purity Bi). Global (criticality, kinetic parameters) and local parameters (fission rates distributions, rod worth, Bi void, fuel temperature effects) were measured. For selected parameters, comparison between the experiment and MCNP calculations using various versions of the JEFF, JENDL and ENDF nuclear data libraries are presented. Special attention is given to the impact of the delayed neutron data (6- or 8-group precursors) applied in the experimental data analysis on the measured reactivity effects.

Testing of resonance parameters of actinides

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Abstract: The quality of the resonance parameters of actinides contained in evaluated data files has been tested in the energy range from 0.5-100 eV. The transmission factors of well characterized samples have been measured and the isotopic fractions in the material have been determined. Two sets of samples were investigated, one set containing predominantly uranium the other on plutonium isotopes. For both sets the isotopic ratios are certified to a high precision.

The measurements were done at GELINA, the neutron time-of-flight facility operated by the Joint Research Centre in Geel. This facility is based on an electron linear accelerator, in which electrons are accelerator up to energies of 150 MeV. The electron beam is steered on mercury-cooled uranium target, in which bremsstrahlung is produced and subsequently neutrons are emitted. Measurements stations, located at distances ranging from 5 to 400 m, are used for measurements of total and partial cross section. When performing transmission measurements in the epi-thermal energy range, the samples are inserted in the neutron beam at a distance of approximately 7 m, while the neutron passing through the samples are detected at a flight path distance of 12 m, the presented experiment employed a Li glass scintillator.

For the analysis the resonance shape analysis code REFIT was used to determine the isotopic fraction for a given sample. The deviation of the determined isotopic fraction from the certified value will provide information on the quality of the resonance parameters used in the analysis region. The performance of different evaluated data files can be compared.

Inelastic scattering on 232Th: measurements and beyond

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Abstract: In order to increase the reliability of reactor simulation tools, accurate nuclear data are mandatory. Especially, reactions induced by fast neutrons, comparatively less studied, are of utmost importance in fast neutron reactor. Among these reactions, neutron inelastic scattering is important for criticality of fast neutron reactors, power distribution in reactor core, etc. In addition, (n,2n) and (n,3n) reactions are important as well. We aim to increase cross section accuracy for these three reactions, thereafter collectively referred as (n,xn) reactions, in an energy interval ranging from reaction threshold to 20 MeV. Isotope studied is ^{232}Th, a potential constituent of nuclear fuel.
Cross sections of (n,xn) reactions on ^{232}Th are measured by means of prompt gamma spectroscopy associated with time of flight measurements. It consists in detecting γ-rays emitted over the course of the de-excitation of residual nucleus after (n,xn) reaction occurred. Pulsed neutron source used is produced by the GELINA facility located at JRC Geel, Belgium. Experimental setup includes four planar HPGe counters for gamma spectroscopy, and one fission chamber monitoring neutron flux through the target. Accurate measurements of these γ-rays production cross sections, thereafter referred as (n,xnγ) cross sections, should constrain the description of reaction mechanisms enough to obtain accurate (n,xn) cross sections.

Gamma production cross sections have been obtained for 81 transitions of ^{232}Th from (n,n'γ), 11 of ^{231}Th from (n,2nγ) and 7 of ^{230}Th from (n,3nγ). A lot of care is taken in the determination of all sources of uncertainty and their correlation. Total uncertainty on cross sections range from 3 to 20% depending on γ-rays intensities. Collaboration work is in progress to test and constrain theoretical predictions, using these measurements.
In this presentation, I will begin to present principles and limitations of our experimental method. Then, the uncertainty determination approach followed here will be explained, as it would be crucial for evaluation or model optimization stemming from our measurements. Obtained cross sections will then be compared to prior experiment and calculations from the last version of reaction code TALYS. Finally, future work complementing our measurements, in progress or considered, will be mentionned.

Fission program at n_TOF

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Abstract: Since its start in 2001 the n_TOF collaboration developed a measurement program on fission, in view of advanced fuels in new generation reactors. A special effort was made on measurement of cross sections of actinides, exploiting the peculiarity of the n_TOF neutron beam which spans the hugest energy domain, from the thermal region up to GeV. Moreover fission fragment angular distributions have also been measured.I will present an overview of the cross section results achieved with different detectors, with a special focus on the 237Np case where discrepancies showed up between different detector systems. The results on the anisotropy of the fission fragments and its implication on the mechanism of neutron absorption, and in applications, will also be discussed.
Finally I will show the perspectives in the near future, in particular the need of an absolute reference cross section in the high energy domain, and the refurbishment of the spallation target which will deliver a higher quality intense beam at the vertical short distance station.

Preliminary results on the 233U neutron-induced capture cross section measured at the n_TOF facility at CERN with the fission tagging technique

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Abstract: 233U is of key importance among the fissile nuclei in the Th-U fuel cycle. A particularity of 233U is its small neutron capture cross-section, which is on average about one order of magnitude lower than the fission cross-section. Therefore, the accuracy in the measurement of the 233U capture cross-section depends crucially on an efficient capture-fission discrimination, thuss a combined set-up of fission and γ-detectors is needed. A measurement of the 233U capture cross-section and alpha ratio was performed at the CERN n_TOF facility. The Total Absorption Calorimeter (TAC) of n_TOF was employed as γ-detector coupled with a novel compact ionization chamber as fission detector. A brief description of the experimental set-up will be given, and the analysis procedure, including extensive Monte-Carlo-simulations, as well as preliminary results of the 233U cross sections are presented and discussed.

Measurement of the ^{244}Cm and ^{246}Cm neutron-induced capture cross sections at the n_TOF facility.

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Abstract: The neutron capture reactions of the ^{244}Cm and ^{246}Cm isotopes open the path for the formation of heavier Cm isotopes and of heavier elements such as Bk and Cf in a nuclear reactor. In addition, both isotopes contribute significantly to the decay heat and to the neutron emission in irradiated fuels proposed for the transmutation of nuclear waste.

There are only two previous experimental data for these two neutron-induced capture cross sections, one measured in 1969 using a nuclear explosion [1] and the most recent data measured at J-PARC [2]. The data for both isotopes are very scarce due to the difficulties in performing the measurements: high intrinsic activity of the samples and limited facilities capable of providing isotopically enriched samples. We have measured both neutron capture cross sections at the n_TOF Experimental Area 2 (EAR-2) with three C_6D_6 detectors, using the same ^{244}Cm and ^{246}Cm enriched samples as in J-PARC. An additional measurement has also been performed at the n_TOF Experimental Area 1 which has a larger flight path and less neutron fluence,with the n_TOF Total Absorption Calorimeter, which will be used to obtain a better normalization and to cross check the results of the measurement in EAR-2.

Preliminary results assessing the quality and limitations (background subtraction, measurement technique and counting statistics) of this new experimental dataset are presented and discussed.

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Towards the adoption of $^{238}U(n,f)$ and $^{237}Np(n,f)$ as primary standards for fast neutron energies

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Abstract: Modelling of Generation-IV nuclear power plants requires highly accurate values of cross sections in the fast energy region. Cross section measurements are usually performed relative to the primary standard $^{235}U(n,f)$, but in environments where the thermal and epi-thermal neutron background is non-negligible, other isotopes with a fission threshold should be preferred. Up to now, none of the isotopes with a fission threshold has been considered a primary standard. Two isotopes, for which the experimental data-base is sufficient, could potentially fill this role: $^{238}U(n,f)$ and $^{237}Np(n,f)$. $^{238}U(n,f)$ has a fission threshold at around E_n=1.6 MeV and is a secondary standard from E_n=2 MeV. Although the JEFF 3.2 evaluation shows discrepancies up to 7% in the range 1.5 MeV < E_n < 5 MeV with respect to ENDF/B-VIII.0, the new JEFF 3.3 evaluation is in agreement with ENDF/B-VIII.0. $^{237}Np(n,f)$ would be more suitable as a standard because of its lower fission threshold (E_n=0.5 MeV) and its higher cross section above E_n=1.0 MeV, but some discrepancies between measurements have been found in the last years.

To address these issues, two experimental campaigns have been performed at the Van de Graaff accelerator of the National Physical Laboratory (NPL,UK) under the European CHANDA project. The Nuclear Metrology group at NPL is known worldwide as being capable to provide very accurate and precise neutron flux measurements by using a well characterized long counter. It also benefits from a large low-scatter target hall ideal for cross section measurements. A twin Frisch-grid ionization chamber was used as fission fragment detector. Measurements were done absolutely, by using the long counter, and relatively, by placing two samples in a back-to-back configuration in the fission chamber. The first campaign was performed in January 2016 and the isotopes measured were $^{235}U(n,f)$, $^{238}U(n,f)$ and $^{237}Np(n,f)$. For the second campaign, performed in January 2017, measurements were performed for $^{235}U(n,f)$, $^{237}Np(n,f)$ and $^{242}Pu(n,f)$.

Results from both campaigns will be presented with emphasis on the data analysis and the uncertainty evaluation.

Development of a gaseous proton recoil detector for fission cross section measurements below En = 1MeV

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Abstract: One of the main challenges in the measurement of neutron induced fission cross sections is the neutron flux determination. Typically well known cross sections such as 235,238U(n,f) and 237Np(n,f) are used as standard to normalize the measured fission rate. The elastic 1H(n,p) reaction is another standard reaction. It provides a cross section accuracy of 0.5% over a wide energy range, and it requires a completely different detection setup. Normalization to (n,p) reaction is commonly performed with Si detector for neutron energies of few MeV. However, it is rather challenging for neutron energies below 1 MeV due to an intense background in the Si detector. A new proton recoil detector, based on a segmented Micromegas detection plane, have been developed at CENBG. A prototype has been built and test experiments have been carried out at the AIFIRA facility. It shows a good behaviour of the detector under irradiation, with a very low sensitivity to parasitic background. Timing properties allows a 3D track reconstruction, and recoil protons with energies as low as 200 keV has been detected.

NUCLEAR FISSION, INCLUDING PROMPT PARTICLE EMISSION AND FISSION YIELDS

Systematic measurements of prompt fission γ-rays – and what they tell us about fission fragment de-excitation

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Abstract: Nuclear fission is a complex process, which is still not fully understood – after almost 80 years since its discovery. One field of research is for instance studies of the de-excitation process of fission fragments, which in the early stages, i.e. within a few nanoseconds after scission, takes place through the successive emission of prompt neutrons and gamma rays. For nuclear applications, information about the prompt neutrons is crucial for calculating the reactivity in reactors, while precise knowledge about the prompt gamma rays is important for the assessment of the prompt heat released in the reactor core. Concerning the latter we have contributed in the past years with a number of precise measurements of prompt gamma-ray spectra from various fissioning systems. From those we determined average characteristics like multiplicity, mean energy per photon and total gamma-ray energy released in fission.

The obtained results were investigated for their dependences of mass and atomic numbers of the fissioning system as well as the dissipated excitation energy. The purpose of this endeavor was to find a description that allows predicting prompt gamma-ray spectra characteristics for cases that cannot be studied experimentally.

In this talk we will give an overview on the latest measurements of prompt fission gamma ray spectra. We will also present first results from a recent angular correlation measurement between these gamma rays and fission fragments from the spontaneous fission of ^{252}Cf and infer what can be learned from the observed angular distributions. For instance, the relative contributions of dipole and quadrupole photons were deduced and compared to results of very recent calculations with the Monte Carlo Hauser-Feshbach code FIFRELIN, developed at CEA Cadarache.

Fission-fragment dependent prompt and isomeric gamma-ray emission in nuclear fission

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Abstract: Precise knowledge about prompt gamma-ray emission in nuclear fission is important for the assessment of the prompt heat released in a reactor core. During the last years we have contributed with a number of precise measurements of prompt gamma-ray spectra from various neutron and particle induced fission reactions as well as spontaneous fission decay. From those we determined average characteristics like multiplicity, mean energy per photon and total gamma-ray energy released in fission.
In a next generation of experiments we are investigating prompt fission gamma-ray spectral (PFGS) characteristics as a function of fragment properties, as e.g. mass and total kinetic energy, and also the time-dependence of PFG emission relative to the moment of fission. Together with the competition between prompt neutron and gamma-ray emission our data might turn useful to understand the de-excitation process of fission fragments in more detail and, therefore, may help testing fission model calculations.

I will give an overview about our present activities and the contemplated experiment programme.

Performance validation of the FALSTAFF first arm: 252Cf and 235U fission fragment characterization

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Abstract: The renewed interest for the study of nuclear fission is mainly motivated by the development of GEN-IV reactor concepts, mostly foreseen to operate in the fast neutron energy domain. To support this development, new high quality nuclear data are needed. The incoming Neutrons For Sciences (NFS) facility, built in the framework of the SPIRAL2 project at GANIL (Caen, France), will open new opportunities to measure these data. High intensity neutron beams will be produced over a large domain in energy, from some hundreds of keV up to 40 MeV. In this context, a new experimental setup, called FALSTAFF, dedicated to the study of fission is under development.

The FALSTAFF setup will permit to study the fission of actinides in the fast energy domain. This two-arm spectrometer will detect both fragments in coincidence and will measure their time-of-flight (TOF) and their kinetic energy. Thanks to the 2V and EV methods, the fission fragment mass before and after neutron evaporation will be deduced and the correlation between neutron multiplicity and fragment mass will be determined. The first step of the FALSTAFF project, consisting in building one arm of the spectrometer development, has been reached. This first arm is composed of two SED-MWPC detectors (combination of a foil to produce Secondary Electrons and a Multi-Wire Proportional Chamber to detect them) and an axial ionization chamber. The SED-MWPC give access to the velocity (V) thanks to measurements of time-of-flight and positions. The ionization chamber measures the fragment kinetic energy (E) and the energy loss profile.

In this paper, the FALSTAFF setup will be described. 252Cf fragment kinetic energy and velocity distributions obtained with the first arm of FALSTAFF will be presented. Results of the mass determination will be shown. We will compare our data with simulations and data from the literature. Preliminary results from the 235U thermal neutron-induced fission experiment performed at the Orphée reactor will be presented.

Prompt fission neutron investigation in $^{235}U(nth,f)$ reaction

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Abstract: The prompt neutron emission in thermal neutron induced fission of 235U and 252Cf(sf) has been investigated applying digital signal electronics. Using a twin Frisch grid ionization chamber for the fission fragment detection and a NE213 equivalent neutron detector in total about 106 neutron coincidences have been registered. The fission fragment kinetic energy, mass and angular distribution has been investigated along with prompt neutron time of flight and pulse shape using a six channel synchronous waveform digitizer with sampling frequency of 250 MHz and 12 bit resolution. The signals have been analyzed using digital pulse processing algorithms, developed by authors. Experiments in $^{252}Cf(sf)$ was performed in 2007-2008 in EC-JRC-IRMM and was already reported in ND2010. In this research we reanalyzed data, of IRMM experiment using the same data analysis software, where we have implemented recent achievements in fission fragment spectroscopy with Frisch-gridded twin ionisation chamber.

Status of fission fragments observables measured with the LOHENGRIN spectrometer

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Abstract: Nuclear fission yields are key parameters to evaluate reactor physics quantities, such as decay heat, criticality or spent fuel radiotoxicity, and for understanding fission process. Despite a significant effort allocated to measure fission yields during the last decades, the recent evaluated libraries still need improvements in particular in the reduction of the uncertainties. Additional kinds of measurements provide complementary information in order to test the models used in the nuclear data evaluation. Moreover, some discrepancies between these libraries must be explained.
A common effort by the CEA, the LPSC and the ILL aims at tackling these issues by providing precise measurement of isotopic and isobaric fission yields with the related variance-covariance matrices. Nevertheless, the experimental program represents itself a large range of observables requested by the evaluations:

- Complete range of isotopic distribution per mass allows the determination of the charge polarization which have to be coherent between complementary masses (pre-neutron emission)
- Odd-even effect and Kinetic energy dependency of isotopic yields in order to test the sharing of total excitation energy;
- Ionic charge distributions are indirect measurements of ns isomeric ratio as a probe of the nuclear de-excitation path in the (E^*, J, π) representation.

Measurements for thermal neutron induced fission of ^{241}Pu have been carried out at the ILL in Grenoble, using the LOHENGRIN mass spectrometer. Methods, results and comparison to models calculations will be presented corresponding to a status on fission fragments observables reachable with this facility.

Computational Approaches to Whole Process of Nuclear Fission

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Abstract: Nuclear fission is a unique large-amplitude collective motion of nuclei which gives rise to a transition from a single to two (or rarely three) nuclei. In spite of a long history of research, however, understanding of the fission mechanisms, especially the part from the compound nucleus to scission point, is still very poor. It is poor also in terms of the fact that predictions of the nuclear theory is not accurate enough to be used in application fields.

We have been doing a comprehensive approach to understand and describe the whole process of nuclear fission, starting from the compound nuclei, evolution of elongation leading to scission, emission of prompt neutrons and finally beta decay of fission products. In the first part when elongation evolves from the compound nucleus, we use Langevin model as well as microscopic theories. Much insights into the mechanisms of nuclear fission have been accumulated with these theories. The emission of prompt neutrons is described by Hauser-Feshbach decay model. That information is connected to beta-decay by either using decay data file or gross theory of beta data, and the decay heat and reactor antineutrino problems are also touched. We wish to present our resent progress and discuss problems we are facing with and ways to go.

Prompt and Delayed Neutron Emission and Fission Product Yield Calculations with Hauser-Feshbach Theory and Beta Decay

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Abstract: The fission product yields, prompt and β-delayed neutron and photon emissions,and the decay heat are key ingredients for studying nuclear reactor applications.

Essentially study of such applications requires a comprehensive set of nuclear data that are completely consistent each other. However, these nuclear data are often compiled separately based on experimental data or some model assumptions practically.
We have developed a Hauser–Feshbach Fission Fragment Decay model, HF3D, which can be applied to the statistical decay of more than 1,000 primary fission fragments formed by the neutron induced fission of 235U. Those primary fission fragments emit prompt neutrons and photons to reach their ground states or long-lived isomeric states. In order to calculate the prompt neutron and photon emissions, unlike the Monte Carlo simulations for sampling the primary fission fragment distributions, i.e. mass, charge, excitation energy, spin and parity (*Y(A, Z, Eex, J, π)*), the HF3D model numerically integrates the distributions for all of 1,000 primary fission fragments and calculates the prompt neutron multiplicities, independent fission product yields and isomeric ratios, which are fully consistent each other. The fission products then β-decay, which leads to the β-delayed neutron and photon emissions. We combine the β-decay calculation with the HF3D model calculation to obtain the cumulative fission product yields, decay heat and delayed neutronyields. The calculated fission observables are compared with available experimental data.

Advances in the Simulation of Correlated Fission Data

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Abstract: The study of prompt fission neutron and photon observables has received much attention in recent years thanks in large part to the development of codes that simulate the decay of primary fission fragments on an event-by-event basis. Often limited to average quantities such as the average prompt fission neutron spectrum (PFNS) and average prompt fission neutron multiplicity (ν), previous physics models and tools were unable to account for the complex correlations that exist between all particles ejected in a fission event.
The CGMF code [1], under development at Los Alamos, follows the decay of all fragments produced in a particular fission reaction and in various initial conditions of mass, charge, kinetic energy, spin and parity, by computing the probabilities of emitting a neutron or a γ-ray at each stage of the decay until a stable configuration is reached. It has now been incorporated [2] in the MCNP transport code to allow for the simulation of correlated fission observables in any experimental setup.
In this talk, I will review some recent advances implemented in CGMF and applied to specific problems: Role of fission fragment isomers in the determination of γ-ray multiplicities [3]; Influence of the entrance channel to γ-ray observables [4]; Use of fission fragment yields calculated in the macroscopic-microscopic approach in CGMF calculations [5]; Consistent study of fission yields and prompt and β-delayed neutrons and γ rays [6,7].

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Constraining Fission Yields Using Machine Learning

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Abstract: Having accurate measurements of fission observables is important for a variety of applications, ranging from energy to non-proliferation and defense to astrophysics. However, not all of these data can be measured, whether due to the sheer number of observables this encompasses or the feasibility of the studies that must be performed. Therefore, it is important to be able to accurately calculate these observables, both where data exists and where it presently does not. Several codes exist to perform these calculations, including CGMF, a Monte Carlo code developed at Los Alamos that follows the decay of each fission fragment in order to calculate any prompt fission observable of interest.

A variety of models must be included in order to properly describe the fission fragment yield distributions and subsequent decay paths, the first of which is the description of the fission fragment yields in mass, charge, kinetic energy, spin, and parity. While many microscopic and macroscopic models are being developed with reasonable success, phenomenological models are still widely used, including within the current implementation of CGMF. Within these models, parameters are often optimized for each nucleus of interest, which allows the models to describe the data well but does not provide a means to make accurate predictions outside of this set of nuclei.

In this work, we exploit probabilistic and Bayesian techniques, using Neural Networks, to optimize the current parameterizations for these phenomenological yield models. We begin with spontaneous fission of Cf-252, where there is abundant experimental data, in order to understand the method, its challenges, and results, before applying these techniques to construct a global yield model. The new yield models will be used to improve the current calculations for observables such as average neutron multiplicity as well as higher moments of the multiplicity distribution, which can be validated against experimental data. With global yield models, we will also be able to predict these quantities for systems where data does not exist.

Residual temperature distributions and systematic behaviours of residual quantities following the sequential emission of prompt neutrons

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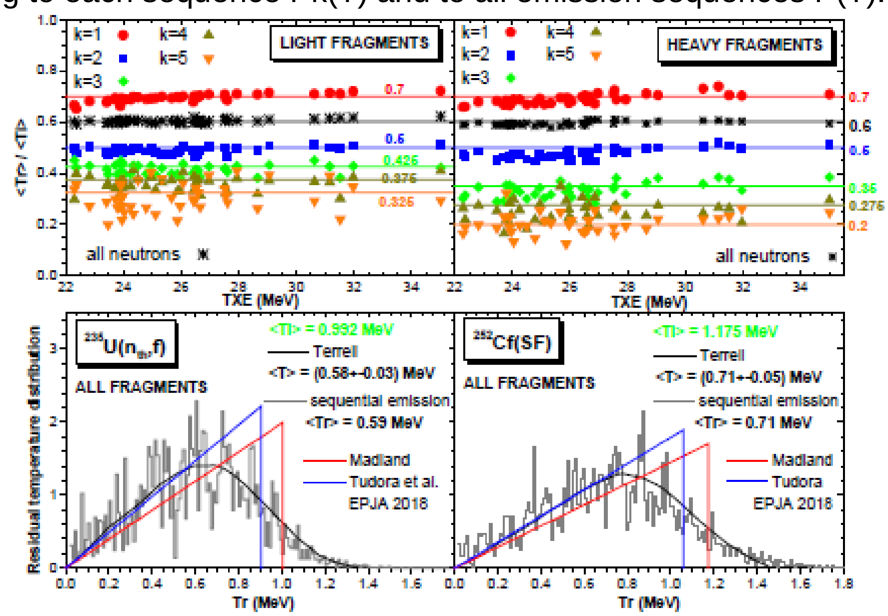
Abstract: A deterministic treatment of sequential neutron emission based on recursive equations of residual temperatures [1] was applied to numerous cases of spontaneous and neutron induced fission. This emphasizes systematic behaviours and correlations between average quantities of initial and residual fragments. An example (upper part of the figure below) is the constant ratio of the average residual temperature following the emission of each neutron (indexed k) and of all neutrons (black stars) to the initial temperature, irrespective to the prescriptions used for the compound nucleus cross-sections and the level density parameters. This finding (allowing to express the average residual temperatures only as a function of the initial temperature) leads to the determination of general triangular forms of the residual temperature distributions corresponding to each sequence Pk(T) and to all emission sequences P(T).



These distributions are used in the prompt emission models with a global treatment of the sequential emission (e.g. Los Alamos and Point-by-Point). Such P(T) are illustrated with blue lines in the lower part of the figure, together with P(T) resulting from the sequential emission treatment of Ref.[1] (grey lines), the P(T) of Terrell [2] (black lines) and the triangular form of Madland and Nix (red lines).

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Advanced Modelling of ^{238}U(n,f) in a Fast Reactor Application

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Abstract: Fast neutrons reactors based on fission process of ^{238}U, as a possible future solution on energy demand of the human society, request new and reliable nuclear data necessary for the new generation reactors design. In this respect, a detailed modelling and theoretical predictions on the ^{238}U(n_{fast},f) reaction induced by fast neutrons were performed. Fission cross sections, charge and mass distribution of fission fragments, prompt neutron emission, isomer's ratios and corresponding uncertainties were calculated using Talys computer code or programs realized by authors. Production of isotopes of interest like ^{133}Xe, ^{99}Mo, and ^{131}I, as well as yields of fissile nuclei were evaluated as well. The detailed input and results of the ^{238}U(n_{fast},f) modelling provide an opportunity to gauge the use of the author's computer code in a fast reactor application. The nuclear data obtained by using the advanced modeling developed through the present research on ^{238}U - fuelled fast reactor were compared and validated by the very good agreement with the existing experimental values. The present investigations on fast neutron induced fission of ^{238}U fuel in the frame of nuclear data research program of JINR Dubna basic facilities were effectuated.

PROCESSING AND BENCHMARKING

Current status of the verification and processing system GALILÉE-1

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Abstract: GALILÉE-1 is the new verification and processing system for evaluated data, developed at CEA.
This paper describes the current status of GALILÉE-1. Three main components are under development.

- GALION component (GALilée Input Output for Nuclear data) is dedicated to read (resp. write) input (resp. produced) data. Input format can be ENDF-6 or GND format while output format is GND format.
- GALVANE component (GALilée Verification of the Accuracy of Nuclear Evaluations) is dedicated to verify nuclear evaluations that are GALILÉE-1 input data. This verification is achieved by comparing structure data found in the evaluations with reference ones given in NUBASE or ENSDF for instance.
- GTREND component (GALilée TReatment of Evaluated Nuclear Data) is dedicated to provide continuous-energy and multigroup data as well as probability tables.

All these components are written in C++ language and share the same objects which hierarchy is very close to the GND object hierarchy.
Nowadays, GALION can read evaluated data in ENDF-6 format. The new GNDS format is being supported from the recent ENDF/B-VIII library. All the physical data are stored in C++ objects, named GBASE objects, independent of the evaluation format. GALVANE and GTREND components only work on these GBASE objects, which allows the same verification and processing stages, whatever the evaluation format is. GALVANE can diagnose inconsistencies in general information, resonance parameters, Q reaction values, thresholds, excited level schemes, kinetic data of emitted particles, thermal scattering laws. GTREND can reconstruct continuous energy cross-sections in the resolved resonance range, provide a linearization grid, broaden exact or linearized cross-sections and calculate moment based probability tables. In the unresolved energy range, GTREND has the capability to calculate the average cross sections and to generate random resonance parameters. We will present the first results of the probability table calculations in this domain.
At each processing step cross-comparisons are made with NJOY, CALENDF or PREPRO codes. Some results of such comparisons are provided.

Nuclear data analyses for improving the safety of advanced lead-cooled reactors

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Abstract: The Lead-cooled Fast Reactor (LFR) is one of the three technologies selected by the Sustainable Nuclear Energy Technology Platform (SNETP) that can meet future European energy needs**.** However, the main drawbacks for the industrial deployment of LFR are the lack of operational experience and the impact of uncertainties. In nuclear reactor design the uncertainties mainly come from material properties, fabrication tolerances, operative conditions, simulation tools and nuclear data. Even thought the uncertainty in nuclear data is one of the most important sources of uncertainty in reactor simulations, significant gaps between the uncertainties and the target accuracies have been systematically shown in the past.

In this work, a target accuracy assessment of the effective neutron multiplication factor, k_{eff}, for MYRRHA (Multi-purpose hYbrid Research Reactor for High-tech Applications) lead-bismuth cooled fast reactor is performed with JEFF-3.3 and ENDF/B-VIII.0 state-of-the-art nuclear data libraries and the SUMMON system. Uncertainties in k_{eff} due to uncertainties in nuclear data have been assessed against the target accuracies defined by SG-26 of the WPEC of OECD/NEA in 2008 for LFR. Results show that k_{eff} target accuracy is still exceeded by more than a factor of two using the latest nuclear data evaluations released in 2018. Consequently, nuclear data assimilation has been carried out using criticality experiments from the International Criticality Safety Benchmark Evaluation Project for which correlation coefficient data of uncertainties is available publicly and are representative of MYRRHA.

The results from this work show that the level of accuracy needed in nuclear data cannot be obtained using only differential experiments and that the combination of experimental covariance data and integral experiments together with Generalised Least Squares technique, can provide adjusted nuclear data capable of predicting reactor properties within the target design accuracy and consistent with differential data.

THERMAL SCATTERING LAWS

Validation of Thermal Neutron Scattering Cross Sections for Heavy Water based on Molecular Dynamics Simulation

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Abstract: Recently, the thermal scattering libraries of ENDF/B-VIII.0 for light and heavy water have been released with the new water model (CAB model) proposed by Damian using NJOY2012 code. In case of CAB model, parameters and functions associated with the molecular vibrations and the coherent effect of heavy water have been calculated using the molecular dynamics simulation to more accurately describe the realistic motions of water molecule. In this paper, we generated the thermal scattering cross section of deuterium and oxygen bound in D_2O molecules that the coherent component is considered. The effect from the coherent scattering of heavy water molecules is taken into account by applying the Sköld approximation. Therefore, we calculated the Sköld correction factor which is needed for applying the Sköld approximation by using GROMACS v.5.1.4. code and EPSR (Empirical Potential Structure Refinement) code. In addition, the frequency spectrum which represents the diffusion, intermolecular vibration and intramolecular vibration of heavy water is also calculated by using GROMACS v.5.1.4. code. The thermal scattering cross sections based on newly calculated Sköld correction factor and the frequency spectrum are generated by NJOY2016 code. Finally, the performances of generated thermal scattering cross sections are validated by performing ICSBEP benchmark simulation using MCNPX 2.7.0 code.

Impact of Thermal Scattering Law for Light Water on the French Plutonium Temperature Effect Experimental Program

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Abstract: A series of experiments were conducted by the *Institut de Radioprotection et de Sûreté Nucléaire* (IRSN) at the 'Apparatus B' facility of CEA VALDUC to study the positive temperature effect for diluted plutonium solutions [1-2]. These experiments demonstrated a positive temperature effect within the temperature range of 22 °C to 28 °C, resulting from various physical effects (expansion of solution, thermalization of neutron spectrum and Doppler effect). Haeck *et al* have suggested that the Thermal scattering law (TSL) of light water at appropriate temperature is necessary to observe such a positive temperature effect. A negative effect was seen when using the closest temperature of light water TSL data available in the JEFF-3.1.1 library [3-4]. Apart from the unavailability of TSL data in the evaluations at the temperatures where experiments were performed, a possible reason to observe the negative effect can be due to the TSL model itself. In order to quantify the temperature effect, IRSN used the experiments evaluated in the ICSBEP handbook at 22°C and 28°C within the PU-SOL-THERM-038 and PU-SOL-THERM-039 series and calculated the benchmark model with its own codes and processed libraries. The most recent TSL evaluation in the JEFF-3.3 and ENDF/B-VIII.0 library have been utilized in this study to carry out Monte Carlo simulations with the MORET 5.D.1 Monte Carlo code. In addition, two new TSL evaluations of light water have been developed at IRSN, one from the TOF measurements at ILL [5] and the other from molecular dynamics simulations using highly accurate polarizable water potential [6], efficient to describe most of the water properties over a large temperature scale. The full paper will show how the thermal scattering law of light water can have an impact on the positive temperature effect of these benchmarks.

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UNCERTAINTIES AND COVARIANCE MATRICES (METHODOLOGY AND IMPACT ON REACTOR CALCULATIONS)

Influence of nuclear data parameters on integral experiment assimilation using cook’s distance

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Abstract: Nuclear data used in the core design of fast reactors is improved by using integral experiments. To utilize the past critical experimental data to the reactor design work, a typical procedure for the nuclear data adjustment is based on the Bayesian theory (least-square technique or Monte-Carlo). In this method, the nuclear model parameters are optimized by the inclusion of the experimental information with the Bayesian parameter estimation. The selection of integral experiments is based on the availability of well-documented specifications and experimental uncertainties. Data points with large uncertainties or large residuals (outliers) may affect the accuracy of the adjustment. Hence, in the adjustment process, it is very important to study the influence of experiments as well as nuclear data a priori on adjustment results.

In this work, the influence of each individual ingredient (experiment/nuclear data) will be analyzed using the concept of Cook's distance. First, JEZEBEL (Pu239, Pu240 and Pu241) will be considered and the transposition of the result on ASTRID Fast reactor concept will be discussed.

Covariance Matrices of Neutron-Induced Integral and Differential Scattering Cross Sections of ^{56}Fe

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Abstract. Correlation about the angular distribution of inelastic scattering reactions is not included in any established nuclear data library, while the correlation is one of the key issues for the propagation of uncertainties. The angular distributions are usually presented with the coefficients of Legendre polynomials. By respecting the integral cross sections in JEFF-3.1.1, the present work generates the complete correlation matrix through nuclear models, including the correlations of between different cross sections, correlations between Legendre coefficients and different integral cross sections, and correlations between different differential cross sections of 56Fe. Strong correlations between integral and differential cross sections are found for some reactions, such as the Legendre coefficients of MT2 and integral cross sections of MT1, MT2, and MT4. Correlations between differential cross sections are important, especially for low order Legendre polynomials. The determination of complete correlation matrix is thus necessary for further nuclear data evaluations and their applications.

JEFF-3.3 covariance application to ICSBEP using SANDY and NDaST

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Abstract: SANDY is a stochastic sampling code for the production of perturbed nuclear data files based on evaluated covariances provided in ENDF-6 formatted files. Amongst the nuclear data that SANDY can perturb there are cross sections, fission neutron multiplicities, neutron angular and energy distributions. In addition, the code introduces extra levels of correlations enforcing conservation equations to the samples, such as renormalization of probability distributions or summation cross section reconstruction.
In this work, SANDY was used to produce perturbed files for the major evaluated nuclear data libraries. The resulting files can be used for uncertainty propagation studies applying a straightforward brute force propagation method, which does not entail the development of sensitivity capabilities in the selected codes.

The effects of the perturbed files produced for JEFF-3.3 were also propagated to several ICSBEP criticality benchmarks using the NEA Nuclear Data Sensitivity Tool NDaST. The benchmark selection and perturbation effects were based on the k_{eff} sensitivity profiles available in the DICE database. Finally, SANDY's sampling method was also compared to a linear covariance propagation approach by NDaST for several types of nuclear data showing consistent results.

About the using of the MERCI-1 experiment analysis for the uncertainty quantification on the DARWIN2.3 decay heat calculation

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Abstract: Decay heat is a crucial issue for PWR in-core safety after reactor shutdown and back-end cycle. It is a dimensioning parameter for SCRAM devices and procedures. The DARWIN2.3 calculation package is the French reference for fuel cycle studies, including the evaluation of decay heat; it is developed by CEA with the support of its industrial partners. It is associated with a VVUQ (Verification, Validation and Uncertainty Quantification) process. Its current experimental validation for decay heat is quite limited; it relies on elementary fission burst experiments and two integral experiments (the French MERCI-1 experiment and the Swedish experiment performed in the CLAB facility). The uncertainty associated with the DARWIN2.3 decay heat calculation is mostly estimated by nuclear data covariance propagation. The DARWIN2.3 package uses a first-order formalism to propagate uncertainties on the different physical quantities by using sensitivity profiles. This estimation highly depends on input data, that is to say the covariance matrix accuracy and completeness. The experimental validation can also be used to quantify the calculation uncertainty on decay heat. To this end, a first work was performed on the basis of the MERCI-1 experiment. This experiment consisted in measuring with a calorimeter the decay heat released at short cooling times (45 minutes to 42 days) after irradiating a PWR-UOX fuel sample (with a burnup of 3.5GWd/t and 3.7% enrichment in 235U). The representativity of the MERCI-1 experiment was analyzed (based on three parameters – the burnup reached at the end of irradiation, the initial 235U enrichment, and the cooling time): the goal was to determine to what extent it is possible to transfer the information we have on discrepancies between the Calculation and the Experiment to a particular reactor application – in our case, PWR-UOX assemblies, representative of the French nuclear reactor fleet. The covariances are taken from the European evaluation JEFF-3.1.1 for the decay data and fission yields, and the CEA/Cadarache covariance matrix database COMAC-V2 for cross sections. Several methodological impacts were evaluated on the representativity calculation, and will be presented in this paper: the Boltzmann/Bateman coupling for the sensitivity coefficient calculation, the implementation of correlations between independent fission yields (no correlations are considered in JEFF3.1.1 for fission yields) in covariance matrices, and the impact of other international covariance matrices (ENDF/B-VII.1).

Construction of model defect priors inspired by dynamic time warping

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Abstract: Fe^{56} is particularly important for radiation-induced material damage calculations since it is the primary component of most stainless steels. However, Fe^{56} has traditionally been challenging to model. In this work, we propose a method to address model defects in reaction models using Fe^{56} as an example. A model defect is usually a function of energy and describes the difference between the model prediction and the truth. Of course, neither the truth nor the model defect are accessible. A Gaussian process (GP) enables to define a probability distribution on possible shapes of a model defect by referring to intuitively understandable concepts such as smoothness and the expected magnitude of the defect. However, standard specifications of GPs impose a typical length-scale and amplitude valid for the whole energy range, which is often not justified, e.g., when the model covers both the resonance and statistical range. In this contribution, we show how a GP with energy-dependent length-scales and amplitudes can be constructed from available experimental data. The proposed construction is inspired by a technique called dynamic time warping used, e.g., for speech recognition. The essential idea is to learn a metric on the input space, i.e. the energy axis, which effectively stretches and shrinks distances between energies. This metric can then be used in a standard specification of a GP. These GPs, beside their potential to improve evaluations for reactor relevant isotopes, such as Fe^{56}, may help to better understand the performance of nuclear models in the future.

Formulation of Model Defects Suitable for the Resonance Regime

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Abstract: Recently it was pointed out that there is a significant impact of model defects on the final results of a nuclear data evaluation. Especially, the uncertainties are strongly underestimated if model deficiencies are not taken into account. Recently Schnabel introduced the concept of Gaussian processes for a statiscally well defined formulation of model defects. The method was successfully applied to experimental data. However, the associated form of the covariance matrix is only applicable for smooth cross sections and cannot be applied in the resonance region. Based on Gaussian processes we present a novel method suitable for the description of model defects in the resonance regime. The novel approach is tested in applications to realistic examples of schematic resonance data.

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Monte Carlo integral adjustment of nuclear data libraries – experimental covariances and inconsistent data

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Abstract: Integral experiments can be used to adjust ND-libraries. In this work we show how we can use integral experiments in a consistent way to adjust the TENDL library. A Bayesian Monte Carlo method based on assigning weights to the different random files using a maximum likelihood function [1] is used. The proposed method does not violate constraints set by the differential data. Emphasis is put on the problems that arise from multiple isotopes being present in the integral experiment [2]. The challenges in using multiple integral experiments are also addressed, including the correlation between the different experiments. Inconsistent experimental data affects the results and this is further discussed in the presentation.

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Examples of Monte Carlo Techniques Applied for Nuclear Data Uncertainty Propagation

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Abstract: The aim of this work is to review different Monte Carlo techniques used to propagate nuclear data uncertainties.
Firstly, Total Monte Carlo methodology (TMC) has been used in a Bayesian approach in which each individual sample is ultimately weighted by a corresponding scalar likelihood factor. This is presented as the Bayesian Monte Carlo (BMC) approach based on integral criticality-safety benchmarks to assess the performance of the nuclear data and predicting correlations which do not appear with differential data [1,2]. Here, we explore the impact of other benchmarks (shielding/transmission benchmarks) within this approach.
In addition, many uncertainty quantification studies which rely on techniques based on stochastic Monte Carlo sampling to propagate nuclear data covariance and uncertainties have been published. Here, we present code SANDY which is developed to generate random samples of nuclear data from evaluated covariance data and available in ENDF-6 format. In this work, an example of this technique is presented to assess the impact of JEFF-3.3 nuclear data uncertainties for PWR Cycle Operation [3].

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NEA Nuclear Data Services, an update on relevant activities

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Abstract: The presentation will provide an update of recent in work carried out by the NEA Nuclear Data Service and the projects it co-ordinates, including the Joint Evaluated Fission and Fusion (JEFF) Nuclear Data Library project, and the work carried out for the development of NDEC (Nuclear Data Evaluation Cycle), a platform for the verification, processing and benchmarking of evaluated nuclear data.