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P01

High Fiedility Simulations For Inovative Nuclear Energy Systems

P02

Site and Technology Assessment Co-optimisation for Nuclear Safety: Lessons from Kenya's Nuclear Power Programme.

Acholla, Emmanuel Mulehane [1]

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Kenya is implementing its Nuclear Power Programme [1] in line with the International Atomic Energy Agency (IAEA) Milestones Approach [2], spearheaded by the Nuclear Power and Energy Agency (N*u*PEA). Kenya is currently in Phase II of its nuclear power programme, with the key objective of undertaking project development activities, which entails finalization of key infrastructure, including site characterization and technology selection. Kenya has established national user requirements for both large reactors and SMRs [3], [4] and has conducted a preliminary reactor technology assessment (RTA) using the IAEA INPRO methodology [5]. The methodology applies defined exclusion criteria and key indicators addressing reactor design maturity, safety philosophy, unit size, siting flexibility, and deployment readiness. In parallel, Phase I site studies entailed comprehensive desktop and field data collection on natural and human-induced hazards, heat sink, electricity grid, population distribution, emergency planning, nuclear security and logistical infrastructure [6]. The data was used to derive site parameters from which were used as input in the RTA process. As a result, Kenya identified a list of candidate large NPP and SMR technologies. A detailed Request for Information (RFI) [7] was subsequently drafted and issued to the respective vendors to obtain further data on technology aspects, including site–facility interface parameters, to be used in the finalization of the RTA. During Phase II, detailed site characterization will be undertaken at the preferred NPP sites, leading to preparation of a Site Evaluation Report (SER) and definition of site parameter envelope and design bases. The SER will enable site licensing through site safety analysis, and be incorporated into the NPP Bid Invitation Specifications (BIS), enabling final reactor technology selection and licensing, through reactor safety analysis. This experience demonstrates that early and systematic integration of site studies with reactor technology assessment enables risk-informed decision-making through objective safety analyses, thus enhancing nuclear safety.

Key words: IAEA, N*u*PEA, safety, technology selection, RFI, site studies, safety margins

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P03

Integration of Nuclear Energy in Ghana's Energy Mix: Institutional Readiness and Capacity Building Efforts.

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Abstract

The integration of nuclear energy into Ghana’s energy mix offers a strategic way toward sustainable development providing reliable, low-carbon baseload electricity to complement renewable resources. This study examines the role of nuclear power, including emerging reactor technologies, in addressing Ghana’s growing energy demand, climate change mitigation commitments, and long-term energy security objectives. Key enabling factors such as national energy planning, regulatory and institutional readiness, human capacity development, and public stakeholder engagement are analyzed. The paper further evaluates economic and financing considerations, highlighting the importance of innovative investment models and cost-benefit assessments to ensure project viability. Reactor technology assessment is emphasized as a critical decision-making tool for selecting technologies that align with Ghana’s infrastructure, safety requirements, and grid characteristics. In addition, international cooperation and regulatory harmonization are identified as essential for leveraging global best practices and strengthening national nuclear governance.

P04

MULTI-FIDELITY NEURAL SURROGATE FOR RAPID K-EFFECTIVE PREDICTION IN SMALL MODULAR REACTOR (SMR) LATTICE CONFIGURATIONS

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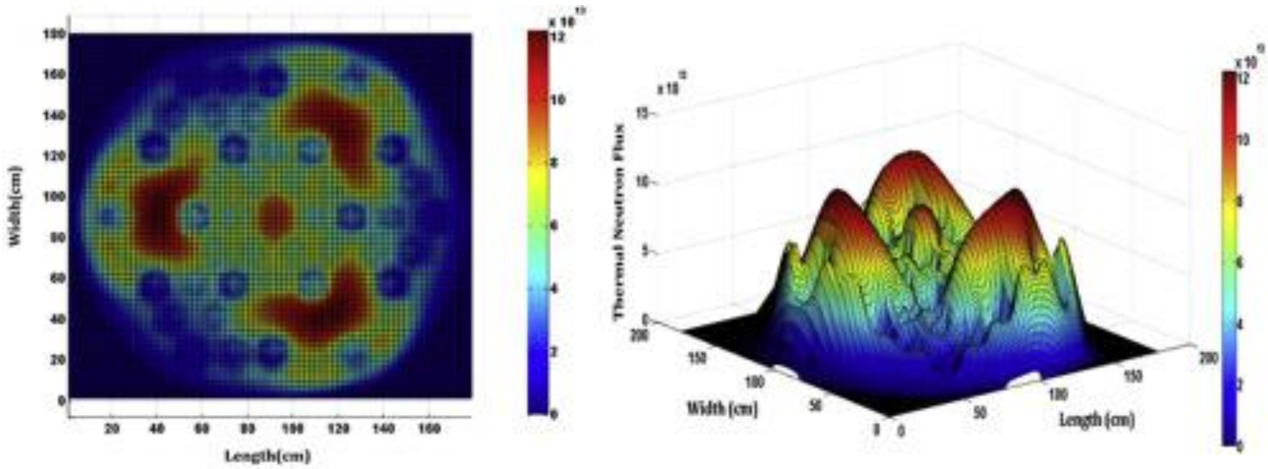
Accurate prediction of the neutron multiplication factor (k-effective) is fundamental to the design and safety analysis of Small Modular Reactors (SMRs). However, high-fidelity Monte Carlo transport simulations remain computationally expensive for large parametric studies. This work presents a multi-fidelity neural network framework for rapid and accurate k-effective prediction in SMR lattice geometries incorporating UO_2, ThO_2 mixed fuel with gadolinium as a burnable absorber and coolant water density. A low-fidelity dataset, generated using coarse OpenMC Monte Carlo simulations with reduced batch counts and particle populations, is integrated with a high-fidelity OpenMC reference dataset within a hierarchical multi-layer perceptron (MLP) architecture. The model employs a shared latent feature representation with fidelity-dependent correction layers to learn systematic discrepancies between low- and high-fidelity responses. The proposed approach demonstrates strong predictive performance, achieving an average high-fidelity test accuracy of approximately ~600 pcm RMSE, with cross-validation errors as low as ~100 pcm, with strong agreement between predicted and reference k-effective values across the parameter space. The remaining prediction error can be attributed to the limited feature space and the simplified formulation of low-fidelity approximations. Comparative evaluation against conventional machine learning models, including linear regression, random forests, and gradient boosting (XGBoost), shows that the multi-fidelity framework provides competitive and, in most cases, improved accuracy while maintaining a structured representation of fidelity relationships. This model shows effective utilization of low-cost approximate data, reducing dependence on computationally expensive high-fidelity simulations. This approach addresses a critical gap in SMR design workflows by enabling scalable and efficient neutronic surrogate modeling compatible with early-stage design optimization and rapid reactivity evaluation.

P05

Neutron Transport Simulation in Innovative Nuclear Reactor Systems



The photo shows a nuclear reactor core simulation or neutron flux distribution, where different colors represent neutron intensity or flux levels inside the reactor. It illustrates how neutrons move and distribute in the reactor core, which is important for reactor physics, safety analysis, and reactor design optimization.

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Neutron Transport Simulation in Innovative Nuclear Reactor Systems

Panji Dillsuz

Innovative nuclear energy systems play an important role in the development of sustainable, safe, and efficient energy production technologies. Advanced nuclear reactor designs require accurate analysis of neutron transport, reactor core behaviour, fuel composition, and safety parameters. Computational reactor physics and simulation tools are widely used to model and analyse these complex systems.

This study focuses on neutron transport simulation and reactor physics analysis in innovative nuclear reactor systems using Monte Carlo methods. The research investigates important reactor physics parameters such as neutron flux distribution, effective multiplication factor (k-effective), and the impact of fuel composition on reactor performance. Computational tools and Monte Carlo neutron transport methods are used to simulate reactor core behaviour and analyse system performance.

The objective of this work is to understand the role of computational modelling and simulation in the design, analysis, and optimisation of innovative nuclear reactors and advanced fuel cycle systems. Simulation results demonstrate the importance of modelling tools in reactor safety analysis, performance evaluation, and reactor design improvement.

This work is related to innovative reactor physics, advanced modelling, and multiphysics simulation tools, which are important research areas in innovative nuclear energy systems. The poster presents the simulation methodology, modelling approach, and analysis of neutron transport behaviour in innovative reactor systems.

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P06

Comparative Analysis of Reactivity Feedback in iPWR Transients: Steam Line Break and Rod Withdrawal

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Small modular reactors (SMRs), particularly integral pressurized water reactors (iPWRs), have gained increasing attention as next-generation nuclear energy systems due to their compact design and enhanced passive safety features. Understanding how these reactors respond to unplanned disturbances is essential for verifying their safety performances. In nuclear reactors, reactivity feedback mechanisms play a central role in self-regulating reactor behavior during transient events.
This study investigates two representative transient scenarios: Steam Line Break (SLB) and Control Rod Withdrawal. A Steam Line Break causes a rapid cooldown of the coolant, which can insert positive reactivity and drive the reactor power upward. Control Rod Withdrawal, on the other hand, directly reduces neutron absorption, also leading to a sudden increase in reactivity and power. Although both events result in a power excursion, the underlying physical mechanisms and the timeline of reactor response differ significantly between the two.

To examine these differences, an iPWR simulator is employed to track the time-dependent behavior of key parameters, including reactor power, fuel temperature, and coolant temperature. Through this analysis, the distinct roles of two major feedback coefficients are identified: the Moderator Temperature Coefficient (MTC), which reflects how changes in coolant temperature affect reactivity, and the Fuel Temperature Coefficient (FTC), also known as the Doppler coefficient, which captures the effect of fuel temperature on neutron absorption.

By comparing how MTC and FTC respond under each transient scenario, this study aims to clarify which feedback mechanism acts first, how quickly each respond, and how their combined effect ultimately stabilizes the reactor. The findings contribute to a deeper understanding of iPWR safety characteristics and provide physical insight into the self-limiting nature of reactivity feedback during abnormal transient conditions.

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P07

Helium Trapping in γ''-Ni_3Nb and NbC Precipitates in Ni-Based Alloys: A Density Functional Theory Study

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As advanced nuclear reactors operate under extreme conditions, understanding the microstructural evolution of structural materials is critical to ensuring reactor safety and long-term performance. Helium generated under irradiation can degrade the mechanical integrity of Ni-based alloys by promoting bubble formation and embrittlement [1]. In Ni-based systems, secondary phases or precipitates such as γ''-Ni_3Nb and Nb-containing carbides may act as trapping sites for helium and thereby influence its transport and accumulation behavior [2]. This study employs density functional theory (DFT) calculations using the Quantum ESPRESSO package to investigate helium trapping in γ''-Ni_3Nb and NbC precipitates, with fcc Ni used as a reference matrix. Helium solution energies and He-vacancy binding energies will be evaluated and compared to assess the relative trapping capability of each phase [3]. Particular attention is given to the role of local free volume in helium stabilization. The findings are expected to provide atomistic insight into helium behavior and to support the development of radiation-tolerant materials for advanced nuclear energy systems.

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P08

Genetic Algorithm-Based Optimization for Advanced Reactor Core Loading Patterns

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Abstract

This study presents the application of a Genetic Algorithm (GA) to optimize the loading pattern of the Advanced Power Reactor 1400 (APR1400). The arrangement of fuel assemblies in the reactor core plays a crucial role in determining both operational efficiency and safety margins. Traditional optimization methods often depend on heuristic techniques, which, while effective, may not consistently achieve globally optimal configurations. In contrast, the GA approach systematically explores a wide solution space, evolving candidate loading patterns over successive generations guided by defined fitness functions.

The optimization targeted multiple objectives: maximizing cycle length, minimizing the assembly power peaking factor, and ensuring compliance with operational constraints. The GA incorporated specialized operators to preserve core symmetry and fixed positions while maintaining feasibility. Each candidate pattern was evaluated using the MASTER core physics code, with assembly cross sections generated by CASMO-3, ensuring accurate assessment of reactivity, peaking factors, and burnup behavior.

Results indicate that the GA-based approach can identify loading patterns that extend cycle length and improve power distribution more efficiently than manual search methods. These findings highlight the potential of evolutionary algorithms to enhance fuel management strategies and support advanced nuclear reactor core design.

Keywords: APR1400, Genetic Algorithm, Fuel Management Net Graph, Reactor Core Optimization, Cycle Length, Power Peaking Factor

P09

APPLICATION OF MUON TOMOGRAPHY FOR NON-INVASIVE IMAGING IN NUCLEAR ENERGY SYSTEMS

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Abstract template for 5th Joint ICTP–IAEA Workshop on Physics and Technology of Innovative Nuclear Energy Systems — SMRs

Muon tomography is an emerging technique that uses naturally occurring cosmic-ray muons to probe the internal structure of dense objects non-invasively. In the context of nuclear energy systems, this method can be applied to inspect reactor components, monitor shielding integrity, and verify material distributions without interrupting operations.

In this work, we present computational simulations and experimental studies of muon flux attenuation through representative reactor materials. Using muon tracking and flux analysis, we reconstruct density maps and identify structural variations, demonstrating the feasibility of real-time monitoring.

This approach offers a complementary tool for enhancing safety, quality assurance, and non-destructive evaluation in innovative nuclear systems such as Small Modular Reactors. The results highlight the potential of integrating particle physics techniques into nuclear energy applications, contributing to safer and more reliable reactor designs.

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P10

Impact of Different Accident Tolerant Fuel Coatings on the Neutronic Behavior of the Nuscale Reactor Core

Abstract

The high design flexibility of small modular nuclear reactors, when coupled with accident-tolerant fuel concepts, significantly enhances the coping time during severe accident conditions associated with the loss of active cooling, while preserving fuel performance comparable to that of conventional systems under normal operating conditions. The NuScale reactor adopts a fully integrated design with fuel enrichment below 5%, a thermal power rating of 160 MW, and a 24-month operating cycle, ensuring reliable and inherently safe shutdown through multiple layers of redundant passive safety systems.

In this study, the NuScale reactor core is rigorously modeled using the MCNP Monte Carlo code to perform an in-depth steady state neutronic analysis. A comprehensive comparative assessment is conducted using U3Si2 fuel combined with various coating materials. Key core parameters are evaluated, including the effective neutron multiplication factor, boron concentration variation as a function of fuel burnup, reactivity coefficients (Doppler, moderator density, and temperature), radial and axial power distributions, and neutron energy spectra for each fuel–coating configuration. The results demonstrate that reactor operation with the investigated materials is neurotically stable, safe, and technically feasible, satisfying all evaluated safety-related neutronic criteria.

P11

Comparison of Generation IV Reactor Concepts for Small Modular Reactors and Their Potential Applications

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This study focuses on comparing different Generation IV reactor concepts commonly considered for Small Modular Reactor (SMR) deployment, including thermal-spectrum and fast-spectrum designs. The comparison is made from a technical perspective, highlighting the main characteristics, advantages, and challenges of each concept, such as safety features, fuel types, and operational flexibility.

The study also explores the potential applications of these SMR concepts as power plants in developing countries, considering factors such as scalability, resource availability, and suitability for diverse energy needs. By providing a general overview of advanced SMR technologies and their practical implications, this work aims to contribute to a broader understanding of innovative nuclear energy systems and support informed decisions in the development and deployment of SMRs globally.

P12

Application of Genetic Algorithms for Weibull Parameter Estimation in Nuclear Component Reliability Analysis

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Abstract:

The reliability of safety-critical structures, systems, and components (SSCs) in Nuclear Power Plants (NPPs) is a cornerstone of operational safety and security. Accurate modeling of component degradation is essential for robust Probabilistic Safety Assessment (PSA) and risk-informed decision-making. This study investigates the application of Genetic Algorithms (GAs) to optimize the estimation of Weibull distribution parameters shape and scale for reliability analysis.

To validate this heuristic approach, a controlled synthetic dataset (N=200) was generated based on established IAEA-TECDOC-1668 reference parameters for Alloy-600 MA steam-generator tubing. The performance of a custom Genetic Algorithm, implemented in FORTRAN to minimize the Chi-squared goodness-of-fit statistic, was compared against traditional Maximum Likelihood Estimation (MLE) and Rank Regression on Y (RRY) techniques.

The results demonstrate that the GA-based optimization achieved the highest accuracy among the tested methods. The GA yielded a score of 3.24, representing a 3.6% improvement over the MLE benchmark with a Chi-squared of 3.36 and a significant 22.1% improvement over Rank Regression with a Chi-squared of 4.16. The evolutionary approach successfully reconstructed the empirical failure density with high precision, avoiding the systematic residual deviations observed in classical regression methods. These findings confirm that Genetic Algorithms offer a robust, derivative-free alternative for modeling component wear-out, providing critical inputs for predictive maintenance and enhancing the overall safety culture of nuclear facilities.

P13

Information-Theoretic Frameworks for Strategic Planning of Sustainable Nuclear Energy Systems: Quantifying Decision Requirements and Robustness Under Uncertainty

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Long-term strategic planning for sustainable nuclear energy systems requires decision-makers to navigate deep uncertainty across technology development, economic parameters, and policy landscapes[1]. While the IAEA's INPRO methodology provides comprehensive sustainability assessment tools, current decision-support frameworks inadequately characterize what information is needed to make robust policy choices and how decision confidence responds to different uncertainty sources[1,2]. This work applies information-theoretic methods to develop a rigorous mathematical framework for quantifying information requirements in nuclear energy system planning. We propose using Shannon entropy and mutual information to: (1) measure decision robustness under uncertainty via entropy-based metrics;[3] (2) identify critical information gaps affecting long-term strategy formulation through mutual information analysis $I(X;Y) = H(X) - H(X|Y)$ [4] and (3) design optimal monitoring and data collection strategies using value-of-information frameworks and Bayesian decision analysis[5,6]. The methodology establishes theoretical foundations connecting information science with INPRO's multi-dimensional sustainability assessment[7], providing policy-makers quantitative tools to evaluate whether proposed data collection efforts justify their costs relative to decision improvements[8, 9]. We demonstrate the approach through a case study comparing SMR versus conventional large-reactor deployment strategies, quantifying strategy robustness across technology learning curves and computing the information value of early technology demonstration projects[3,10]. This work directly supports the INPRO School's mission to develop competencies for long-term strategic planning by providing decision-makers with mathematically rigorous, practically implementable tools for assessing information sufficiency and decision confidence, key questions facing IAEA Member States formulating national nuclear strategies.

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P14

Selective Adsorption of Actinides and Lanthanides over γ-Graphyne: A First-Principles Investigation for Nuclear Waste Separation

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The growing demand for clean and sustainable energy has significantly increased the reliance on nuclear power, which currently contributes a substantial share to global electricity production. However, the expansion of nuclear energy has led to the accumulation of radioactive nuclear waste, necessitating efficient strategies for its safe management and separation. The effective separation of actinides from lanthanides remains a major challenge in nuclear waste processing due to their similar chemical properties. In this study, the adsorption behavior of representative actinide and lanthanide ions, namely UO_2^{2+}, Pu^{3+}, and Eu^{3+}, on pristine and doped γ-graphyne has been systematically examined using density functional theory. The results show that pristine γ-graphyne exhibits relatively weak interactions with all considered ions, primarily driven by electrostatic forces. In contrast, introducing heteroatom dopants significantly enhances adsorption strength by creating localized electronic states and increasing surface reactivity. A comparative analysis of adsorption energies indicates a preferential binding affinity of actinide species over lanthanides on doped γ-graphyne. This increased selectivity is attributed to the partial covalent character arising from the involvement of actinide 5f orbitals in bonding, whereas Eu^{3+} interacts predominantly through ionic interactions due to the localized nature of 4f orbitals. Charge transfer and electronic density of states analyses further support the stronger coupling between actinide ions and the doped surface. Moreover, variations in charge density distribution and work function upon adsorption reveal distinct interaction mechanisms between the ions and the substrate. Overall, these findings demonstrate that electronic structure modification through doping plays a vital role in controlling adsorption behavior and selectivity in nuclear waste systems. This study highlights the promising potential of γ-graphyne as an advanced material for next-generation nuclear waste separation technologies.

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P15

Radiation-Resistant Beryllides as Advanced Neutron Reflectors for Research Reactors

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Beryllium is widely employed as a neutron reflector in research reactors because of its low neutron absorption cross section and favorable moderating properties [1]. However, its long-term application is constrained by radiation-induced swelling, embrittlement, and significant tritium retention [2,3]. At fast neutron fluences of the order of $\sim 1.8 \times 10^{22}$ cm^{-2} ($E > 1$ MeV), pure beryllium can exhibit volumetric swelling approaching 50%, mainly due to helium and tritium accumulation, leading to degradation of mechanical integrity and reduced service lifetime [4].
Intermetallic beryllides, particularly titanium beryllide ($Be_{12}Ti$) and chromium beryllide ($Be_{12}Cr$), have been proposed as promising alternatives because of their enhanced radiation tolerance, reduced gas-induced swelling, and improved structural stability [5,6]. An industrial fabrication route for $Be_{12}Ti$ and $Be_{12}Cr$ has been developed at the Ulba Metallurgical Plant, enabling production of bulk components suitable for reactor applications [6].
Irradiation experiments performed in the WWR-K research reactor demonstrated substantially lower swelling (~2.7%) and suppressed helium bubble formation in beryllides compared with metallic beryllium, confirming their superior dimensional stability under neutron exposure. To quantify the neutronic implications of replacing metallic beryllium with beryllides, Monte Carlo neutron transport simulations were carried out using the MCNP code. The calculations evaluated changes in neutron spectrum, reflector worth, and activation behavior for metallic Be, $Be_{12}Ti$, and $Be_{12}Cr$ reflector configurations.
Comparative activation analyses indicate a reduction in long-lived activation products in beryllides relative to pure beryllium, contributing to lower radiological burden and extended reflector service life. These results support the feasibility of beryllides as advanced neutron reflector materials for research reactors and irradiation facilities, and demonstrate their relevance for reflector optimization in innovative and next-generation nuclear energy systems.

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P16

Transient Neutronic–Thermal–Hydraulic Analysis of Innovative Reactor Cores Using the OpenNode Code

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Innovative nuclear energy systems, including Small Modular Reactors (SMRs), require reliable and computationally efficient simulation tools capable of capturing strong multiphysics coupling during transient operating conditions. In this work, we present the development and validation of a coupled neutronic–thermal–hydraulic framework implemented in the OpenNode reactor physics code, with a focus on transient reactor dynamics relevant to safety analysis [1].

OpenNode solves the time-dependent multi-group neutron diffusion equations using an advanced nodal expansion method, coupled with simplified thermal–hydraulic models for fuel heat conduction and coolant enthalpy evolution [2]. The coupling explicitly accounts for key reactivity feedback mechanisms, including Doppler effects, moderator temperature, coolant density variations, and control rod motion, which are essential for accurately modeling reactivity-initiated accidents [3]. Fully implicit numerical schemes are employed to ensure numerical stability during rapid reactivity insertion scenarios such as control rod ejection events.

The methodology is validated against well-established international benchmarks, including the NEACRP and LMW 3D LWR Core Transient Benchmarks, under both Hot Zero Power and Hot Full Power conditions [4, 5, 6]. Simulated scenarios include critical boron concentration search and control rod ejection transients, with detailed analysis of neutron flux evolution, power distribution, fuel temperature, and coolant thermal response. The results show good agreement with reference nodal and industrial reactor codes, with deviations generally within 2%, confirming the accuracy and robustness of the proposed coupling approach [1].

Although the validation is performed on light water reactor configurations, the developed framework is directly applicable to innovative and SMR reactor concepts, where compact cores, strong feedback effects, and transient behavior play a dominant role. OpenNode thus provides an accessible and efficient platform for reactor physics research, safety-oriented studies, and capacity-building activities aligned with ICTP–IAEA objectives.

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P17

Design and Development of a Vacuum Photodiode Detector for Plasma Diagnostics Applications: Advanced diagnostics for Nuclear Energy Generating Systems

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The precise measurement of plasma radiation in the vacuum ultraviolet and soft X-ray spectral range is critical in the studies of energy transport, impurities, and temperature profiles in magnetic confinement devices [1]. In this context, the development of a cost-effective diagnostic system based on the vacuum photodiode (VPD) technique has been carried out and implemented for the studies of VUV/SXR radiance in the ADITYA-U Tokamak (IPR, Gandhinagar) device. The VPD-based diagnostic system comprises an electropolished stainless steel photocathode and pinhole-based line-of-sight geometry [2-3]. The VPD diagnostic system was subjected to calibration in the range of 10-300 eV using synchrotron source at INDUS-I beamline BL-4 ,Raja Ramanna Centre for Advanced Technology, Indore. The precise determination of the absolute responsivity, quantum efficiency, linearity, and spectral sensitivity of the VPD diagnostic system was carried out. The initial experimental results obtained for ADITYA-U have confirmed the potential of the VPD system in capturing time-resolved emission spectra associated with plasma dynamics. The signal trend is in good agreement with other established diagnostics, namely the bolometer and soft x-ray (AXUV diode-based) detectors [4]. Moreover, the H_α and impurity line emissions trends, confirm the reliability and spectral selectivity of the system. The developed VPD system provides several advantages in terms of radiation hardness, immunity to magnetic fields, and spectral range. This makes it a reliable and cost-effective solution for plasma diagnostics. The next step is to extend the VPD system in its multichannel configuration.

The relevance of this work extends to the broader context of advanced nuclear energy systems. Diagnostic techniques developed for fusion plasmas contribute to radiation measurement, material response studies, and high-energy environment monitoring, which are also critical in the innovative fission systems and hybrid energy concepts. Thus, the developed VPD diagnostic represents a versatile and robust tool with applications spanning both in fusion and next-generation nuclear energy technologies.

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P18

Modelling the Behaviour of Non-Condensable gases in Reactor Containment & Coolant Systems: From Proprietary to Open-Source Platforms

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Non-condensable gases (NCGs) are important factors that lead to heat transfer degradation, condensation inhibition, and pressure build-up in the containment and coolant systems of nuclear reactor during normal and severe accident conditions. Hence the accurate modelling of the NCG behaviour is crucial for a dependable safety analysis, especially for advanced reactor concepts and small modular reactors (SMRs) whose containment designs might be radically different from that used in large reactors.

This poster presents ongoing research related to modelling the release, dissolution & transport of non-condensable gases with established system-level codes like TRACE, APROS, in addition to with high-fidelity CFD simulations in ANSYS FLUENT. Building upon these methods, the project considers open-source options for the NCG modelling like OPENFOAM which may provide increased flexibility and collaborative development.

A few benchmark test cases relevant to containment thermal-hydraulics are suggested and some of the issues associated with interfacial mass transfer as well as system codes & CFD coupling are discussed. The study illustrates the potential of open-source platforms to enable better validation workflows and accessibility, especially for academic research organizations that are resource-constrained to utilize commercial tools.

This work contributes to the advancement of modelling capabilities for innovative nuclear energy systems, supporting robust safety assessment and facilitating broader international collaboration in nuclear reactor and fuel-cycle research.

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Abstract template for “Modelling and dynamic simulation of the primary circuit of a Lead-cooled Fast Reactor”

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According to the Italian National Platform for Sustainable Nuclear (PNNS) [1], the Lead-cooled Fast Reactors (LFR) are among the most promising nuclear systems, since they aim to enhance efficiency, safety, proliferation resistance, economic competitiveness, and waste minimisation of nuclear power plants.

In modern energy systems, the integration of non-dispatchable renewable energy sources and nuclear power plants will be paramount: therefore, developing simulation methodologies and tools capable of replicating the steady-state behaviour and dynamic response of nuclear power plants is crucial.

This work presents the integration between the detailed 3D modelling of the fast core (performed with the Monte-Carlo software *OpenMC* [2]) and the detailed 1D modelling and dynamic simulation of the primary circuit (*Dymola*) of a small-sized LFR. A modular approach was adopted, in which the main components of the primary circuit (reactor core, pumps, steam generators, hot and cold pool sides) were modelled and interconnected to reproduce their physical coupling. Particular attention was devoted to coherently modelling the primary side of the peculiar spiral-tube steam generators in order to describe their thermal-hydraulic behaviour: this is a key aspect for predicting plant operational transients. To accurately determine the dynamic behaviour of the nuclear core, the main coefficients and parameters for the *Point-Kinetics Equations* were calculated in *OpenMC* and implemented in *Dymola*.

The simulations performed provide a foundation for more advanced analyses in the future, which will also incorporate the plant control strategies and the secondary circuit, as well as the interaction with the broader energy system in which the reactor will operate.

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P20

Machine-Learned Surrogate Modelling for Digital Twin Applications and Accelerated Monte Carlo Neutron Transport in Generation IV Reactors

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Digital twin architectures are increasingly being investigated for Generation IV nuclear energy systems to enable real-time reactor-state estimation, predictive monitoring, anomaly detection, and advanced operational support. However, high-fidelity digital twin realization remains computationally constrained by the intensive multiphysics calculations required for continuous reactor analysis and forecasting.

Continuous-energy Monte Carlo neutron transport methods provide highly accurate reactor physics simulations but remain prohibitively expensive for real-time deployment. In codes such as OpenMC, macroscopic cross-section evaluations constitute a dominant computational bottleneck, accounting for nearly 80% of total runtime through repeated microscopic cross-section searches, resonance-resolved interpolation across complex energy grids, and material-wise accumulation. XSBench, developed at Argonne National Laboratory, reproduces this kernel and serves as a representative benchmark for studying performance optimization strategies relevant to full Monte Carlo transport simulations.

This work presents a physics-informed, data-driven surrogate framework, validated against reference Monte Carlo solutions, for accelerated neutron transport calculations and digital twin applications in Generation IV reactor systems. The proposed approach replaces conventional microscopic cross-section lookup operations with a GPU-accelerated surrogate model trained on neutron energy, material composition, and reactor-state parameters. Particular attention is given to maintaining surrogate fidelity across the epithermal resonance region, where highly nonlinear cross-section structure and Doppler broadening effects present the most demanding accuracy requirements for data-driven approximations. By transforming irregular memory-bound lookup operations into dense neural inference while rigorously bounding prediction uncertainty, the framework aims to substantially reduce computational overhead without compromising physical correctness. The current implementation focuses on fuel pin-cell configurations and is being extended toward full-core reactor applications through incorporation of nuclide-wise material representations, energy-dependent spectral features, and thermal-hydraulic coupling parameters, with the Hoogenboom-Martin benchmark serving as the primary validation anchor. SFR applications introduce harder neutron spectra with more stringent energy-grid coverage requirements, while MSR online reprocessing necessitates robust surrogate tracking of continuously evolving fuel compositions, both representing active areas of framework development.

The framework is further developed to support near-real-time reactor-state reconstruction and short-timescale predictive forecasting across advanced reactor concepts including Sodium-cooled Fast Reactors (SFRs), High-Temperature Gas-cooled Reactors (HTGRs), and Molten Salt Reactors (MSRs). Subject to rigorous uncertainty quantification and validation of surrogate confidence bounds, the proposed approach enables anomaly-aware monitoring with characterizable detection reliability and early identification of abnormal neutronic or thermal behaviour, establishing a physics-grounded pathway toward scalable digital twin infrastructures for next-generation nuclear energy systems.

ULTRASOUND-ENHANCED DECONTAMINATION AS AN INNOVATIVE PRE-TREATMENT TECHNOLOGY FOR WASTE MINIMIZATION AND SUSTAINABLE RECOVERY OF MEDICAL RADIOISOTOPES WITHIN ADVANCED NUCLEAR FUEL CYCLES

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This contribution outlines a conceptual study on the feasibility of incorporating the UKNNL's 2025 low-frequency ultrasound decontamination technology into advanced nuclear fuel cycles, particularly in the pre-treatment of nuclear materials in advanced nuclear fuel cycles.

This study is centered on improving the overall extraction and processing of actinium-225, a significant alpha emitter in targeted alpha therapies for cancer treatment and research.

Current limitations in the extraction and processing of actinium-225 from existing Th-229 decay and IAEA-supported recycling of Ra-226 are attributed to contaminants on the surface, affecting purity and overall yields.

Ultrasound technology is a proven surface cleaning technology that improves feedstock quality, reduces processing times by over 90% compared to chemical processing, and eliminates secondary waste, thereby reducing the need for new uranium mining activities, in line with sustainable and circular economy principles.

Through a review of case studies from UKNNL, IAEA, and DOE, and a conceptual model, there is a strong potential for incorporating this technology in advanced nuclear fuel cycles and nuclear waste management strategies, bridging the gap between nuclear waste management and medical isotope applications for climate and health benefits.

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