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Book of Abstracts

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Relationship with JENDL and Expectations for Possibilities of Opening up Nuclear Data/JENDL との係りと核データの切り開く可能性への期待

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I am grateful for the opportunity to present about my work. I have been keeping relationship with Japanese Evaluated Nuclear Data Library (JENDL) [1-11] for around 28 years. During this period, I have been contributing code developments, for example, Preequilibrium Nuclear Reaction Calculation Code (ALICE-F)[12] and Particle and Heavy Ion Transport code System (PHITS) [13]. I have also been assisting for EXFOR activities, and promoting Evaluated Nuclear Data Processing Code (FRENDY)[14] and Multiphase Multicomponent Detailed Thermal Fluid Analysis Code (JUPITER)[15].
In this presentation, introduced are origin of the author's motivation on the fields relating to nuclear data and outline the author's work. Also reported are the author's perspective and challenge for the future form of nuclear data

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Oral presentation / 2

Reactor physics for innovative reactor development and applications in space, medical and planetary science/革新炉開発と宇宙、医学、惑星科学への応用のための炉物理

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The author has studied various types of innovative fission reactors with different nuclear fuels and expanding applications of nuclear energy released by neutron reactions or disintegrations of radioisotopes, aiming at utilization in many fields including space, medical diagnostics/therapy, planetary science, etc. In the vast of those studies with different purposes, unique materials and rather different neutronic principles which are rarely applied in traditional light water reactors are required in analyses. Enriching transmutation chains, nuclear data for minor nuclides and cross sections for various nuclear reactions are essentials to move forward with extended nuclear applications for such emerging fields. The author's presentation will cover following topics.

Innovative power reactors; Long-life fast reactor, thorium-fueled water-cooled breeder reactor[1], beryllium-moderated thermal breeder using natural uranium, MA burner, fast reactor for LLFP transmutation[2], breed/burn reactor (CANDLE reactor), and high flux irradiation reactor.

Medical isotopes production; Ac-225 production for targeted alpha therapy (TAT) by fast neutrons, Ac-225 production by multiple neutron captures in thermal spectrum, Mo-99/Tc-99m production in commercial LWRs, Lu-177 production in commercial LWRs, assessment of dose absorbed by cancer cell.

Fission systems for space; Radioisotope thermoelectric generator (RTG), nuclear thermal rocket, alpha-particle propulsion, rotary space reactor, micro reactor on the moon, neutron propulsion, heavy-particle propulsion.

Planetary science; Geo-reactor worked in the primitive earth, geo-fast reactor in center of the earth.

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Oral presentation / 3

Experiences of HTTR Critical Approach Calculation and Nuclear Data/HTTR の臨界近接での経験と核データ

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The High Temperature Engineering Test Reactor (HTTR) has achieved at its first criticality in 1998. Before the first criticality approach, first criticality prediction had been carried out using Monte Carlo code MVP with JENDL3.3 but it missed the first criticality point.
The MVP of that time could not treat stochastic geometry model. Therefore, the effect of double heterogeneity of coated fuel particle was evaluated by SRAC code. The evaluated double heterogeneity reactivity effect was added to the calculated result of MVP with homogenized compact model. Moreover, there were other causes for misprediction such as number of history, amount of materials, etc. After the first criticality, the MVP was improved to treat stochastic geometry model. Revised first criticality prediction with improved MVP showed better prediction but it could not predict the first criticality accurately.
The JENDL-4 library was necessary for accurate first criticality prediction by MVP. The capture cross section of carbon was revised in JENDL-4. The change in capture cross section is not so large but it gave fairy large change in calculation because of large amount of graphite in the HTTR core.
The detail in improvements at the first criticality prediction will be presented.

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Measurement of photoneutron production cross section for mono-energetic linearly polarized photon/単一エネルギー直線偏光光子に対する光中性子生成断面積の測定

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The reaction between photons and nuclei, photonuclear reaction, has been studied in nuclear physics research since the 1950s. The experimental data on its reaction cross-section have been obtained for photons with energies from reaction thresholds to several tens of MeV[1]. In addition, energy and angular distribution have been obtained for the secondary particles from the reaction. The data have been used not only for understanding nuclear structure but also for developing nuclear data and empirical reaction models used to evaluate radiation transport and activation production in applications of electron accelerators.
To measure the data, photons were generated with Bremsstrahlung and positron annihilation in flight techniques that provide mono-energetic photons with subtraction. Since the 1990s, the laser inverse Compton scattering (LCS) technique became available for the experiment, which allowed us to obtain new data for mono-energetic linearly polarized photons, unlike previous experiments. NewSUBARU BL-01 is a unique facility in Japan for studying photonuclear reactions with various energies using the LCS technique[2]. At the facility, for example, polarization dependence on the azimuthal angle was revealed for neutrons from 17 MeV photons on medium-heavy targets[3].

Using the unique photon beam, we conduct experiments to obtain neutrons production double differential cross sections (DDXs) of tens of MeV photons on medium-heavy targets. The data from this experiment can be used to evaluate parameters of nuclear reaction models implemented in codes that handle energetic electron and photon transport, such as general-purpose Monte Carlo code and a code used for nuclear data evaluation.
Until now, we obtained DDXs of Ti, Fe, Cu, Sn, Au, Ta, W, Bi, and Pb targets from 13 to 20 MeV photons at horizontal angles (θ) of 30, 60, 90, 120, 180 degrees and vertical angles (ϕ) of 90 degrees with respect to the incident photon beam axis[4-6]. The data shows two components having different angular dependence, angular independent low energy, and angular dependent high energy components. The angular dependence depends on neutron emission angles and the direction of photon polarization. The dependence is found to be expressed as a function of the angle between polarization and neutron emission. The degree of dependence and amount of high-energy components varies according to both incident photon energy and the mass of target nuclei.
In this talk, an overview of the experimental procedure and results will be given in comparison with that of the previous.

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Oral presentation / 5

Nuclear data generation by combining machine learning and nuclear reaction models/機械学習と核反応模型を組み合わせた核データ生成

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In recent years, there have been many attempts to generate nuclear data using machine learning. Following this trend, we are trying to generate nuclear data by combining machine learning and nuclear reaction models. In this presentation, we will present the award-winning paper " Nuclear data generation by machine learning (I) application to angular distributions for nucleon-nucleus scattering" [1] and our recent research results.

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Oral presentation / 6

Crucial importance of correlation between cross sections and angular distributions in nuclear data of ^{28}Si on estimation of uncertainty of neutron dose penetrating a thick concrete/厚いコンクリートを透過する中性子線量の不確かさの評価における ^{28}Si 核データの断面積と角度分布の相関の重要性

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Accurate estimation of uncertainties, induced by those of nuclear data through error propagation, of quantities calculated by neutron transport calculation is crucially important for Verification and Validation (V&V) of design and safety analysis of nuclear systems. In this paper, uncertainty in neutron dose after penetrating a 3-m-thick concrete with ^{235}U thermal fission neutron source was estimated based on three different kinds of Total Monte Carlo methods under random sampling methodology. A thousand random nuclear data files were generated for ^{28}Si by T6 by perturbing underlying model parameters. In the first method, these files were used directly to yield processed library preserving all the correlations among different physical quantities. In the second method, a covariance file in the ENDF-6 format was generated and 1000 random files were produced based on the covariance file. In the third method, the random files populated by T6 were used but the angular distribution data were kept fixed to non-perturbed nominal ones. It was found that the second and third methods gave equivalent variance of the neutron dose after deep penetration of a concrete, and this uncertainty was larger than the uncertainty given by the first method. It shows that the positive correlation between total cross section and angular distribution of elastic scattering, which stems from Wick's inequality derived from the optical theorem affects uncertainty of the calculated neutron dose. The correlation of uncertainties of such different quantities is not represented properly in the ENDF-6 format, hence this correlation is normally ignored. It could be concluded that the uncertainty obtained by using the covariance files given in the ENDF-6 format may not give correct results for the uncertainty of deep neutron penetration calculation [1].

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Oral presentation / 7

Overview of FRENDY version 2/FRENDY 第二版の概要

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Nuclear data processing has an important role to connect evaluated nuclear data libraries and neutronics calculation codes. JAEA has developed the nuclear data processing code FRRENDY since 2013 to generate a cross section file from an evaluated nuclear data file with a simple input file. FRENDY version 1 was released in 2019 [1]. It only generates an ACE formatted cross section file for the continuous energy Monte Carlo calculation codes such as PHITS, MCNP, Serpent, and OpenMC. After we released FRENDY version 1, many functions were implemented such as multi-group cross section file generation function [2], adaptive setting of background cross section [3], ACE file perturbation [4], statistical uncertainty quantification of probability table [5], and modification of ENDF-6

formatted files. We released FRENDY version 2 including these new functions in 2022 [6]. This presentation explains the overview of FRENDY version 2 and the newly implemented functions in this code.
FRENDY is an open-source software under 2-clause BSD license. Everyone can freely use FRENDY and implement the modules of FRENDY in their code without any restriction. It can be downloaded from the JAEA website [7].
FRENDY can treat two input formats. One is the FRENDY's original input format. It is very simple and it does not require expert knowledge of nuclear data processing. For example, FRENDY can generate a cross section file with an evaluated nuclear data file name and processing mode. The other is the NJOY compatible input format. The available NJOY input is MODER, RECONR, BROADR, PURR, UNRESR, THERMR, ACER, GROUPR, and MATXSR.
FRENDY version 2 has original functions to generate a multi-group cross section file, e.g., explicit consideration of the resonance interference effect of the compound of different isotopes such as UO2, automatic background cross section set with the minimum number of background cross section, and resonance upscattering correction [8]. These functions are only available for the FRENDY's original input format. The sample input to use these functions are found in the manual of FRENDY [6]. These functions will improve the prediction accuracy of the multi-group neutronics calculation code.
We are now developing the heat production cross section calculation function, multi-group covariance matrices function, and treatment of the GNDS format. FRENDY version 3 will be released including these functions in the future.

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Oral presentation / 8

Nuclear data measurement by surrogate reactions using ion beam/イオンビームを用いた代理反応による核データ測定

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We are promoting surrogate reaction study to obtain nuclear data. Neutron-induced reactions are the fundamental reaction process important for nuclear reactor design and nucleosynthesis in stars. For many nuclei, taking data using neutron beam is difficult or practically impossible due to short half-lives of nucleus. Preparing enough target material of enriched isotopes would constrain the measurement. Alternatively, in the surrogate reactions, neutron-induced data are taken using transfer or multinucleon transfer reactions, by impinging accelerated ion-beams to a available target material, to populate the same compound nucleus and measure the radioactive decays, such as fission and γ-ray emission. At the tandem accelerator facility of JAEA, we have measured fission-fragment mass distributions (FFMDs) for many actinide nuclei and their excitation-energy dependence [1]. Here, effects of multichance fission on the FFMDs are quantified [2]. For the cross section measurement, we obtained neutron-capture cross sections for short-lived nucleus important for s-process [3] using surrogate ratio method. We are promoting experiment to determine (n,n') cross sections for the first time in the direct surrogate reaction. For this, we will measure probability of one-neutron emission channel from compound nucleus. In our setup, probabilities leading to all the decay channels will be measured to simultaneously obtain the cross sections of (n,f), (n,γ), and (2,2n) in addition to (n,n'). The data would provide each fission-chance probabilities with high accuracy.

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Oral presentation / 9

Muon Nuclear Data/ミューオン核データ

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The nuclear muon capture reaction is the capture of a negative muon by a proton in a nuclear medium via weak interaction from the 1s state of the muonic atom. When the muon stops in the matter, the muonic atom is formed and deexcited to the 1s state by emitting muonic X-rays. The reaction probability of the muon capture is more than 90% for the muonic atom with heavy elements. The excitation energy populated by the reaction is distributed around 10–50 MeV, producing several radioactive nuclei followed by particle emissions.
The importance of nuclear muon capture is now focused in the many fields of the natural sciences and applications, such as nuclear physics, nuclear transmutation for nuclear waste, muon-induced radioactive isotope production for medical use, radiation safety data in the muon facilities, cosmic muon-induced soft error in modern semiconductor devices, and cosmogenic production of radioactive nuclides for geological studies. Despite those demands, nuclear data of this muon-induced reaction is rarely known or investigated thus far.
Thanks to the recent advances of the low-energy muon facilities, Muse at J-PARC MLF, RIKEN-RAL muon facility, and MuSIC at RCNP, new muon nuclear data has become available. A new methodology called the in-beam activation method was invented to obtain the production yields of radioactive nuclei by muon capture, and a state-of-the-art detector system with digital pulse shape analysis has been developed to obtain energy spectra of charged particles from the reaction. Those new nuclear data will provide crucial information for understanding the reaction mechanism and for progress in many applications.
In the workshop, I will overview recent experimental and theoretical activities related to the muon-induced nuclear reaction and introduce the nuclear muon data project in Japan.

Oral presentation / 10

Research and Development on Recycling of Radioactive Waste in JAEA/日本原子力研究開発機構における放射性廃棄物の再資源化研究開発

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JAEA has set the following three items as the pillars of research and development; (a) Nuclear × Renewable synergy, (b) Making nuclear energy itself sustainable, and (c) Diversification of nuclear energy use (Ubiquitous). In particular, to make the use of nuclear energy sustainable, research and

development on recycling of radioactive waste has been initiated.
In this presentation, the following three contents will be presented; (1) the use of unburnable uranium in storage batteries, (2) the use of elements in spent fuel, and (3) power generation from heat and radiation. In addition, ideas for recycling that may be related to nuclear data will be introduced.

Oral presentation / 11

Features of the Supercritical Water-cooled Reactor (SCWR) and the Reactor Physics Issues/超臨界圧軽水冷却炉の特徴と炉物理に関連する課題

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Supercritical Water-cooled Reactor (SCWR) is the only water-cooled reactor among the six Generation IV reactor concepts. It is the logical evolution of the current Light Water-cooled Reactor (LWR) as it follows the historical development of fossil fuel-fired power plant, which has been operating at the supercritical pressure of water since 1970s. The SCWR plant concept may be characterized by low capital cost and high flexibility. The single-phase cooling nature of SCWR realizes the once-through direct cycle plant system, which has the potential to dramatically reduce capital (construction) cost of the plant by simplification and elimination of components. Moreover, the large temperature and density changes of the coolant allow designing of flexible cores with outlet temperature ranging from 500 to over 600 C and neutron spectrum ranging from thermal to fast. The existing SCWR design concepts include both the pressure vessel type (China, the EU, Japan, Russia) and pressure tube type (Canada) reactors. For the pressure vessel type, most design concepts adopt the PWR-like Reactor Pressure Vessel (RPV) and control rod drive together with the BWR-like containment with suppression chamber and safety systems. Passive safety systems and Small Modular Reactor (SMR) design concepts have also been proposed and analyzed.
Generally, the thermal and fast reactors are loaded with enriched uranium and plutonium fuels, respectively. The fuel enrichments tend to be higher than those of the current LWRs, because of larger neutron absorption cross sections of candidate cladding materials. The major reactor physics issues may include, but not limited to accurate evaluations of the core power distributions and the coolant density reactivity feedbacks. The core power distribution directly influences the core average outlet temperature, while the coolant density changes by about ten times from the core inlet to the outlet. Such issues may be more important for the fast reactors, which have the following characteristics. Namely, the pseudo-fast neutron spectrum; the large heterogeneity with the seed and blanket fuels; use of solid moderator (ZrH) in some designs to achieve negative void reactivity characteristics.
Currently, at Waseda University, the concept of fast reactor concept of SCWR is further being developed with multi-level physics modeling, which covers from the core design (including fuel performance modeling), transient and accident plant behavior, and severe accident management. The unique and challenging issue of the water-cooled fast reactor is preventing re-criticality of the fuel debris, when total loss of coolant must be assumed.

Oral presentation / 12

Overview and future of JENDL-5/JENDL-5 の概要と今後

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The latest Japanese Evaluated Nuclear Data Library, JENDL-5 [1], was released in 2021. JENDL-5 was developed with the amin of providing nuclear data to a wide range of application fields. JENDL-

5 includes nuclear reaction data not only of neutron but also of proton, deuteron, alpha-particle, and photon. JENDL-5 also provides the data of thermal neutron scattering, fission product yields, and nuclear decay. These data were stored in separate sub-libraries; JENDL-5 consists of 11 sub-libraries. Regarding the neutron reaction data, the energy region was extended up to 200 MeV for 73% of storing 795 target nuclei. The number of nuclei increased almost double of that of the previous version JENDL-4.0, covering most of the nuclei with the half-lives longer than 1 day. The revision of the data covered a wide range of nuclei from light to heavy ones reflecting the latest experimental data. The charged particle and photon induced reaction data were based on the data of the JENDL special purpose files released so far with revisions. JENDL-5 is expected to be applicable for large parts of computer simulations for radiations.
For future, the covariance data in JENDL-5, whose updates and coverage are limited, will increase. Those data would be used to estimate reliability and uncertainties in the computer simulations originated from the nuclear data. The number of target nuclei of charged particle reaction data will also increase to satisfy future needs of wide area.

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Oral presentation / 13

Experiments for nuclear data using RCS at J-PARC and HiRadMat at CERN/J-PARC RCS と CERN HiRadMat を用いた核データのための実験

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Since the experimental nuclear data is scarce in the kinetic energy region around GeV, which is crucial for high-power proton accelerators, such as Accelerator Driven System (ADS), 3-GeV synchrotron called RCS (Rapid Cycling Synchrotron) and beam transport facility 3NBT, conveying the beam from 3-GeV synchrotron to Material Life Science Experimental facility (MLF), play important roles in obtaining nuclear data. J-PARC is an incredibly attractive facility for studying nuclear data for ADS because J-PARC can only provide the users with the proton beam in the GeV region in Japan. RCS and 3NBT can easily vary the extracted beam's kinetic energy, and many experimental data were obtained.
From the early stage of J-PARC, the experiment was conducted to obtain the cross section of nuclide production with a projectile of protons. To validate the calculation model of spallation neutron spectrum produced at the most backward angle, which is essential to evaluate the shielding for ADS due to the duct streaming for the proton beam transport, the spectrum of the neutron from mercy target placed at MLF was observed using a small scintillator. Experiments for the displacement cross section were conducted to validate the material damage model. This talk will present the future experiment plan by using 440 GeV protons at HiRadMat CERN to obtain the displacement cross section for extremely high-energy regions.

Oral presentation / 14

Activities of Investigation Committee on Nuclear Data of AESJ/日本原子力学会シグマ調査専門委員会の活動

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Investigation Committee on Nuclear Data (Sigma Committee), established by the Atomic Energy Society of Japan (AESJ) in 1963, has been active for more than 50 years in order to contribute to the further development of nuclear data activities in Japan by examining nuclear data activities in Japan from a broad perspective while keeping an eye on global trends, and by communicating and exchanging information not only with nuclear energy but also with academic institutions in a wide range of other fields. The Sigma Committee has been worked closely with the Nuclear Data Division of AESJ, the JENDL Committee and Nuclear Data Center of Japan Atomic Energy Agency.
In recent years, as direct relation to future nuclear data research activities, operation of the nuclear data request list site leading to exploring the potential contribution of nuclear data, writing a textbook on nuclear data as human resource development, and creating the roadmap as the foundation for scientific and technological research and development surrounding future nuclear data. An overview of the Sigma Committee and its recent activities will be given in the presentation.

Oral presentation / 15

Current development status of simulation code for physical and chemical processes in PHITS/PHITS における物理・化学過程のシミュレーションコードの開発状況

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Radiation-induced biological effects such as cell death and mutation are believed to be attributed to initial DNA damage induction by energy deposition (physical processes) and reactions of chemical products to DNA (chemical processes). The mechanisms of DNA damage induction remain unclear. To make clear the mechanisms, the development of the simulation codes for physical and chemical processes is of great importance. In this presentation, we introduce the current development status of the simulation codes for physical and chemical processes in Particle and Heavy Ion Transport code System (PHITS) [1]. The physical processes of radiation in the human body can be simulated using track-structure modes in PHITS, i.e., PHITS-ETS and PHITS-KURBUC modes [2]. These track-structure modes consider each atomic interaction (such as elastic scattering, ionization, electronic excitation, dissociative electron attachment, vibrational excitation, photon excitation, rotational excitation, electron capture, and electron loss). Meanwhile, to simulate the physicochemical and chemical processes, we recently developed the PHITS-Chem mode [3]. The PHITS-Chem mode allows the calculation of the G values of 16 products (e.g., •OH, e^-_{aq}, H_2, and H_2O_2) and 35 chemical reactions as a function of time after irradiation (> 1 ps). Through this presentation, we present the features of PHITS-ETS, PHITS-KURBUC, and PHITS-Chem, and discuss the application of these codes to biological research.

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Oral presentation / 16

Details of the PHIG-3D's visualization functions/PHIG-3D 可視化機能の詳細

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PHIG-3D, developed as a 3D visualization tool for PHITS[1] input data, is oriented toward an intuitive interface. However, its internal structure and functions, which are difficult to describe in the document, have not been described clearly.
The most significant limitation in PHIG-3D is the large amount of memory used when using the lattice structure. After reading the input file, PHIG-3D eliminates cell dependencies and flattens the hierarchy structure in the preprocessor before creating cell objects. In this process, the lattice structure cells are replaced by cells of each lattice element, e.g., if there is a $100 \times 100 \times 100$ Lattice, 10^6 cells are generated. In 3D visualization, the data size is proportional to the number of polygonal vertices, therefore, visualization of the lattice structure geometry requires a large amount of memory.
One of the most important features of PHIG-3D is cross-platform capability. However, the biggest problem in implementing it is character encoding. Among the libraries used by PHIG-3D, Qt[2] uses UTF-8, but the character encoding of the C++ runtime library is compiler-dependent. Although VTK[3] can switch character encoding, it is necessary to load a multibyte-capable font to display multibyte characters. For these reasons, there is some overhead, especially in the Windows environment, as conversion from UTF-8 to the system character code (e.g., SJIS) and the removal of CR for line feed code are required.
In this presentation, I will describe and demonstrate these details.

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Oral presentation / 17

Geometry Design of Complex Entities into the PHITS Computational Space by using 3D-CAD/CG and Solid meshing/3D-CAD/CG 及びソリッドメッシングを用いた PHITS 計算空間への複雑なエンティティに対する体系設計

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Radiation and particle transport calculations based on the Monte Carlo method have been effectively used in various fields and are also now indispensable tools for the researchers and engineers involved. We have utilized the PHITS code mainly for medical applications, radiation protection, radiation activation evaluation for medical accelerator, and next generation fusion reactor development. One of the most important issues of Monte Carlo simulation calculations, including PHITS, is to set up a highly accurate and precise model of human bodies and structures equivalent to a real entity in a three-dimensional virtual space, i.e., a complex topology in which the real entities do not have to be actually constructed (manufactured or built). To fulfill this issues, it had been impossible to reproduce such complex topology groups in the 3D virtual calculation space due to very low software performance associated with older computer hardware performance (poor CPU and GPU performance and data transfer hardware) and insufficient 3D-CAD processing performance such as pre-post processing and solver processing using the finite element method, which is essential for 3D processing, and algorithms with slow code compilation processing. Recently, however, the drastic improvement in computer performance combined with the growing needs of general-purpose industries in the automotive, aerospace, shipbuilding, and creator industries has led to remarkable technological progress in 3D-CG production, an integrated simulation environment that includes pre-post processing, solver processing, and 3D-CAD (Computer Aided Design)/CAE (Computer Aided Engineering). In the PHITS calculation code, which should be positioned as a solver processing for radiation and particle transport behavior, it has been now possible to export Nastran bulk data (.bdf format as solid meshing typical in use) corresponding to its code by integrating the latest methods for pre-post processing of complex entities (human body, structures, etc.), namely, integrating MSC Apex modeler with a design development platform with general-purpose 3D-CAD/CG functionality, and we will introduce on these technical workflows at this meeting. Our presentation has provided the workflow from capturing the 3D-CG outer complex entities with a 3D scanner to building them into Nastran bulk data. And also, the results of exporting the T-track data obtained from the PHITS calculation to openFOAM format and then using the data to create a 3D display of the tetrahedral discrete mesh bins in Paraview.

Appendix: The press release article is titled as "Tokushima University and Helical Fusion Co., Ltd. Utilize MSC Apex in 3D Neutron Transport Monte Carlo Simulation Calculations for Next Generation Helical-type Fusion Reactor ". Please check the URL link "https://hexagon.com/ja/company/newsroom/press-releases/2023/tokushima-university-and-helical-fusion".

Oral presentation / 18

A Proposal for the Development of Boron Neutron Capture Therapy Agents based on Simulation Studies using PHITS Microdosimetry/PHITS を用いたシミュレーション解析に基づくホウ素中性子捕捉療法用ホウ素剤の開発指針

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BNCT is a radiation methodology utilizing the BNCR mechanism, depicted by ^{10}B(n.α)^{7}Li, to accurately target and eradicate tumor cells. The required ^{10}B level for its therapeutic effect in BPA-BNCT is usually between 15-40 ppm. While this concentration serves as a standard for several boron derivatives, the precise ^{10}B demand for alternative boron compounds, with their distinct accumulation patterns, remains unclear. To examine this, we crafted a virtual cell model with organelles. Using the particle and heavy ion transport system, we calculated the BPA equivalent dose concentration for the cell nucleus. Additionally, we presented the idea of the intranuclear minimal region (IMR), referring to the domain in the microdosimetric kinetic mode, and projected the BPA's equivalence dose concentration to the IMR. The findings suggest that the required boron dosage can change markedly based on how boron molecules settle within cell components. It appears important to consider the BNCR's impact, emphasizing the individual accumulation profiles, rather than strictly using the 15-40 ppm as a reference.

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Oral presentation / 19

Simulation Analysis of Cosmic Ray Muon Penetrating Subsurface of Huge Mountain/巨大山体の表層を透過する宇宙線ミュオンの挙動解析シミュレーション

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We conducted a simulation analysis of the behavior of cosmic-ray muons (CRM) passing through a huge mountain using the newly established cosmic-ray muon mode in PHITS. In this analysis, we investigated various aspects, including the altitude and angle dependence of incident and transmitted muons, further attenuation of decelerated muons after transmission of massive mountain by atmosphere, and asymmetry of multiple scattering.
The phreatic eruption at Mt. Ontake in 2014, which is the worst volcanic disaster causing the largest postwar victims, had taken place abruptly without any precursor. The goal of this study is to explore a new approach to catch a sign of anomaly preceding such eruption at Mt. Fuji by using CRM. The CRM radiography is a unique method to provide transmission image of internal structure inside of active volcano[1]. Mt. Fuji, however, is too large even for high-energy CRMs to probe dynamics of magma directly. Motivated by the seasonal variation
observed in CRM intensity after passing through Mt. Kurokura in West Iwate mountains and its correlation with frequency of steam explosion[2], we focused on subsurface structure of Mt. Fuji where hidden underground streams must be running to supply spring water of over 5 million m^3/day around the foot of the mountain.
In this study, we investigate attenuation and scattering behavior of CRM passing through Mt. Fuji with the position sensitive detector which set up at the Gotemba Tarobo at 1290m in altitude nearby the Fuji Skyline Roadway. From here, Mt. Fuji is an object with wide dynamic range of elevation angles from 0 to 20 deg, altitudes from 1300 to 3776 m, and path lengths from zero to several kilometers, even limited to subsurface and near the summit. After transmission through mountain, CRMs travel long distance from dozens of meters to several kilometers to the detector in atmosphere.
Intensity of the incident CRM at h=3776 m is 1.8 times larger than at 0 m, in θ=15~20 deg, while it is almost same at h=0 and 1300m. After transmission through standard rock, it was found that the ratio decreases down toward unity up to 200 m.w.e. , then increases again, resulting in 1.2 to 1.3 times at 2000 to 5000 m.w.e. On the other hand, we confirmed that the further attenuation by atmosphere during the flight to the detector is negligible. Multiple scattering will be discussed in comparison with Moliere's approximation.

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Oral presentation / 20

Fundamental study on responsiveness of gel dosimeters to carbon-ion beams and applicability of measurement of three-dimensional

dose distribution/ゲル線量計の重粒子線に対する応答性と三次元線量分布測定への適用性に関する基礎的研究

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In carbon-ion radiotherapy, the treatment plan is verified by comparing the planned dose and measured dose before treatment. The dose measurement is limited to a portion of the three-dimensional (3D) irradiation region of interest. Gel dosimeters can measure the 3D absorbed dose distribution using the reaction between radiation-sensitive dye and radicals generated by radiation exposure. However, gel dosimeters have the problem of low responsiveness in the position where the dose is localized, such as Bragg peak [1]. In order to solve the problem, this study was performed. Specifically, carbon-ion beams were irradiated onto micellar gel dosimeter and poly(vinyl alcohol) and iodide (PVA-I) gel dosimeter under various beam conditions, and then an optical absorbance and an optical density distribution was measured [2-3]. The results were compared with the absorbed dose distribution estimated by Particle and Heavy Ion Transport code System ("PHITS") [3]. The radical concentration distribution generated in the gel dosimeters was also estimated by liner energy transfer (LET) calculated using "PHITS". Based on these results, we considered the responsiveness of the gel dosimeters at Bragg peak and spread-out Bragg peak (SOBP). We will also introduce examples of the use of "PHITS"in a series of studies.
This collaborative study involved Yokohama National University, Toshiba Energy Systems and Solutions Corp. (Kawasaki, Japan), and Kanagawa Prefectural Hospital Organization. This study underwent an ethical review process and received approval from the Yokohama National University Institutional Review Board (YNU IRB 2020-7-9) on May 13, 2020, and from the Kanagawa Cancer Center (KCC IRB 2020-46), which is part of the Kanagawa Prefectural Hospital Organization, on June 29, 2020.

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Oral presentation / 21

Nuclear heating and damage data in JENDL-5 neutron ACE file/JENDL-5 中性子 ACE ファイルの核発熱、損傷データ

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The Japanese Evaluated Nuclear Data Library JENDL-5 was released in 2021. The neutron ACE file of JENDL-5 was mainly produced with the FRENDY2 code, while nuclear heating and damage data were done with the NJOY2016.65 code modified for JENDL-5, and it was released as one of ACE-J50 [1] in 2022. This presentation explains the nuclear heating and damage data in it.
Neutron ACE files have heating numbers and damage production energy cross sections as the nuclear heating and damage data. Note that heating numbers are deduced from KERMA factors and

DPA cross sections are produced from damage production energy cross sections. KERMA factors are calculated with two methods: the energy balance method and kinematics method. Original NJOY stores heating numbers from KERMA factors with the energy balance method to ACE files, but the KERMA factors can be negative or too large because of the energy balance problem.
In JENDL-4.0 heating numbers from KERMA factors with the kinematics method were stored to all the neutron ACE files by using modified NJOY99 in order to sidestep the energy balance problem. However the modification of NJOY99 was not adequate, which produced negative probability table (p-table) of heating number. Heating values in PHITS heating calculations by using ACE files with the negative p-table become "NaN"(Not a Number). Thus all p-table data were deleted from neutron ACE files of JENDL-4.0 with negative p-table. Heating numbers from KERMA factors with the kinematics method were stored to all the neutron ACE files in JENDL-5 because of the energy balance problem, by using adequately modified NJOY2016.65, which produced no negative p-table of heating number.
It was known that damage production energy cross sections in neutron ACE files of JENDL-4.0/HE dropped down above 20 MeV because of no energy distribution data of several residual nuclei above 20 MeV in JENDL-4.0/HE [2]. JENDL5 includes energy distribution data of all residual nuclei above 20 MeV, which solves this issue.
Recently Chinese researchers reported that the HEATR module of NJOY had a fatal bug that it calculated KERMA factors and DPA cross section data without multiplying secondary gamma yield when secondary gamma data were stored in File 6 [3]. JENDL-5 was processed with the bug fixed NJOY.

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Oral presentation / 22

Simulation of aluminum activation experiment at CERN/CHARM / CERN/CHARM でのアルミニウム放射化実験の模擬計算

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An activation experiment using aluminum samples at the CHARM [1] of CERN was simulated for a benchmark on the radiation shielding of high energy accelerators. Na-24 production in aluminum samples set in the irradiation room and corridor of the CHARM was calculated. Additionally, measurable feasibility of other activated nuclides produced in the aluminum sample was checked.
Two radiation transport codes: PHITS [2] and GEANT4 [3] were used to simulate the experiment at the CHARM. The calculation geometry was largely constructed with the size of 21 m wide, 30 m long, and 16 m height. Spherical output regions of 10 cm in diameter were located in the irradiation room and corridor. Proton beam with the energy of 24 GeV/c was hit on a copper target of 8 cm in diameter and 50 cm length, and produced particles were transported. As output of calculation, particle type, position, direction, energy, and weight of a particle were acquired when the particle was incident into the output region.
Production of activated nuclides in aluminum was derived by a connective calculation using acquired data and activation cross sections. Cross sections of ^{7}Be, ^{22}Na, ^{24}Na, and ^{27}Mg productions

for neutron and proton incidences were checked by comparing experimental, evaluated, and calculated values, and were implemented into the connective calculation. The particles based on acquired data were produced as a source, and attenuated in the geometry consisting of a cylindrical aluminum set in a sphere filled with air. The productions were calculated in attenuation.
As an example of PHITS calculations, productions (1/atom/primary) at 1.5 m far away from the copper target are 2×10-32 for ^{7}Be, 7×10-31 for ^{22}Na, 4×10-30 for ^{24}Na, and 5×10-30 for ^{27}Mg. Productions of ^{24}Na and ^{27}Mg are suitable to observe neutron streaming since neutron induced productions of were occupied. Be-7 production is good to observe the contribution by proton from production ratio of ^{7}Be to ^{24}Na, however, the measurement would be difficult due to the long half-life. Na-22 which has a longer half-life than 7Be cannot be measured.

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Oral presentation / 23

Test of ^{107}Pd transmutation with macroscopic quantities/^{107}Pd 核変換実証試験

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Palladium is one of the nuclides targeted for recycling from spent nuclear fuel. Reasonable nuclear reaction paths for ^{107}Pd and the cross-sections for proton- and deuteron-induced spallation in inverse kinematics have been investigated [1]. However, a transmutation experiment using long-lived fission products as the target would be required for an actual system. To experimentally demonstrate the feasibility of ^{107}Pd transmutation by deuteron irradiation under continuous irradiation with the existing azimuthally varying field (AVF) ring cyclotron at RIKEN RIBF, we conducted a test with macroscopic quantities to transmute ^{107}Pd by deuteron beams produced by the accelerator [2].
To effectively detect the reaction products of the ^{107}Pd + d reaction, we prepared a material with a ^{107}Pd-concentration of almost 100\% by ion implantation. The implanted samples were irradiated for several days with deuterons produced by the AVF ring Cyclotron at RIKEN RIBF. After cooling, gamma-ray measurements of the irradiated sample were conducted. ^{105}Pd and ^{106}Pd produced from transmutation of ^{107}Pd were estimated using DCHAIN. The ^{107}Pd in the irradiated samples were measured by ICP-MS. The isotopic ratios ^{105}Pd/^{107}Pd and ^{106}Pd/^{107}Pd obtained from the experimental results were compared with those obtained by calculation using PHITS.
In this paper, an outline of the test of ^{107}Pd transmutation with macroscopic quantities is presented and certain experimental results are reported.

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Oral presentation / 24

Design of radiation shield for RI production beam line by PHITS/PHITS を用いた RI 製造ビームラインの遮蔽設計

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A new beamline for the production of astatine-211 is to be installed at the RIBF accelerator facility. At the beamline, an alpha beam is irradiated to the bismuth target. The RI production requires a high intensity beam, which generates a large amount of radiation from the target. A shielding system with sufficient capacity is necessary to keep the radiation dose rate in the surrounding area low. In this study, the shield was designed with the goal of achieving a radiation dose rate of less than 10 μSv/h on the outside of the shield using the Particle and Heavy Ion Transport code System (PHITS)[1], and the compact shield was designed by combining multiple shielding materials.
Because a shield made of one type of material is not effective enough against high energy neutron, its weight and size became large. A size of the shield is limited because the beamline is installed in a narrow space. In addition, the floor load-bearing capacity is also limited due to the structure of the building.
The radiation dose rate around the target by an alpha beam with an energy of about 29 MeV and an intensity of 100 pμA is estimated to be 300 Sv/h. Thus, the shielding to reduce the radiation dose rate by seven orders of magnitude is requested. Neutrons with energies up to 15 MeV are generated by the objective reaction. In general, neutrons less than a few MeV are shielded well by hydrogen-rich materials, such as polyethylene or water. To shield high-energy neutrons, firstly inelastic scatterings with heavy materials are applied to reduce the neutron energy. Secondary, polyethylene is used outside the heavy material part to shield low-energy neutron effectively. Primary and secondary gamma rays are shielded by metals such as iron and lead.
The optimized shielding configuration was 30 cm thicknesses of iron, 10 cm of polyethylene containing 10 % B_2O_3, and 40 cm of polyethylene. Since neutrons are absorbed with protons in the polyethylene shield and it provides secondary gamma rays, lead was additionally installed to reduce the gamma rays.
In addition to the shielding around the target, the shielding of the concrete building was also taken into consideration to keep the radiation dose rate on the border of the radiation-controlled area below the legal limit.

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Oral presentation / 25

Calculation of the Skyshine Radiation Measurement Experiment in Kansas by PHITS/PHITS によるカンザスでのスカイシャイン線測定試験の線量評価

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Kansas State University measured exposure dose from skyshine radiation using ^{60}Co sources at Kansas in 1977 [1]. Mitsubishi Nuclear Fuel (MNF) performed calculation of this experiment with a

three-dimensional Monte Carlo code PHITS v3.24 (Particle and Heavy Ion Transport code System), which is developed by JAEA [2].

For nuclear facilities, it is necessary to calculate the exposure dose objective to public which is calculated direct and skyshine radiation. Direct radiation is gamma rays emitted from the ^{60}Co source that penetrate the shielding wall and reach the detector location directly. Skyshine radiation is gamma rays emitted from the ^{60}Co source that penetrate the relatively thin ceiling, are reflected in the sky, and reach the detector location.
The analysis method for the skyshine radiation is adopted the single-scattering calculation method which is used by G33 code. For the skyshine calculation of the MNF facilities, G33 code or PreGAM/S code, which has equivalent performance, is used.
However, the single-scattering calculation method has the disadvantage that it cannot correctly calculate the effect of attenuation of gamma rays for a concrete ceiling condition. In addition, due to the characteristics of the calculation method, it cannot accurately simulate the geometry of the analysis system. Furthermore, gamma rays that are scattered multiple times in the sky cannot calculated.

A solution of this situation is to use a Monte Carlo code in three dimensions that can handle a wide variety of geometries, and PHITS code can treat accurate model geometries. The radiation behavior can be correctly analyzed by applying a relatively accurate geometry model.
However, PHITS code makes the analysis more complicated. In particular, the shielding analysis, in which the number of particles decreases due to transmission through concrete and other materials, has a large uncertainty. And large number of histories is necessary to obtain sufficiently reliable analysis results. Recent advances in computers have made it possible to achieve this.

In this study, MNF performed a calculation of the skyshine measurement experiment and found that PHITS code can predict results in ±10% error between experiment and calculation.
In contrast, the conventional code, G33 overestimated the results of the experiment by about five times.
This result confirms that PHITS code can simulate the skyshine measurement experiment more accurately than G33 code.
In the future, MNF expects to improve the accuracy of shielding calculation changing the analysis code to PHITS. As a result, more rational shielding design will be expected.

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Oral presentation / 26

Introduction to PHITS-UDM (User Defined Model)/PHITS-UDM（ユーザー定義モデル）の紹介

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We have developed a feature that allows users to implement any interaction and particle in PHITS. New sections [User Defined Interaction] and [User Defined Particle] are introduced to control them. PHITS can expand its application possibilities by allowing users to easily implement minor physical processes, user-defined processes, or new particles. We will introduce this new feature in this talk.

Poster presentation / 1

Improving Accuracy of Fission Product Yields by Bayesian Neural Network/ベイジアンニューラルネットワークによる核分裂生成物収率の高精度化

Author: Jingde/敬徳 Chen/陳[1]

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Fission product yields (FPY) is one of the most important nuclear data, which provide information necessary for applications like burn-up calculation in nuclear reactors, environmental assessment of radioactive waste disposal and evaluation for production of valuable isotopes such as ^{99}Mo for medical imaging. Traditionally, major nuclear data libraries (such as JENDL, ENDF, JEFF, etc.) have established FPY databases for neutron-induced fission at only thermal energy, 0.5 MeV, and 14 MeV where many experiments have been carried out, and the data in the energy range lacking experimental data are obtained by an interpolation in linear-linear basis, which is not consistent with energy dependence of experimental data for many FPY. Therefore, evaluated data on FPY are still insufficient. However, consistent and systematic evaluations of energy dependence of the FPY have been proven to be quite challenging by both theoretical calculation and experimental measurement. In order to improve this situation, we utilized a two hidden-layer Bayesian neural network (BNN) model with data augmentation technique to train and predict evaluated and experimental fission product yields with high accuracy to obtain energy dependence of mass distributions of the FPY. Similar to the normal deep learning, the JENDL-5 FPY data were divided into 2 groups, 80% for training and 20% for validation. Additionally, the training data included several experimental cumulative yields and theoretically calculated values. The number of units in each layer and activation function were selected carefully to reproduce the global and fine structure of the mass distribution data. Additionally, the data augmentation is particularly valuable for enhancing the accuracy of specific nuclides. Finally, the predicted results for energy dependence of mass distribution for ^{232}Th, ^{233,235,238}U and ^{239,241}Pu for incident energy ranging from 1MeV to 5MeV in BNN model exhibited certain peak structures at fission product mass numbers A = 134 and A = 140-144, which agreed with enhancement due to the shell and even-odd effects, and the standard I and II asymmetric modes of fission from Brosa model [1]. The capability of our BNN to reproduce the mass distributions including the fine structure is evaluated to be an advancement of the similar approach by a group of Beijing University [2], which aims at description of an overall 2-peak structure of the FPY and the fine structure.

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Poster presentation / 2

4-D Langevin trajectory analysis using machine learning/機械学習を用いたランジュバン軌道解析

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Nuclear fission serves as the most fundamental physics phenomenon underlying nuclear energy. It is quite a complex process of large-amplitude collective motion of finite quantal systems, which produces a huge energy of around 200 MeV/fission, and fission neutrons, which are used to sustain chain reactions, and over 1,000 kinds of fission products are produced. Understanding nuclear fission is also essential in describing the nucleosynthesis in the cosmos since fission recycling is believed to occur in r-process sites. The mass distribution of fission products in the actinide region is known to be characterized by 2 asymmetric peaks, and it turns to a sharply symmetric shape for nuclei heavier than ^{257}Fm while it is mildly symmetric for pre-actinide nuclei. This change in the mass distributions between the dominance of the asymmetric and symmetric distributions gives us an important clue to understanding how fission proceeds. Even though there is a long history of research on the fission mechanisms, the essential mechanisms are still a big mystery, and many studies are necessary to understand them from both experimental and theoretical approaches.
On the theoretical side, a large number of macroscopic, macro-microscopic, and microscopic theories have been proposed to describe nuclear fission. As dynamical theories, time-dependent density functional theories have been recognized to be the most advanced method. However, the calculations start outside the barrier in this kind of calculation since trajectories do not come out if these calculations are started from the ground state or the second-minimum of the potential energy surface inside the saddle point. In this approach, therefore, we cannot understand how the system overcomes the barriers after forming a compound nucleus. On the other hand, the Langevin approach, where the nuclear fission is treated as a Brownian motion of nuclear shape degree-of-freedom, can describe the dynamical process of the system starting from the ground state or the second minimum.
The purpose of this study is to elucidate the basic mechanism of nuclear fission, namely, how a compound nucleus overcomes the barrier and how it leads to population of symmetric or asymmetric fission fragments. For this sake, we calculate fission trajectories in the 4-dimensional Langevin model[1] and analyze the trajectories as time-sequential data by using a Long Short-Term Memory (LSTM) method of Recursive Neutral Network (RNN). The time-series data used for training are preprocessed and labeled by time steps, the atomic number Z and the mass number A of a fissioning nuclide, event numbers, values of four collective variables, and corresponding momenta. Then, classification of symmetric or asymmetric fission is performed using multiple all-coupled layers. This presentation will report a progress of this study and preliminary results.

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Poster presentation / **3**

Study of INC model for α incident reaction at 230 MeV/u / 230MeV/u の α 粒子入射反応に対する INC 模型の研究

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The Intranuclear Cascade model have been improved for calculation of alpha incident reaction. The fragmentation reaction which is dominant in alpha incident reactions is calculated using the model that incident alpha particle is broke up according to the probability, which show cluster state in the alpha particle, and density distribution of targets. In addition, the direct knockout reaction is adopted to explain composite particles in low energy region. The calculation results were compared with experimental data of double differential cross sections of charged particles produced from the reaction of alpha particles at incident energy of 230 MeV/u on ^{12}C, ^{27}Al, and ^{59}Co. As a result, good agreements are obtained.

Poster presentation / 4

Calculation of Fission Fragment Yields for thermal neutron reaction of ^{239}Pu/熱中性子核反応における ^{239}Pu の核分裂収率の計算

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^{239}Pu の熱中性子核反応における核分裂収率を核反応モデル計算コード CCONE[1] を用いて計算した。実験データと比較して検証した結果を本発表で行う。

これまでの核データにおける核分裂収率の評価は、主に独立核分裂収率と積算核分裂収率を基にして評価値が決められてきた。しかし核分裂収率は、全運動エネルギー (TKE) や即発中性子、崩壊熱など様々な観測量と関連する物理量であり、それらも考慮した評価値の決定が求められている。JENDL-5[2] でいくつかの改良が実施されたが、依然として核分裂に伴う観測量との相関は総合的に考慮されていない。この問題を解決するために、我々は核反応モデル計算コード CCONE[1] を用いた核分裂収率計算システムを開発した。この計算システムでは、核分裂直後の収率分布を５-Gaussian モデルで近似することで、TKE や即発中性子、崩壊熱などの核分裂に伴う観測量を系統的に計算することができる。本発表では、この^{239}Pu の核分裂収率を計算した結果について紹介する。

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Poster presentation / 5

Development of a New Web Services and RESTful APIs for for Experimental Nuclear Reaction Database (EXFOR)/原子核反応実験データベース（EXFOR）の新 Web サービスと RESTful API の開発

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Experimental nuclear reaction data are essential for understanding nuclear reaction phenomena, developing nuclear theories and models, and evaluating data for nuclear data libraries. Efficient data mining from the Experimental Nuclear Reaction Database (EXFOR)[1] has a potential for utilization of modern computational analysis techniques to find trends, shortcomings, and hidden patterns in the database, which in turn helps improve our knowledge of nuclear physics. The IAEA Nuclear Data Section (NDS) is entrusted with the responsibility of maintaining and facilitating user-friendly access to this data. To fulfill this mandate, the NDS has developed several services, such as the EXFOR web retrieval system, however, the rapid advances of compute infrastructure and the increasing demand to process nuclear data at scale in the context of ML and AI applications enforces us to adhere the FAIR (Findable, Accessible, Interoperable, Reusable) principles for the service implementation.
To facilitate more advanced method in the nuclear data field, we have developed two EXFOR parsing computer programs (EXFOR Parser) to convert the data in the EXFOR format into the widely adopted JSON format. The converted JSON data are used for further processing to extract individual

physical observables and generate tabulated data (x, y, dx, dy) where all units of measurement are standardized. Furthermore, we have developed REST APIs and an open web system for easy access and quick visualizations of these converted datasets.

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Poster presentation / **6**

Neutron total and capture cross section measurements of ^{nat}Er at ANNRI

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The extension of the nuclear fuel life has always been seen as an effective method to improve the economic viability of nuclear reactors. Nonetheless, this has always been hampered by the ^{235}U 5 wt% limitation due to criticality concerns [1].An increase in ^{235}U above the 5 wt% threshold would mean a major reformulation of both reactor criticality and safety assessments for the present nuclear reactors. The Erbia-credit super high burnup (Er-SHB) fuel is an innovative configuration that allows for the fuel life to be extended while providing several physical improvements (i.e., less downgrade of the flux distribution, improving the intrinsic reactor safety parameters, better control of the transient power phase). Moreover, the negative reactivity introduced by Erbium offers a means to increase the enrichment of ^{235}U > 5 wt% while treating the fuel as if the enrichment of ^{235}U were to be lower than 5 wt% as in present LWR reactions. Meaning that, this new fuel configuration could be used in present LWR reactors without any major modification to the facilities [1,2]. Nonetheless, for this to be achievable, the accuracy of the nuclear data for the neutron capture cross section of Erbium needs to be improved [3].
The present experiments were performed in the Accurate Neutron-Nucleus Reaction Measurement Instrument (ANNRI) at the Materials and Life Science Facility (MLF) of the Japan Proton Accelerator Research Complex (J-PARC) using Li-glass detectors to measure the neutron total cross section; and NaI(Tl) and Ge spectrometers to determine the neutron capture cross section in separate measurements. Several samples of ^{nat}Er with different thicknesses were measured in the present experiment to improve the accuracy for the cross sections.
In this study, the preliminary results for the ^{nat}Er neutron total and capture cross sections measured with Li-glass, Ge and NaI(Tl) detectors are presented and compared. For the neutron capture cross section, the preliminary results were obtained relative to the incident neutron flux determined with a boron sample measurement and normalized using an Au sample measurement.

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Poster presentation / 7

Measurement of the very-forward-angle neutron elastic scattering and PHITS simulation for neutron shielding/超前方散乱角における中性子弾性散乱の測定と PHITS による中性子遮蔽計算

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In equation of state of nuclear matter, constraints on parameters of the symmetry energy $S(\rho)$ are important for understanding of the nuclear many-body system which is related to various astrophysical phenomena. The symmetry energy is essential for the neutron matter ($\delta \sim 1$ where δ is degree of asymmetry), but it is less certain than the symmetric nuclear matter ($\delta \sim 0$). It is known that there is a linear correlation between the slope parameter and neutron skin thickness δR in ^{208}Pb[1]. δR can be written as the difference of the neutron and proton RMS radii. However, the uncertainty of the neutron radius in ^{208}Pb is still large, while its proton radius is precisely determined by electron scattering.
Proton elastic scattering (PES) is one of the powerful probes in determining the density distributions. In the case of PES, the cross sections at very forward angles which is sensitive to the nuclear radius, are mainly caused by the Coulomb scattering. It is difficult to extract the information of the neutron radius. Therefore, we proposed an experiment of the neutron elastic scattering (NES) to precisely determine the neutron radius in ^{208}Pb.
Recently, we have performed a measurement of the NES at very forward angles (4, 7 degrees) in ^{208}Pb and ^{40}Ca. We have designed a new setup with neutron beams at 63 MeV generated by the ^{7}Li(p, n)^{7}Be reaction. To identify scattered neutrons of NES, Time of Flight (ToF) method and Pulse Shape Discrimination (PSD) method have been applied to the measurement of NES.
We have performed the measurement of angular distribution of NES at $\theta_{c.m.} = 4.098,\ 4.571,\ 6.981,\ 7.458$ degrees of ^{208}Pb(n, el) scattering and $\theta_{c.m.} = 4.199,\ 4.683,\ 7.152,\ 7.640$ degrees of ^{40}Ca(n ,el) scattering. However, angular distribution that we have measured has large statistical errors compared to theoretical requirements. Major factor in the large statistical errors is background neutrons. To distinguish between the background neutrons and the scattered neutrons in ^{208}Pb and ^{40}Ca, beam collimation and neutron shielding were essential. Neutron shielding for the experimental setup was calculated by Particle and Heavy Ion Transport code System (PHITS)[2]. In this poster presentation, the details of the experimental setup and feasibility test with PHITS will be discussed.

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Poster presentation / 8

Evaluation of Photonuclear Reaction Data ^{209}Bi at 13 and 17 MeV photon energy

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Photonuclear reaction cross-section data are essential for electron accelerator shielding design and possibly nuclear transmutation. So far, photonuclear cross-sections of various target materials have been evaluated up to a photon energy of 200 MeV within the nuclear data libraries, such as JENDL-5 [1]. Almost all data in JENDL-5 have been evaluated based on the experimental reaction cross-section data. However, the evaluations using to the reaction cross-section data are inadequate to provide all information about the emitted secondary particles, for instance, their energy distributions. Recently, the photoneutron energy spectra on the medium and heavy targets at 13 and 17 MeV photon energies have been measured [2–4]. The 13 and 17 MeV photon beams are nearly monoenergetic and have high intensity. Among the data given in [2–4], ^{209}Bi is one of monoisotopic elements with the available data at both 13 and 17 MeV photon energies, we evaluated the photonuclear data of this target based on the reaction cross-section data by I. Gheorghe \textit{et al}. [5], and the measured photoneutron energy spectra in [2–3]. The evaluation procedure was conducted on the CCONE code system [6]. The photo-absorption cross-sections were evaluated with the giant dipole resonance (GDR) and quasi-deuteron (QD) models. The emission of photoneutrons is described by the exciton model for the preequilibrium process and the statistical model for the compound process. The photoneutron energy spectra have been compared to the experimental data [2–3] to find the connection between the theoretical reaction models and experiment. An adjustment of the multiplying factor for the single neutron average density in the exciton model was made to improve the energy distribution calculations. For 13 MeV photon energy, this evaluation by CCONE code can reach the measured data well. In contrast, the underestimate to the experimental data is still observed for 17 MeV photon energy when the preequilibrium photoneutrons increase.

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Poster presentation / 9

Measurement of double differential cross sections of charged particles produced by 100 MeV/u ^{12}C beam nuclear reactioins/100MeV/u ^{12}C ビーム入射荷電粒子生成二重微分断面積の測定

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Nuclear data on carbon ion induced reactions in a wide range of energy are needed for purposes such as improvement radiation protection for space exploration and evaluation systems for secondary exposure on radiation therapy. However, it is reported that there is no measurement data on double differential cross sections of incident high energy ^{12}C particles between 100 MeV/u and 500 MeV/u. Therefore, there is a need to obtain measurement data of double differential cross sections in high energy regions.
In this study, we measured double differential cross sections of charged particles produced by 100 MeV/u carbon ions on ^{12}C, ^{27}Al and ^{59}Co targets. Obtained data are compared with data previously measured by other researchers and a moving source model. Overall good agreements are shown.

Poster presentation / 10

Production of Np isotopes from ^{238}U beam at BigRIPS in RIKEN/理研 BigRIPS での ^{238}U ビームからの Np 同位体の生成

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A variety of unstable nuclear beams with atomic numbers (Z) up to 92 can be produced by the projectile fragmentation and in-flight fission from high intensity U beams at RIBF. Recently, it was found that $^{234-238}$Np can be created by a proton pickup reaction on 1 GeV/nucleon ^{238}U beam. Owing to the recent developments of the high-Z beams at BigRIPS, energy dependence of the proton pickup reaction on ^{238}U can be obtained at RIBF. Thus, we conducted an experiment to determine the energy dependence of the production cross section of ^{237}Np. A test of the production of Np isotopes was performed by using the BigRIPS spectrometer at RIBF in March 2022.
Secondary beams around Z = 90 were produced by a ^{238}U beam with energies of 345 and 250 MeV/nucleon impinging on a 1-mm-thick ^{9}Be production target at F0 in BigRIPS.
The particle identification (PID) of the secondary beam was performed using the TOF-Bρ-ΔE method. To validate the production of the ^{237}Np^{91+}, a two dimensional (2D) Gaussian fitting approach was conducted in accordance with the distribution patterns of neighboring ions of ^{234}U^{90+}, ^{235}U^{90+}, and ^{232}Pa^{89+}. It is found that Np isotope can be counted up with contaminated U/Pa isotopes using the 2D Gaussian fitting technique. The production cross sections of ^{234}U, ^{235}U, ^{236}U, ^{232}Pa, and $^{233}Pa as well as Np isotopes were derived.
In this presentation, we will report the analysis status of 345 MeV/nucleon.

Poster presentation / 11

Isotopic production of high-radiotoxic nuclide ^{90}Sr via proton- and deuteron-induced reactions and new analytical model for its longitudinal momentum distribution/高放射性核種 ^{90}Sr の陽子・重陽子入射反応による同位体生成とその縦運動量分布に対する新しい解析モデル

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We have been systematically measuring the isotopic production cross sections via proton- and deuteron-induced spallation reactions of ^{90}Sr and other radionuclides using the inverse kinematics method.

Utilizing this technique, we can also measure the longitudinal momentum distribution of each residual product. The distribution has asymmetry with a tail on the low momentum side, and it is known to become more pronounced at lower incident energies[1]. Until now, the asymmetry has been mainly evaluated using Gaussian functions with different standard deviations σ_{High}, σ_{Low} on the higher and lower sides of the distribution[2]. Although this approach reproduces experimental results, it is significantly artificial, and such obtained asymmetry is difficult to correlate directly with physical phenomena. Therefore, we attempted to devise a new analytical model.

In contrast to the conventional method, the proposed equation reproduces the distribution with higher accuracy and can be applied to a wide range of isotopes with a certain asymmetric parameter α'. Furthermore, the relationship between the skewness expressed in terms of third-order moments and the introduced α' is made explicit. It is shown that the longitudinal momentum distribution can be explained by introducing an asymmetry of about α' < 0.15 for the proton- and deuteron-induced spallation reactions. In the future, this model will be used to analyze reaction data of other radionuclides more precisely.

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Poster presentation / 12

Measurement of the spallation neutron spectrum by unfolding at 180° from 3-GeV protons and natHg with the ^{209}Bi(n,xn) reactions/^{209}Bi(n,xn) 反応を用いたアンフォールディング法による 3-GeV 陽子と水銀の反応で 180 度方向に生成する核破砕中性子スペクトルの測定

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Neutron source term is required for shielding design of accelerator facilities. A Time-of-Flight (TOF) technique is applied to get the source term. When TOF cannot be utilized at accelerator facilities (e.g., Continuous Wave (CW) operation), an unfolding method is useful. However, the validity of unfolding is not completely understood.
One of the accelerator facilities where CW operation will be used is Accelerator-Driven System (ADS) [1]. At the ADS facility designated by JAEA, 1.5-GeV proton beam is provided to a Lead-Bismuth Eutectic (LBE) alloy target. Transmutation of minor actinides is performed by neutrons produced by reactions between incident protons and nuclei in LBE. As with other accelerator facilities, radiation shielding is important for the facility.
According to the shielding design [2], there is a high-dose-rate area at 180° attributed to a streaming of a beam duct. Thus, it is required that we study the neutron spectrum at 180°.
At J-PARC, high-intensity neutron beam at 180° is available by the reaction between 3-GeV protons and $^{\mathrm{nat}}$Hg. The neutron spectrum at 180°, which is similar to the spectrum by the reaction between proton and LBE, was already measured with TOF [3].
To evaluate the neutron source term at facilities where TOF cannot be used, confirming the reliability of the unfolding is necessary. Thus, the purpose of this study is to acquire the neutron energy spectrum by the unfolding with the ^{209}Bi(n,xn) reactions and response functions (JENDL/HE-2007

[4] or TALYS [5]).
In our poster, we present our activation measurement at J-PARC, the unfolding, and the comparison with the TOF-spectrum and calculation results.

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Poster presentation / 13

C/Be neutron converter design for increasing production amount of medical radioisotopes in accelerator neutron method/重陽子加速器中性子源を用いた医療用 RI 製造量増加を目的とした C/Be 複合コンバータの設計

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Accelerator neutrons generated by deuterons are new source for radioisotope (RI) production. In this system, neutrons are produced by the (d, n) reaction by deuteron irradiation on a thick neutron converter made of single element light nuclide such as C or Be. Generated neutrons irradiate the nuclides in a raw material turned into medical RI by direct reactions.
As a feature of the neutron converter, the single-material Be can generate higher neutron yield, but it starts expanding during deuteron irradiation and finally breaks. This effect is known as blistering, which must be suppressed as much as possible for long life of the converter. Thus, we focused on a new target converter design. The converter consists of two materials, Be and C. On the deuteron incident side beryllium is placed to generate intense neutrons but the length is 0.1 mm shorter than incident deuteron range to dispose the deuterons downstream. The penetrated deuterons are absorbed inside thick carbon part with generating feint neutrons. The thickness of the Be is determined to dispose 99.7% of deuterons. Using the range R_{Be} and its deviation σ_{Be} calculated by the SRIM [1] code, beryllium having $R_{Be} + 3\sigma_{Be}$ mm thick target can achieve the disposing condition. We select thick carbon absorber because the length must be simply determined to shield almost all of deuterons.
We conducted a neutron generation experiment to investigate the performance of the designed C/Be converter at the JAEA tandem accelerator. Deuterons were accelerated to 20 MeV and bombarded on C/Be target to produce neutrons via the C(d, n) reaction. The neutron yield was measured by multiple-foil activation method. The unfolding code GRAVL [2] was used to analyze the neutron yields using the obtained activities of products of the specific reactions. Response functions were analytically determined from cross sections stored in JENDL5 [3]. Initial guess spectrum derived by the Monte Carlo simulation code PHITS [4]. The results are compared with our previous experiment results using a beryllium-single material converter performed at CYRIC.
As a result, a composite converter was designed using the SRIM code, which has a durability 769 times longer than Be converter. The neutron yield when using the combined converter was derived from the experimental results of the JAEA tandem.

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Poster presentation / 14

Small-angle neutron scattering and neutron transmission of hardened cement paste/硬化セメントペーストの中性子小角散乱と中性子透過率

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We have measured a small-angle neutron and X-ray scattering (SANS and SAXS) of cement paste to investigate a nanoscale structure of cement paste [1, 2]. Through the in-situ SAXS measurements of cement paste, in particular, we have focused on a fine nanostructure that emerged with time as a shoulder on the SAXS profiles at the high-q region of around 3 nm^{-1}. Based on a microstructure model of cement paste [3] and a previous SANS work [4], it is expected that the fine nanostructure consists of calcium silicate hydrate (C-S-H) gel and pore water, where C, S, and H stand for respectively CaO, SiO_2, and H_2O in a conventional notation of cement chemistry. The C-S-H is a major hydrate among the cement hydrates and relates closely with the compressive strength of hardened cement paste (HCP).
Recently, to obtain information of the elemental composition of the fine nanostructure, SANS measurements of HCP samples were conducted using a contrast variation method in BL15 TAIKAN of MLF at J-PARC. In addition, the neutron transmissions of the saturated and dried HCP samples were also measured because water contents including these samples were evaluated for subtracting the background due to incoherent scattering of hydrogen from the SANS profiles. In the data analysis for the water contents evaluation, the neutron transmissions which were calculated using the PHITS code were compared with the measured neutron transmissions, where the JENDL-5 ACE library (ACE-j50; neutron induced nuclear data and thermal scattering law data for hydrogen and deuterium in water) [5] was applied to the PHITS calculations. The data analysis of the neutron transmissions is presented together with the data of SANS measurements.

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Poster presentation / 15

Feasibility test of cavity exploration using a prototype muography detector/ミュオグラフィ試作検出器を用いた空洞探査の実現可能性試験

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Shield tunneling methods are widely used to construct large underground tunnels. Although the methods are considered safe, subsidence accidents due to underground cavities created during tunnel excavation have occurred in recent years. To prevent such accidents, it is necessary to detect the cavities and take some measures. Various exploration methods such as ground-penetrating radar have been used to detect such cavities. However, it is difficult to detect cavities deeper than 10 m underground using those conventional methods.
To resolve this issue, we propose an exploration method using the muography technique [1]. Muography is a noninvasive exploration method that utilizes cosmic-ray muons. By measuring muon fluxes at multiple locations underground, information on underground density distribution can be obtained as in a CT scan. Our goal is to develop a disaster prevention system for cave-ins. As a first step, a dedicated prototype muography detector is being developed.
In this presentation, we will report the results of the feasibility test using the prototype detector capable of determining the direction of incoming cosmic rays. For the test measurement, clay bricks were piled above the detector to form a cavity. The muon flux was measured with and without the cavity, and the spatial distribution of transmittance was determined. In the obtained distribution, there was a high-transmittance region corresponding to the cavity position. The size of cavity was estimated from that of the high-transmittance region. A PHITS simulation [2] was then performed incorporating a realistic building structure, and it reproduced the experimental results well.

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Poster presentation / 16

Development of New Technique to Measure Neutron-Induced Charged-Particle Emission Reactions Using Sample-Added Scintillator/試料添加シンチレータを用いた中性子誘起荷電粒子放出反応の新しい測定法の開発

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The cross sections of neutron-induced charged-particle emission reactions such as (n,p) and (n,α) for many nuclides have not been measured as well as those of the neutron capture reaction. In the present work, new measurement technique for neutron-induced charged-particle emission reactions were developed. The new method uses plastic scintillator added with sample material for measurement. The sample-added scintillator attached on a photomultiplier tube is irradiated with neutrons

and charged-particles emitted from neutron-induced reactions are detected at the same time. Scintillators including sample materials were fabricated and the fabricated scintillators were tested in irradiation test experiments conducted with the Accurate Neutron Nuclear Reaction Instrument (ANNRI) of the Japan Proton Accelerator Research Complex (J-PARC). Boron nitride (BN) and lithium fluoride (LiF) were chosen as sample materials to mix with scintillator for the test experiments. The 10B(n,α)7Li and 6Li(n,t)4He reactions occur in scintillators added with BN and LiF, respectively. The cross sections of the reactions are high and the Q-values are also high. Thus, charged particles from the reactions are easy to detect and these reactions are good for test. To identify charged particles, the pulse shape discrimination (PSD) was also employed. The pulse shape discrimination technique is based on the property of organic scintillators that the decay constant of light output changes depending on the mass and charge of charged particles. Signals from the photomultiplier tube were fed into the CAEN waveform digitizer V1720 that enables us to process signal onboard for the PSD parameter. In addition to the PSD parameter, the time-of-flight and the pulse heights of events were recorded sequentially. As a result, charged-particles were detected and identified successfully. The present contribution will report the results of the test experiments.

Poster presentation / 17

Study on Neutron Activation Method Using UV-curing Resin Scintillator/光硬化型プラスチックシンチレータを用いた中性子放射化法の研究

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Neutron activation analysis is used in a variety of many fields, such as geology, medicine, and archaeology. In the neutron activation method, a sample is irradiated with neutrons and activated via neutron-induced reactions. The elemental composition is determined from the radioactivity of the sample. In measurement of radioactivities, -rays are often detected with a germanium detector which provides high enough energy resolution to identify γ-rays from the sample. If the resultant radioactive nuclides emit only β-rays, β-rays are detected with a β-ray detector such as gas counter for high energy β-rays or liquid scintillator for low energy β-rays. β-rays have shorter range than γ-rays. This property requires the sample to be thin enough for β-rays to emerge from the sample or to be dissolved into solvent of liquid scintillator. Thus, sample preparation including chemical processing is usually necessary. Measurement of pure β nuclides is not as easy as that of γ-ray emitting radionuclides. To make neutron activation analysis for pure β nuclides easier, a new technique is being developed in the present research. In the new method, sample material for analysis is added to plastic scintillator and the scintillator including the sample is irradiated with neutrons. The irradiated scintillator is attached to a photomultiplier tube to count β-rays to determine the radioactivity. This technique does not require complicated chemical process before or after irradiation. To study the feasibility of this technique, the method to fabricate scintillator including sample material was developed and test irradiation experiments were carried out at the Tokyo Institute of Technology (Tokyo Tech). A gold foil was irradiated with neutrons from the ^{7}Li(p,n)^{7}Be induced by bombarding a lithium target with a proton beam from a Pelletron accelerator of Tokyo Tech. β-rays from the gold foil were detected with the fabricated scintillator. This contribution will report the current progress and results of the development.

Poster presentation / 18

Performance evaluation of an EJ-276 plastic scintillator using ^{252}Cf neutron source/^{252}Cf 中性子線源を用いた EJ-276 プラスチックシンチレータの性能評価

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Fast neutron measurements are indispensable technique in the field of experimental nuclear physics and nuclear data measurement. In order to discriminate gamma rays generated by the production of fast neutrons, it is necessary to use a detector capable of discriminating between gamma rays and fast neutrons. As a typical detector for fast neutrons, organic liquid scintillators are widely used. Although liquid scintillators are used in metal containers, there is a problem that the volume decreases over time. Furthermore, import and export procedures are complicated because they are toxic and flammable liquids. On the other hand, plastic scintillators are convenient due to their physical hardness, non-toxicity, and lower flammability. One of the latest pulse-shape discriminating plastic scintillators is EJ-276, but there are few measurements of detector characteristics. This study aims to evaluate the capability to discriminate between neutrons and gamma rays and derive the neutron response function.
In the experiment, the time of flight of neutrons from ^{252}Cf neutron sources was measured. A Φ5 inch x 2 inch EJ-276 coupled to a photomultiplier tube (PMT) Hamamatsu photonics R1250 and two Φ2 inch x 2 inch EJ-301s coupled to a PMT Hamamatsu photonics R7724 were used. Signals from the PMTs were fed to a digitizer, CAEN-V1730SB, to convert the analog waveforms into digital data. The discrimination capability of EJ-276 and EJ-301 was compared. The response functions obtained from the experimental data were compared with PHITS SCINFUL calculations [1]. The comparison results will be reported in the presentation.

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Poster presentation / 19

Development of a PHITS simulation technique and a numerical method to optimize measures against radioactive sources/線源対策最適化のための PHITS シミュレーション技術及び数値計算手法の開発

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We have developed the radiation dose evaluation system for indoor environments named 3D-ADRES-Indoor, which is especially designed for two applications: the estimation of radioactive source distributions with the machine learning technique and the planning of measures against estimated radioactive sources [1,2]. 3D-ADRES-Indoor mainly uses Particle and Heavy Ion Transport code System (PHITS) [3] for the ambient dose rate calculation required for these applications. In this work, a PHITS simulation technique and a numerical method for the latter application have been developed.
For a better planning of measures against radioactive sources, it is necessary to repeat the simulations with different geometrical models that incorporate various measures. However, it generally takes long computational times to execute whole new PHITS simulations with different geometrical models. Therefore, we have developed a PHITS simulation technique to construct the dose rates

with specific models using those obtained from the smaller scale simulations that only account for the difference of models. The point of the technique is the decomposition of dose rates using the "counter"feature and the simulation using the "dump source"feature. While this technique requires the decomposed dose rates and dump data obtained by the normal simulation with a prior model, the computational times of the simulations with the models that incorporate various measures to the prior model are significantly reduced compared to the normal simulations.
We consider four kinds of measures against radioactive sources: the decontamination, removal, relocation of contaminated structures, and shielding. Except for the decontamination, the technique explained above is applied. As for the decontamination, a numerical method to optimize decontamination rate of each radioactive source that achieve target dose rates with the minimum cost for the decontamination. In this method, a constrained optimization problem with the loss function composed of the cost function and the penalty function to achieve target dose rates is solved. To solve this numerical optimization problem, Particle Swarm Optimization (PSO) method has been employed.
Users of 3D-ADRES-Indoor can set conditions of four kinds of measures by GUI operations.
PHITS input required for the present techniques are automatically generated depending on the measures. Users can try more variations of measures with the advantage of the present technique.

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Poster presentation / 20

Design of new brachytherapy source using PHITS code/Phits を用いた新規密封小線源治療用線源の設計

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The purpose of this research is to design new radionuclides suitable for brachytherapy sources. Brachytherapy is one of the radiotherapy technique which sealed radionuclides in capsules are inserted into the body. Photons, electrons from a sealed source are usually used. Dose rate can be controlled using a variety of irradiation setup, such as placement and irradiation time to the patients. In some case for the treatment, the biological effects of high-dose-rate irradiation of brachytherapy are superior to those of external beam using medical LINAC. Currently, only a limited number of nuclides such as ^{60}Co, ^{90}Sr, ^{106}Ru, ^{125}I, ^{137}Cs, ^{192}Ir, ^{198}Au are used in clinical practice. Other radionuclide may also have useful dose effects for the treatment, but not all have been investigated yet. In this study, we simulate the dose properties of radionuclides for brachytherapy based on the AAPM (American association of physicists in medicine) TG-43 method [1,2] using PHITS code [3].

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Poster presentation / 21

Direction Vector Visualization of Scattered Radiation for fluoroscopy by PHITS/PHITS による透視検査のための散乱線の方向ベクトルの可視化

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In fluoroscopy, radiation shielding effectively reduces radiation exposure to medical staff [1]. However, it is still unclear how to understand where the scattered radiation comes from and how to properly use radiation shields. The purpose of this study is to clearly visualize the direction of scattered radiation to assist in the optimal use of radiation shields.
The Monte Carlo code PHITS [2] was used to simulate the behavior of scattered radiation under fluoroscopy. The direction vector was obtained by counting the number of photons passing through the plane of each voxel. The voxel space divides the entire fluoroscopy room at regular intervals. The simulations included the x-ray tube, C-arm, water phantom, and couch of the C-arm fluoroscopy system. Scattered photons from the patient were depicted by 3D arrows. Cross sections of the dose distribution were superimposed on the direction vectors. A surgeon model was also included to observe the direction of the scattered rays when the height of the protective plate was adjusted.
The directional vector of the radiation radiating around the patient could be visualized; changing the angle of the C-arm affected the direction and intensity of the radiation. The protective plate effectively shielded the surgeon's head, especially when placed at a height of 130 cm from the floor, resulting in a 99.1% dose reduction.

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The ambient dose in TrueBeam LINAC: Measurement and PHITS simulation with JENDL-5.0/TrueBeam リニアックにおける周辺線量：JENDL- 5.0 による測定と PHITS シミュレーション

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Purpose: At energies above the (γ,n) threshold, photons can interact with the nuclei of high-Z materials, liberating fast neutrons. The aim of this study was to validate the Monte Carlo simulation with the PHITS code for the TrueBeam LINAC using mono-energy. Additionally, we examined the photon neutron dose surrounding the TrueBeam LINAC's head and investigated the influence of field size on the distribution of photon neutron ambient dose.
Method and Materials: Research group used PHITS codes version 3.29, Japanese Evaluated Nuclear Data Library (JENDL-5.0), and the training data of Varian to simulate the head of TrueBeam LINAC 10 MV. The simulated Percentage Depth Dose (PDD) (Field size 10 x 10 cm^2, Source Surface Distance (SSD) 100 cm), and simulated crossline at 5, 10, 20 cm depths were compared with the measured data.

And then research group used these PHITS codes to calculate photon and neutron dose at twenty-five points around the head of TrueBeam LINAC 10 MV with both two field size 20 x 20, and 0.5 x 0.5 cm^2.
In measurement: PDD and crossline were measured with Blue Phantom, CC13S ion chamber, TrueBeam LINAC 10 MV photon; Photon and neutron dose at each point of twenty-five points around the head's TrueBeam LINAC were measured with three Radio-photoluminescence for photon dose and three CR-39 detectors for neutron dose, when TrueBeam LINAC radiated 50 Gy in both field size 20 x 20, and 0.5 x 0.5 cm^2.
Results: The measured neutron doses were in the range 0.4 –12.53 mSv (0.5 x 0.5 cm^2), the range 0.43 –12 mSv (20 x 20 cm^2); The measured photon doses were in the range 0.63 –177.00 mSv (0.5 x 0.5 cm^2), and 2.23 –183.33 mSv (20 x 20 cm^2). The simulated neutron doses were in the range 0.11 - 26.65 mSv (0.5 x 0.5 cm^2), the range 0.06 –14.36 mSv (20 x 20 cm^2); The simulated photon doses were in the range 0.16 –182.63 mSv (0.5 x 0.5 cm^2), and 1.58 –178.84 mSv (20 x 20 cm^2).
Conclusion: Measured and simulated photon neutron dose showed larger field size increased photon ambient dose distribution, and decreased neutron ambient dose distribution. In vice, smaller field size decreased photon ambient dose distribution and increased neutron ambient dose distribution.

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Real-time scattered radiation exposure estimation system during X-ray fluoroscopy using PHITS results/PHITS の結果を用いた X 線透視検査時のリアルタイム散乱線被ばく推定システム

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The control of occupational exposure in fluoroscopy and interventional radiology is critical due to the risk of radiation exposure. Current dosimetry measurements have limitations, such as incomplete whole-body measurements and lack of real-time measurements. To improve radiation awareness, we have developed a system that displays 2D scattered radiation distribution and estimates the surgeon's radiation dose in real time.
Using the Monte Carlo code PHITS [1], a scattered radiation simulation for X-ray fluoroscopy during the examination was performed. Three-dimensional data of scattered radiation was mapped into the X-ray room using an AR marker. A two-dimensional scattered radiation display was created. A real-time scattered radiation exposure system has been developed to track a surgeon's body using AzureKinect. The system's accuracy in distance and dose was tested, comparing measurements with a laser rangefinder and dosimeters.
The system estimated the dose and accurately visualized the radiation distribution. Distance accuracy improved as the surgeon moved closer to the camera. Distance accuracy decreased as the distance from the camera exceeded the body tracking capability. Dose estimates were 0.6 to 1.2 times higher than actual measurements. Dose accuracy was lower in the chest and pelvis areas, likely due to surface mounting of the dosimeters and body tracking of the system for torso measurements.

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Estimation of deposition positions of α-emitters in the body by L

X-ray analysis/LX 線解析による α 線放出核種の体内沈着位置の推定

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In the case of accidents involving inhalation of radioactive materials, there is a need for rapid evaluation of the amount of transuranium (TRU) nuclides such as Pu ingested in the body by measurement from outside of the body. In the measurement, L X-rays, which have energies of 10 to 30 keV and are emitted by internal conversion electrons due to α-decay, are used. In order to measure the L X-ray energy spectrum of TRU, the use of a transition edge sensor (TES)-type microcalorimeter with an energy resolution of less than 100 eV is being considered. It is because each daughter nuclide emits a couple of L X-rays between 10 to 30 keV. In this case, the attenuation of L X-rays in body tissues must be taken into account, and information on the position of deposition in the lungs is important. The deposition positions of α-emitters are estimated by the radiation transport code PHITS with a tetrahedral mesh phantom model published by ICRP and a TES-type microcalorimeter using a Sn absorber. The energy spectra deposited to the absorber are calculated. As a result, it was found that the deposition position in the lung was estimated from the intensity ratio of each L X-ray peak in the energy spectrum.

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Estimation of Radioactivity Depth Distribution of Concrete in a BNCT Facility/BNCT 施設におけるコンクリートの放射化の深さ分布の推定

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In Boron Neutron Capture Therapy (BNCT) facilities, concrete of the treatment room is activated by neutrons and dose rate in the treatment room is still high after the end of neutron irradiation. The concrete wall surrounding the room is thick enough to reduce dose rate outside the room to a safety level. The concrete in certain area of the wall is highly radioactive. Estimating the amount of radioactivity depending on the depth from the concrete surface will provide important information for evaluating the safety and economic efficiency of future building demolition. In this study, we estimated the depth distribution of the amount of activation in concrete due to neutrons in a BNCT treatment room, the nuclides that contribute to the dose rate after the end of neutron irradiation, and the dose rate in the treatment room by the Monte Carlo radiation transport code PHITS. The wall behind the water phantom, which simulates a patient, is made of ordinary concrete and boron-containing concrete. The calculation results showed that the amount of concrete activation decreased for the boron-containing concrete, and that the depth up to about 50 cm from the surface made a large contribution to the dose rate in the treatment room.