H2020 ARIEL Hands-on school on nuclear data from Research Reactors

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Book of Abstracts

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NIRR-1'S FUEL ODYSSEY: EXPLORING BURNUP ESTIMATES, DECOMMISSIONING PLANS, AND RADIOACTIVE WASTE MANAGEMENT

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This abstract presents a detailed and comprehensive analysis of various aspects related to the Nigerian Research Reactor-1 (NIRR-1), including fuel burnup estimation, decommissioning strategies, radioactive waste management, and proliferation concerns. The study utilizes the WIMS-ANL computer code to assess the fuel burnup characteristics of both the existing Low-Enriched Uranium (LEU) fuel and the decommissioned Highly Enriched Uranium (HEU) fuel in the NIRR-1 reactor. The study focuses on understanding the fluctuation in atom density of key isotopes, namely U235, U238, Xe135, and Pu239, over the burnup time for both the HEU and LEU cores. A comprehensive evaluation of these isotopes' concentrations provides valuable insights into the reactor's operational dynamics and potential safety implications. In terms of fuel consumption, the analysis reveals that during the entire burnup period, the HEU fuel (UAL 4-Al) utilized 9.817257202 grams of U235, while the LEU fuel (UO2) consumed 12.04913749 grams of U235. Similarly, the HEU fuel generated 0.2042625596 grams of Pu239, whereas the LEU fuel produced 0.052503096 grams of Pu239. These findings contribute to a better understanding of the fuel utilization efficiency and the overall performance of the NIRR-1 reactor. The study also highlights the significance of the NIRR-1 reactor as a commercial version of the Miniature Neutron Source Reactor (MNSR) developed by the China Institute of Atomic Energy (CIAE). Operated by the Centre for Energy Research and Training (CERT) at Ahmadu Bello University, Zaria, the NIRR-1 reactor plays a crucial role in neutron activation analysis and minimal radioisotope generation. In line with international efforts to reduce the use of highly enriched uranium in civil nuclear applications, the HEU core of the NIRR-1 reactor has been converted to LEU fuel. Consequently, understanding the current LEU fuel's burnup characteristics is essential for effective fuel management and long-term reactor operation. The analysis confirms the reliability of the WIMS-ANL code in accurately estimating the fuel burnup for the potential LEU core, providing valuable insights for decision-making processes. Furthermore, the study addresses decommissioning strategies for the NIRR-1 reactor, considering its design features and core lifetime. The reactor is designed to operate at maximum flux for 2.5 hours per day, five days per week, with a core lifetime of 10 years. To compensate for the loss of reactivity due to core burnup, controlled beryllium shims are added to the top aluminum tray. This practice leads to increased fuel burnup and the accumulation of fission products throughout the reactor's operation. Considering the potential for proliferation concerns, the study examines the accumulation of fissile materials, particularly Pu-239, in the NIRR-1 reactor. The findings indicate that the accumulation of Pu-239 is insufficient to compensate for the reactivity loss caused by U-235 depletion. Additionally, the concentration of Pu-239 in the spent fuel remains below levels that raise concerns about nuclear weapon design and production. To ensure the safe and efficient operation of the NIRR-1 reactor throughout its life cycle, precise fuel burnup estimation is crucial. This knowledge helps monitor reactivity parameters, neutron fluxes, and power distributions within the reactor core. It also facilitates the estimation of the radioactive source term for safety analysis during accidental scenarios, safeguards requirements for fissile material monitoring, and the determination of cooling and shielding needs for spent fuel storage and transportation. In conclusion, this study provides a comprehensive analysis of fuel burnup estimation, decommissioning strategies, radioactive waste management, and proliferation concerns in the Nigerian Research Reactor-1 (NIRR-1) using the WIMS-ANL code. The findings contribute to a better understanding of the reactor's long-term operation, fuel utilization, and potential safety implications. The results validate the reliability of the WIMS-ANL code for fuel burnup estimation and underscore the importance of accurate fuel management for safe and cost-effective reactor operation.

Keywords: NIRR-1, Fuel odyssey, Burnup estimates, Decommissioning plans, Radioactive waste management

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Development and characterization of a novel single plane Compton gamma camera

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The detection of radiation in the environment is crucial, often requiring compact and portable instruments. One effective method for detecting gamma-ray sources is through the use of a Compton gamma camera. Unlike Anger cameras, Compton gamma cameras employ source localization techniques based on Compton scattering kinematics. Various types of Compton gamma cameras have been developed, including those based on semiconductor detectors, which offer excellent spatial resolution but limited efficiency and high costs, and those based on scintillator detectors, which utilize either photo-multiplier tubes or Silicon photomultipliers. Scintillator-based Compton gamma cameras generally provide lower angular resolution than their semiconductor-based counterparts, but they offer greater efficiency and cost-effectiveness. Most scintillator-based Compton gamma cameras consist of two separate detector readout planes: the scatterer and the absorber, or they implement complex designs to determine the depth-of-interaction of incoming gamma radiation.

In this study, we introduce a novel concept for a Compton gamma camera, utilizing segmented scintillators read out on a single side by silicon photomultipliers. Each detector element comprises two scintillator crystals optically coupled by a light guide. We employed GAGG:Ce scintillators measuring 3 mm x 3 mm x 3 mm and 3 mm x 3 mm x 20 mm plexiglass light guides. These detector elements were arranged in an 8x8 matrix with a 3.2 mm pitch, separated by ESR reflectors. In this configuration, the front scintillator layer acts as the scatterer, and the back scintillator layer acts as the absorber, both read out by the same silicon photomultiplier array coupled to the back side of the matrix. This distinctive feature minimizes the number of read-out channels, essential for a compact and portable device. The length of the 20 mm light guide was chosen based on a compromise between detector intrinsic efficiency and angular resolution, as determined through GEANT4 simulation studies.

Following the design, we constructed the Compton gamma camera and conducted laboratory tests to characterize its performance. The tests involved Cs-137 and Na-22 sources, with the setup temperature maintained between 18-20℃, and the silicon photomultiplier array read out by the TOFPET2 data acquisition system. Energy resolution characterization was performed individually for the front and back detector layers by irradiating the respective layers with the collimated source. The average energy resolution at 662 keV for the front and back GAGG:Ce layers was found to be $8.9 \pm 1.9\%$ and $10.8 \pm 1.6\%$, respectively, accounting for variations within the module.

To analyze the Compton events, we applied specific conditions based on kinematic criteria, such as the energy of each detector element and the geometry of the detector (e.g., distance between fired elements). These conditions ensured a clean sample of Compton events. The basic imaging test using a Cs-137 source (diameter $\approx$ 3 mm) positioned 50 mm in front of the module, employing a back-projection algorithm, revealed a distinct peak corresponding to the source, with a spatial resolution of $\sigma = 5.1 \pm 0.2$ mm. In this work, we will present detailed results characterizing the detector's performance, including efficiency estimation and imaging capabilities for gamma sources of different energies at varying distances and positions within the field-of-view. Finally, we will discuss the potential of this designed detector for highly compact and portable Compton gamma camera applications in environmental gamma-ray detection and localization.

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Comprehensive Review and Application of Neutron Activation Analysis for Elemental Concentration Analysis of Coffee Samples

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Neutron Activation Analysis (NAA) is a vital tool in the qualitative and quantitative measurement of elemental concentrations, with far-reaching applications in archaeology, geography, and agriculture, among others. This study applies NAA to quantitatively evaluate elemental concentrations in coffee samples and offers a comprehensive review of NAA —its operational principles and associated methodologies. Utilizing the TRIGA Mark II reactor at the Atominstitut in Vienna and using apple leaves powder samples as a standard reference, we meticulously prepared and irradiated the coffee samples for 180 seconds. The reactor has a power of 250 kW and a maximum thermal neutron flux density of 1x1013cm-2s-2. The samples are fixed in predetermined locations in the reactor core using a pneumatic transport system. Following irradiation, a High Purity Germanium (HPGe) detector was employed to measure the samples. Spectral analysis and comparison of elemental activity enabled the calculation of elemental concentrations. The radionuclides identified are Na, Ca, K, Cl and Mn with masses for each at 1.28 µg, 73 µg, 1349 µg, 10.13 µg and 2.07 µg, respectively. The findings reiterate the effectiveness of NAA in elemental composition quantification, with a spotlight on coffee samples, and underscore the study's contribution to consolidating knowledge around NAA.

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Development of Experimental Methods for Investigating Innovative Approaches to Nuclear Waste Management and to Nuclear Safety

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What happens to spent nuclear fuel? One promising future option might be fast (spectrum) reactors operated as a nuclear waste burners, i.e. fissioning plutonium and minor actinides (having half-lifes of tens of thousands of years) and essentially leaving fission products with storage times in the range of hundreds of years. One such fast reactor concept is a molten salt fast reactor (MSFR). The project "NAUTILUS" aims at developing experiments to contribute to the nuclear data base that is needed for the assessment of the parameters and the safety of that reactor concept. The experiments will be conducted at the education and research reactor (AKR-2) of the TU Dresden, Germany.

The three main prerequisites of that goal, which represent major tasks of the project, are (1) the determination of the neutron spectrum of the AKR-2, (2) the simulation of the neutron spectrum of the AKR-2 as well as certain experimental devices, and (3) the development/establishment of the pile-oscillator and the neutron transition method at the AKR-2. The latter methods (3) in combination with the well-characterized neutron field (1, 2) will be employed aiming at reducing uncertainties in the nuclear data base of chlorine-35/37. The gained knowledge will be used to investigate the feasibility of the chloride-based MSFR as a waste burner and assess the safety of the system.

The flash talk motivates the MSFR concept in the given context and gives an overview about the project. Furthermore, the experimental and computational steps gone so far and upcoming work are highlighted.

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Nuclear data measurement using accelerator-based neutron sources

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In order to develop new applications and advanced technologies the worldwide scientific community requires very precise and highly reliable cross-section measurements. In this talk, I will present a measurement of neutron-induced (n,γ), (n,p), & (n,2n) reaction cross sections in the fast neutron energy region using FOTIA and PURNIMA facilities at BARC, India [1-2]. I will further talk about the work carried out at IFIN-HH related to preparation of a neutron capture cross-section measurement at the n_TOF facility at CERN [3].

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^{50,53}Cr (n,γ) cross section measurement

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Chromium is a very relevant element regarding criticality safety in nuclear reactors because its presence in stainless steel, used as structural material. There are serious discrepancies between the different evaluated data of ^{50}Cr and ^{53}Cr neutron capture cross section which are not present in the corresponding estimated uncertainties. The Nuclear Energy Agency (NEA) opened an entry in their High Priority Request List (HPRL) to measure these reactions between 1 and 100 keV within 8-10% accuracy. Two experiments have been performed for this matter: one based on the time-of-flight technique at the n_TOF facility of CERN (Geneva, Switzerland) and another based on activation of ^{50}Cr at the HiSPANoS facility of CNA (Seville, Spain). The experimental set-up and the first results will be presented here.

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A New Irradiation Workbench Enabling Quantitative Measurements in the Fast and Thermal Neutron and Photon Field of a Polyethylene Shielded Americium-Beryllium Source

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The construction of a new irradiation workbench is almost completed. Once finished, it will enable the conduct of quantitative neutron and photon experiments in the field of an americium-beryllium source. Therefore, different radiation field quantities, i.e. spectral fluence rates, fluence rates, ambient dose rate equivalents and directional dose rate equivalents, were determined for the individual fast and thermal neutron and photon components of the source radiation field. This was done by means of evaluated detector measurements and FLUKA simulations at given distances from the source. All results all catalogued in an own database.

The presentation only concentrates on the quantification of the fast neutron field by means of spectral fluence rates. Methods include the calibration of a stilbene detector for neutron energy depositions using a pulse-shape-discrimination-aided Time-of-Flight setup and the unfolding of pulse-charge spectra via a simple unfolding technique. The validity of the results determined by measurements will be discussed by a comparison to analogue FLUKA simulation data.

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Opening session of the school, welcome

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Four Dimensional Isotope Tracking with the N4DP Instrument at MLZ

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Neutron Depth Profiling (NDP) is a non-destructive, element-specific, high-resolution nuclear analytical technique, which is often used to probe concentration profiles of lithium, boron, nitrogen, helium, and several other light elements in different host materials. The N4DP instrument is located at

the Prompt Gamma Activation Analysis (PGAA) beamline of Heinz Maier-Leibnitz Zentrum (MLZ), which provides a cold neutron flux up to $5 \times 10^{10}\,\mathrm{s}^{-1}\mathrm{cm}^{-2}$. When a neutron is captured by a specific nuclide, ions with well-defined energies are emitted. The energy loss of the charged particles traveling through the host material is related to the depth of origin at a resolution level up to tens of nanometers.

We developed a detector system based on double-sided silicon strip detectors (DSSSD) with extremely thin and homogeneous entrance windows to provide a new quality of NDP measurements for the N4DP instrument. A highly segmented DSSSD with 32×266 stripes, including integrated, self-triggering electronics, was successfully assessed and tested at the research reactor RID in Delft, in the Netherlands. Using a two-detector camera-obscura geometry set-up, we were able to image and reconstruct Li containing targets with space resolution down to ∼100\,μm×200\,μm and time resolution down to 10 ms. This project is supported by the BMBF, Contract No. 05K16WO1, 05K19WO8.

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Nuclear data for applications: Evaluation methodology and recent examples

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Radiative neutron capture cross section measurements in neutron beams

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Demonstration of gamma spectrum evaluation program(s)

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Development of a Time-Dependent Random Ray Method for High-Accuracy Neutron Flux Simulations

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The most essential quantity that characterises the behaviour of a nuclear reactor is the neutron flux. Especially for the simulation of its time-dependent behaviour, which is crucial for the safety assessment, relatively coarse simplifications are often applied to make the neutron transport equation accessible to numerical calculations. The presented work introduces a high-fidelity, time-dependent neutron transport solver based on the novel random ray method, a stochastical variation of the method of characteristics. So far, this approach has been tested for a positive reactivity insertion into a homogeneous, infinite model problem, and the obtained results matched those of the analytical solution. The random ray method has a firm potential to provide precise solutions for more complex transients, which shall be investigated in future studies.

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Short ABSTRACT of Participant's research field

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Over the coarse of my faculty studies in the field of physics I was introduced to plethora of different subjects. My background when it comes to Bachelors revolves around Medical Physics. There I was introduced in subjects and techniques that utilize concepts of modern physics and medicine. Here I did research and practical work in the field of RadioTherapy/Diagnostics where my B.Sc. was devoted to optimizing the shielding in the case of PET diagnostics. However, I always showed interest in a more fundamental approach to physics, and during my Masters this interest was recognized by my then mentor and I was employed in CERN within the NICOLE research group (ISOLDE Collaboration). Here we tackled in detail the topic of nuclear structure by exploring certain properties of some exotic nuclei to which my Master Thesis was dedicated. This only further sparked my interests and would send me off to pursue even more ambitious goals towards my future career as a nuclear physics researcher. During my PhD studies I developed high interest in tools for large data analysis such as ROOT, but more importantly I got inspired by the world of Monte Carlo simulations and their applications, which is what I'm predominently working on ever since and which will also be the topic of my Doctoral Dissertation. My efforts in these fields were recognized both by my research group from Faculty and the group I'm working with where I'm employed at the Institute for Physics in Belgrade, where amongst other works we did some novel research in the study of different uranium samples. Some of my international collaborations amongst other include DUNE Collaboration, co-joint research with group for Centre for Energy Research (Budapest) and work at the EU-JRC Institutes with which I honed and improved my skills. Aside efforts at CERN, I've spent time both in JRC-Geel, where I've worked on the study of spontaneous fission of Cf-252 - more precisely investigating the multiplicity of prompt gammas in regards to TKE (but we also tackled some topics concerning neutrons from this process as well), and in JRC-Karlsruhe, where I worked with both Nuclear Forensics Group and the Gamma Spectrometry Team - more precisely we're investigating certain materials and how they could be used as personal dosimeters in case they were to be used as Retrostective Dosimeters, and Monte Carlo modeling of known plutonium samples afterwhich we're aiming at predicting the unknown matrices, which are both considered hot topics in these areas. I would be more than happy to present and expand more in detail my up-to-date work, what my PhD thesis will include and revolve around as well as what plans and motivations I hold for my future career.

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Efficiency Calibration of a Large Volume Well-Type HPGe Detector

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The Canberra GCW6023 well-type HPGe detector is one of the instruments at the Nuclear Security Department that is used to measure the gamma energy spectrum of low activity and typically small-sized environmental samples, e.g., slag, soil, or dust. Small samples in vials can be fitted into the 16 mm wide well providing high efficiency detection and consistent geometry. The detector is also needed for the measurement of other environmental samples which are typically large samples and need to be placed on top of the aluminum cup. Accurate activity determination requires an adequate calibration process for each sample type and sample-detector configuration.
The goal of this project is to determine the full-energy peak efficiency (FEPE) of the GWC6023 detector as a function of gamma-ray energy for different measurement geometries and samples. For this purpose, certified reference materials are assayed, and the Monte Carlo simulation technique is applied to calculate theoretical efficiency values.