Book of Abstracts

6th International Workshop On Nuclear Data Evaluation for Reactor applications (WONDER)

5th June – 9th June, 2023

Aix-en-Provence, France

Microscopic and integral nuclear data measurements

Recent Nuclear Data Activity at the RPI Gaerttner LINAC Center

Y. Danon[1)], P. Brain[1)], K. Cook[1)], D. Fritz[1)], A. Golas[1)], G. Siemers[1)], S. Singh[1)], B. Wang[1)], D. Barry[2)], A. Daskalakis[2)], B. Epping[2)], A. Lewis[2)], M. Rapp[2)], and T. Trumbull[2)]

1) Gaerttner LINAC Center Rensselaer Polytechnic Institute, Troy, NY, 12180, USA
2) Naval Nuclear Laboratory, P.O. Box 1072, Schenectady, NY 12301, USA

danony@rpi.edu

Abstract: The nuclear data group at the RPI Gaerttner LINAC Laboratory uses a 60 MeV pulsed electron LINAC to produce short pulses of neutrons and perform cross section and other nuclear data measurements in a wide energy range from thermal to about 20 MeV. This talk will cover several recent activities that are of interest to nuclear applications.

Recently there has been a growing activity in thermal neutron scattering evaluations which prompted the need for accurate thermal total cross section measurements for validation. To improve the neutron flux in the sub-thermal region (below 0.01 eV) a cold moderator was designed and installed. A polyethylene moderator operating at about 26K resulted in a factor of 8 increase in neutron flux below 0.01 eV. Using this new capability, several transmission measurements were performed with samples of polyethylene, polystyrene, Plexiglas, and yttrium hydride. A second thermal system designed to measure thermal neutron die-away is under development and will provide experimental validation for new evaluations.

New neutron capture and transmission measurements in the keV energy range were made for Fe-54, which are aiding an evaluation effort that is underway. Capture measurements were collected on an array of C_6D_6 detectors that was expanded from 4 to 7 detectors, a complementary transmission measurement was also performed.

A new project aimed at validation of capture gamma production is underway. This project measures the energy dependent capture gamma cascades with the RPI 16-segment gamma multiplicity detector. Measurements are compared to cascades generated from nuclear structure evaluations processed with DICEBOX and transported with a modified version of MCNP. This system provides important information on the completeness of primary gamma-ray databases. The group is also working on evaluations of Pb and Zr isotopes. Lastly, the capabilities for analysis of unresolved resonance experiments were improved by adding a resonance self-shielding correction module to the SAMMY code that enables fitting of transmission and capture yield data.

Total and Double Differential Scattering Cross-Section Measurements of Isotropic Graphite

Atsushi Kimura[1), Shunsuke Endo[1), Shoji Nakamura[1)

1) Japan Atomic Energy Agency, 2-4 Shirakata, Tokai, Ibaraki 319-1195, Japan

kimura.atsushi04@jaea.go.jp

Abstract: Nuclear-grade graphite has been considered as an efficient neutron moderator and reflector for the following reasons: its low atomic mass number, small neutron capture cross-section, high neutron scattering cross-section, easy availability, and comparatively low cost. However, the reported total cross-sections of graphite [1] are inconsistent in the low-energy region. To examine the origin of these discrepancies, the total and double differential scattering cross sections (DDSCS) of graphite have been measured in the Materials and Life Science Experimental Facility (MLF) in the J-PARC.

Isotropic graphite has higher homogeneity and mechanical properties than those of traditional extruded graphite and has been widely used in recent years. Five isotropic graphite samples with different densities and grain sizes were prepared for the experiments. The DDSCS were measured using the Beam Line #14 (AMATERAS) in the MLF. The measurements were performed for nine incident neutron energies ranging from 1.2 to 94 meV. The scattering cross-section data were normalized using those of vanadium as a standard. Small angle scattering and Bragg edges were observed in the derived cross sections, and their intensities varied among the samples. The neutron total cross sections in the energy region from 1 to 100 meV were measured using Beam Line #04 (ANNRI) in the MLF. In the epithermal range (over 40 meV), all samples tended to have values. close to the free atom cross-section of graphite. However, at the first Bragg edge, the deduced total cross sections by those samples started to separate from each other. The difference became more significant with decreasing the neutron energy, and the value tended to increase with the grain size of the sample. The results of these measurements suggest that the discrepancies between the derived total cross sections in the low-energy region are due to small-angle scattering caused by grains of graphite with uniform size.

The present study includes the result of ‘Development of Nuclear Data Evaluation Framework for Innovative Reactor’ founded by the Ministry of Education, Culture, Sports, Science and Technology of Japan. The neutron experiments at the MLF of the J-PARC were performed under the user programs (Proposal No. 2022B0109, 2022P0301).

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Experimental setup of the ^{239}Pu neutron capture and fission cross-section measurements at n_TOF,

A. Sánchez-Caballero[1], V. Alcayne[1], J. Andrzejewski[2], D. Cano-Ott[1], T. Cardinaels[3], P. Dries[3], A. Gawlik[2], E. González-Romero[1], J. Heyse[4], G. Leinders[3], T. Martínez[1], E. Mendoza[1], A. Moens[4], A. Pérez de Rada[1], J. Perkowski[2], A. Plompen[4], C. Paradela[4], P. Schillebeeckx[4], G. Sibbens[4], K. Van Hecke[3], K. Vanaken[3], D. Vanleeuw[4], K. Verguts[3], M. Verwerft[3], R. Wynants[4], and n_TOF collaboration[5]

1) Centro de Investigaciones Energéticas, Medioambientales y Tecnológicas (CIEMAT), Av. Complutense 40, Madrid, 28040, Spain
2) University of Lodz, Pomorska 149/153, Lodz, 90-236, Poland
3) SCK CEN, Belgian Nuclear Research Centre, Boeretang 200, Mol, B-2400, Belgium
4) European Commission, Joint Research Centre (JRC), Retieseweg 111, Geel, B-2440, Belgium
5) CERN, Switzerland

Adrian.sanchez@ciemat.es

Abstract: More accurate neutron capture cross-section data for ^{239}Pu are required for the design and operation of both current and future nuclear systems. As a result, the capture and fission cross-sections of ^{239}Pu are included in the NEA/OCDE High Priority Request List. To meet these needs, a new capture measurement was performed in the neutron time-of-flight facility n_TOF at CERN in late 2022 as part of the scientific program approved by the European Commission *H2020 Supplying Accurate Nuclear Data for energy and non-energy Applications* (SANDA). The experiment aims to improve previous ^{239}Pu capture measurements and obtain new α-ratio and fission cross section data to reduce current uncertainties in nuclear data evaluations. The detector system consists of the n_TOF Total Absorption Calorimeter (TAC) with 40 BaF_2 crystals and a new ionization chamber for the fission fragments, developed specifically for this measurement. The fission chamber operates in coincidence with the TAC and is used as a fission tagging detector to strongly reduce the background from fission reactions. In addition to the cross-section data, the measurement will also provide valuable information on the distribution of the *γ*-rays cascades emitted in ^{239}Pu(n,*γ*) and ^{239}Pu(n,f) reactions, as experienced in previous experiments performed with the TAC. This conference presentation will describe experimental activities performed during the ^{239}Pu campaign at n_TOF and present some first results from the data analysis, including fission yield obtained with the new fast fission detector and preliminary results on the capture yield.

$^{50,53}Cr(n,\gamma)$ cross section measurement at n_TOF

P. Pérez-Maroto[1,2], C. Guerrero[1,2], A. Casanovas[3], J. Andrzejewski[4], V. Alcayne[5], V. Babiano[6], J. Balibrea[6], L. Caballero[6], O. Cabellos[7], F. Calviño[3], D. Cano-Ott[5], R. Capote[8], M.A. Cortés[1], S. Cristallo[9], C. Domingo-Pardo[6], B. Fernández[1], A. Gawlik[4], E. González[5], S. Kopecky[10], I. Ladarescu[6], J. Lerendegui-Marco[6], L. Leal[11], T. Martínez[5], E. Mendoza[5], M.A Millán[1,2], C. Paradela[10], J. Perkowski[4], J. M. Quesada[1], T. Rodríguez[1,2], P. Schillebeeckx[10], A. Tarifeño[3] & n_TOF Collaboration[12]

1) Centro Nacional de Aceleradores, Universidad de Sevilla, 41092 Sevilla, Spain
2) Dept. de Física Atómica, Molecular y Nuclear, Universidad de Sevilla, 41012 Sevilla, Spain
3) Universitat Politècnica de Catalunya, 08034 Barcelona, Spain
4) University of Lodz, 90-136 Lodz, Poland
5) Centro de Investigaciones Energéticas, Medioambientales y Tecnológicas (CIEMAT), 28040 Madrid, Spain
6) Instituto de Física Corpuscular (IFIC-CSIC), 4680 Valencia, Spain
7) Universidad Politécnica de Madrid (UPM), 28040 Madrid, Spain
8) International Atomic Energy Agency (IAEA), 1220 Vienna, Austria
9) INFN Sezione Perugia, 06123 Perugia, Italy
10) EC Joint Research Centre – JRC, 2440 Geel, Belgium
11) Institut de Radioprotection et de Sûreté Nucléaire (IRSN), 92260 Fontenay-aux-Roses, France
12) European Organization for Nuclear Research (CERN), CH-1211 Geneva, Switzerland

ppmaroto@us.es

Abstract: Chromium is a very relevant element regarding criticality safety in nuclear reactors because of its presence in stainless steel, an important structural material. Currently, there are serious discrepancies between the different evaluations regarding the neutron capture cross sections of ^{50}Cr and ^{53}Cr, most probably related to the difficulties in the corresponding measurements due to the reduction and estimation of the very large neutron scattering effects.

In this context, the Nuclear Energy Agency (NEA) opened an entry in their High Priority Request List (HPRL) to measure these reactions between 1 and 100 keV with an accuracy of 8 to 10%. In response to this request, we have performed an experiment based on the time-of-flight technique at the n_TOF facility of CERN (Geneva, Switzerland) in Summer 2022. The first results and the data analysis strategy will be presented, including as well preliminary results from the ^{50}Cr neutron activation experiment at the HISPANoS facility of CNA (Seville, Spain).

Toward the improvement of the ^{238}U level scheme thanks to γ-spectroscopy ν

Carole Chatel[1], Jonathan Wilson[2], Maëlle Kerveno[1], Philippe Dessagne[1], Greg Henning[1]

1) Université de Strasbourg, CNRS, IPHC/DRS UMR 7178, 23 Rue du Loess, F-67037 Strasbourg, France
2) CNRS, IJClab Orsay, bât 100, 15 rue G. Clemenceau, 91406 Orsay Cedex, France

carole.chatel@iphc.cnrs.fr

Abstract: To improve the accuracy of neutronics simulations of actual and future reactor cores, a better knowledge of the neutron population is required. This population is, among others, driven by (n, xn) reactions, including inelastic scattering. Indeed, these reactions change the number of neutrons in a reactor core and their speed. However, their cross sections are, still nowadays, not precisely known. Hence, the neutron inelastic scattering cross section off ^{238}U features in the High Priority Request List [1]. One method to obtain this cross section is to use the prompt γ-ray spectroscopy coupled to time-of-flight measurements. This allows, from the measured (n,xnγ) cross sections and the level scheme information, to infer the total (n, n') cross section [2]. In the case of the ^{238}U, the knowledge of the level scheme is still very incomplete: the discrete states are assumed to be fully known up to 1.3 MeV only and the average uncertainties on branching ratios in ENSDF [3] are of 8%. Yet, sensitivity calculations performed with the TALYS code [4] showed that modifying the branching ratios of 10% in the input's code can have an impact of up to 4% on (n, n'γ) cross sections [2].

For all these reasons, improving the level scheme knowledge has become of high importance. This can be done thanks to the coupling between the ν-Ball γ-spectrometer [5] and the LICORNE neutron source [6, 7] of the ALTO facility. Indeed, the LICORNE source allows the production of a pulsed quasi-mono-energetic kinematically focused neutron flux thanks to the p(^{7}Li, n)^{7}Be inverse reaction, the produced ^{7}Li beam impinging on a ^{1}H-gas cell. The neutron flux impinged then on the ^{238}U target and the γ produced have been collected thanks to the two rings of 12 HPGE-Clover detectors composing the ν-Ball γ-spectrometer. These 24 detectors allow a data analysis by γ-γ coincidences.

Two ν-Ball campaigns have been led in 2018 and 2022. The analysis of the γ-γ coincidences matrix obtained during the first ν-Ball campaign with a neutron flux of a mean energy of 2.1 MeV is now performed thanks to the Radware software [8]. Until now, 59 γ and 43 levels registered in ENSDF have been confirmed and 55 new γ and 26 new levels have been found. Once finished, analyzing the data acquired during the ν-Ball2 campaign will allow to consolidate and improve the results of the analysis of the data acquired during the first campaign.

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Measurement of ^{242}Pu(n,f) in the [1;2MeV] energy range

L. Mathieu[1), M. Aïche[1), P. Marini[1), O. Bouland[2), C. Chatel[3), S. Czajkowski[1), D. Kattikat-Melcom[1), T. Kurtukian[1), S. Oberstedt[4), I. Tsekhanovich[1)

1) Univ. Bordeaux, CNRS, LP2I, UMR 5797, F-33170 Gradignan, France
2) CEA, DES, IRESNE, DER, SPRC, Physics Studies Laboratory, Cadarache, F-13108 Saint-Paul-lez-Durance, France
3) Université de Strasbourg, CNRS, IPHC/DRS UMR 7178, 23 Rue du Loess, F-67037 Strasbourg, France
4) European Commission, DG Joint Research Centre, Directorate G - Nuclear Safety and Security, Unit G.2 SN3S, 2440 Geel, Belgium

mathieu@lp2ib.in2p3.fr

Abstract: The design of new generation nuclear reactors using fast neutrons requires highly accurate cross-section measurements in the MeV energy range. The ^{242}Pu fission cross section is of particular interest for nuclear waste production, as this isotope is a gateway to the production of heavier actinides like ^{243}Am. There are discrepancies around 1 MeV incident neutron energy between libraries and among experimental data. Some data suggest the presence of a strong structure between 1 and 1.2 MeV whereas such structure is barely visible on data from P. Salvador [1]. The shape of this structure is also very different between ENDF/B-VIII and JEFF-3.3.

The large majority of the ^{242}Pu(n,f) measurements have been carried out relative to the ^{235}U(n,f) secondary-standard cross section. This introduces a strong correlation between independent measurements based on the same standard. Moreover, the ^{235}U(n,f) cross section also exhibits structures, in particular a steep increase of +10% at 1 MeV. Therefore, we aim to measure the ^{242}Pu(n,f) cross section relative to the primary-standard H(n,n) cross section, by using a proton recoil detector. This standard has a very high accuracy (0.4%) [2], is independent of other measurements, and is structureless.

An experiment has been carried out in October 2022 at the MONNET facility [3] in JRC-Geel. This facility can produce a quasi-monoenergetic neutron beam via the T(p,n)^{3}He reaction in the MeV energy range. The experimental setups consisted in a ^{242}Pu sample positioned in front of two solar cells in order to detect fission fragments. The neutron flux was converted to protons via an elastic scattering nuclear reaction on a H-rich foil. The proton recoil detector was a Si detector placed downstream the target. Measurements have been performed with incident neutron energies from 0.9 MeV to 2.0 MeV. The experimental setup will be presented, and the analysis procedure will be detailed.

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Investigation of (n,x) reactions with enriched Ge targets at 15.7 MeV at the upgraded facility of NCSR "Demokritos"

S. Chasapoglou[1), R. Vlastou[1), M. Diakaki[1), M. Kokkoris[1), L. Amanatidis[1), M. Axiotis[2), S. Harissopulos[2), A. Lagoyannis[2)

1) Department of Physics, National Technical University of Athens, Zografou Campus, Athens, 15772, Greece
2) Tandem Accelerator Laboratory, Institute of Nuclear and Particle Physics, N.C.S.R. "Demokritos", Athens, 15341, Greece

sotirischasapoglou@mail.ntua.gr

Abstract: Studies of neutron induced reactions are of considerable interest, not only for their importance to fundamental research in Nuclear Physics and Astrophysics, but also for practical applications in nuclear technology, dosimetry, medicine and industry. These tasks require improved nuclear data and higher precision cross sections for neutron induced reactions.

The 5.5 MV Tandem T11/25 Accelerator of NCSR "Demokritos", has been recently upgraded to a pelletron charging system, with new ion sources, gas stripper and beam optics. Quasi-monoenergetic neutron beams are produced in the energy range ~ 16-19 MeV using a Ti-tritiated target of 373 GBq activity, consisting of 2.1 mg/cm2 Ti-T layer on a 1 mm thick Cu backing, by means of the $^{3}H(d,n)^{4}He$ reaction. The maximum flux has been determined to be of the order of 10^{5}-10^{6} n/cm^{2}s, implementing reference reactions, while the flux variation of the neutron beam is monitored by using a BF3 detector. An investigation of the energy dependence of the neutron fluence has been carried out via the multiple foil activation technique in combination with the SAND unfolding code as well as by means of MCNP5 Monte Carlo simulations. The 15.7 MeV neutron beam has been used for the measurement of the $^{70}Ge(n,2n)^{69}Ge$, $^{76}Ge(n,2n)^{75}Ge$, $^{73}Ge(n,p)^{73}Ga$, $^{72}Ge(n,p)^{72}Ga$, $^{73}Ge(n,np/d)^{72}Ga$, $^{74}Ge(n,np/d)^{73}Ga$, $^{74}Ge(n,\alpha)^{71m}Zn$, $^{72}Ge(n,\alpha)^{69m}Zn$, $^{73}Ge(n,n\alpha)^{69m}Zn$ reaction cross sections with the activation technique, using monoisotopic Ge targets which significantly improve the accuracy of the results, in comparison with the ones derived from ^{nat}Ge targets. The preliminary results from these cross section measurements will be presented, along with the results from the neutron beam characterization.

Study of ^{149}Sm capture and total cross sections for burnup credit applications

***A. Stamatopoulos*[1)]**, *P. Koehler*[1)], *E. Leal-Cidoncha*[1)], *A. Couture*[1)], *G. Rusev*[2)], *J. Ullmann*[1)],

1) Physics Division, Los Alamos National Laboratory, 87545, NM, USA
2) Chemistry Division, Los Alamos National Laboratory, 87545, NM, USA

thanos@lanl.gov

Abstract: The performance of evaluated cross sections of fission products (FP) in the thermal, resolved (RRR) and unresolved resonance region (URR) for burnup credit analysis has been found to be insufficient (ORNL/TM-2005/65). Throughout the years, the evaluations were focused on actinides such as Uranium and Plutonium, whereas efforts on evaluating cross sections of FP were significantly less thorough. More specifically, there is a lack of measurements and R-matrix analyses in the RRR and URR for various FP nuclei such as ^{143}Nd and ^{149}Sm. ^{149}Sm in particular, has a thermal capture cross section of 40 kb and is an important stable FP for burnup credit. ^{149}Sm builds up like ^{135}Xe in power reactor fuel, however, it does not decay out of spent nuclear fuel. The capture and total cross sections have been identified as insufficient (ORNL/TM-2005/65). The 4π Detector for Advanced Neutron Capture Experiments (DANCE) and the new neutron-transmission Device for Indirect Capture Experiments on Radionuclides (DICER) have been used at the Los Alamos Neutron Science Center (LANSCE) to study the capture and total cross section, respectively. Experimental details and efforts on the ongoing analysis will be presented.

Study of (n,α) reactions of interest for nuclear energy

A.Chevalier(1), FR. Lecolley(1), JL. Lecouey(1), N. Marie-Nourry(1), G. Lehaut(1) , L. Manduci(2), X. Ledoux(3), R. Beyer(4), A. Junghans(4), O. Bouland(5), O. Serot(5)

(1) LPC Caen, 6 Bd Maréchal Juin, Caen 14000, France
(2) EAMEA, BP 19 50115, Cherbourg Armées 50100, France
(3) GANIL, Bd Henri Becquerel, Caen 14000, France
(4) HZDR, Bautzner Landstraße 400, Dresden 01328, Germany
(5) CEA, DES, IRESNE Cadarache, Saint-Paul-lez-Durance 13108, France

chevalier@lpccaen.in2p3.fr

In reactors, oxygen is present in abundance in the form of water, and/or in the form of oxide in the fuel used (in the case of Pressurised Water Reactors and Fast Reactors). These oxygen nuclei are responsible for 25% of helium formation in nuclear reactors due to the reaction $^{16}O(n,\alpha)^{13}C$. However, this reaction still shows significant discrepancies between experimental and evaluated data that can go up to 30% for some energy ranges. This is why the NEA (Nuclear Energy Agency) has issued several requests included in the HPRL (High Priority Request List)[1] and confirmed by the WPEC 40 (CIELO, 2014)[2] for this reaction in the incident neutron energy range from threshold energy to 20 MeV. Sensitivity analyses conducted by the WPEC 26 (2008)[3] showed that these discrepancies induced significant uncertainties on some nuclear reactors parameters such as helium production (± 7%) and keff (± 100 pcm)[3].

Regarding other (n,α) reactions in light target nuclei, the $^{19}F(n,\alpha)^{16}N$ cross section is of great interest for the development of the next generation IV reactors that could potentially use molten salt mixtures. Significant differences (up to a factor of 3) have been observed for this nucleus with regards to the (n,α) channel.

In view of improving our knowledge on the (n,α) reactions, the GrACE group (Groupe Aval du Cycle Electronucléaire) of the LPC Caen has developed a new detector named SCALP[4] (Scintillating ionization Chamber for ALPha particle detection in neutron induced reactions). The first two experiments with this new detector carried out at the new NFS facility of GANIL in Caen and at the nELBE facility of HZDR in Dresden were successful.

During this conference, the operational principle of the SCALP detector will be presented and discussed, as well as the experiments that have been conducted using it. Furthermore, insights into the data acquired during this experiment, as well as the ongoing processing and multi-channel analysis of it, will be provided.

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Validation of the Monte-Carlo efficiency calculation of the LOENIEv2 long counter for delayed neutron measurements

Pierre Leconte[1), Benoit Geslot[2), Timothee Kooyman[2)

1) CEA Cadarache, DES/IRESNE/DER/SPRC/LEPh, 13108 Saint Paul Lez Durance Cedex, France
2) CEA Cadarache, DES/IRESNE/DER/SPESI/LP2E, 13108 Saint Paul Lez Durance Cedex, France

pierre.leconte@cea.fr

Abstract: LOENIEv2 is a long counter detector designed by CEA for delayed neutron (DN) yield ν_d and group constant (a_k,λ_k) measurements. It is composed of sixteen ^{3}He tubes embedded in a cylindrical high-density polyethylene (HDPE) matrix. The general formulation linking ν_d to the detector counting rate requires a prior knowledge of the DN energy spectra χ_d(E) as a weighting function of the detection efficiency $\varepsilon(E)$. As these data are poorly known, except for the 20 most abundant DN precursors, it is recommended to minimize the energy dependence of $\varepsilon(E)$, so that their contribution vanishes. Design calculations of the LOENIEv2 detector were performed with the TRIPOLI-4® Monte-Carlo to meet this requirement. Thanks to a special arrangement of the ^{3}He tubes in three concentric rings, variations of the total efficiency as low as 2% can be reached over the [0.1 – 1 MeV] energy range. The purpose of this paper is to confirm these design calculations thanks to calibrated neutron source measurements, performed at the NPL institute. These sources are in the form of small cylinders containing either a spontaneous fission material (^{252}Cf) or a radioactive material producing neutron through (α,n) reactions (AmLi, AmB, AmF, AmBe). These sources are well characterized in emission rate, spectrum and anisotropy so that they can used as standards for efficiency calibration. Moreover, the availability of several sources of the same materials with a diversity of emission rates, provides a convenient way to validate the dead-time correction model up to 5.10^4 c/s. The study concludes that the agreement between the TRIPOLI-4® simulated and measured efficiencies is better than 1% and that the best agreement is reached with the JEFF-3.3 library. The impact of the nuclear data cross section and thermal scattering data is tested with the JEFF-3.1.1, JEFF-3.3, JENDL-4.0 and ENDF/B-VIII.0 libraries and is shown to be low. At last, the TRIPOLI-4® model of LOENIEv2 is applied to compute the detection efficiency for delayed and prompt neutron measurement from the thermal neutron induced fission of ^{235}U. A Total Monte-Carlo approach is applied to propagate the uncertainty due to the energy spectra and due to the technological data of the long-counter.

Sample Worth Measurement of Calcium Hydride

Tadafumi Sano[1)], Takashi Kanda[1)], Jun-ichi Hori[1)]

1) Atomic Energy Research Institute, Kindai University, Kowakae, Higashi-Osaka, Osaka, 577-8502, Japan
2) Institute for Integrated Radiation and Nuclear Science, Kyoto University, 1010-2, Asashiro-Nishi, Kumatori-cho, Osaka, 590-0494, Japan

t-sano@kindai.ac.jp

Abstract: A small modular reactor (SMR) employing a solid moderator have attracted attention to the passive safety. A one of the solid moderator is calcium hydride (CaH2). Some integral experiments should be carried out to validate nuclear data of CaH2, however, no integral experiment by reactor for the validation have been reported. Therefore, reactivity worth measurements of CaH2 samples as the integral experiments were conducted in a university training and research reactor (UTR-KINKI) of Kindai University.

The UTR-KINKI reactor is a light-water-moderated and graphite-reflected two-core (coupled-core) reactor. A graphite region between the two cores has been employed as a standard irradiation field which has a standard neutron spectrum consisting of 1/v and Maxwell distributions. The CaH2 sample was placed in a cavity located at the center of the irradiation field.

The sample reactivity worth was determined from a difference between two excess reactivities of the respective reactors with and without the sample. The respective excess reactivities were measured using positive period method. The reactivity worth measurements were repeated for 21.20 g, 41.00 g, 72.78 g and 164.4 g of CaH2 sample. From the present experiment, the following reactivity worth could be determined.

1) 21.29 g of CaH2, -0.0042±0.0003 [%Δk/k]
2) 41.00 g of CaH2, -0.0086±0.0003 [%Δk/k]
3) 78.72 g of CaH2, -0.0159±0.0004 [%Δk/k].
4) 164.4 go of CaH2, -0.0322±0.0007 [%Δk/k].

Furthermore, the sample reactivity worth was calculated using the continuous energy Monte Carlo codes MVP3.0 with the nuclear library JEFF-3.1, where each cross section of H and Ca constituting a molecule of CaH2 was taken into the thermal neutron scattering law (S(α,β)). The ratios of calculated to experimental values (C/E) were 1.19±0.72, 0.806±0.347, 0.874±0.189 and 0.894±0.100 for 21.29 g, 41.00 g, 78.72 g and 164.4 g of the samples, respectively.

Nuclear Fission

Dependence of total kinetic energy of fission fragments as function of excitation energy and neutron excess for U and Pu isotopes

Satoshi Chiba[1,2], Kazuya Shimada[2,3], Chikako Ishizuka[2]

[1] NAT Research Center, Tokai, Naka Ibaraki 319-1112, Japan
[2] Tokyo Institute of Technology, Ookayama, Meguro, Tokyo 152-8550, Japan
[3] Present address: Japan Atomic Energy Agency, Tokai, Naka, Ibaraki 319-1195, Japan

Abstract:
Total kinetic energy (TKE) of fission fragments consists of about 80 % of energy released by nuclear fission. Therefore, understanding of the behavior of TKE is crucial in designing nuclear energy systems as well as understanding local heat source in r-process nucleosynthesis where fission recycling plays an important role. We have investigated fragment mass and TKE correlation as functions of excitation energy and neutron excess for a series of U and Pu isotopes in terms of a 4D Langevin dynamical model developed at Tokyo Tech. We found that decrease of TKE as a function of the excitation energy of fissioning nucleus could be understood by considering the change of shape of the heavy fragments from spherical to prolate deformation, which makes distance between the center-of-mass of the 2 nascent fragments longer, then the Coulomb repulsion of 2 fragments to decrease [1]. Then, we have done similar systematic calculations of mass-TKE correlation ranging from proton drip to neutron drip line of U and Pu. We found that the TKE trend deviates from the 1/A^{1/3} law as indicated by Viola and Unik systematics. The reason of such a deviation was again accounted for by change of the fragment deformation.

^{235}U fission fragment study with Falstaff at NFS

D. Doré[1)], D. Ramos[2)], E. Berthoumieux[1)], J.-E. Ducret[2)], X. Ledoux[2)], P. Marini[2,3)], S. Oberstedt[4)], J. Pancin[2)], M. Ballu[1)], P. Herran[1)], G. Kaur[1)], A. Letourneau[1)], T. Materna[1)], P. Miriot-Jaubert[1)], B. Mom[1)], L. Thulliez[1)], M. Vandebrouck[1)], A.-M. Frelin[2)], P. Sharma[2)], I. Jangid[2)], A. Cheboubbi[5)], O. Litaize[5)], O. Serot[5)], A. Porta[6)], M. Estienne[6)], M. Fallot[6)], E. Bonnet[6)], J. Pépin[6)], L. Matthieu[3)], T. Kurtukian-Nieto[3)]

1) Irfu, CEA, Université Paris-Saclay, 91191 Gif-sur-Yvette, France
2) GANIL, Caen,14000, France
3) Univ. Bordeaux, CNRS, LP2I, UMR 5797, F-33170 Gradignan, France
4) European Commission, Joint Research Centre (JRC), 2440 Geel, Belgium
5) CEA, DES, IRESNE, DER, SPRC, Cadarache, 13108 St. Paul lez Durance, France
6) Laboratoire Subatech, University of Nantes, CNRS/IN2P3, Institut Mines Telecom Atlantique, 44307 Nantes, France

diane.dore@cea.fr

Abstract: Nowadays the fission process still presents a great interest from both theoretical and experimental points of view. New developments on microscopic calculations and expected improvements of nuclear reactors are among the main motivations for new experimental programs devoted to the study of nuclear fission. The FALSTAFF spectrometer aims at providing constraining data that may significantly contribute to an accurate description of the fission process. In its future two-arm configuration, the goal of the FALSTAFF program will be to determine the evolution of prompt neutron multiplicity and the fragment characteristics (mass, charge and kinetic energy) as a function of the compound nucleus excitation energy, by studying neutron-induced fission of specific actinides in the MeV range. Recently FALSTAFF in its one-arm configuration was used in an experiment dedicated to the study of ^{235}U(n,f) at NFS (Neutrons for Science, SPIRAL2/GANIL).

The white energy spectrum of incident neutron beam provided by reactions of deuterons on a thick ^{9}Be production target at NFS allows us to study ^{235}U post-neutron evaporation fission fragments over the incident neutron energy range from 0.5 to 40 MeV. The fragment velocities were measured thanks to two MWPC-SED detectors giving access to the time and position of the fragments crossing an emissive foil while an axial ionization chamber measured the residual energy and the energy loss profile of fragments. LaBr3 detectors were coupled to FALSTAFF to provide an absolute time reference point allowing the determination of the incident neutron energy. The evolution of the fragment characteristics can then be studied as a function of the incident neutron energy.

In this paper, the motivations for the FALSTAFF@NFS experiment will be detailed and the experimental setup will be described. Preliminary results for the fragment velocity, energy, mass and charge distributions will be presented. Foreseen experiments will be discussed.

Energy Dependence of Prompt Fission Neutron Multiplicity in the ^{239}Pu(n, f) Reaction

P. Marini[1,2,*] *, J.Taieb*[2,3] *, D. Neudecker*[4] *, G. Bélier*[2,3] *, A. Chatillon*[2,3] *, D. Etasse*[5] *, B. Laurent*[2,3] *, P. Morfouace*[2,3] *, B. Morillon*[2] *, M. Devlin*[4] *, J. A. Gomez*[4] *, R. C. Haight*[4] *, K. J. Kelly*[4] *, and J. M. O'Donnell*[4]

1)LP2I Bordeaux, UMR5797, Université de Bordeaux, CNRS, F-33170, Gradignan, France
2)CEA, DAM, DIF, F-91297 Arpajon, France
3)Université Paris-Saclay, CEA, LMCE, 91680 Bruyères-le-Châtel, France
4)Los Alamos National Laboratory, Los Alamos, NM-87545, USA
5)Normandie Univ, ENSICAEN, UNICAEN, CNRS/IN2P3, LPC Caen, 14000 Caen, France

* paola.marini@cnrs.fr

Prompt neutron emission is a challenge in nuclear fission research. Accurate values of the number of prompt fission neutrons emitted in fission reaction and their kinetic energy distributions are essential for fundamental and applied nuclear physics. Indeed, they provide valuable information on the amount of excitation energy of the heated fissioning system transferred to the primary fragments. Moreover, these data, for the fissile ^{235}U and ^{239}Pu isotopes and the fertile ^{238}U nuclide, are vital inputs to calculate next-generation nuclear reactor neutronics.

Measuring them to high precision for radioactive fissioning nuclides remains, however, an experimental challenge. We present here a recent and novel measurement of the average prompt-neutron multiplicity from the ^{239}Pu (n,f) reaction as a function of the incident-neutron energy, over the range 1-700 MeV. The experiment was carried out at the Los Alamos Neutron Science Center of the Los Alamos National Laboratory. An innovative setup, coupling the Chi-nu liquid scintillator array to a newly developed, high-efficiency, fast fission chamber was used. The combined setup, the double time-of-flight technique and the high statistics collected allowed to minimize and correct for the main sources of bias and thus achieve unprecedented precision. Corrections needed to account for neutron angular and energy distributions, as well as detector dead-time and beam characteristics will be discussed in details.

Our data were compared to the most recent ENDF/B-VIII.0 and JEFF3.3 nuclear data evaluations. We will show that, at low energies, our data validate for the first time the ENDF/B-VIII.0 evaluation with an independent measurement and reduce the evaluated uncertainty by up to 60%. This work opens up the possibility of precisely measuring prompt fission neutron multiplicities on highly radioactive nuclei relevant for an essential component of energy production.

Revisiting prompt emission of $^{252}Cf(SF)$ with focus on post-neutron fragment distributions and different correlations

Anabella Tudora

University of Bucharest, Faculty of Physics, str. Atomistilor 405, Magurele, Ro-77125, Romania

anabellatudora@hotmail.com

Abstract: The present investigation, focusing on post-neutron fragment distributions and different correlations between pre- and post-neutron fragment quantities and their distributions, is performed by using the Deterministic Sequential Emission (DSE) model developed at the University of Bucharest. The impact of energy partition in fission on prompt emission and post-neutron fragment distributions is investigated by using two different methods: one is based on modeling at scission and another consists of the sharing of total excitation energy (TXE) according to the temperature ratio $R_T=T_L/T_H$ of fully accelerated complementary fragments, which is parameterized as a function of A_H (by a few jointed segments). Three pre-neutron fragment distributions Y(A,TKE), measured during the time at JRC-Geel (i.e. by Hambsch and Oberstedt in 1997, Göök et al. in 2014 and in the VESPA experiment in 2021) are used in order to investigate their influence on post-neutron fragment distributions and different correlations.

The model calculations are validated by the good description of all experimental prompt neutron and γ-ray data of $^{252}Cf(SF)$ by the DSE results obtained with both TXE partitions and three Y(A,TKE) data. The results of independent FPY $Y(Z,A_p)$ and $Y(A_p)$ are also in good agreement with the experimental data from EXFOR. Compared to the previous investigated case of $^{235}U(n_{th},f)$ [1], the influence of Y(A,TKE) on post-neutron fragment yields is more pronounced, different Y(A,TKE) data leading to changes of both the position and the magnitude of visible peaks and dips in the $Y(A_p)$ structure. The even-odd effect in fragment charge still plays a role in the $Y(A_p)$ structure but it is less pronounced than in Ref.[1] because the even-odd effect in Y(Z) is 10 times lower in the case of $^{252}Cf(SF)$ compared to $^{235}U(n_{th},f)$. The correlation between the excitation energy E* of fully accelerated pre-neutron fragments and the kinetic energy KE_p of post-neutron fragments, ascertained in Ref.[2], is maintained in the case of $^{252}Cf(SF)$, too.

The investigation is extended to the distributions of pre-neutron fragment energies $Y_\nu(E^*)$ leading to the number of emission sequences (or prompt neutrons) ν = 0, 1, 2, 3, 4, etc. The correlation between E* and ν is well reflected in the almost linear increase exhibited by the first moments of these distributions $\langle E^* \rangle$ (all fragments) and $\langle E^* \rangle_{L,H}$ (separately for light and heavy fragments) as a function of ν. The slope of $\langle E^* \rangle_H(\nu)$ being visible lower than that of $\langle E^* \rangle_L(\nu)$.

The mass, charge and TKE distributions of pre-neutron fragments ($Y_\nu(A)$, $Y_\nu(Z)$, $Y_\nu(TKE)$) which lead to each number of emission sequences (or prompt neutrons) ν = 0, 1, 2, 3, etc. are also investigated, showing that the highest $Y_\nu(A)$, $Y_\nu(Z)$, and $Y_\nu(TKE)$ distributions are those corresponding to ν = 2, 3 and 4. This fact confirms the Gaussian shape of experimental P(ν) data which is centered on $\langle \nu \rangle$ values placed between 3.7 and 3.8. The differences between the three Y(A,TKE) data are reflected in visible differences between the DSE results of $Y_\nu(A)$, $Y_\nu(Z)$, and $Y_\nu(TKE)$, respectively. It is also observed that in all cases the $Y_\nu(A)$ and $Y_\nu(Z)$ distributions of light fragments are higher than those of heavy fragments confirming again the usual statement that the light fragment group emits more prompt neutrons than the heavy fragment group.

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First results on $^{235}U(n_{th},f)$ independent isotopic fission yields using prompt gamma rays at FIPPS

P. Herran [1), T. Materna [1)

1) Irfu, CEA, Université Paris-Saclay, F-91191 Gif-sur-Yvette

pierre.herran@cea.fr

Abstract: Although nuclear fission has been known and studied for more than 80 years, it remains a very active field of research. Knowledge of the fission process can be improved by studying the prompt gamma-ray cascades emitted by fission fragments.

We will present the results of a measurement campaign with the FIPPS gamma-ray spectrometer and the use of fission prompt gamma rays for the determination of independent isotopic yields. The yield of a set of well-produced even-even nuclei for the $^{235}U(n_{th},f)$ reaction was extracted from the data. It includes the specific case of the doubly magic nucleus ^{132}Sn, for which an anomaly in the yield was observed for the $^{238}U(n_{fast},f)$ reaction.
The FIPPS gamma-ray spectrometer is installed at the end of a thermal neutron guide in the research reactor of the Institute Laue-Langevin (Grenoble). The neutron beam interacts with an active target consisting of a solution of ^{235}U diluted in a scintillating liquid. The active target allows us to discriminate fission events from fragment beta decays or (n,γ) reactions on the target support. It also allows us to determine precisely the total number of fissions that occurred during the experiment. The spectrometer consists in a set of 16 HPGe clovers that were placed around the target to detect the prompt and delayed gamma-ray cascades emitted by the fission fragments.

The obtained fission yields are compared with evaluated values from JEFF-3.3 and ENDF/B-VIII nuclear databases. The observed differences are interpreted with the FIFRELIN code, which simulates the fission process and the de-excitation of the fission fragments.

Computer Simulation of Prompt Fission Neutron Detector

Olga Sidorova [(1)], Shakir Zeynalov [(1)]

1) Joint Institute for Nuclear Research, Dubna, Russia

Abstract:
Investigation of prompt fission neutron properties is of great importance for understanding the nuclear fission process and for study of kinetic energy sharing between fission fragments (FF). For detailed study of mass and kinetic energy distributions along with prompt fission neutron (PFN) emission properties in fission process of 235U, 237Np and 239Pu induced by resonance neutrons and the spontaneous fission reaction of 252Cf a new setup with combination of double Frisch gridded chamber with the neutron detector (ND) was developed. The ND was multi detector system consisted of 32 BC501 scintillation liquid filled modules. For each fission event, the following simulated information was recorded : correlated fission fragments emission time stamp, and their emission angles in respect to the selected coordinate frames along with the pulse heights and shapes of the neutron detector signals. Cross talks between individual modules composing the detector imitates multiplicity cannot be separated by apparatus, so the actual number of detected neutrons is less then registered by the apparatus. The aim of the work was to estimate the systematic inaccuracy.

Investigation of the structure of $^{235}U(n_{th}$,fission) prompt gamma energy spectrum by FIFRELIN

Tatsuhiko Ogawa[1,2], Olivier Litaize[3], Davide Mancusi[2], Abdelhazize Chebboubi[3], Olivier Serot[3]

1) Japan Atomic Energy Agency, 2-4, Shirakata, Tokai, Naka, Ibaraki 319-1195, Japan
2) Université Paris-Saclay, CEA, Service d'Etudes des Réacteurs et de Mathématiques Appliquées, Gif-sur-Yvette 91191, France
3) Commissariat à l'énergie atomique et aux énergies alternatives, DES, IRESNE, DER, Cadarache, Saint-Paul-Lez-Durance 13108, France

ogawa.tatsuhiko@jaea.go.jp

Abstract: The gamma-ray spectrum up to 20 MeV of the $^{235}U(n_{th}$,fission) reaction is calculated by FIFRELIN, the fission event generator based on phenomenological reaction models, and nuclear structure data. Comparisons are made with a measurement performed at the Institut Laue Langevin (ILL) to verify the accuracy of FIFRELIN and study the mechanisms responsible for high-energy gamma-ray emission.
The spectra featured by a bump at 4 MeV, a shoulder from 6 to 8 MeV, and a hump around 14 MeV agree well whereas the spectrum of FIFRELIN is lower than that of the measurement in the gamma energy range from 8 MeV to 13 MeV. FIFRELIN shows that the bump around 4 MeV is attributed to the deexcitation of ^{132}Sn which does not have excited states below 4 MeV owing to it closed shells. The shoulder from 6 to 8 MeV is explained by the neutron separation energy of fission fragments from 6 to 8 MeV. The fragments with excitation energy lower than the neutron separation energy emit gamma-rays whereas those with higher excitation energies favor neutron emission. The gamma-ray spectrum consequently drops in this energy range.
The hump at 14 MeV is attributed to the first gamma de-excitation of light fission fragments to the states close to the ground state. Heavy fission fragments, whose mean excitation energy before neutron emission is about 9 MeV according to FIFRELIN, scarcely contributes to this hump. In contrast, the mean of excitation energy of light fission fragments are 14 MeV. Their transitions to low-energy states are responsible for the gamma-rays of the hump. Owing to this mechanism, the height of the hump depends on the level density model. The hump simulated using the Composite Gilbert-Cameron Model was lower than that using Hartree-Fock-Bogoliubov model, which can take into account for the nucleus-specific characteristics in a more sophisticated manner.
This study illustrated that FIFRELIN is robust tool which can reproduce and identify the origins of the features characterizing the gamma-ray spectrum up to 20 MeV of ^{235}U thermal fission.

Production and Fission of ^{236}Pu by Fast Neutrons

Cristiana Oprea, Ioan Alexandru Oprea

Romanian Scientific Research Agency, Bucharest (ANCS), County Department For Education Bihor, Mihai Eminescu 11, Oradea 410010, Bihor, Romania

coprea2007@yahoo.co.uk

Abstract: A trace element with spin and parity 0^+, ^{236}Pu, is not a naturally occurring isotope of Plutonium, but an isotope formed by neutron-induced fission of ^{235,238}U. It can also be obtained by photon-induced processes (γ, n), fast neutron reactions (n, 2n) of ^{237}Np followed by beta decay, ^{236}U beta decay, and other ways. The ^{236}Pu nucleus may be useful for studying the fuel cycle's environmental impact. As part of the design of the new basic facilities, an analysis of the separation and influence of different fission products, including ^{236}Pu, will be necessary.
The present study investigated ^{236}Pu fast neutron fission. The Talys code and programs written by authors were used to evaluate fission observables such as cross-sections, mass and charge distributions for fission fragments, neutron spectra emitted by fragments, and isotope production yields of interest for applications in medicine, electronics, and nuclear technology. The contributions of different reaction mechanisms to fission and production of ^{236}Pu were examined. In the incident, emergent, and fission channels, level density, and Wood-Saxon potential were extracted for each process.
We compared experimental data from the literature with our theoretical evaluations of fission observables. It is necessary to note that in the case of fast neutron-induced fission of ^{236}Pu nuclei, there are very few experimental data regarding fission observables, and therefore their evaluation is critical both for fundamental research and nuclear reactor technical development.

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Theory

Microscopic prediction of gamma-ray strength function

Thomas Duguet [(1)], Mikael Frosini [(2)], Andrea Porro [(1)]

(1) CEA Saclay, DRF/IRFU, F-91191 Gif sur Yvette Cedex, France
(2) CEA Cadarache, DES/IRESNE/DER/SPRC/LEPh, F-13108 Saint Paul Lez Durance Cedex, France

thomas.duguet@cea.fr

Abstract:
Recent advances in theory of nuclear structure open the way to the use of microscopic models of nuclei for evaluated data in reactor physics. In this framework, nuclei are considered as a collection of nucleons in strong interaction and their properties are coherently described via the use of so-called A body methods. They represent therefore a tool of choice to describe within a single framework (and without readjustment) phenomena that are very different in nature (spectroscopy, reaction, fission) over the whole nuclear chart. They are expected to improve the predictive power of theoretical models and allow a better extrapolation to systems or observables that are difficult to access experimentally.
In this talk, predictions of gamma-ray strength functions based on microscopic models will be presented. Gamma-ray strength functions constitute a key ingredient of Hauser-Feshbach's statistical model for the determination of the neutron capture (emission) cross section given that this physical process is in direct competition with the emission of a photon. Traditionally, the Quasi Random Phase Approximation (QRPA) (that explores harmonic vibrations around the ground state) is considered to be the tool of choice for systematic microscopic calculations over the nuclear chart. We will list the advantages and limitations of QRPA and see how these drawbacks (that are usually corrected a posteriori with ad hoc external parameters) can affect the input used for nuclear cross-sections.
This will motivate the introduction of a theory going beyond standard QRPA, i.e. the Projected Generator Coordinate Method (PGCM), that has been employed for many years in the nuclear structure community to describe the low-lying spectroscopy of nuclei, but rarely for strength functions. We will illustrate how the PGCM overcomes most of QRPA's limitations on the basis of monopole and dipole strength functions in selected sd-shell nuclei.
This preliminary study, performed within the PAN@CEA collaboration, paves the way for the systematic generalization of these tools to all nuclei in the coming years.

Fundamentals and progress of theoretical models for the evaluation of photonuclear reaction data in CNDC

Yuan Tian [1], Zhigang Ge [1], Xi Tao [1], Jimin Wang [1], Ruirui Xu [1]

1) China Institute of Atomic Energy, P. O. Box 275- , Beijing 102413, China

tiany@ciae.ac.cn

Abstract:
The photonuclear reaction cross section is important for energy security and medical applications, as well as for nuclear and astrophysics. Firstly, by investigating the widely used methods for calculating photon absorption cross sections in the international arena and taking into account our progress in photon absorption cross section. We decided to use a simplified version of the modified Lorentzian model (SMLO) and the microscopic relativistic quasiparticle random phase approximation (RQRPA) method to calculate photon absorption cross sections. Secondly, to calculate the sub-photon neutron emission cross section, we adopt the MEND-G photonuclear reaction program which has been jointly developed by the China Nuclear Data Centre and Nankai University. By taking the photon absorption cross sections calculated by the SMLO and RQRPA methods as input quantities. Then, the optical model parameters and the corresponding energy level density and pair correction parameters of the reaction channel are adjusted using the automatic parameter tuning procedure.

Fisrt study of the ^{235}U multi-chance fission with the FIFRELIN code

M. Sabathé[1], O. Litaize[1], P. Tamagno[1] and T. Ogawa[2]

[1] CEA, DES, IRESNE, DER, SPRC, Cadarache, 13108 Saint-Paul-lez-Durance, France

[2] JAEA, 2-4, Shirakata, Tokai, Naka, Ibaraki, Japan, 319-1195

Neutron emission is a physical process which can occur at the different steps of the nuclear fission. The first source of neutron emission comes from the primary fission fragments after full acceleration. Indeed after scission, fragments are generally excited and neutron, gamma and conversion electron emission are the different ways to release this excess of energy. Another source of neutron emission comes from the deexcitation of the nucleus before scission. When the excitation energy is sufficient, neutron emission is energetically allowed. This phenomenon generates a competition between fission and neutron emission. This physical mechanism is called multi-chance fission. It becomes signifiant when the incident neutron has enough energy to eject one – or more – neutrons while leaving the residual nucleus with excitation energy larger than its fission barrier height. For energies lower than this limit, multi-chance fission is only possible by tunnel effect. Pre-scission neutron, which comes from multi-chance fission, can be released through two different mechanisms. The first ones are emitted from the compound nucleus. In this case, neutron emission mechanism is similar to the fragment neutron evaporation process. The second ones are emitted during the pre-equilibrium phase and the proportion of these neutrons depends on the excitation energy.

Multi-chance fission process is not yet implemented in the FIFRELIN code. The aim of this work is to add the possibility for FIFRELIN to simulate this phenomenon. At first, only the second and the third chance fission will be considered. It will allow to extend the scope of FIFRELIN for incident neutron energies up to 20 MeV approximately.

In this conference, we will discuss the physical models used in FIFRELIN and how they have been implemented into the code by two different algorithms. The purpose is to calculate the probability for the nucleus to undergo fission or to emit neutrons. The first algorithm uses energy dependent partial width ratio Γ_n/Γ_f and the second one uses ratio of evaluated multi-chance fission cross sections. We will also discuss how these methods calculate the spectrum of emitted neutrons, how we treat the input data during the deexcitation, and how this new functionality has been integrated in the existing code.

Assimilating fission-code FIFRELIN using Machine Learning

Guillaume BAZELAIRE[1]*, Abdelhazize CHEBBOUBI*[1]*, David BERNARD*[1]
Geoffrey DANIEL[2]*, Jean-Baptiste BLANCHARD*[2]

1) CEA, DES, IRESNE, DER, SPRC, Cadarache, Physics Studies Laboratory, Saint-Paul-lès-Durance, 13108, France
2) Université Paris-Saclay, CEA, Service de Génie Logiciel pour la Simulation, 91191, Gif-sur-Yvette, France

guillaume.bazelaire@cea.fr

Abstract: The CEA and furthermore the laboratory where the work has been carried develop several codes of nuclear fission including FIFRELIN. FIFRELIN is a Monte-Carlo code which describes the fission process in two steps. Firstly, the fission fragments are generated according to their mass, nuclear charge, kinetic energy, excitation energy, angular momentum and parity. Secondly, the de-excitation of the fragments is performed. FIFRELIN relies on four free parameters and outputs are calculated with their respective statistical uncertainty. Those free parameters are tuned in order to reproduce the average neutron multiplicity. In this work, FIFRELIN which relates input data to output data is considered as a black box. This work's goal is to find a suitable list of free parameters in order to obtain specific output data. Due to the Monte-Carlo method, the computation times are relatively high in regards on the uncertainty. In fact, an execution of FIFRELIN with reasonable uncertainty takes more than five minutes. Therefore, finding the good free parameters can take a long time since the input space is too big to be explored randomly. In this talk we propose to use Machine Learning to overcome such issue. Due to the small size of the database used to train models and the almost linear variation between inputs and outputs the Machine Learning algorithm used is the Krigeage. Actually the method works well; with a small amount of time – less than a few hours starting from scratch using 20 CPU – the algorithm developed produces a list of the four free parameters that gives the desired outputs data using FIFRELIN.

Theoretical study of forbidden non-unique beta transitions

Xavier Mougeot

Université Paris-Saclay, CEA, List, Laboratoire National Henri Becquerel (LNE-LNHB), F-91120 Palaiseau, France

xavier.mougeot@cea.fr

Abstract

Over the past decade, forbidden non-unique beta transitions have been identified to be of critical importance in several fundamental physics topics, such as the modelling of antineutrino flux from nuclear reactors or from the Earth, and the background modelling in dark matter experiments. The decay half-lives of long-lived beta-emitting radionuclides, which are typically highly forbidden transitions, are of importance for geo- and cosmo-chronology. These half-lives are determined by means of activity measurements that must be very accurate. In radionuclide metrology, the BIPM (International Bureau of Weights and Measures) has recently developed a new extended international reference system (ESIR) for establishing primary standards of pure beta-emitting radionuclides, based on the liquid scintillation counting technique. The beta spectrum is an essential input to decisively establish the primary activity of the sample. More generally, a better knowledge of these transitions is also of interest to improve the quality, the completeness and the accuracy of nuclear decay data.

Contrary to allowed and forbidden unique transitions, it is essential to take into account the nuclear structure of the initial and final states involved in forbidden non-unique transitions. This situation significantly complicates the formalism and the calculations. Realistic nuclear structure calculations are usually time demanding and require a certain expertise. The transitions can strongly depend on relativistic matrix elements that necessitate relativistic nuclear wave functions, while nuclear structure models are usually non-relativistic. A code that treats these transitions, which is fast and simple to use for a non-expert, is currently unattainable.

The present contribution will describe our recent work on the calculation of the forbidden non-unique beta transitions. Realistic nuclear structure information (one-body transition densities) has been determined in the non-relativistic shell model with the NushellX code, which makes use of Hamiltonians fitted to experimental data in different mass regions. The Behrens and Bühring formalism has been studied in detail and specific codes have been developed in order to treat the forbidden non-unique transitions. Several forbidden non-unique transitions, of differing degrees, have been compared to precise measurements. The sensitivity of the spectra to different assumptions has been studied: simplified or full numerical lepton current; determination of the relativistic matrix elements with the conserved vector current (CVC) hypothesis; and methods for determining Coulomb displacement energies. A summary of these results will be presented.

Systematic large scale Quasiparticle Random Phase Approximation calculations with Relativistic and Chiral interactions

L. González-Miret[1,2)], S. Péru[1,2)], JP. Ebran[1,2)], S. Hilaire[1,2)]

1) CEA, DAM, DIF, F-91680 Arpajon, France
2) Université Paris-Saclay, CEA, LMCE, F-91680 Bruyères-le-Châtel, France

luis.gonzales-miret@cea.fr

Abstract: The atomic nucleus is a quantum many-body system. This implies that nontrivial collective behaviors may arise. One of them are giant resonances, which consist of harmonic oscillations of all the nucleons as a whole [1]. These excitations are present on all nuclei along the nuclear chart, and as a fundamental excitation modes of the nucleus, they are involved in a plethora of nuclear phenomena.

To study giant resonances, some of the most used models are Random Phase Approximation (RPA) for nuclei without pairing and Quasiparticle Random Phase Approximation (QRPA) when pairing is accounted. This methods are able to describe collective excitations from a microscopic point of view. However, they are very computationally costly. For this reason, there is a lack of systematic QRPA calculations in the literature with interactions other than Gogny D1M [2]. In this project, we take advantage of the supercomputing resources and robust codes available at CEA-DAM to perform systematic strength function calculations for all nuclei using several state-of-the-art interactions. Our codes implement the QFAM method [3], which allows us to significantly reduce the calculation time of strength functions by linearizing the QRPA equations.

An extensive study has been done with the covariant interaction DD-PC1 [4], calculating strength functions of giant resonances of all multipolarities. We have observed a dependence of the strength function with the basis parameters of the underlying HFB starting point. To overcome this, we have developed a series of strategies that allow us to produce robust results.

On the other hand, as part of the PAN@CEA collaboration [5], we have studied the impact of triaxiality in strength functions of giant resonances of ^{24}Mg by using a computationally heavy chiral interaction [6]. This nucleus plays a pivotal role in many astrophysical processes, and a good theoretical characterization is key to explain them.

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Evaluation of nuclear data

Update of the CIELO U238 resonance evaluation to improve LWR performance with burnup and LEU lattice criticality

J. Malec,[1] A. Trkov,[1] R. Capote,[2] M. Pigni,[3] P. Schillebeeckx,[4] S. Kopecky[4], M. Hursin[4], and D. Rochman[5]

[1]*Jozef Stefan Institute, Ljubljana, Slovenia*
[2]*NAPC-Nuclear Data Section, International Atomic Energy Agency, Vienna, Austria*
[3]*Oak Ridge National Laboratory, TN, USA*
[4]*European Commission, Joint Research Centre Geel, Belgium*
[5]*Paul Scherrer Institute, Zurich, Switzerland,*

New U-235 and U-238 evaluations [1-3] were undertaken within the OECD/NEA Data Bank CIELO Project [4] and were adopted for the ENDF/B-VIII.0 library, which was released in 2018 [5]. Since then, several reports and publications were released that showed serious discrepancies with the light water reactor (LWR) performance of the previous ENDF/B-VII.1 library [6] in criticality studies as the function of the burnup, e.g., see Ref. [7]. A slight increase of the LWR reactivity was observed at the Beginning of Cycle (BOC) with a severe lost of reactivity at large burnups observed for the ENDF/B-VIII.0 library. Sensitivity studies showed some compensation effects at the BOC, but uniquely identified the U-238 evaluation as the responsible for the reactivity loss [7].

In this work we focused on studying changes in resonance cross sections of U-238 that may improve the observed trend as a function of burnup. It was found that capture cross section from 0.1eV up to 10eV was reduced in ENDF/B-VIII.0 evaluation by about 2% [,4,5,9] compared to the ENDF/B-VII.1 evaluation [6,10] as shown in Figure 1.

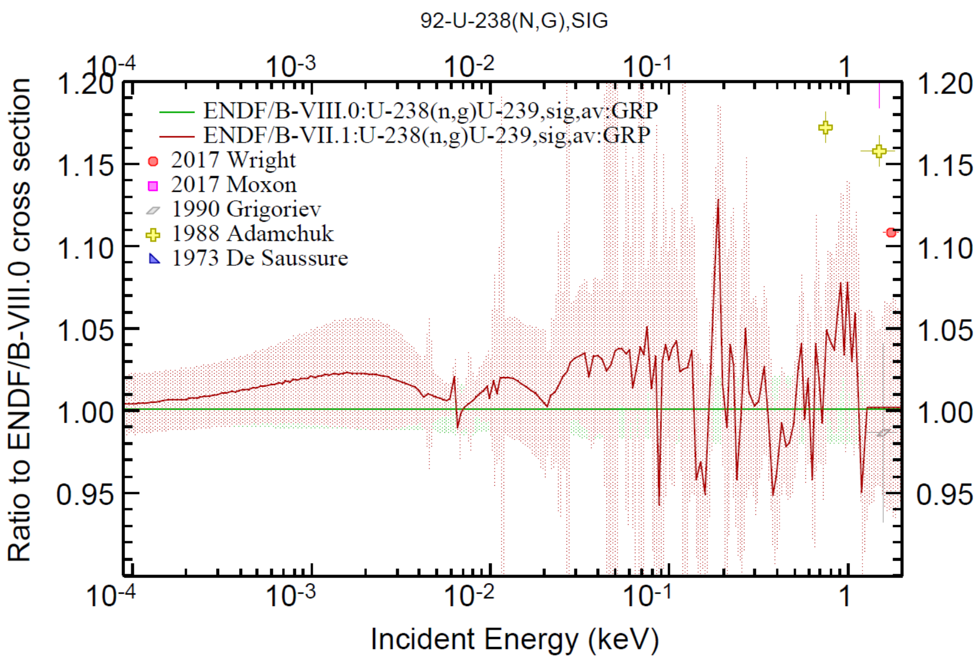

Figure 1. ENDF/B-VII.1 to ENDF/B-VIII.0 capture cross-section ratio in the resonance region.

The observed changes may explain the burnup trend as lower capture cross sections in U-238 below 10eV leads to increased criticality at the BOC, but lower capture above 100 eV results in lower criticality at higher burnup due to the reduced production of Pu-239. There is a new solution proposed by Japanese colleagues for the JENDL-5 library [11]. There is also a new RRR ev;uatin proposed by EC

JRC Geel colleagues. We would like to compare different solutions to check the impact on burnup as well as on LEU lattice criticality.

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Looking for one integral reference for the (n,f) reaction in actinides above 1 MeV

Ignacio Durán[1], Roberto Capote[2]

1) Universidad de Santiago de Compostela, Santiago de Compostela, 15701-SPAIN
2) International Atomic Energy Agency, AA-1400-Vienna, AUSTRIA

ignacio.duran@usc.es

Abstract:
The neutron-induced cross-section of actinides at energies above threshold (~1MeV) is evaluated based on the experimental data obtained from different experimental setups, having so systematic uncertainties that are difficult to minimize. One of the most important source of systematic error in the measured cross section is the absolute normalization of every dataset which is often performed by measuring simultaneously other isotope used as reference. In other experiments the shape of the cross-section spectrum is normalized using as reference the corresponding datafile retrieved from an evaluated library. The choice of the energy interval used as reference has been left up to the experimentalist criteria, leading to a hardly assessable uncertainty.
In this work the experimental datasets of the (n,f) cross section of principal actinides are reviewed looking for the best suited energy interval to be recommended for renormalization purposes. Using standard integration intervals, wide enough to get very low statistical uncertainties, should lead to a better normalization of every experimental dataset, reducing so the associated total uncertainty of the evaluated datafiles.

Contribution to the JEFF project

S. Kopecky[1), M. Hursin[2,3), C. Massimi[4), R. Mucciola[5), A.Oprea[1), ***C. Paradela[1)***, *D. Rochman[2),P. Schillebeeckx[1),*

1) European Commission Joint Research Centre, Retieseweg 111, Geel, 2440, Belgium
2) Paul Scherrer Institut, Nukleare Energie und Sicherheit PSI Villigen 5232, Switzerland
3) Ecole Polytechnique Fédérale de Lausanne, Laboratory for Reactor Physics and Systems behaviour, Lausanne 1015, Switzerland
4) Department of Physics and Astronomy, University of Bologna and Sezione INFN of Bologna, Via Irnerio 46, Bologna, 40216, Italy
5) Department of Physics and Geology, University of Perugia and Sezione INFN of Perugia, Via A.Pascoli, Perugia, 06123, Italy

Stefan.Kopecky@ec.europa.eu

Abstract: The contribution of JRC Geel to the JEFF project is twofold. At first we operate two accelerator facilities to provide a wide range of experimental data, such as energy dependent cross section data or prompt gamma spectra data. Another important aspect of our work is to provide evaluated parameters for the resonance region.
Here the focus lies on including as much as possible reliable and well described data from the literature. This can be either by including published data in the resonance shape analysis or when these data are not avaible, reported capture kernels can be combined with reported resonance parameters. An example is molybdenum. In literature results of measurements with isotopically enriched samples are available, but unfortunately neither the transmission not the capture yields are in the EXFOR database. By combining all published information with new measurements of natural samples, consistent and improved resonance parameters for all of its stable isotopes could be obtained.
For other elements we are developing a methodology where at first the limitations of energy dependence cross section data are carefully mapped. If information on parameters is not directly accessible though energy dependent data, choices on parameters are then guided by results of integral benchmarks. As example, for ^{238}U in recent works a significant over prediction of the reactivity loss (by around 800pcm) during depletion calculation when comparing JEFF-3.3 to the JEFF-3.1.1 library has been linked to biases in the radiation widths of the resonances. However from energy dependent cross section data an unambiguous determination of both the neutron and radiation width is not possible. As the neutron width is smaller than the radiation width, the radiation width has to be derived from the width of the observed resonance profile and the result is strongly influenced by corrections, such as Doppler broadening and experimental resolution. Progress on the work on such methodology will be reported.

Direct radiative capture calculations on ^{56}Fe

G. Gkatis[1,2], M. Diakaki[2], A. Mengoni[3,4,5], G. Noguere[1]

1) CEA/DES/IRESNE/DER/SPRC/LEPh, Cadarache, F-13108 Saint Paul Lez Durance, France
2) Department of Physics, National Technical University of Athens, GR-15780 Athens, Greece
3) European Organisation for Nuclear Research (CERN), CH-1211 Geneva, Switzerland
4) Istituto Nazionale di Fisica Nucleare (INFN), IT-40127 Sezione di Bologna, Italy
5) Agenzia nazionale per le nuove tecnologie (ENEA), IT-00196 Roma, Italy

georgios.gkatis@cea.fr

Abstract: Iron is one of the main structural materials used in nuclear infrastructures and technological applications. In the case of nuclear reactors, steel alloys are used in several parts, mainly for the pressure vessel, for building the core structures and in some reactors as reflectors. For this reason, accurate neutron cross section data are indispensable for the design and reliable operation of such facilities.

^{56}Fe is the most abundant naturally occurring isotope, it amounts to 92% of natural iron. The neutron cross section data of iron were studied under the CIELO project [1]. Based on this study, two main issues were discovered in the evaluated cross sections of the $^{56}Fe(n,\gamma)$ reaction. Those issues were addressed by implementing changes in the ENDF/B-VIII.0 evaluation [2]. Specifically, an artificial background was added in the energy region of 10 eV to 100 keV, in order to properly reproduce integral measurements in this energy range, and also the cross section above 850 keV was increased based on experimental data provided by the RPI [3].

This work aims to provide a physical interpretation behind these changes by exploring the direct radiative capture mechanism for ^{56}Fe. For the calculations, a dedicated code was utilized [4,5]. In this presentation, the first results of the direct capture cross section of ^{56}Fe will be presented.

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Ongoing developments at the Decay Data Evaluation Project

S. Leblond, C. Dulieu, M. A. Kellett, X. Mougeot

Université Paris-Saclay, CEA, List, Laboratoire National Henri Becquerel (LNE-LNHB), F-91120 Palaiseau, France

Sylvain.leblond@cea.fr

Abstract:

The Decay Data Evaluation Project (DDEP) is an international collaboration providing easy-to-use and reliable decay data recommendations. It was originally founded in 1994 between two National Metrology Institutes (NMIs) for ionising radiation: the *Laboratoire National Henri Becquerel* (LNE-LNHB) in France and the *Physikalisch-Technische Bundesanstalt* (PTB) in Germany. The collaboration was later strengthened through the addition of laboratories from the US, China, Romania, Russia, Spain and the UK. Beyond the initial scope of ionising radiation metrology, the DDEP recommendations are used for a wide variety of topics, from fundamental physics to nuclear medicine applications. Most notably, almost all 220 DDEP evaluations are included in the version 3.3 of the radioactive decay data library from the Joint Evaluated Fission and Fusion (JEFF) file project of the OECD Nuclear Energy Agency.

An overview of the DDEP collaboration will be presented. The evaluation process and recommendations will be detailed as well as the various dissemination media available. In particular, the recent update of the LNHB website, and the associated online tool *Laraweb* will be highlighted. The current status of DDEP, as well as the future updates and developments will be presented, in particular the possible inclusion of total absorption gamma-ray spectroscopy (TAGS) measurements. The production of the future JEFF-4 Radioactive Decay Data Library will also be discussed.

Evaluation of nuclear data using the Half Monte Carlo technique

Erik Andersson Sundén[1], Tyra Axén[1], Olle Diös[1], Alf Göök[1], Viktor Lindström[1], Henrik Sjöstrand[1], Axel Wohlin[1]

1) Uppsala University, Uppsala, Lägerhyddsvägen 1, 752 37, Sweden

erik.andersson-sunden@physics.uu.se

Abstract: The Total Monte Carlo (TMC) technique has proven to be a powerful tool to propagate uncertainties in nuclear data to the uncertainty in macroscopic quantities, such as neutron fluxes at detector positions and the criticality of reactor cores, and radiation damages in materials in the presence of radiation.

The uncertainties of nuclear data can be used to create self-consistent sets of cross-sections. Each set contains files that are generated by variations of nuclear model parameters to properly fit the model to the nuclear data, accounting for their uncertainty. These files are called random files. The random files will reflect the covariances of cross sections due to the variations of model parameters.

TMC uses Monte Carlo transport codes to transport particles through arbitrarily complex geometries. Each set of random files is used in a separate Monte Carlo transport code run. This allows for the propagation of uncertainties in nuclear data, which otherwise could be hard to account for in the transport codes. However, Monte Carlo techniques are well-known to be computationally expensive.

The Half Monte Carlo (HMC) technique uses the random files of the TMC technique but does not rely on Monte Carlo transport codes to propagate the uncertainties of nuclear data to the uncertainty of the sought macroscopic quantity. Instead, it uses pre-calculated sensitivity matrices to calculate the difference in a macroscopic quantity, given the difference of the random files relative to the best estimate of the nuclear data evaluation. This calculation is based on the assumption that the Monte Carl transport code can be replaced with a first order Taylor expansion.

In this work, we demonstrate how to use the HMC technique to calculate the deviation of macroscopic quantities in integral experiments for a set of random files relative to the best nuclear data evaluation. Furthermore, we show how these results can be used to incorporate integral experiments into an automated nuclear data evaluation.

Progress towards the ENDF/B-VIII.1 release

Gustavo P.A. Nobre[1], David A. Brown[1]

1) National Nuclear Data Center, Brookhaven National Laboratory, Upton, New York 11973-5000, USA

gnobre@bnl.gov

Abstract: The efficient, safe, and high-performance design and operation of nuclear reactors, and other nuclear applications, rely on complete and accurate nuclear data files. The ENDF/B library is the main nuclear reaction data library in the United States and one of the main ones in the world [1]. It encompasses sub-libraries for neutron, proton, alpha, deuteron, ^{3}He, and gamma projectiles, in addition to thermal neutron scattering law, fission product yields and atomic ones. The last release of ENDF/B was its VIII.0 version in 2018 [1]. Since then, many important and impactful developments have been made, in particular for actinides, structural materials and light elements. Some of these contributions are part of the IAEA-coordinated International Nuclear Data Evaluation Network (INDEN) [2]. This warrants a new library release, namely ENDF/B-VIII.1, which is scheduled for early 2024. This release is planned to happen in both ENDF-6 and GNDS-2.0 formats. The present work will describe the updates expected to be present in ENDF/B-VIII.1, as well as the review process and quality control procedures developed and implemented at the National Nuclear Data Center (NNDC). We will also show preliminary results from the Beta versions already released.

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A Nuclear Data Evaluation Pipeline for the Fast Neutron Energy Range - using heteroscedastic Gaussian processes to treat model defects

Alf Göök[1), Henrik Sjöstrand[1), Joachim Hansson[1), Erik Andersson-Sunden[1)

1) Department of Physics and Astronomy, Uppsala University, Box 516, 751 20, Uppsala, Sweden

alf.gook@physics.uu.se

Abstract: In this paper, we discuss the development of a nuclear data evaluation pipeline, created at Uppsala University. The pipeline focuses on the evaluation of the fast neutron energy range, above the resolved resonances. The evaluation methodology is based on the Levenberg-Marquardt algorithm, a natural extension of the Generalized Least Squares method to non-linear models. The nuclear model code TALYS is combined with relevant experimental data to produce nuclear data evaluations. A strong focus in the development lies on automation and reproducibility to enable rapid testing of new algorithms and modified assumptions. Several novel concepts for nuclear data evaluation methodology are implemented in the pipeline. This includes automated procedures to identify and correct unrecognized sources of uncertainty in experimental data. Additionally, ways to treat model defects using Gaussian processes on energy-dependent model parameters are implemented.

A particular problem in evaluating the neutron-induced reaction cross-section using optical and statistical models, as implemented in TALYS, relates to the intermediate energy range. While TALYS only predicts the smooth average cross-section, experiments reveal unresolved resonance-like structures. We explore ways to treat this type of model defect using heteroscedastic Gaussian processes to automatically determine the distribution of experimental data around an arbitrary, smooth cross-section curve. We will discuss the practical implementation of these concepts in the context of a tentative evaluation of ^{52}Cr neutron-induced reaction cross-sections.

Challenges in Nuclear Data Evaluations of Light Nuclear Systems

Helmut Leeb[1), Thomas Srdinko[1)

1) Atominstitut, TU Wien, Stadionallee 2, Vienna, 1230, Austria

Helmut.leeb@tuwien.ac.at

Abstract: Reactions of light nuclear systems play an important role in the development of nuclear technologies and design of nuclear facilities, but also in various fields of science e.g. in nuclear astrophysics, medicine, materials science, space science etc. Therefore a good knowledge of the relevant reaction cross sections on the basis of reliable and consistent nuclear data evaluations are required. At present the situation of evaluated nuclear data files for light nuclear systems is not fully satisfactory. There are several difficulties in the evaluation of reaction data of light nuclear systems, e.g. the extended resonance range up to high energies, the lack of quantitative microscopic models, the occurrence of dominant breakup channels and open questions concerning uncertainty information. Usually R-matrix theory is applied to describe the resonance range. Albeit non microscopic, R-matrix theory satisfies the conservation rules and yields a consistent and quantitatively reliable set of reaction cross sections. However, R-matrix theory is limited to two-body channels. The description of breakup channels is frequently given within a perturbative approach, whose applicability to a dominant channel is questionable. Another problem represents the determination of reliable uncertainty information from R-matrix analyses. In this contribution various challenges of nuclear data evaluations of light nuclear systems are addressed and possible solutions are discussed. Especially, we will consider at the example of specific light nuclear systems the treatment of dominant breakup channels within an R-matrix analysis, the extension of R-matrix analyses to higher energies and a suggestion for the generation of uncertainty information associated with R-matrix analyses.

Advancing the theory of nuclear data evaluations[1*]

Goran Arbanas[1)], Jesse Brown[1)], Dorothea Wiarda[1)]

1) Oak Ridge National Laboratory, Oak Ridge, TN 37831, United States

arbanasg@ornl.gov

Abstract: We present recent advances in the R-matrix formalism as well as the Bayesian evaluation framework for improved nuclear data evaluations. The advances in the R-matrix formalism include: 1) direct processes, 2) doorway, as well as multistep, processes, and 3) various forms of the Reich-Moore approximation for eliminated capture channels. Furthermore, to address unreasonably small posterior uncertainties often encountered in nuclear data evaluations of large data sets using the conventional form of the Bayes' theorem, we introduce *imperfections* (of the data or the model) as a formal evaluation tool for taming the evaluated uncertainties in harmony with Bayes' theorem. These theoretical advances were motivated by the nuclear data evaluations of differential resolved resonance cross section data using the code SAMMY, as well as the integral benchmark experiments using the SCALE code system, being performed at Oak Ridge National Laboratory for the Nuclear Criticality Safety Program. Some pedagogical applications of the new formalism, as well as a snapshot of the SAMMY modernization efforts, will be presented.

Acknowledgment: This work was supported by the Nuclear Criticality Safety Program, funded and managed by the National Nuclear Security Administration for the Department of Energy.

Study of (n,2n) reaction cross section of fission product based on neural network and decision tree models

Xiaodong Sun[(1)], Zihao Wei, Yuan Tian, Duan Wang, Ruirui Xu

[(1)] China Institute of Atomic Energy, P. O. Box 275, Beijing 102413, China

1214518959@qq.com

Abstract:
The neutron induced nuclear reaction cross section of fission products is related with the neutron flux and the reactor burnup, which plays an important role in accurately designing nuclear engineering. To predict (n,2n) reaction cross sections especially those without experimental data, the relevant features were analyzed and the experimental data set were established on the basis of sorting out the experimental data recorded in EXFOR library. This work includes 5294 (n,2n) cross section measured results, among which a lot of experimental data concentrate around 14 MeV incident energy. Moreover, there may be divergence between measurements due to system error and negligence error. Faced with these real and defective data, researchers discovered laws behind them and established the compound nucleus reaction models. However this means a heavy workload. It would be surprising if machine learning could reach a quantitative level close to the evaluation results. The 8 features include the proton number Z, the mass number A, the single nucleon separation energy of both proton and neutron, the Casten factor, the level density, the pairing correction, and the incident energy. The back propagation artificial neural network (ANN) and decision tree (DT) models were built to learn the experimental data set, respectively, adopting PyTorch and XGBOOST toolboxes. Draw lessons from the variational auto-encoder network, the 2 sub-networks with the same internal structure, which contains 128 neurons in 2 hidden layers, were designed to learn the mean and variance respectively. The boosting model integrates 16 decision trees. The training set includes 4 000 uniform and randomly selected data, while the remaining data constitutes the test set. The results show that both ANN and XGBOOST models describe the experimental cross section data well, moreover model gives a smooth and continuous curve, indicating a certain predictive ability. For the case of lack of experimental data, the predictions are also basically consistent with the evaluation nuclear reaction data libraries. Compared with the XGBOOST model, the ANN model has somewhat better generalization ability in the range of neutron incident energy above 20 MeV. In the test set, the ANN predictions with a mean absolute percentage error (MAPE) deviation less than 10% from the experimental data account for more than 85%. Therefore we successfully established machine learning models to analysis and predicate (n,2n) reaction cross section. On the other hand, through traditional nuclear reaction models, one can intuitively understand the relationship among physical quantities and build a picture of the reaction mechanism. However, machine learning models are often regarded as difficult to understand black boxes, which is questionable for physicists. In future it is planned to further apply machine learning for the nuclear data research to verify the rationality of the method.

Uncertainties and covariance matrices

Development of a new module to process covariances in the IRSN nuclear data processing code GAIA

Pierre SOLE[1,2], Vaibhav JAISWAL[1], Vivian SALINO[1], Cédric JOUANNE[2]

1) Institut de Radioprotection et de Sûreté Nucléaire, PSN-RES/SNC/LN, Fontenay-aux-Roses, 92260, France
2) Université Paris-Saclay, CEA - Centre de Saclay, Service d'Etudes des Réacteurs et de Mathématiques Appliquées - SERMA, Gif-sur-Yvette, 91191, France

pierre.sole@irsn.fr

Abstract:

It is crucial to account for uncertainties related to nuclear data when analysing and interpreting neutronics simulation results. The uncertainties associated with the nuclear data are available in the standard nuclear data libraries, usually in the form of covariance matrices [1]. However, using these matrices for uncertainties propagation in the neutronics simulations is not always straightforward and requires the data to be processed.

IRSN is working on the development of the nuclear data processing code GAIA [2]. GAIA has various modules to process cross sections, like the DOP module for reconstruction and Doppler broadening, TOP module for treating probability tables in the unresolved resonance region, and SAB for calculations related to the neutron thermal energy region. However, GAIA does not yet have a full capability to process covariances. To address this limitation, a new module named COP is under development, which will process covariance matrices and provide comprehensive capabilities for processing cross section (File 33), angular distribution (File 34), and resonance parameter (File 32). This paper presents the development carried out in the COP module. Also, preliminary results obtained using the COP module are presented, and are compared with those obtained using the ERRORR module of NJOY [1] and PUFF module of AMPX [3].

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Producing uncertainties and covariance matrix from intermediate data using a Monte-Carlo method

Greg HENNING [(1)], Francois CLAEYS [(1)], Nicolas DARI BAKO [(1)], Philippe DESSAGNE [(1)], Maelle KERVENO [(1)]

[(1)] Institut pluridisciplinaire Hubert Curien, F-67037 Strasbourg, France

Abstract: The necessary improvement of evaluated nuclear data for nuclear applications development is possible through new and high quality experimental measurements. In particular, improving (n, n') cross-section evaluations for faster neutrons is a goal of interest for new reactor fuels.

Our group at CNRS-IPHC developed an experimental program to measure (n, n' γ) cross-section using prompt gamma-ray spectroscopy and neutron energy determination by time-of-flight [1-3], with a focus on reaching the highest achievable level of accuracy. The collected partial cross-section can then be used to infer the total (n, n') one and contribute to evaluation improvement [1]. The extraction of the partial (n, n'γ) cross-sections from the recorded data involves using many external parameters (detector efficiencies, distance of flight...). Additionally, the steps of the data processing (event selection, calibration...) may introduce extra uncertainties and correlations between data points.

The usual method for combining and computing uncertainties is to use analytical developments based on the perturbation theory (e.g. $u_{f(x)}^2 = (\partial f/\partial x)^2 \times u_x^2$).
With multiple parameters and sources of uncertainty, deriving the final total combined uncertainty can be long and complex. This method makes the calculation of covariance hard and the inclusion of some unusual form of uncertainty (asymmetric, non-Gaussian) even more difficult. To overcome this issue, we developed a process relying on random sampling methods (a.k.a *Monte Carlo*) that processes intermediate analysis data to compute final values (cross-sections), uncertainties and covariance.

As a benchmark, we used the Monte Carlo method on ^{238}U [4] and reproduced the central values and uncertainties calculated using the analytical method. In addition, we were able to produce covariance matrices for (n, n'γ) cross-section. In some particular cases, the random sampling method is able to produce finer uncertainties that reflect the original data, while the analytical method smooths some features.

After explaining the method, presenting the results, we will discuss possible applications and extension of the method to other data set.

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Target Accuracy Requirements (TAR) Exercise within WPEC/SG46 and Feedback on Nuclear Data Needs

O. Cabellos on behalf of the SubGroup46 members

Department of Energy Engineering and Instituto de Fusion Nuclear - "Guillermo Velarde", Universidad Politécnica de Madrid, 28006, Madrid, Spain

e-mail : oscar.cabellos@upm.es

Abstract: A Target Accuracy Requirements (TAR) exercise is presented which aims at quantifying nuclear data needs, in terms of uncertainty reduction, to meet target accuracies on specific integral parameters driven by reactor and fuel cycle designs.

A first TAR exercise was performed at NEA in the period 2005-2008 within the framework of WPEC/SG26 on “Uncertainty and target accuracy assessment for innovative systems using recent covariance data evaluations” [1]. The WPEC/SG26 exercise did provided a total of 15 entries to the NEA High Priority Request List (HPRL) [2] which served as guidance for new experiments and data evaluations.

In 2018, a second TAR Exercise was launched in the framework of the WPEC/S46 on “Efficient and Effective Use of Integral Experiments for Nuclear Data Validation” [3].

Firstly, participants in the 2nd TAR Exercise reviewed the status of design target accuracies and their potential evolution for both traditional systems and new reactors concepts. Participants provided the definition of models and sensitivity profiles for the key integral values in a set of new reactor designs.

Then, a new TAR methodology was defined within WPEC/SG46 considering nuclear data corelations in energy, reactions and isotopes in the inverse-optimization problem. Moreover, covariance data from recent nuclear data evaluation projects were processed in only seven energy groups which were defined on physical considerations.

Results of the 2nd TAR Exercise will be presented demonstrating the effectiveness of the TAR outcomes to provide new requirements of nuclear data uncertainty reduction for the NEA- HPRL. The work will have a significant impact in prioritizing new experiments, both differential and integral; as well as in fostering international collaboration.

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Nuclear Data Uncertainty Quantification for Reactor Physics Parameters in Fluorine-19-based Molten Salt Reactors

Sigfrid Stjärnholm[1), Zsolt Elter[1,2), Henrik Sjöstrand[1)

1) Division of Applied Nuclear Physics, Department of Physics and Astronomy, Uppsala University, Box 516, Uppsala, 751 20, Sweden
2) Seaborg ApS, Titangade 11, 2200 Copenhagen N, Denmark

sigfrid.stjarnholm.2141@student.uu.se

Abstract: The use of the F-19 isotope in the nuclear fuel cycle is already well established for fuel enrichment, but future plans for Gen-IV reactors such as Molten Salt Reactors could utilize a fluorine-based salt as a basis for the fuel. It is therefore imperative that an understanding of the characteristics of F-19 is instituted, and one component of key interest is the quantification of reactor parameter uncertainties that arise from the uncertainties in the nuclear data. The results from such analyses can shed light on where experimentalists need to further improve nuclear data for F-19, as well as yielding critical information for developing and optimizing reactor designs thanks to greater knowledge in the uncertainties that result from nuclear data.

In this work, we analysed a molten salt reactor based on the designs made by Transatomic Power. Uncertainty quantification was performed for two operating modes of the reactor – a thermal mode, and an epithermal mode with a faster neutron spectrum compared to the thermal mode due to the use of less moderator rods. We generated nuclear data that was sampled from the covariance matrices in the JEFF-3.3 nuclear data library using SANDY and NJOY. By utilising the Total Monte Carlo-approach, we propagated the uncertainties from the samples to uncertainties in the neutron multiplication by simulating the reactor in OpenMC, a Monte Carlo-based neutron transport code. Individual reaction channels were perturbed while keeping others constant, allowing for quantification of how much a single reaction channel contributes to the overall uncertainty.

For the thermal reactor, the F-19 data sampling resulted in an uncertainty in reactivity of 61.5 pcm. The main contributors of the reactivity uncertainty for the thermal reactor are elastic scattering, neutron capture and alpha production. The epithermal reactor, with a reactivity uncertainty of 213.4 pcm, is mostly affected by elastic scattering, inelastic scattering, and alpha production. The alpha production channel had an unexpectedly large contribution, and it should be investigated further. Quantitatively, we observe that scattering plays a bigger role for the uncertainty in the epithermal system, a phenomenon which could be explained by the fact that with less moderation in the form of moderator rods, the role of F-19 in thermalization of neutrons is greater, and hence its contribution to the uncertainty is greater.

Application of nuclear data covariance matrices to representativity calculations for fast reactors

Federico Grimaldi[1,3], Patrick Blaise[2], Luca Fiorito[1], Antonin Krása[1], Anatoly Kochetkov[1], Pierre-Etienne Labeau[3], Pablo Romojaro[1], Gert Van den Eynde[1], Guido Vittiglio[1], Jan Wagemans[1]

1) Belgian Nuclear Research Centre, SCK CEN, Mol, Belgium
2) CEA Saclay, DES/Scientific Division Energies, Gif-sur-Yvette, France
3) Service de Métrologie Nucléaire, Université libre de Bruxelles, Bruxelles, Belgium

Federico.Grimaldi@sckcen.be

Abstract:
In the framework of the Multipurpose HYbrid Research Reactor for High-tech Applications (MYRRHA), the Belgian Nuclear Research Centre has carried out several experimental programmes since 2011 at the VENUS-F zero power reactor [1]. Those experiments are meant for nuclear data and code validation, while also aiming at the representation of the neutronic phenomena happening in a fast spectrum facility. Throughout the years, the VENUS-F core has been loaded with a number of configurations (critical and subcritical) in a process of progressive adaptation to the evolving MYRRHA core designs.
MYRRHA is a 30%-enriched MOX fuelled reactor design, foreseen to be cooled by Pb-Bi eutectic (LBE). Thanks to the versatility of VENUS-F, a wide range of materials has been loaded in the VENUS-F core, mainly: Pb and Bi to simulate LBE and U fuel (metallic, 30% w.t. enrichment). In order to improve what was done so far in terms of similarity to MYRRHA, possible MOX core loadings at VENUS-F are now under investigation. This analysis started with a representativity study performed at the level of the 2D assembly, assessing a number of possible MOX assemblies to be loaded in the VENUS-F core and how they compare to an analogous model of MYRRHA. The representativity r of one model to another is computed as the model correlation coming from the nuclear data, which is a measure of how coherently the two designs react to the same change in the nuclear data themselves. This measure was used for model comparison of thermal reactors in the past [2]. Despite the effect of the nuclear data evaluation being limited on the sensitivity profiles, the impact of different covariance evaluations on r is strong. This work aims at the investigation of the impact of the nuclear data library choice on r as well as at a definition of which steps should be taken to allow for complete representativity analyses.

[1] J. Wagemans, «Integral experiments in the VENUS-F reactor», Annals of Nuclear Energy, 2021
[2] S. Cathalau, «An application of sensitivity and representativity approach for the design of a 100%MOX BWR experimental program in ZPR», Annals of Nuclear Energy, 2021

Processing and benchmarking

Radiation Safety Information Computational Center (RSICC): An Information Analysis Center for Nuclear Science

Timothy E. Valentine

Nuclear Energy and Fuel Cycle Division, Oak Ridge National Laboratory, One Bethel Valley Road, Oak Ridge, Tennessee 37831 USA

valentinete@ornl.gov

Abstract: The Radiation Safety Information Computational Center (RSICC) was founded in 1962 at Oak Ridge National Laboratory (ORNL) and is a specialized information analysis center under the auspices of the Office of Science and Technical Information within the U.S. Department of Energy (DOE). RSICC has served the international nuclear community for over six decades through its efforts to collect, archive, and distribute information, data, and modeling and simulation (M&S) tools for a broad range of nuclear technology applications. RSICC is the sole organization responsible for the distribution of the MCNP® Monte Carlo code that is a reference tool for testing and evaluating nuclear data libraries. MCNP® is one of the primary tools utilized by the Validation of Nuclear Data Libraries (VaNDal) subgroup under the Working Party on Evaluation and Cooperation (WPEC) of the Nuclear Energy Agency (NEA) of the Organization for Economic Cooperation and Development (OECD). Since 2010, RSICC has distributed over 23,000 copies of the MCNP® code to our Worldwide customer base. Over the past decade, RSICC has distributed over 1,000 copies of MCNP® annually to our customers located in the U.S. and abroad. Approximately 50% of the MNCP® software packages distributed have been provided to U.S. universities and sponsored organizations whereas over 27% of the packages have been provided to foreign organizations and over 22% to domestic organizations that are not supported by RSICC's sponsors. While there is high demand for the code, the distribution of the code is limited to only approved countries because the code is regulated by the U.S. National Nuclear Security Administration. This paper provide will provide a general overview of RSICC's activities, services, and systems; provide information regarding Federal export control regulations for codes such as MCNP®; and provide recommendations for the control and use of RSICC software.

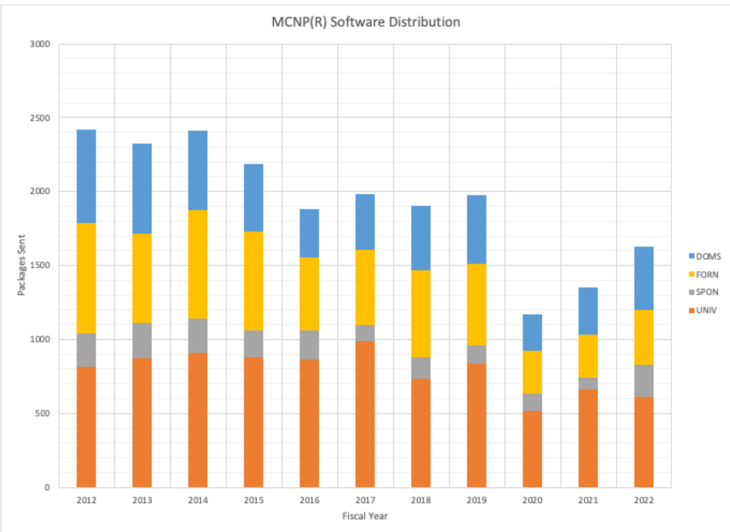

Figure 1. MCNP® Annual Distribution Statistics

Integrated, Automated, and Reproducible Nuclear Data Processing at the NEA

Daniela Foligno [1)]

1) OECD/Nuclear Energy Agency, 2 rue André Pascal, 75775 Paris Cedex 16, France

Daniela.FOLIGNO@oecd-nea.org

Abstract:
The preservation of the data is a priority at the NEA Data Bank. In this context, “preserving data” means: verifying, processing, sharing, improving, and storing the data. The NEA wants to automatize this process as much as possible in order to provide the JEFF community with reproducible good quality data. The Data Bank decided to work with GitLab, a web-based distributed Version Controlled System that allows tracking changes over time and across different users. The author will present the current status of the NEA pipeline, an ongoing collaborative effort to standardize the way nuclear data is processed..

Preliminary investigation of nuclear data sampling for the new Monte Carlo code TRIPOLI-5®

C. Larmier[1]

1) Université Paris-Saclay, CEA, Service d'Etudes des Réacteurs et de Mathématiques Appliquées, 91191, Gif-sur-Yvette, France

coline.larmier@cea.fr

Abstract: A new massively parallel Monte Carlo particle transport code devoted to novel HPC architectures is currently under development at CEA: TRIPOLI-5®. As a preliminary step towards extended verification of the implemented routines, we have accurately tested the sampling laws for the neutron physics as a function of nuclear data, first within the so-called « free-gas » model and then including thermal neutron scattering to probe crystallography or molecular bond-effects. For this purpose, a code-to-code comparison has been performed between TRIPOLI-5® and two other reference Monte Carlo transport codes, TRIPOLI-4® and OpenMC, over around 560 isotopes from the JEFF-3.3 nuclear data library. For the sake of simplicity, probability tables for the treatment of unresolved resonance range were initially inhibited. More than 5000 configurations have been tested for a simple benchmark, consisting in a sphere filled with a single isotope, with a pointwise, isotropic, single-energy source located at the center of the sphere (ten representative incident energies have been considered). The fiducial quantity for the benchmark is the flux per unit of lethargy: the tallies obtained with the three Monte Carlo codes have been compared using a Holm-Bonferroni statistical test. In order to systematically analyze the detected discrepancies, the energy and angle distributions for each isotope and reaction have been compared at various incident energies thanks to the Kolmogorov-Smirnov statistical test. Additionally, the microscopic cross sections and multiplicities read by each code in the nuclear data library have been carefully checked. TRIPOLI-5® and OpenMC rely on ACE files, which makes the comparison easier. TRIPOLI-4®, on the other hand, uses directly ENDF files without pre-processing, which is responsible of an increased number of discrepancies with respect to the two other codes. Overall, a very good agreement has been found between the three codes. This work has allowed validating the implementation of the free-gas model and thermal scattering laws in TRIPOLI-5®, mainly based on the almost perfect statistical agreement with respect to OpenMC. Besides, we have been able to highlight some inconsistencies in the nuclear data library and to detect and fix a few implementation errors in the sampling algorithms. Work is ongoing to perform a similar analysis on the unresolved resonance range for neutron physics, and on photon physics.

Recent development in the GALILÉE-1 processing code

Mireille Coste-Delclaux[1], Cédric Jouanne[1], Stéphane Mengelle[1], Frédéric Moreau[1], Claude Mounier[1], Thierry Visonneau[1],

[1] Service d'Études des Réacteurs et de Mathématiques Appliquées (SERMA), CEA, Université Paris-Saclay, F-91191, Gif-sur-Yvette, France

cedric.jouanne@cea.fr

Abstract: GALILÉE-1 is the new verification and processing system for evaluated data, developed at CEA. The main features of GALILÉE-1 are the following ones. We have already presented in the past the reconstruction and Doppler broadening of cross-sections implemented in the GTREND module of GALILÉE-1. The results obtained are of very good quality and we can explain the discrepancies observed with respect to other processing codes such as NJOY and PREPRO. This intensive comparison phase has increased the reliability of the GTREND module. GTREND is also able to handle the R-Matrix Limited format (LRF=7) with a large number of channels in the Resolved Resonance Region (RRR) and to calculate the angular distributions from the resonance parameters in this energy range.

The treatment of the Unsolved Resonance Range is a more complex problem because it is difficult to obtain experimental or theoretical references to evaluate the different results of the various processing codes. We have chosen to implement two approaches in order to produce usable data for the deterministic and Monte Carlo transport codes, APOLLO3® and TRIPOLI-4® developed at CEA, respectively. These transport codes use URR cross-section data in the form of multi-group probability tables on an energy mesh chosen by the user. This was previously done by the CALENDF code. The description of these probability tables is different from that of the NJOY PURR module which produces pointwise probability tables on a coarse energy mesh.

GALILÉE-1 has the capability to generate probability tables for all reactions. The competitive reaction given in resonance parameters section may be taken into account and that can have a significant influence on certain calculations. In order to generate probability tables for transport codes, such as MCNP or TRIPOLI-5, GALILÉE-1 can also produce pointwise TPs and thus replace the PURR module of NJOY. In the thermal energy range, GALILÉE-1 produces double differential sections, like NJOY's THERMR module, with refinement adjustment options on the energy grids.

In this paper we will focus on the following points. The R-Matrix Limited format is increasingly used for resolved domain resonance parameters. In particular, it allows several reaction channels to be specified, such as inelastic scattering, (n,2n), (n,alpha) or (n,p) reactions. We have analysed the different cases and more particularly those for which charged particles are produced. Comparisons with NJOY2016 and PREPRO21 will be presented for these cases. We will also look at the reconstruction of the anisotropies of the particles on the exit channel for these reactions and we will present the method developped in GALILÉE-1 to define a linearization grid of the anisotropy. We will show results comparing the reconstructed anisotropies to those available in the initial evaluation files.

We will present analyses of the normalisation of cross-sections in URR according to the options declared in the evaluations. GALILÉE-1, in order to process evaluated, performs many tests on the evaluations. We will finally present these tests and those dealing with the coherence of the reaction thresholds considered.

Improvements on the damage calculations using evaluated nuclear data and NJOY

Shengli Chen[1], David Bernard[2]

1) Sino-French Institute of Nuclear Engineering and Technology, Sun Yat-sen University, Zhuhai 519082, China
2) CEA, DES, IRESNE, DER, SPRC, LEPh, F-13108 Saint-Paul-Lez-Durance, France

chenshli23@mail.sysu.edu.cn

Abstract: NJOY is the only open-source nuclear data processing code allowing calculating neutron-induced displacement damage cross sections from evaluated nuclear data. However, there are many issues related to NJOY and/or evaluated nuclear data for the damage cross section calculation, such as the inconsistent DPA cross sections and KERMA factors induced by neutron capture reaction with photon data given in MF6 vs. MF12-15, incorrect recoil nuclear data in MF6, and the discrepancy of DPA cross sections using different approaches/nuclear data. The present work summarizes these issues and proposes the corresponding improvements.

Wonder 2023

Thermal scattering laws

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Advancements in Validation of TSLs through Inelastic Neutron Scattering and Transmission Measurements

Kemal Ramić[1], Chris W. Chapman [1], Goran Arbanas[1], Jesse Brown[1], Luke Daemen[1], Doug Abernathy[1], Alexander Kolesnikov[1], Yongqiang Cheng[1], Anibal Ramirez Cuesta[1], Daniel Siefman[2], Yaron Danon[3], Dominik Fritz[3]

[1]Oak Ridge National Laboratory
[2]Lawrence Livermore National Laboratory
[3]Rensselaer Polytechnic Institute

Abstract:

Historically, the free gas approximation has been used to treat the thermal scattering of neutrons with energies below a few electron-volts (eV) in unevaluated materials. However, this method inadequately reproduces neutron scattering at these energies. Until recently, only a limited number of materials had available thermal scattering law (TSL) files/libraries in the ENDF nuclear data libraries in this energy range. With advancements in atomistic modeling techniques, such as molecular dynamics, ab-initio molecular dynamics, and density functional theory, TSL libraries have become available for many more materials. This is particularly relevant due to the rising interest in several advanced reactor systems that require novel moderator and reflector materials.

While quasi-integral and integral benchmarks have been designed to validate historically important moderator materials (such as light water and polyethylene), there is currently a lack of standard validation methods for TSLs, especially when multiple conflicting TSLs exist. To address this issue, the Oak Ridge National Lab Nuclear Data group has been working on utilizing inelastic neutron scattering (INS) measurements combined with transmission (i.e., total cross section) measurements to evaluate and validate TSLs for different materials. We plan to demonstrate how this method has worked on materials such as polyethylene, lucite, and polystyrene. In addition, we will compare the newly created libraries to ENDF libraries for these materials and explain why integral benchmarks should not be used for validation when multiple TSLs exist.

This work was supported by the Nuclear Criticality Safety Program, funded and managed by the National Nuclear Security Administration for the Department of Energy.

Evaluation of thermal neutron scattering law of nuclear-grade isotropic graphite

Shinsuke Nakayama[1], Osamu Iwamoto[1], Atsushi Kimura[1]

1) Japan Atomic Energy Agency, 2-4 Shirakata, Tokai, Ibaraki 319-1195, Japan

nakayama.shinsuke@jaea.go.jp

Abstract: We have started a project entitled "Development of Nuclear Data Evaluation Framework for Innovative Reactor" in 2021. The objective of this project is to establish a scheme to improve the accuracy of nuclear data required in the development of innovative nuclear reactors within a short time period through collaboration between experiments and evaluations. Graphite is a candidate of moderator in innovative nuclear reactors such as molten salt reactors. Scattering of thermal neutrons by the moderator material has a significant impact on the reactor core design. Currently, ENDF/B-VIII.0 provides practically the only thermal scattering law (TSL) data for nuclear-grade graphite, and JENDL-5 adopts them. However, it has recently been pointed out that the TSL evaluation for nuclear-grade graphite employed in ENDF/B-VIII.0 have several concerns [1]. Under these circumstances, we newly evaluated TSL for nuclear-grade graphite.

The inelastic scattering component due to lattice vibration was evaluated based on the phonon density of states computed with first-principles lattice dynamics simulations. The simulations were performed for ideal crystalline graphite. This is based on the assertion in Ref. [1] that the vacancies in nuclear-grade graphite are larger in size than the crystals and other non-vacant region are highly crystalline. This is also in contrast to the modelling in ENDF/B-III.0 evaluation, in which carbon atoms are randomly removed from the crystal. The present evaluation and that of ENDF/B-VIII.0 were compared with the double-differential cross sections we have recently measured in the Materials and Life Science Experimental Facility (MLF) in the J-PARC in the temperature up to 500 K.

The coherent elastic scattering component due to crystal structure was evaluated based on neutron scattering and transmission experiments we recently performed in the MLF in J-PARC. The intensities of the individual Bragg peaks were evaluated through comparison with the experimental angular distribution of scattered neutrons. The sum of the inelastic and coherent elastic scattering components evaluated by the methods described above reproduced the experimental total cross sections well in the incident energy range above 10 meV. Below 10 meV, however, the experimental values were significantly underestimated. To resolve this discrepancy the small-angle neutron scattering (SANS) component was quantified. By adding the SANS component, the evaluated values reproduced the experimental total cross sections well.

This work was supported by MEXT Innovative Nuclear Research and Development Program.

[1] I. Al-Qasir et al., Ann. Nucl. Energy 161, 108437 (2021).

Experimental Validation of Thermal Neutron Scattering Law Data for Innovative Reactor

Jun-ichi Hori[1), Yoshiyuki Takahashi[1), Hiroshi Yashima[1), Kazushi Terada[1),
Tadafumi Sano[2), Yasunori Matsuo[2)

1) Institute for Integrated Radiation and Nuclear Science, Kyoto University, 1010-2, Asashiro-Nishi, Kumatori-cho, Osaka, 590-0494, Japan
2) Atomic Energy Research Institute, Kindai University, Kowakae, Higashi-Osaka, Osaka, 577-8502, Japan

hori.junichi.6e@kyoto-u.ac.jp

Abstract: Improvement of accuracy of thermal neutron scattering law data of hydrides as a candidate moderator is required for developing innovative reactor system. In order to meet the requirement, the project entitled as “Development of Nuclear Data Evaluation Framework for Innovative Reactor” is ongoing in collaboration with Kyoto University, Japan Energy Agency, Tokyo Institute of Technology and Kindai University. In the project, the validation of thermal neutron scattering law data of candidate moderator materials is one of the most important technical issues. In JENDL-5, the thermal scattering law data for 16 materials containing light water, heavy water were newly evaluated with the molecular dynamics simulations and the data for 17 materials containing yttrium hydride and zirconium hydride as a candidate solid moderator were taken from ENDF/B-VIII.0.

Therefore, we performed the integral experiments to validate the thermal neutron scattering law data of JENDL-5 using a 46-MeV electron linear accelerator pulsed neutron source at Institute for Integrated Radiation and Nuclear Science, Kyoto University. Fast neutrons from a Ta target as a photo-neutron source without a moderator were directly used. The collimated fast neutrons were incident on the sample assembly located at a 12m flight station. The neutrons moderated in the assembly were measured by a ^{6}Li-glass detector with a TOF method. The distance between the center of the assembly and the detector was 40 cm. On the other hands, the time-dependence of 2.2-MeV capture gamma rays from hydrogen in the assembly was also measured by a BGO detector. In this study, we used light water, calcium hydride, and zirconium hydride as samples. We compared the experimentally obtained time dependences of neutron and capture gamma-ray with the calculated ones using the Monte-Carlo simulation code and the evaluated data of JENDL-5.