FY2024 (2024.4-2025.3) NSEC Annual Report

Nuclear Science and Engineering Center
Nuclear Science Research Institute
Japan Atomic Energy Agency

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Preface

TSUJIMOTO Kazufumi

Director General, Nuclear Science and Engineering Center

The Nuclear Science and Engineering Center (NSEC) of the Japan Atomic Energy Agency (JAEA) aims to conduct research and development to advance the science and technology that supports the use of nuclear energy and radiation. This annual report provides research highlights and an overview of the research groups' activities in the NSEC for Fiscal Year 2024. We hope this annual report will increase your understanding of the NSEC.

The use of nuclear energy and radiation is supported by the underlying basic science and various technologies that link science and engineering. As such, we conduct fundamental research to elucidate various phenomena involving atomic nuclei, radiation, and radioactive materials using our innovative techniques for measurement and analysis. Based on modeling of the observed phenomena, we develop computer simulation codes and databases for predicting the behavior of energetic particles, heat and fluid in a reactor core, performance of nuclear fuel and reactor structural materials, properties and functions of radioactive materials related to their physical and chemical states, migration behavior of radionuclides in the environment, and the effects of radiation on the human body.

It is our responsibility to provide the results of our research and development activities to society in ways that are transparent and have high quality and impact. We have been disseminating our innovative technologies for resolving challenges in various fields, such as industry, environment, and medicine. The NSEC is a key research center for supporting the nuclear energy infrastructure through our nuclear science and engineering research. We strive to become a leading center for research collaboration, using our fundamental research and development capabilities to contribute to advances in science and technology.

We seek your understanding, support, and encouragement in our research and development activities.

September 2025

FY2024 NSEC R&D Highlights

The following 6 highlights are selected among various outcomes of the R&D activities accomplished by the NSEC in FY2024.

- *"Development of a Deep-Learning-Based Bubble Detector Using the Swin Transformer"*
- *"Reactivity Worths of TRU Oxide Samples Measured in FCA-IX Assemblies with Systematically Varied Neutron Energy Spectra"*
- *"Origin of the Unique Mechanical Properties of Refractory High-Entropy Alloys - Mechanical-Properties Design Based on Electronic-Structure Calculations - "*
- *"Development of Adult Japanese Polygon Mesh-type Human Models for Accurate Dose Assessment"*
- *"Sensitive Detection of Nonfluorescent Solutes in Small Amounts of Dilute Aqueous Solutions through Photothermally Induced Reflectivity Modulation"*
- *" Development of a Dissolution Method for Analyzing the Elemental Composition of Fuel Debris Using the Sodium-Peroxide Fusion Technique"*

Development of a Deep-Learning-Based Bubble Detector Using the Swin Transformer

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Research Group for Reactor Physics and Thermal-Hydraulics Technology

This study proposes a novel bubble-detection method for gas–liquid two-phase flows using the latest deep-learning technology, the Shifted Window Transformer (Swin Transformer)[1]. The goal is to achieve high-precision detection and segmentation of bubbles. Parameters such as bubble size, distribution, and void fraction (the volume ratio of bubbles in the fluid) are essential for validating computational fluid dynamics simulations, especially in thermal-hydraulic applications, including nuclear engineering[2].

Traditionally, bubble detection has relied on rule-based image-processing techniques that use brightness differences. Representative methods include the Hough transform for detecting circular objects, the breakpoint method for extracting contour features, and the watershed algorithm, which uses brightness gradients to separate regions. However, these methods struggle when bubbles are deformed or overlapping, resulting in reduced accuracy.

To improve the detection of deformed and overlapping bubbles, this study adopted a deep-learning-based image-recognition model, the Swin Transformer. The Swin Transformer splits an image into small regions (windows) and uses a technique called self-attention to identify important patterns within each region. This enables efficient and accurate feature detection. By shifting the windows during processing, the model captures inter-regional dependencies, achieving high recognition accuracy while reducing computational cost.

The Swin Transformer was used to build a bubble detector[3]. For training, several dozen synthetic bubble images generated by generative AI, together with experimental bubble images, were used. These synthetic images eliminated labeling errors and enabled stable training, while the experimental images ensured that real-world observation conditions were reflected.

Figure 1 compares the proposed method with traditional rule-based detection techniques (background removal, brightness-based contour extraction, and the watershed algorithm). The Swin Transformer-based detector accurately separated bubbles even in shadowed regions or in cases of overlap. Validation against conventional rule-based image-recognition methods demonstrated superior performance in detecting bubbles obscured along the line of sight.

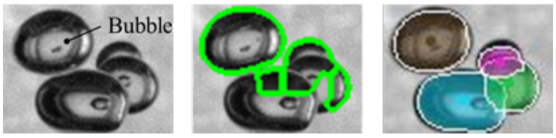

Fig. 1 Detection of overlapping bubbles along the line of sight (Left: captured image, center: detection results (green lines) using rule-based image recognition, right: detection results (white lines) using the proposed method)

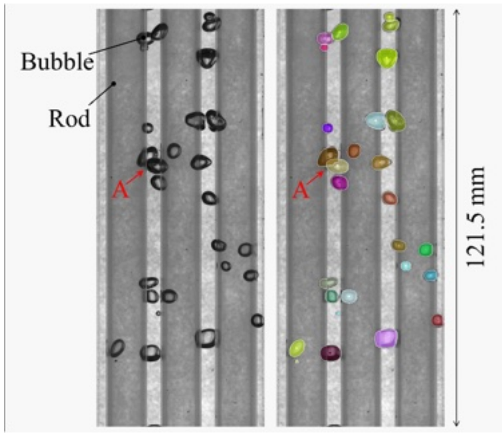

Fig. 2 Visualization in a rod-bundle flow channel (left) and bubble-detection results (right)

The detector was applied to a more complex flow geometry—a 3×3 rod bundle (Fig. 2). Overlapping bubbles in narrow subchannels were successfully detected (A in Fig. 2), and bubble-size distribution was also estimated. These results indicate that the detector is suitable for analyzing bubbly flow in rod bundles.

In conclusion, the bubble-detection method using the Swin Transformer maintains high accuracy even with limited training data and adapts well to complex fluid structures and experimental conditions. This approach is expected to provide a reliable image-analysis tool for future thermal-hydraulic studies and computational fluid dynamics model validation.

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Reactivity Worths of TRU Oxide Samples Measured in FCA-IX Assemblies with Systematically Varied Neutron Energy Spectra

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1 Research Group for Nuclear Transmutation System, 2 Retired (JAEA)
3 Retired (Japan Atomic Energy Research Institute)

Geological disposal of high-level radioactive waste requires containment for hundreds of thousands of years. To address this challenge, partitioning and transmutation (P&T) technologies offer a promising solution by transmuting long-lived fission products and transuranic (TRU) nuclides into short-lived or stable ones, thereby reducing the radiotoxicity and heat generation of the waste. Accurate nuclear data for TRU nuclides are essential for designing and evaluating such systems safely and effectively.

To improve nuclear-data accuracy, a series of integral experiments were conducted using the Fast Critical Assembly (FCA)[1)]. Seven experimental configurations (IX-1 to IX-7) were developed, covering neutron spectra from intermediate to fast regions (Fig. 1). Unlike single integral experiments, which lose neutron-energy information, this systematic approach enabled energy-dependent validation of nuclear data. While previous studies focused on benchmarking criticality[2)] and TRU fission-rate ratios[3)], the present work further evaluated the reactivity worth of small TRU samples[4)], completing a comprehensive dataset for validating both fission and capture cross sections (Table 1).

Using this dataset, integral validation of the latest nuclear-data library, JENDL-5, was performed. Detailed modeling of the experimental conditions was carried out using the continuous-energy Monte Carlo (MC) code MCNP6.2 to minimize analytical errors. As shown in Fig. 2, the fission-rate ratios and reactivity worth of ^{237}Np were consistently reproduced across all configurations, confirming the reliability of its nuclear data regardless of neutron spectrum. In contrast, spectrum-dependent discrepancies were observed for americium isotopes[4)], indicating the need for further investigation.

This newly established database systematically covers TRU nuclides that lack sufficient experimental data internationally, providing both fission-rate ratios and sample reactivity worth across various neutron spectra. This database is expected to significantly improve the accuracy of nuclear data and support the development of advanced P&T technologies for sustainable nuclear-waste management.

Table 1 Database of TRU integral experiments

Measured item (reaction)	Fission Rate ratio (fission)	Sample reactivity worth (capture, fission)
^{237}Np	Yes	Yes
^{238}Pu	Yes	Yes
^{240}Pu	—	Yes
^{242}Pu	Yes	—
^{241}Am	Yes	Yes
^{243}Am	Yes	Yes
^{244}Cm	Yes	—

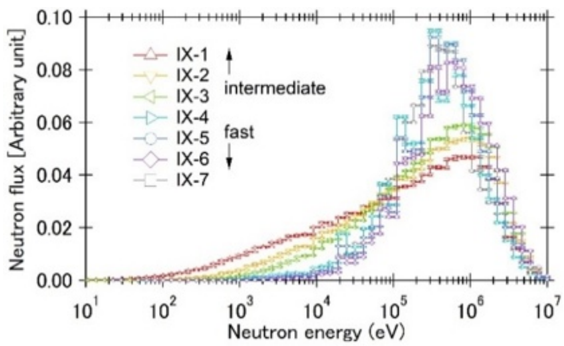

Fig. 1 Systematically varied neutron energy spectra of FCA-IX assemblies

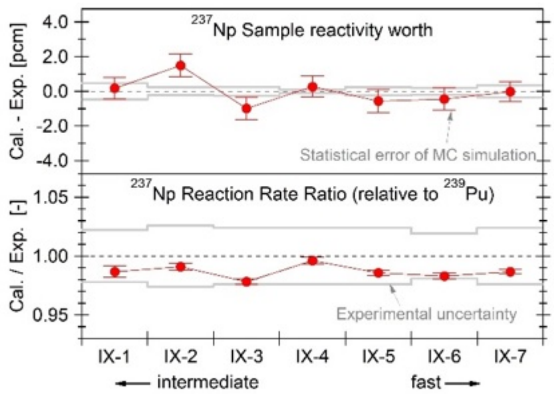

Fig. 2 Example comparison between experiment and calculation (^{237}Np)

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Origin of the Unique Mechanical Properties of Refractory High-Entropy Alloys - Mechanical-Properties Design Based on Electronic-Structure Calculations -

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Research Group for Radiation Materials Engineering

Refractory high-entropy alloys (RHEAs) are of interest for ultrahigh-temperature applications. To overcome their drawbacks — low-temperature brittleness and poor creep strength at high temperatures — an improved fundamental understanding is needed. TiZrHfNbTa (RHEA-Ti) and VNbMoTaW (RHEA-V) alloys, which are expected to be used as new refractory alloys in place of Ni-based superalloys, have been widely studied. Using experiments, theory, and modeling, we investigated prototypical body-centered cubic (BCC) RHEAs. Our experiments found that these two alloys have different strength and ductility properties, but the underlying mechanism remains unclear. The present study aims to clarify the factors responsible for these differences in mechanical properties using electronic-structure calculations.

In metals, lattice distortion is known to correlate closely with strength. Therefore, we evaluated the lattice distortion using an effective parameter, the mean-square atomic displacement (MSAD), which is directly calculated from atomic displacements, as shown in Fig. 1. A clear difference was found in the normalized MSAD values of the two RHEAs. The MSAD of RHEA-V is close to that of FCC HEAs, whereas that of RHEA-Ti is quite large, exceeding 6% of the Burgers vector. This substantial lattice distortion (large MSAD) contributes not only to volumetric strain but also to shear-strain components.

The dislocation structure, which governs ductility, was also analyzed. Typical examples of dislocation core structures in the two RHEAs are shown in Fig. 2, where the dislocation core was identified using differential displacement vectors. The dislocation in RHEA-V has a compact core, as commonly seen in pure BCC metals. By contrast, the dislocation core of RHEA-Ti is heterogeneously spread. This extended core arises from its tendency toward phase instability, which originates in the electronic structure of constituent elements. The core energy is distributed over a wide energy range. More importantly, however, there is a significant difference between the average core energies of the two RHEAs. The dislocation core energy of RHEA-Ti is much lower than that of RHEA-V. This finding indicates that dislocations are easily introduced into the matrix, contributing to the excellent ductility in RHEA-Ti at low temperatures. Since plastic deformability (ductility) depends partly on the ease of dislocation nucleation and partly on the ease of dislocation motion, our results provide a plausible explanation for why RHEA-Ti exhibits superior ductility. Electronic-structure calculations further showed that these properties arise from group IV elements such as Ti, Zr, and Hf, which are expected to play a key role in element-strategy alloy design.

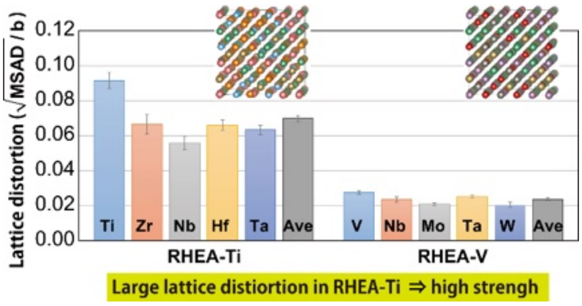

Fig. 1 Lattice distortion of RHEA-Ti and RHEA-V. The mean-square atomic displacement of RHEA-Ti is quite large, exceeding 6% of the Burgers vector, which results in a higher modulus-normalized yield stress for RHEA-Ti.

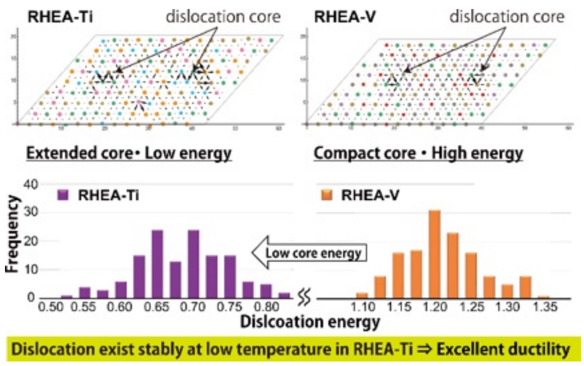

Fig. 2 Dislocation core structure and dislocation energies of RHEA-Ti and RHEA-V. The dislocation core energy of RHEA-Ti is much lower than that of RHEA-V, indicating that dislocations can be introduced more easily in RHEA-Ti.

Reference

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Development of Adult Japanese Polygon Mesh-type Human Models for Accurate Dose Assessment

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Human physical characteristics such as body size and posture generally influence exposure doses. To accurately evaluate exposure doses, it is necessary to assess the detailed behavior of radiation in the body by using a human model that reflects physical characteristics as closely as possible. Thus, a calculation method by combining a Monte Carlo simulation code, including PHITS, with a human model is very useful for evaluating exposure doses.

The International Commission on Radiological Protection (ICRP) has updated knowledge about the effects of radiation on the stem cell regions of the skin and lens, which are highly radiosensitive, and has reduced the equivalent dose limit for the lens. This means that it is important to accurately evaluate the exposure dose to the stem cell regions. However, it was difficult to accurately construct the stem cell region into the body of the earlier human models, because of its less minute and complex structures.

We have newly developed polygon mesh-type human models for male (JPM: the Japanese Polygon mesh-type Male model) and female (JPF: the Japanese Polygon mesh-type Female model) (Fig. 1(a))[1,2]. JPM and JPF have the average body sizes and organ masses of Japanese adults. By employing a polygon technique that can flexibly represent object shapes, we constructed the stem cell regions in JPM and JPF. For example, the micro-sized and complicated structures (cornea, vitreous, aqueous and lens (Sensitive and others)) of the eye tissue were accurately reproduced (Fig. 1(b)).

Figure 2 shows the energy dependence of the organ doses absorbed in the eye tissues of JPF irradiated by electrons in the antero-posterior (AP) geometry. In the irradiation energy range below 0.5 MeV, electrons have extremely short ranges and do not reach the lens, giving almost no dose. By contrast, at higher irradiation energies of 0.6 to 1.5 MeV, more electrons reach the lens (Sensitive), resulting in a significant increase in dose. In addition, at 4 MeV or higher, electrons have ranges that exceed the size of the eyeball, so the eyeball dose becomes consistent with that of the lens (Sensitive). These results demonstrate that JPM and JPF can accurately evaluate the doses to eye tissues, reflecting their fine and complex structures and the physical characteristics of adult Japanese.

Development of a deformation technique for changing the postures and body sizes of JPM and JPF is currently underway. In the future, combining the deformation technique with JPM and JPF will enable evaluation of exposure doses that take into account individual posture and body size. This is expected to improve dose control for medical personnel and patients during medical treatment and for workers in radiation accidents, resulting in optimized radiation protection.

The electronic data of the JPM and JPF are available free of charge on the GitHub repository site[3] from October 24, 2024, and can be obtained from the site without any procedures.

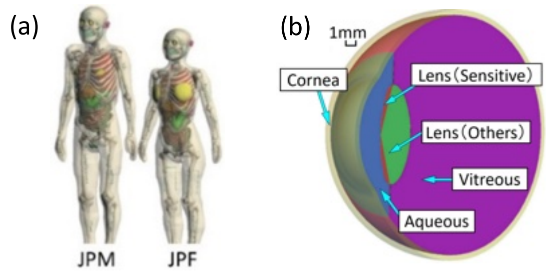

Fig. 1 Polygon mesh-type human models (a) Anterior view of the whole body (b) 3D cross-sectional view of the eye tissue model

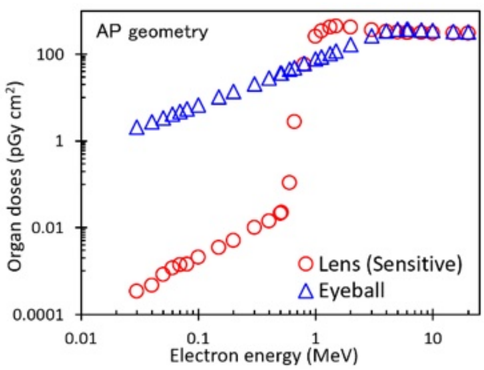

Fig. 2 Organ dose of eye tissues for electron incidents in the AP geometry on JPF

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Sensitive Detection of Nonfluorescent Solutes in Small Amounts of Dilute Aqueous Solutions through Photothermally Induced Reflectivity Modulation

URASHIMA Shu-hei, KUSAKA Ryoji

Research Group for Nuclear Chemistry

Sensitive detection of optical absorption with minimal sample volumes is essential for the chemical characterization of hazardous materials, such as radioactive waste. Photothermal spectroscopy, including techniques such as thermal lens spectroscopy, has been widely employed for this purpose. However, conventional detection methods often require precise optical alignment to achieve reproducible and highly sensitive measurements. In this study, we demonstrate a sensitive and reproducible optical absorption detection method with a simplified experimental setup, based on photothermal reflectivity modulation[1).

In this approach, the sample solution is deposited onto a transparent substrate, such as silica, and two laser beams—a pump beam resonant with the sample and a non-resonant probe beam—are focused onto the sample–substrate interface (Fig. 1(a)). Photoabsorption of the pump beam induces localized heating at the focal spot, leading to a temperature rise. Since the temperature dependence of the refractive index differs between the sample and the substrate, the reflectivity at the interface is modulated by the pump-induced heat. This modulation is detected by monitoring the intensity of the probe beam reflected from the interface. As the heat source originates from the sample's optical absorption, the magnitude of the reflectivity modulation quantitatively correlates with the absorption strength, enabling determination of the sample concentration. As illustrated in Fig. 1(b), this method requires only coaxial alignment and focusing of the pump and probe beams onto the same location at the interface, in contrast to thermal lens spectroscopy, which requires a slight spatial offset between the two focal spots to achieve optimal sensitivity.

Figure 2 presents the relationship between sample concentration and the magnitude of reflectivity modulation, demonstrated using an aqueous solution of Ponceau 4R. A clear linear correlation was observed across a broad concentration range from 0 to 2000 nmol/L (nM), with a distinct signal even between the blank and the 100 nM sample. The limit of detection (LOD), calculated quantitatively based on noise levels, was determined to be 75 nM. Considering that photothermal spectroscopy probes only the molecules within the tightly focused pump beam (sub-micrometer), this concentration corresponds to approximately 75 molecules within the probed volume. Furthermore, based on the molar absorptivity of Ponceau 4R (18800 $M^{-1}\cdot cm^{-1}$), this concentration yields an absorption coefficient of 0.0014 cm^{-1}. This value is comparable to the best LOD reported in the literature (0.0011 cm^{-1}), which requirs a more complex photothermal spectroscopy setup[2)]. These results demonstrate that our method achieves highly sensitive absorption detection with a simplified optical configuration, meeting the requirements outlined for hazardous sample analysis.

This work was supported by the MEXT Leading Initiative for Excellent Young Researchers, Grant Number JPMXS0320230068.

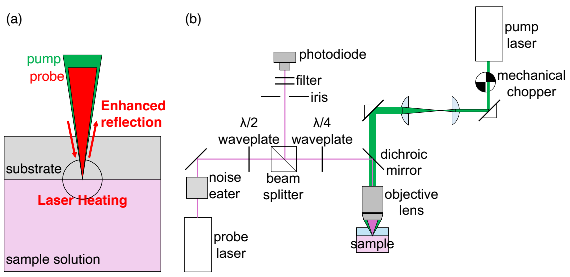

Fig. 1 (a) Schematic illustration of photothermal reflectivity modulation spectroscopy (b) Optical setup used in this study

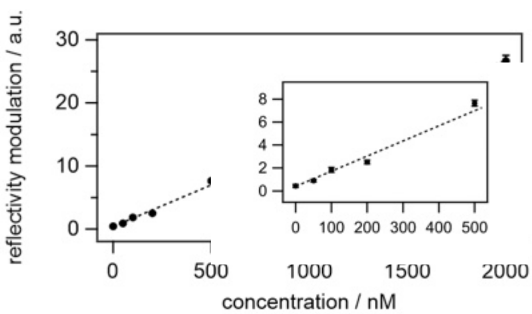

Fig. 2 Concentration dependence of the reflectivity-modulation signal

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Development of a Dissolution Method for Analyzing the Elemental Composition of Fuel Debris Using the Sodium-Peroxide Fusion Technique

NAKAMURA Satoshi[1], ISHII Sho[2], KATO Hitoshi[3], BAN Yasutoshi[1], HIRUTA Kenta[4], YOSHIDA Takuya[4], UEHARA Hiroyuki[4], OBATA Hiroki[4], KIMURA Yasuhiko[4], TAKANO Masahide[5]

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2 Rikohkagaku Co. Ltd
3 Gic Corp
4 Department of Criticality and Hot Examination Technology
5 Nuclear Science and Engineering Center

As a pretreatment for the chemical analysis, it is crucial to dissolve the fuel debris to estimate the elemental and nuclide compositions with high accuracy. However, because fuel debris comprises chemically stable ceramic components, such as oxides, borides, and various alloys, it is barely soluble in nitric acid. In this study, we focus on alkaline fusion using sodium peroxide (Na_2O_2) as the fusing flux. This method has been applied in chemical analysis in the nuclear field as well as for the analysis of materials similar to fuel debris.

In the present work[1)], after studying dissolution methods with various types of simulated debris, a demonstration test with Three Mile Island Unit 2 (TMI-2) debris was conducted. First, we focused on zirconia, the least soluble component of fuel debris in nitric acid, and a solid solution of zirconium oxide and rare-earth oxides ($(Zr,RE)O_2$) was fused with Na_2O_2 at different temperatures in Ni crucibles to determine the optimum temperature. Then, fusion with Na_2O_2 was performed in crucibles of several different materials under the optimum temperature. Subsequently, a series of experiments was conducted using molten core – concrete interaction (MCCI) products that were more representative of actual fuel debris. The objective was to examine the influence of leached elements on analysis and to assess the applicability of the method to MCCI samples using various inexpensive crucibles. The results suggested that Ni crucibles at 923 K provide the optimum testing condition.

The optimum testing condition was then applied in demonstration tests with TMI-2 debris in a shielded concrete cell at the NUclear fuel Cycle safety Engineering research Facility (NUCEF) of the Nuclear Science Research Institute (Fig. 1), thereby achieving complete dissolution of the debris. The elemental composition of TMI-2 debris obtained using the proposed dissolution method showed good reproducibility and only insignificant deviations in the mass balance of the sample. Therefore, this newly developed, reproducible dissolution method can be effectively utilized in practical applications for dissolving fuel debris and estimating its elemental composition.

We recently received fuel debris obtained from the first trial retrieval inside the pedestal of Unit 2 at Fukushima Daiichi Nuclear Power Station. The developed Na_2O_2 fusion technique was applied in the shielded concrete cell, and the experiment was successfully carried out. Complete dissolution was confirmed by visual observation, indicating that the technique is highly promising for subsequent fuel debris analyses.

Reference

1) S. Nakamura, et al., J. Nucl. Sci. Technol. 62(1), 56 (2025).

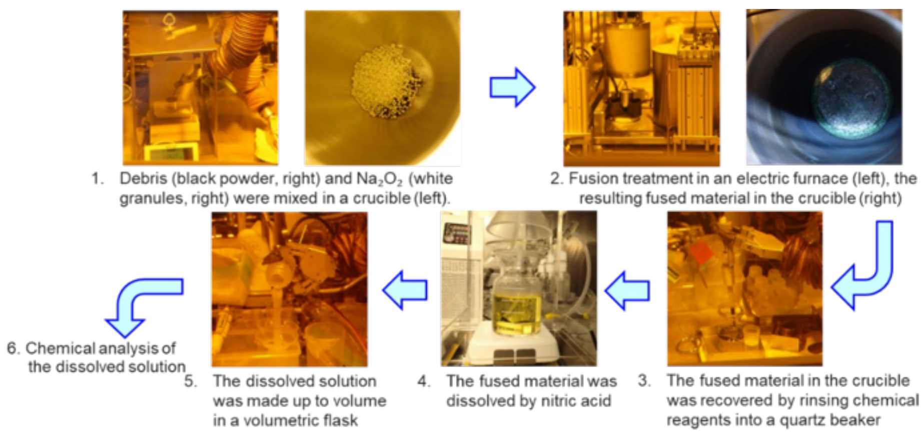

Fig. 1 Scheme of the demonstration tests with TMI-2 debris in a shielded concrete cell

FY2024 NSEC Group Activities

The NSEC of JAEA consists of 11 Groups.

- ✧ *Nuclear Data Center*
- ✧ *Research Group for Reactor Physics and Thermal-Hydraulics Technology*
- ✧ *Research Group for Nuclear Transmutation System*
- ✧ *Research Group for Corrosion Resistant Materials*
- ✧ *Research Group for Radiation Material Engineering*
- ✧ *Research Group for High Temperature Science on Fuel Materials*
- ✧ *Research Group for Environmental Science*
- ✧ *Research Group for Radiation Transport Analysis*
- ✧ *Research Group for Nuclear Sensing*
- ✧ *Research Group for Nuclear Chemistry*
- ✧ *Development Group for Nuclear Engineering Technology*

Organization of NSEC

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  Deputy Director General: NAGAI Haruyasu, OSAKA Masahiko
  - Research Co-ordination and Promotion Office — KATO Chiaki
  - Nuclear Data Center — IWAMOTO Osamu
  - Research Group for Reactor Physics and Thermal-Hydraulics Technology — NAGAYA Yasunobu
  - Research Group for Nuclear Transmutation System — NISHIHARA Kenji
  - Research Group for Corrosion Resistant Materials — IGARASHI Takahiro
  - Research Group for Radiation Materials Engineering — OKUBO Nariaki
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  - Research Group for Nuclear Chemistry — KUMAGAI Yuta
  - Development Group for Nuclear Engineering Technology — KITATSUJI Yoshihiro

URL: https://nsec.jaea.go.jp/organization/en_index.html

Nuclear Data Center

To provide reliable nuclear data, we are engaged in research on theories, measurements and evaluations related to nuclear reactions and structure. The evaluated nuclear data have been compiled in the Japanese Evaluated Nuclear Data Library (JENDL), which is available on our website (http://wwwndc.jaea.go.jp).

Thermal-Neutron Capture Cross-Section Measurements for Nuclear Decommissioning

Thermal-neutron cross sections are important not only for neutronic calculations in operating nuclear reactors but also for radioactivity evaluations during their decommissioning. While a relatively small number of nuclides affect the neutronic calculations, it is necessary to estimate the contribution of radioactivity of many nuclides from the viewpoint of clearance of radioactive materials. The structural materials of reactors contain various impurities that would produce radioactive nuclides by neutron-capture reactions. We are measuring the thermal-neutron capture cross sections for various nuclides related to clearance. In this fiscal year, three scientific papers were published on this topic[1,2,3]. The results for Sc, Cu, Zn, Ag, In, Fe, Er and Hf were obtained. These measurements reveal that many of the results, such as those for the isotopes of Sc, Ag, In, Fe and Er are generally consistent with the latest nuclear data library, JENDL-5. However, they also indicate that the evaluated values of JENDL-5 for ^{63}Cu and ^{180}Hf significantly overestimate or underestimate the present results, respectively. These findings will be considered in nuclear data evaluations for updating JENDL-5.

Time-of-flight Measurements of Neutron Cross Sections

The Accurate Neutron–Nucleus Reaction Measurement Instrument (ANNRI), installed in the Materials and Life Science Facility (MLF) of the Japan Proton Accelerator Research Complex (J-PARC), is one of the most powerful instruments for nuclear data measurements using the time-of-flight method due to its high neutron flux. With ANNRI, we measured the neutron capture cross sections of ^{129}I, which is one of the important long-lived fission products, over the wide neutron energy range from 10 meV to 30 KeV for the first time[4]. It was found that the evaluated data of JENDL-5 overestimates compared with the measured cross sections in the energy region between 1 to 20 eV, which lies between thermal and first resonance energies.

The gamma-rays emitted in neutron capture reactions carry information about the details of the reaction. The polarization of the emitted gamma-rays depends on the reaction system and can be used to identify the spin of the resonance. We developed a polarimeter based on magnetic Compton scattering and tested it using gamma-rays produced with laser Compton scattering, which demonstrated the expected performance[5]. The polarimeter was installed at ANNRI and successfully used to measure the polarization of gamma-rays from neutron-capture reactions on ^{32}S.

Measurement and Evaluation of Thermal-Neutron Scattering Law

The thermal-neutron scattering law (TSL) is a critical nuclear dataset that influences the calculated neutron flux for thermal nuclear reactors. We measured and evaluated the TSL of graphite that is expected to be used in high-temperature gas-cooled reactors and molten-salt reactors. The neutron scattering and total cross sections were measured with Cold-Neutron Disk-Chopper Spectrometer (AMATERAS) and ANNRI at J-PARC, respectively[6]. The first-principles calculation of the phonon density of states for ideal crystalline graphite was used for the evaluation of TSL[7]. It was found that the experimental data support the present evaluation rather than the nuclear data of JENDL-5, which adopted the ENDF-VIII.0 evaluation based on molecular-dynamics calculations for porous graphite.

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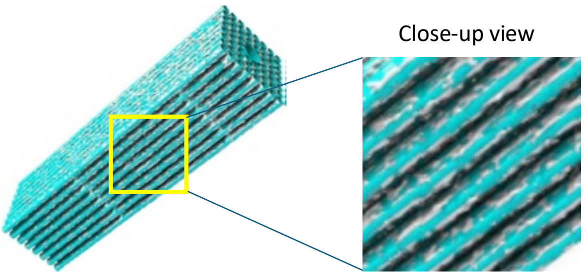
Research Group for Reactor Physics and Thermal-Hydraulics Technology

Our group is working on reactor physics and thermal hydraulics. The group's mission is to develop not only individual technologies but also multi-physics simulation technologies. We are currently focusing on the development of an advanced neutronics/thermal-hydraulics coupling simulation system for the enhanced safety of light-water reactors (LWRs) and the improvement of LWR design.

JAMPAN

Nuclear reactor core design codes account for multiple physical phenomena, such as nuclear reactions and heat transfer within a reactor. To validate core design codes, it is necessary to consider multiple physical phenomena that occur simultaneously in actual reactors. However, it is difficult to reproduce such combinations experimentally. Therefore, we developed a multi-physics simulation platform named JAMPAN[1] that can simulate multiple physical phenomena in combination. Using JAMPAN, we conducted detailed multi-physics simulations targeting fuel in boiling-water reactors (BWRs) (Fig. 1). This large-scale simulation is expected to provide data equivalent to experiments that reproduce multiple physical phenomena. By utilizing the analysis results from JAMPAN, we aim to contribute to improvements in core design, such as enhanced reliability and performance.



Cyan indicates fuel rods, and the light gray indicates steam bubbles.

Fig. 1 Simulation results of an 8 x 8 BWR fuel assembly

Visualization of Atomized Fuel Debris

In a severe nuclear reactor accident, when molten fuel falls into a shallow water pool, some of the molten fuel may split into many tiny droplets. We developed a method to visualize this phenomenon, involving two different liquids (molten fuel and water) in three dimensions through simulated experiments.[2] We applied a visualization method that can accurately measure the size and speed of each of the numerous droplets (Fig. 2).[3] As a result, we clarified that the velocity difference between the two liquids, as well as the centrifugal force and gravity arising from their motion, influence droplet formation. The findings of this study deepen our understanding of the process by which molten fuel fragments into small pieces, cools, solidifies, and forms fuel debris during a severe accident. These findings are expected to contribute to the decommissioning of the Fukushima Daiichi Nuclear Power Station and to improving nuclear reactor safety.

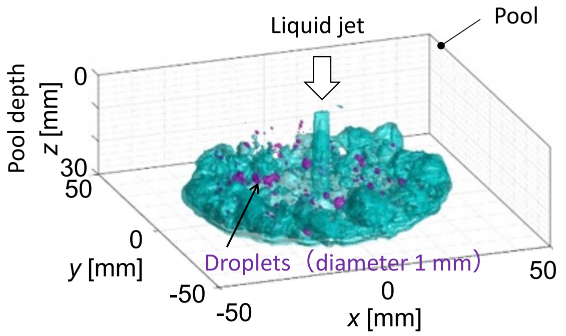


Fig. 2 Three-dimensional visualization of a liquid-jet phenomenon in which the jet splits into nearly 1,000 droplets

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Research Group for Nuclear Transmutation System

The Research Group for Nuclear Transmutation System is developing an Accelerator-Driven System (ADS) dedicated to the transmutation of minor actinides (MAs) into short-lived or stable nuclides. Because the ADS utilizes high-energy protons and unconventional materials such as lead-bismuth eutectic (LBE), fundamental experimental data is insufficient. We are conducting research to expand the existing experimental data using the latest statistical methods and to apply it to the design of the ADS.

Comprehensive Estimation of Nuclide-Production Cross Sections

In the design of an ADS, it is essential to accurately estimate the production yields of nuclides generated by high-energy protons targeting materials inside ADS. To this end, we have developed a machine-learning model to evaluate the nuclide production over a wide range of proton energy and target element. Figure 1 shows the production probability (cross section) of beryllium-7 (^{7}Be) from proton-induced reactions at energies from 0.05 to 3 GeV, plotted as a function of the target mass number. By learning from the existing experimental data shown as dots in Fig. 1, the developed model can predict cross sections even for target nuclei in the mass range of 100–170, for which experiments are scarce or difficult. This technique enables a more reliable design of the ADS.

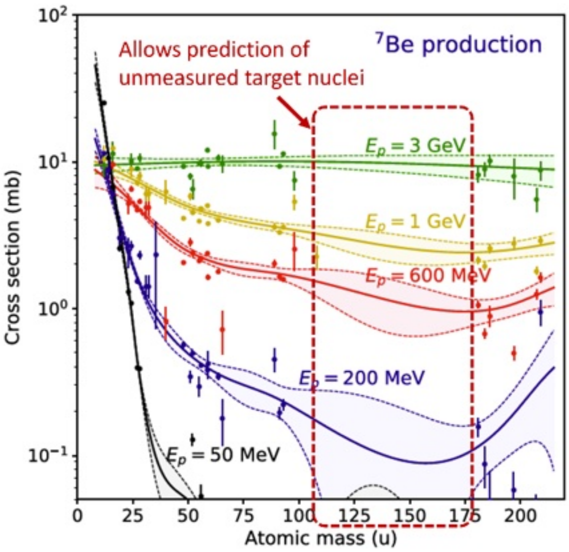

Fig. 1 Proton-induced ^{7}Be production cross section as a function of target mass number[1]. Although the experimental data (dot) provides cross-sections in a very narrow range of proton energies and target materials, the present model expands this to the entire range (solid line).

Uncertainty Reduction Using Sample Reactivity Experiments

The ADS employs liquid-metal LBE as the coolant because of its attractive properties, although LBE has rarely been utilized in reactors. In particular, the reaction data between neutrons and LBE is not well known and has a significant impact on the neutronics design of the ADS. One important core parameter is coolant-void reactivity (CVR), which represents the change in criticality when LBE leaks or boils away from the core. The required CVR uncertainty is 5%, but the current uncertainty estimate slightly exceeds this target. To address this, we applied the data-assimilation technique to the CVR using experiments performed at the Kyoto University Critical Assembly (KUCA)[2]. This technique transfers information from the relevant core (e.g., KUCA) to the target core (e.g., ADS) via common fundamental parameters (e.g., reaction data of neutrons). Although the KUCA experimental core differs substantially from the actual ADS core in terms of size, nuclear fuel, and coolant material, we succeeded in reducing the CVR uncertainty from 6.3% to 3.6% (Table 1).

Table 1 Prior and posterior uncertainties of coolant-void reactivity (%)[2]

Target	Prior	Posterior
5.0	6.3	3.6

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Research Group for Corrosion Resistant Materials

Our research group investigates the corrosion phenomena of metallic materials used in nuclear facilities. Corrosion can degrade reactor components, leading to hole and crack formation. Hence, we aim to elucidate the governing corrosion mechanisms to develop reliable methods for corrosion prediction and prevention. Our principal goal is to enhance the reliability and extend the service life of nuclear infrastructure.

Non-Destructive Evaluation of Corrosion Progression in SM490A Steel Using Image Processing

This study investigates a non-destructive approach for evaluating corrosion progression in SM490A steel by utilizing image-processing techniques without the application of machine learning.[1)] Corrosion images obtained from cyclic wet-dry tests were analyzed using feature-point detection. The method, which incorporates luminance gradients and angular features within the images, demonstrated a strong correlation with actual corrosion depth (Fig. 1). As it does not require training data, this technique offers a rapid and innovative solution for corrosion diagnosis. This method is significant in that it can predict corrosion depth with moderate accuracy without any training. However, for improved precision, it would be preferable to use techniques such as machine learning, even though they may require more time.

Experimental and Modeling Studies on Oxygen Ingression Behavior in Stainless-Steel Crevices in High-Temperature Water

This study investigates the behavior of oxygen ingression and associated water-chemistry changes within crevices of 316L stainless steel under light-water reactor (LWR) conditions (288 °C, 8 MPa).[2)] The limiting distance of O_2 ingress (d_{lim}) was identified by Raman spectroscopy as the depth at which the surface oxide composition transitions from γ(α)-Fe_2O_3 to Fe_3O_4 (Fig. 2(a)). This transition coincides with a sharp increase in electrical conductivity, indicating ion enrichment due to differential oxygen concentration. Experimental variables included crevice gap (g, 5–1000 µm), dissolved oxygen concentration (0.2 and 8 ppm), and immersion time (100 and 1000 h). Results showed that d_{lim} increases with gap size, oxygen concentration, and time (Fig. 2(b)). A finite-element model incorporating oxide-film growth was developed to simulate time-dependent oxygen ingress. The model successfully reproduced experimental trends, demonstrating that oxide layer formation suppresses anodic dissolution and slows oxygen consumption, allowing deeper oxygen penetration. These findings advance the mechanistic understanding of crevice water-chemistry evolution and provide a foundation for predicting stress-corrosion cracking (SCC) behavior in reactor environments.

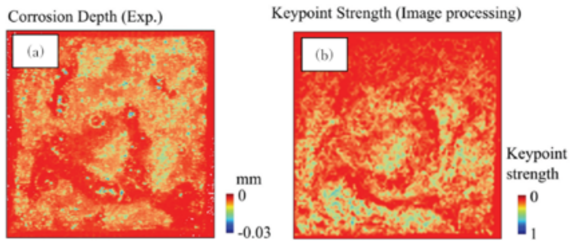

Fig. 1 (a) Heat map of corrosion depth after rust removal (b) Contour map of keypoint strength form a corrosion image obtained by point detection

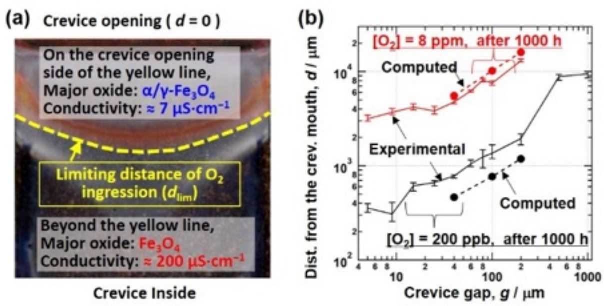

Fig. 2 (a) Crevice surface after the corrosion test (b) d_{lim} on the g (crevice gap)-d (distance from the crevice mouth) plane

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Research Group for Radiation Material Engineering

In the radiation environment of nuclear systems such as light water reactors and advanced nuclear reactors, mechanical properties change as the microstructure of materials evolves. To predict changes in the properties of nuclear materials, we investigate changes in their microstructure using both experimental and computational approaches. In particular, fundamental and engineering studies aimed at elucidating the mechanisms of radiation-induced property changes, such as radiation-induced embrittlement, are important for identifying superior candidate nuclear materials.

Cr-Rich Precipitate Formation in Fe-Cr-Al ATF Alloys: Effects of Composition and Dose Rate via Regression Analysis

To improve nuclear safety, Fe-Cr-Al alloys have been developed as promising accident-tolerant fuel (ATF) cladding candidates. However, irradiation-induced Cr-rich precipitates (CrRP), which cause embrittlement, remain a critical issue. This study[1) examines CrRP formation in 14 Fe-Cr-Al model alloys with systematically varied Cr and Al contents, irradiated at 350 °C using 10.5 MeV self-ions to 0.24 dpa. Three dose rates (8×10^{-6} to 8×10^{-4} dpa/s) simulated neutron irradiation conditions.

Three-dimensional atom probe (3DAP) quantified CrRP number density and size, which were used to evaluate Vickers hardness. A multiple regression model examined the effects of composition, dose rate, and dose. Figure 1 shows the predicted increase in Vickers hardness, derived from CrRP metrics, as contour maps in Cr–Al space, revealing increases with higher Cr, lower Al, and slower dose rates. The dataset and model provide a robust basis for numerical simulations and alloy design to improve irradiation tolerance.

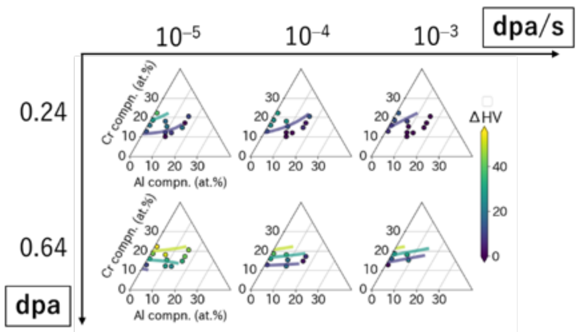

Fig. 1 Increase in Vickers hardness evaluated from CrRP metrics obtained by 3DAP (dots) and model predictions (contours) across Cr–Al compositions.

Machine Learning and First-Principles Study of Al Effects on CrRP Formation in Fe-Cr-Al Alloys

To deepen the fundamental understanding of CrRP formation in Fe-Cr-Al alloys under irradiation, this study[2) examined the effect of Al additions through long-term thermal aging experiments analyzed with a machine learning model. The total dataset included 590 samples—190 from this study and 400 from 28 previous reports. Complementary first-principles calculations were performed to clarify atomistic formation mechanisms via static energy analysis.

As shown in Fig. 2(a), the machine learning model indicates that CrRP formation is enhanced in alloys with less than 10 at.% Al but suppressed above this level. First-principles results (Fig. 2(b)) explain this non-linear trend: Cr-Al-vacancy complexes are more stable than Cr-Cr pairs and promote nucleation, whereas Al-Al pairs are unstable and destabilize CrRP at high Al concentrations.

These insights contribute to the rational design of Fe-Cr-Al alloys with improved irradiation resistance, especially for ATF cladding applications.

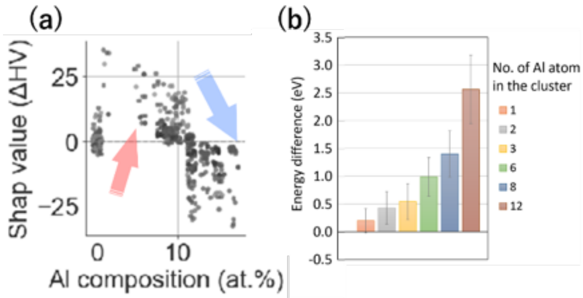

Fig. 2 (a) Increase in hardness Δ HV after thermal aging as a function of Al content, based on a machine learning model. Shaded areas show the 95% confidence intervals. (b) Stability of CrRPs with Al addition calculated using first-principles methods. Lower energy indicates more stable structures.

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Research Group for High Temperature Science on Fuel Materials

We conduct research and development on advanced nuclear fuels, fuel debris characteristics, and fission product (FP) behavior during severe accidents (SAs) to support the sustainable development of nuclear energy. Our research involves both cold- and hot-material experiments, utilizing various high-temperature heating devices, analytical instruments, and property analysis tools. We also integrate computational modeling to complement our experimental findings, applying first-principles calculations, thermodynamic principles, and fuel performance codes, along with our databases on fuel properties.

Advanced Fuel Development

Toward the practical implementation of nitride fuels for minor actinide (MA) transmutation, irradiation tests are essential to verify transmutation performance and demonstrate fuel integrity. Accordingly, an irradiation test of MA-containing nitride fuels is being planned using the experimental fast reactor JOYO.

We have therefore begun fuel performance analysis using the FEMAXI code[1], which has been improved to handle nitride fuel analysis, to assess fuel irradiation conditions and specifications. The analysis was conducted for irradiation conditions consistent with the fuel compositions and irradiation period feasible in the JOYO Type-B capsule fuel assembly. The calculations successfully predicted key fuel behaviors during the feasible irradiation period, including changes in central temperature, pellet-cladding gap, and FP gas release (Fig. 1). Based on these results, we plan to examine the fuel specifications and post-irradiation experimental items necessary to gain insights into MA transmutation rates and key issues specific to nitride fuel, such as pellet-cladding mechanical interaction.

Furthermore, as part of the R&D toward the practical implementation of a nitride fuel cycle, we are advancing technological developments such as fuel pellet fabrication using the external gelation method and the enhancement of dissolution rates for pyroprocessing.

FP Behavior

Radioactive cesium (Cs) at the TEPCO Fukushima Daiichi Nuclear Power Station (1F) is a major source of radiation. Understanding its distribution and water solubility is crucial for rational safety assessments of fuel debris retrieval, decommissioning, and waste management.

To improve the accuracy of the FP chemistry database ECUME, we investigated the chemical behavior of Cs in the reactor and evaluated its transport under SA conditions. Internal surveys of 1F have revealed that highly radioactive deposits remain on the control rod drive rails within the primary containment vessel. To identify the cause, we studied the chemical reactions between Cs and the thermal insulation material.

By analyzing the reaction products between Cs and calcium silicate—the main component of the insulation—we identified for the first time that water-soluble Cs compounds exist in the form of Cs_2SiO_3. Thermodynamic equilibrium calculations further showed that this compound could significantly contribute to Cs retention in the insulation material[2].

These findings indicate that, through reactions with the thermal insulation material, Cs may form relatively water-insoluble cesium silicates such as $Cs_2Si_4O_9$ and $Cs_2Si_2O_5$, in addition to the water-soluble Cs_2SiO_3, and remain in the deposited material.

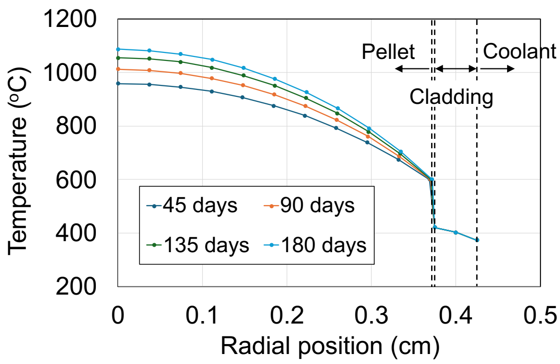

Fig. 1 Example of radial temperature distribution in nitride fuel

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Research Group for Environmental Science

Our group studies the dynamics of radioactive materials in atmospheric, terrestrial, and marine environments to improve technologies for assessing their environmental effects. We have developed and validated computer models to predict the dispersion of radioactive materials in the environment. Examples of the computer models developed include atmospheric dispersion prediction systems (WSPEEDI-DB etc.) and an oceanic dispersion prediction system (STEAMER). We also conduct field observations to collect environmental samples in terrestrial and marine environments. Our advanced analytical technologies enable us to elucidate the dynamics of radioactive materials and validate these computer models.

Discharge of ^{129}I in Riverbank Sediment after the Fukushima Accident

The Fukushima Daiichi Nuclear Power Station (1F) accident occurred in March 2011. As a result, the 1F accident released anthropogenic radioactive materials into the environment (e.g., 1.45 × 10^{16} Bq for ^{137}Cs [1], 8.06 × 10^{9} Bq for ^{129}I [2]). ^{137}Cs has a long half-life (30.1 years) and remains in the environment for extended periods. Therefore, many observational and numerical studies have focused on ^{137}Cs. In contrast, there has been relatively little research on ^{129}I. However, ^{129}I is useful for tracing the behavior of radioactive materials in the environment over long timescales because its half-life is about 15.7 million years. ^{129}I is also important for assessing the long-term radiation effects on marine ecosystems.

This study investigated the transport process of ^{129}I based on surveys conducted from 2013 to 2015 in the Tomioka River watershed (watershed area: 63 km^{2}) in Fukushima Prefecture[3]. The ^{129}I/^{137}Cs activity ratios in the surface soil were relatively low in the mountain areas and relatively high in the plains (Fig. 1). The ^{129}I/^{137}Cs activity ratios on the levee crown remained similar to those in the surrounding areas in 2011 until 2015. In contrast, the ^{129}I/^{137}Cs activity ratios in the surface riverbank sediment were low, suggesting that radioactive materials from the mountain areas were transported to the plains. The vertical distribution of the ^{129}I/^{137}Cs activity ratios in the riverbank sediment also showed that part of ^{129}I and ^{137}Cs remained in the lower layer but was mostly eroded shortly after the 1F accident.

As noted above, the ^{129}I/^{137}Cs activity ratios in the riverbank sediment remained constant until 2015 after the 1F accident. Therefore, based on the ^{129}I/^{137}Cs activity ratios and the previously estimated ^{137}Cs discharge amount[4], the ^{129}I discharge amount to the ocean was estimated in this study. The ^{129}I discharge from the study area to the ocean was estimated to be 1.8 × 10^{5} Bq. Moreover, for the wider river watersheds (Abukuma River and Fukushima coastal rivers, including the study river), it was estimated to be 1.2 × 10^{7} Bq. This amount corresponds to 0.3 % of the amount discharged shortly after the 1F accident (3.4 × 10^{9} Bq[2]). These results indicate that the long-term ^{129}I discharge from the river watersheds in Fukushima Prefecture would have little effect on the amount in the seabed sediment along the Fukushima coastal areas.

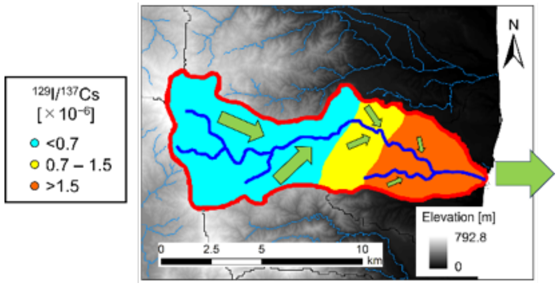

Fig. 1 Spatial distribution of the ^{129}I/^{137}Cs activity ratios in the surface soil in the Tomioka River watershed, Fukushima Prefecture. The arrows indicate the migration of radioactive materials.

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Research Group for Radiation Transport Analysis

Computer simulation is an essential tool for research and development in the field of nuclear and radiation sciences. The Particle and Heavy Ion Transport code System (PHITS) is a radiation transport simulation code developed by our group to meet various societal needs. This paper summarizes our progress in FY2024.

Upgrades of the PHITS Code

A new version of PHITS[1] (version 3.35, https://phits.jaea.go.jp) was developed and released to the public. The number of newly registered PHITS users in FY2024 was 1,515, including 692 Japanese users. The following are the major upgrades in the latest version.

Activation cross sections

The cross sections used to calculate activation by neutron-, proton-, deuteron-, and photon-induced nuclear reactions have been updated to those based on JENDL-5. With this update, the accuracy of induced activity calculations following long-term accelerator irradiation has been improved.

Variance reduction techniques

The variance reduction techniques implemented in PHITS have been enhanced to improve the efficiency of radiation shielding calculations. For example, a unique function was introduced into the weight-window generator of PHITS to guide particles toward regions of user interest[2]. The performance of this new function was evaluated through benchmark simulations for both idealized and practical shielding problems. The results indicated that the new algorithm could reduce computational time by up to an order of magnitude.

Indirect DNA damage

We investigated indirect DNA-damage effects by developing a dynamic Monte Carlo code for the chemical process. The reaction probabilities and the spatial distribution of lesions were theoretically derived as a function of the spur radius and the distance between DNA and the initial ionisation position. We suggested that a hydroxyl radical and a hydrated electron from a single spur can concomitantly react within 10 base pairs of DNA to induce a multiple DNA-damage site comprising a DNA single-strand break and reductive nucleobase damage; however, the reaction probability is 0.4% or less. Once this combination occurs, it may result in a DNA double-strand break (DSB). DSBs are difficult to repair, which may lead to cell death or misrepair, and can cause point mutations in the genome[3].

Graphical Interface of PHITS-Chem

The PHITS-Chem code[3], which allows the simulation of the dynamics of chemical products generated by water radiolysis, has been developed and included in the PHITS package. The results can be graphically illustrated using the PHIG-3D software (Fig. 1), contributing to an intuitive understanding of the temporal reactions of radicals and the subsequently induced DNA damage.

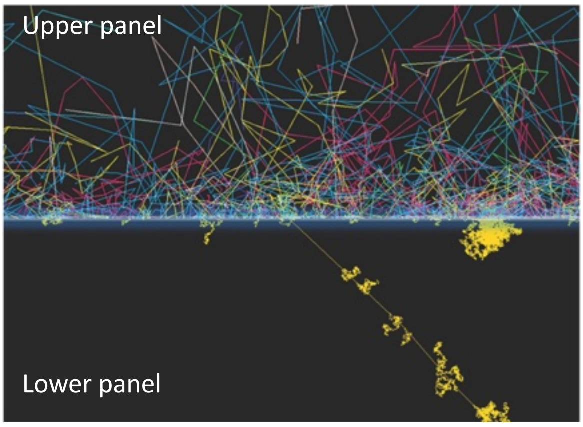

Fig. 1 10 MeV proton track and dynamics of radicals (OH radical, hydrated electron, hydrogen, hydrogen peroxide, etc.) at the DNA scale in the upper panel. The proton passes through the center from left to right. The yellow lines in the lower panel represent electron tracks.

References

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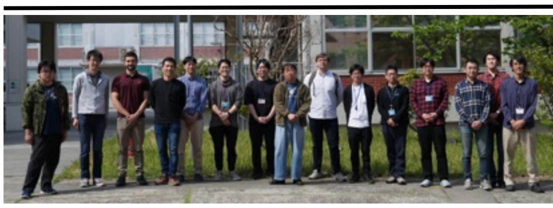

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Research Group for Nuclear Sensing

In FY2024, we achieved significant progress in both accurate temperature measurement technologies and the observation of high-energy natural radiation phenomena. We proposed a novel remote temperature-measurement method using radiation, which demonstrated higher sensitivity and accuracy than conventional techniques and showed potential for application to nuclear-fuel monitoring. In addition, long-term observations of thundercloud-related radiation events over the Tibetan Plateau revealed a periodic variation that may be linked to solar activity, providing valuable insights into high-energy atmospheric phenomena. These results represent steady advancement in both the practical application of radiation measurement and fundamental understanding. The following summarizes the main research achievements.

Development of Neutron Self-Indication Thermometry[1)]

Accurate temperature determination with small uncertainties is essential in neutron thermometry for both fundamental research and industrial applications. Conventional neutron resonance transmission thermometry (NRTT) estimates the sample temperature from small changes in neutron transmission at resonance energies induced by the Doppler-broadening effect. However, its sensitivity is inherently limited because the temperature-sensitive component is small compared with the primary neutron signal. To overcome this limitation, we developed a new method, neutron self-indication thermometry (NSIT), which combines the Doppler-broadening effect with a self-indication technique. In NSIT, both the sample and an indicator containing the same nuclide are irradiated, and prompt gamma rays are measured, effectively using the same resonance twice to enhance the temperature-sensitive component.

To verify the method, experiments were performed with tantalum samples, selected for their resonance energies, which are close to those of uranium and plutonium, over a temperature range of 23.0 °C to 492.6 °C. The results showed that NSIT achieved approximately 1.6–2.4 times higher temperature sensitivity and 3–9 % lower uncertainty compared with NRTT. These improvements indicate that NSIT can provide more accurate and reliable remote temperature measurements. This method has strong potential for widespread applications, including nuclear-fuel monitoring and other situations where direct temperature measurement is difficult.

Long-duration Bursts of High-Energy Radiation Observed on the Tibetan Plateau[2)]

High-energy radiation has been observed from thunderclouds at both ground level and high-mountain regions. While its origin is believed to lie in the electron acceleration in thunderclouds, the underlying mechanisms remain elusive. To help address this mystery, we analyzed long-term data from a cosmic-ray detector installed on the high-altitude region of the Tibetan Plateau. From 1998 to 2017, a neutron monitor located at 4.3 km above sea level in Yangbajing on the Tibetan Plateau, detected 127 long-duration bursts of high-energy radiation associated with thunderclouds. These bursts typically lasted between 10 and nearly 60 minutes, significantly longer than those observed in winter at the coastal area of the Sea of Japan. This indicates that continuous electron acceleration to >10 MeV can occur within thunderclouds. The bursts occurred more frequently at night (especially between 18:00 and 06:00) and showed clear diurnal and seasonal variations, consistent with regional lightning and precipitation activity. Notably, the annual burst frequency exhibited a distinct 16-year periodicity with an approximately 3-year phase lag relative to the 11-year solar cycle, suggesting a possible link between solar activity and the production of long-duration bursts. These findings demonstrate the importance of long-term observations for understanding the temporal behavior and production mechanisms of high-energy radiation emitted from thunderclouds. They also highlight the value of integrating such data with meteorological information to clarify the links between cosmic rays, solar activity, and particle acceleration in thunderclouds.

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Research Group for Nuclear Chemistry

Understanding the chemical behavior and speciation of radionuclides in solution is essential for addressing key challenges in the nuclear field, particularly in the back-end processes related to the safe treatment and disposal of nuclear waste. To advance these chemical investigations, computer simulations and analytical techniques play critical roles. In this report, we briefly introduce a new computational approach for the early-stage radiolysis in liquids and a new automated data-processing method for mapping solid samples by laser ablation–inductively coupled plasma–mass spectrometry (LA–ICP–MS).

Understanding Radiation Chemistry through Dielectric Response

Radiation in liquids creates highly reactive charged species, leading to complex chemical reactions. This study explores how the dielectric response—the time-dependent change in a liquid's ability to screen electric charges—affects these reactions[1]. Using advanced Monte Carlo simulations, we modeled how electrons and ions interact and recombine in various solvents (Fig. 1).

We found that if recombination occurs faster than the dielectric response, traditional models using static permittivity may fail to predict outcomes accurately. This insight is especially important for high-energy radiation or laser-induced processes, where reactions occur on ultrafast timescales. By incorporating dynamic dielectric behavior, our study provides a more precise understanding of radiation-induced chemistry, paving the way for improved predictions in fields such as nuclear science, medical radiation, and materials research.

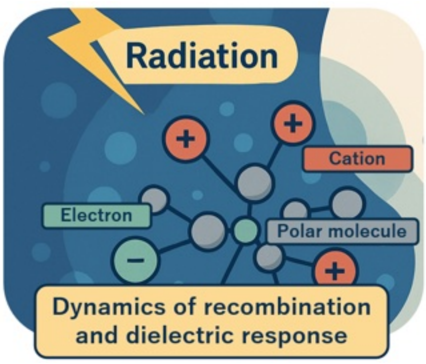

Fig. 1 Illustration of electron-cation recombination influenced by the dielectric response

Automated and Rapid Data Processing for Elemental Mapping by LA-ICP-MS

LA–ICP–MS is widely used for elemental mapping of solid samples in various fields. However, processing the large data volumes generated during laser scanning is time-consuming and labor-intensive. To address this issue, we developed an automated data processing program that uses penalized asymmetric least squares to detect peaks and integrate intensities. The system, built in Python with a user-friendly GUI, reads the raw data, identifies peaks, integrates the intensities, links them with laser position data, and visualizes the distribution of measured isotopes (Fig. 2).

We applied the program to process 547,200 data points obtained from elemental mapping of a mouse molar by online isotope-dilution LA–ICP–MS. The process was completed in ~30 seconds, compared with 20 hours for manual processing, and successfully visualized the distribution of strontium isotopes[2]. We also demonstrated the accuracy of quantification using the certified glass standard (SRM 612). The measured strontium concentrations were consistent with certified values. This approach significantly reduces processing time and manual effort and is applicable to other high-throughput analytical techniques.

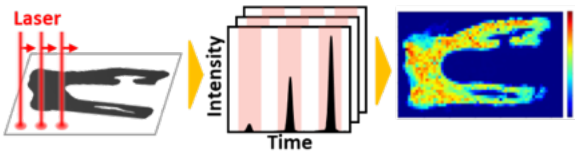

Fig. 2 Illustration of the automated data-processing workflow for LA–ICP–MS analysis

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Development Group for Nuclear Engineering Technology

Our group focuses on advancing chemical research on radioisotopes at the Nuclear Science and Engineering Research Center from a technological standpoint. We possess expertise in safely handling plutonium (Pu) and high-activity radioactive materials in hot cells and glove boxes. Using these capabilities, we are developing separation processes that employ novel extractants and methods to isolate minor actinides (MAs) and fission products.
We collaborate with the NXR Development Center to conduct experimental research aimed at recovering valuable elements from high-level liquid waste. We are also partnering with the Experimental Fast Reactor JOYO to develop separation and purification technology for actinium (Ac)-225, a promising isotope for medical applications. Furthermore, in collaboration with the Collaborative Laboratories for Advanced Decommissioning Science, we are analyzing actual debris from the Fukushima Daiichi Nuclear Power Plant. These efforts contribute to the advancement of fundamental separation and analysis technologies that support the utilization of nuclear energy.

Optimization of SELECT Process

Our research group has proposed a novel hydrometallurgical process called SELECT (Solvent Extraction from Liquid waste using Extractants of CHON-type for Transmutation) to recycle nuclear materials and separate MAs. Among the steps in the SELECT process, an optimized flow sheet was developed for recovering MAs and rare earths (REs) and for separating REs from MAs, using a simulation code called PARC-MA[1]. The extractants used in the former and the latter steps are TDdDGA and HONTA, respectively.
The flow sheet indicated that the weight ratio of MAs in the MA fraction >99%. In addition, the weight ratio of RE to MA distributed to the MA fraction (0.053), the weight ratio of MA to FP in the liquid waste (0.0013), and the volumetric ratio of the raffinate to high-level liquid waste (4.7) all met their respective target criteria: <0.1, <0.002, and <10 (Fig. 1).

Technology for Separating Am-241 in Aged Pu

Radioisotope thermoelectric generators using the decay heat of Pu-238 have been applied in deep-space missions beyond Jupiter, where solar power is limited. However, no facilities exist in Japan to produce Pu-238 for space probes. Moreover, the use of nuclear fuel materials for space exploration is restricted in terms of regulations. Thus, we focused on Am-241 as an alternative heat source.
We investigated the separation of Am-241 produced by the decay of Pu-241 in aged plutonium oxide[2]. Two experiments were conducted: solid-liquid extraction alone and a combined method utilizing both liquid-liquid extraction and solid-liquid extraction. Packed columns were employed in the separations, and the combined method reduced the number of required columns to less than one-fifth that of the solid-liquid extraction alone. Furthermore, the combined method was faster. Six separation experiments were performed, and a total of 0.43 g of Am-241 was collected as oxalate (Fig. 2).

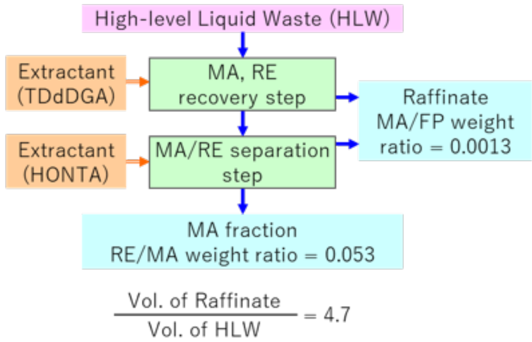

Fig. 1 Outline of the optimized flow sheet

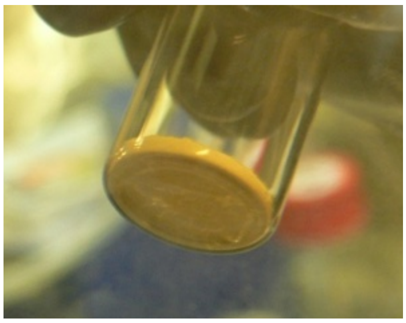

Fig. 2 Precipitation of americium oxalate

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