Second Joint ICTP-IAEA Workshop on Open-Source Nuclear Codes for Reactor Analysis | (SMR 4103)

22 Sep 2025 - 26 Sep 2025
ICTP, Trieste, Italy

P01 - ACHARYA Kshitij Asitkumar

PHWR Fuel Performance Using OFFBEAT: Assessment and Future Directions

P02 - AFIFAH Maryam

Enhancing Core Design and Performance for SMR-CANDLE Reactor with UN Fuel Configuration

P03 - ASUKU Abdulsamad

Neutronics Analysis of Candidate Pin-Cell Configurations for Miniature Neutron Source Reactors using the OpenMC Monte Carlo Code

P04 - AZRAM Shafaq

Steady State Thermal Hydraulics Calculations for PWR core *The final content of abstract will be shared after approval from authority.

P05 - BRAHIM Pablo Ezequiel

REFUELING OPTIMIZATION IN NUCLEAR POWER PLANTS

P06 - FARAJPOUR Maryam

Dynamic Risk Analysis in Small Modular Reactor (SMR) Safety Using Open-Source Codes

P07 - FATIC Enida

High-Fidelity Transient Simulation of Reactivity-Initiated Accident Scenarios in SPERT III-E Reactor Using a Coupled Serpent-SUBCHANFLOW-TRANSURANUS Scheme

P08 - KEDZIERSKA Barbara

Heat transfer simulations of a molten salt reactor with static elements

P09 - KHANSAN Dashnyam

PRELIMINARY NEUTRONIC ANALYSIS OF RESEARCH REACTORS USING VVR-KN AND IRT-4M FUEL

P10 - MOUMNI Sarra

Advanced Modeling and Simulation of Neutronics for the Westinghouse AP300 SMR Using OpenMC.

P11 - MUNOT Samyak Sanjay

SUK-M Micro Modular Reactor: A Conceptual Design for Viksit Bharat 2047

P12 - NDLALAMBI Panashe Lynn

IAEA ONCORE Initiative

P13 - NGUYEN-PHUONG Thao Vy

Enviro-Economic Analysis of Refrigeration Cycle Integration into Ground-Source Heat Pump-Supported Space Heating Systems

P14 - OKTAVIANTO Putra

Simulation of Reactor Core Material Melting in Nuclear Reactor Accident Using MPS (Moving Particle Semi-Implicit) Method

P15 - RAY Dipanjan

Development and Analysis of an OpenMC-Based Model for the Small Modular Heat-Only Reactor TEPLATOR

P16 - SANCHEZ MORA Heriberto

TLANESY: code for the Simulation of Thermalhydrualic systems in Nuclear Reactors

P17 - SHUDDHO Safius Sakib

Neutronic Evaluation of Reflector Materials for the ALFRED Core

P18 - SILVEIRA CASTRILLO Lazara

Simulation of the cooling channel blockage of the IEA-R1 research reactor using the RELAP5

P19 - SOUZA E SILVA Artur

An OpenMC framework for automating the modeling and simulation of small modular light water reactors

P20 - TAIBI Oumaima

Neutronic Feasibility Study of Dual-Cooled Annular Duplex PuO 2 -ThO 2 Fuel for Next-Generation PW-SMRs: Assembly Level Analysis

P21 - THANGARAJ Muthuraj

Stability Mapping of Sodium-Cooled Fast Reactors Using Lyapunov Function and Particle Swarm Optimization

P22 - WADHWA Kavya

FluxGAN: An Open-Source Surrogate Model for Fast, High-Fidelity Neutronics Calculations Using OpenMC with Future Multiphysics Integration.

P01

PHWR Fuel Performance Using OFFBEAT: Assessment and Future Directions

Kshitij Asit Acharya [1,2] , **Dr. Anurag Gupta** [1,2]

[1] *Reactor Physics Design Division, Bhabha Atomic Research Center, Mumbai*
[2] *Homi Bhabha National Institute, Mumbai*

The safe and reliable operation of Pressurised Heavy Water Reactors (PHWRs) requires accurate simulation of fuel performance under normal and off-normal conditions. Fuel performance codes support this goal by providing predictions that assist in fuel design optimization, safety assessment, and failure prevention.

OFFBEAT (OpenFOAM Fuel Behavior Analysis Tool)[1] is an open-source, multi-physics fuel performance code developed primarily for Pressurised Light Water Reactors (PWRs). It employs finite-volume discretization in a flexible OpenFOAM environment to simulate the thermo-mechanical behavior of nuclear fuel. However, key differences between PWR and PHWR fuel systems necessitate adaptation and validation of such tools before reliable application to PHWRs.

PHWR fuel differs significantly from PWR fuel in geometry, operating conditions, and irradiation behavior. Notably:

- PHWR fuel bundles consist of short, thin collapsible elements (e.g., 37-element bundles), unlike long fuel rods in PWRs.
- PHWRs operate with on-power refuelling, leading to frequent power ramps and associated phenomena.
- Lower pressure and temperature conditions alter cladding oxidation, creep, and gap conductance behavior.
- Contact between fuel bundle and pressure tube introduces unique fretting wear mechanisms absent in PWRs.

The objective is to study OFFBEAT to simulate PHWR fuel in order to reinforce its potential as a versatile tool when extended for PHWR-specific physics. Additionally, OFFBEAT's open architecture enables coupling with other physics solvers. A possible integration pathway with other indigenous as well international codes is proposed to advance toward high-fidelity multi-physics simulations for PHWRs.

Conclusion:
This work underscores the value of adapting open-source OFFBEAT to address the specific fuel behavior of nuclear reactors. It also aligns with the ONCORE initiative's goals of expanding access to validated, open-source simulation tools for advanced nuclear analysis in developing countries.

[1] Alessandro Scolaro, Ivor Clifford, Carlo Fiorina, and Andreas Pautz., Nucl. Sci. Eng. 358:110416 (2020)

P02

Enhancing Core Design and Performance for SMR-CANDLE Reactor with UN Fuel Configuration

Maryam Afifah [1], **Zaki Su'ud** [1]

[1]*Department of Nuclear Science and Engineering, Faculty of Mathematics and Natural Sciences, Bandung Institute of Technology, 40132, Indonesia*

CANDLE reactor concept had been widely introduced with several innovation idea to improve neutron economy performance for lead cooled core application. This study considers the Uranium Nitride fuel potential for Small Modular Reactor (SMR) applications in CANDLE core design. A multi core size with various fuel enrichment composisition was compared to analysis the minimum requirements to ensure the reactor remain critical until reached the equilibrium condition. The preliminary result shows that for small size (400 MWth) design, the criticality can be maintain with more than 60 years of burnup without refuelling. The core design has minimum of 4 layers assembly layers with 240 fuel pins per assembly and can achieved reactivity below 2%.

P03

Neutronics Analysis of Candidate Pin-Cell Configurations for Miniature Neutron Source Reactors using the OpenMC Monte Carlo Code

A. Asuku [1,2], **and S. Dalhat** [2]

[1] *(Centre for Energy Research and Training, Ahmadu Bello University, Zaria-Nigeria)*
[2] *Department of Physics, Ahmadu Bello University, Zaria-Nigeria*

Analysis of several candidate pin-cell configurations in specific reactor environments is crucial in finding suitable alternatives to traditional fuels. In this study, six candidate fuels Uranium Oxide (UO_2), Uranium Nitride (UN), Uranium Carbide (UC), Uranium dicarbide (UC_2), Thorium Oxide (ThO_2) and Mixed oxide (MOX), two moderator types light water (H_2O) and heavy water (D_2O), and two clad materials Zircalloy-4 (Zr-4) and Stainless Steel (SS304) were permutated resulting in varying configurations to analyse their neutronic performance in a typical Miniature Neutron Source Reactor (MNSR) environment. The UO_2–Zr_4–H_2O fuel-clad-moderator configuration being the current configuration of Low-Enriched-Uranium-fuelled MNSRs[1] was used as a benchmark model. It was found that the configurations having UO_2, UN, UC, UC_2 as fuel matched the effective multiplication factor of the benchmark model only when the Outer Diameter (OD) of the fuel meat was increased (by reducing the clad thickness) while keeping both the OD of the clad and moderator constant. The results in this work were validated using another Monte Carlo code SCALE (Standard Computational Analysis for Licensing Evaluation). Depletion calculations were also performed to estimate and compare the burnup of ^{235}U and production of ^{239}Pu for the pin-cell configurations.

[1] J. Simon, Y.V. Ibrahim, D.J. Adeyemo, N.N. Garba, A. Asuku. Progress in Nuclear Energy. **141,** 103970 (2021).

P04

Steady State Thermal Hydraulics Calculations for PWR Core

Thermal hydraulics analysis is normally performed to determine the coolant flow parameters and to ensure that there is an adequate heat transfer between the fuel cladding and reactor coolant.

Steady state thermal hydraulics calculations (audit calculations) of PWR core are performed using RTHP computer code. RTHP is a reactor core thermal hydraulics steady state calculation code with advanced single channel model. Trends of fuel centerline temperature, cladding outer surface temperature, and minimum DNBR are analyzed for different BOL and EOL scenarios. Core pressure drop and total pressure drop across vessel are also calculated. All the results are found in good agreement with the acceptance criteria.

[1] Safety Analysis Report

[2] Reactor Thermal hydraulic design report

P05

REFUELING OPTIMIZATION IN NUCLEAR POWER PLANTS

P. Brahim[(a)*]

(a) Department of Core Physics, Nuclear Safety Analysis and Core Design Management, Nucleoeléctrica Argentina S.A, Francisco Narciso de Laprida 3175, Buenos Aires, Argentina.

ABSTRACT

This work presents a methodology for the optimization of the refueling strategy in the PHWR-type core of the Atucha Unit I Nuclear Power Plant. The definition of this strategy is an essential part of core design due to the large impact refueling actions has on the reactor parameters. Its formulation is, in essence, a combinatorial optimization problem that has traditionally relied on the judgment and experience of specialized teams. The remarkable increase in computational capacity has encouraged the adoption of heuristic methods to tackle it; among these, simulated annealing is particularly attractive because of its simplicity and robustness: its systematic character reduces dependence on individual judgment and its random component enables exploration of a solution space larger than that typically considered by expert criteria, helping obtaining optimized strategies that may be non-intuitive yet satisfactory.

A methodology based on time-averaged core parameters was implemented; it showed very good overall performance across the different energy functions and neighbor-state generators considered. The approach proved capable of both improving refueling strategies and reconditioning them under changes in the operational control bank for cores with 252 and 241 fuel assemblies, while keeping computation times reasonable. Thus, the developed methodology constitutes a relatively simple and efficient tool for defining optimized refueling strategies.

In addition, a generator of instantaneous burnup distributions was implemented from the time-averaged calculation results, with the aim of synthesizing an initial state for detailed refueling simulation. This methodology was applied to an optimized strategy, yielding acceptable results within short timeframes. Finally, to complete the design process of an optimized strategy, a refueling simulator representative of the reactor's actual operation was developed to verify the long-term feasibility of the strategy. The simulator evaluation, performed with the optimized strategy and the generated initial state, showed adequate performance and allowed validation of the refueling strategy in a relatively short time.

[*] Corresponding author at: Department of Nuclear Physics, Nuclear Safety Analysis and Core Design Management, Nucleoeléctrica Argentina S.A, Francisco Narciso de Laprida 3175, Buenos Aires, Argentina.
E-mail address: pbrahim@na-sa.com.ar (P. Brahim)

P06

Dynamic Risk Analysis in Small Modular Reactor (SMR) Safety Using Open-Source Codes

maryam Farajpour1 ,

Ph.D. Student in Nuclear Engineering, Shiraz University, Shiraz, Iran

Dynamic risk analysis is an advanced approach that integrates time-dependent modeling and probabilistic risk assessment using open-source reactor analysis codes to accurately evaluate complex and evolving safety scenarios in Small Modular Reactors (SMRs). This study employs dynamic simulations and open-source tools to analyze risk factors and their impacts over time, enabling a more realistic and comprehensive assessment of the safety of this new generation of reactors. The results contribute to enhancing risk-informed decision-making and the development of improved safety management strategies in the design and operation of SMRs.

[

P07

High-Fidelity Transient Simulation of Reactivity-Initiated Accident Scenarios in SPERT III-E Reactor Using a Coupled Serpent-SUBCHANFLOW-TRANSURANUS Scheme

Enida Fatic [1,2] **, Heikki Suikkanen** [1]

[1] *Lappeenranta-Lahti University of Technology*
[2] *University of Ljubljana, Faculty of Mathematics and Physics*

Small Modular Reactors (SMRs) represent a promising pathway for the future of low-carbon energy systems, offering enhanced safety features, economic scalability, and operational flexibility. One of the key objectives of the EURATOM-funded McSAFER project is to demonstrate the viability of high-fidelity, multiphysics simulation tools—combining Monte Carlo neutronics, subchannel thermal-hydraulics, and fuel performance modeling—for advanced SMR safety analysis.

In this context, the Monte Carlo code Serpent 2 (developed by VTT, Finland) has been coupled with the subchannel code SUBCHANFLOW (KIT, Germany) and the fuel performance code TRANSURANUS (European Commission, JRC) to simulate Reactivity-Initiated Accident (RIA) scenarios. The benchmark selected for this work is the Special Power Excursion Reactor Test (SPERT) III-E, a low-power, pressurized water research reactor fueled with UO_2, which operated in the 1960s at the National Reactor Testing Station in Idaho, USA. The SPERT III-E program provides a rich set of experimental data from RIA tests performed under conditions representative of commercial light-water reactors, aimed at evaluating prompt reactivity feedback effects and power excursion dynamics.

The poster will present progress in the coupling implementation of the Serpent 2–SUBCHANFLOW–TRANSURANUS simulation chain developed at LUT University, along with a validation methodology against experimental SPERT III-E data, which represents a key step to assess this proposed high-fidelity simulation scheme.

LUT Group Confidential - Other information (3Y)

P08

Heat transfer simulations of a molten salt reactor with static elements

Barbara Kędzierska, **Andrei Rineiski, Xue-Nong Chen, Yoshiharu Tobita**

Karlsruhe Institute of Technology

Molten-salt reactors (MSRs) are currently intensively studied because of their promising capabilities of burning and breeding fuel to close the cycle and reduce nuclear waste production. Many designs target large cores which might be characterized by a very irregular flow pattern and reactivity oscillations.

An alternative option to eliminate these problems is a reactor built from smaller cylindrical structures filled with molten salt [1]. These elements are placed in a large pool filled with an inert salt and discharge the heat generated inside the cylinders to the inert salt which is pumped upwards. In this situation, the heat transfer from the inner cylinder side is caused by the natural convection, which it is dominated by the forced convection from outside. This behaviour was investigated for different dimensions of the cylinder and different pitches [2]. The simulations of this reactor design were performed in the ANSYS Fluent environment to obtain a correction factor for the heat transfer correlation. Fluid movement inside the cylinders is observed due to the natural circulation. Fluid heats up due to the internal heating and density decrease pushes it up, where it is moved towards the wall. Colder outer fluid absorbs the energy of the hotter fuel which moves down along the cylinder wall. The correlation modified by a factor was introduced into a severe accident tool to perform safety simulations with coupled thermohydraulic-neutronics effects.

[1] B. Kędzierska, A. Rineiski, Y. Tobita and X.-N. Chen, Proc. of GLOBAL 2024, Tokyo, Japan.

[2] B. Kędzierska, A. Rineiski, Y. Tobita and X.-N. Chen, Progress in Nuc. Sci. and Tech. **8** (accepted).

P09

PRELIMINARY NEUTRONIC ANALYSIS OF RESEARCH REACTORS USING VVR-KN AND IRT-4M FUEL

Dashnyam Khansan[1], Tsendsuren Amarjargal[2]

[1,2] *Nuclear Research Center, National University of Mongolia, Ulaanbaatar, Mongolia*

This study aims to provide an accessible reference for early researchers by performing a simple neutronic analysis of research reactor cores. A comparative neutronic analysis of two research reactors using low-enriched uranium (LEU) fuels—VVR-KN and IRT-4M. Using the Monte Carlo-based SERPENT-2.1.30 code [1] and the ENDF/B-VII.1 nuclear data library [2], preliminary simulations were carried out for simplified core models of the VVR-K and LVR-15 reactors [3-4]. Both cores utilize UO_2-Al dispersion fuel enriched to 19.75 wt% ^{235}U. Key parameters such as the effective neutron multiplication factor (k-eff) and axial thermal neutron flux distribution were calculated. For the VVR-K reactor, a k-eff of 1.0822 ± 0.0008 and maximum thermal neutron flux of 3.61343×10^{13} n/cm²·s were obtained. The calculated k-eff value is consistent with the reported startup of the VVR-K reactor, which has a k-eff of 1.0836 ± 0.0002 at the beginning of the cycle [5]. Meanwhile, the LVR-15 reactor, using IRT-4M fuel, showed a k-eff of 1.1247 ± 0.0003 and a thermal neutron flux of 4.38744×10^{13} n/cm²·s. This analysis demonstrates that both reactors are capable of sustained critical operation and highlights the significant impact of core design and fuel configuration on their neutronic behavior.

[1] J. Leppänen et al., *Ann. Nucl. Energy* **82**, 142 (2015).
[2] M. B. Chadwick et al., *Nucl. Data Sheets* **112**, 2887 (2011).
[3] A. Dambrosio, M. Ruščák, G. Mazzini, A. Musa, *Ann. Nucl. Energy* **117**, 145 (2018).
[4] N. A. Hanan, P. L. Garner, *Neutronic, Steady-State, and Transient Analyses for the Kazakhstan VVR-K Reactor with LEU Fuel: ANL Independent Verification Results*, 1 p. (2015).
[5] F. Arinkin *et al.*, *Feasibility Study of the WWR-K Reactor Conversion to Low-Enriched Fuel*, 1 p. (2004).

P10

Advanced Modeling and Simulation of Neutronics for the Westinghouse AP300 SMR Using OpenMC

S. Moumni[a,b], W. Dridi[a]

[a] Research Laboratory for Energy and Matter for the Development of Nuclear Sciences (LR16CNSTN02),
National Center for Nuclear Science and Technology, 2020 Sidi Thabet, Tunisia
[b] Faculty of Sciences of Tunis (FST), 2092 Tunis El Manar, Tunisia

This study presents an analysis of a light water Small Modular Reactor (SMR) design featuring a square-shaped fuel element. The core design is based on the Westinghouse UO_2 SMR, which operates with uranium oxide fuel enriched to less than 5%. To accurately simulate and analyze the reactor's performance, the study employed the open-source Monte Carlo code `OpenMC`, selected for its capability to model complex reactor physics with high precision.

The primary focus was on neutronics analyses of the core containing UO_2 fuel. Key reactor parameters were evaluated, including the effective multiplication factor (k_{eff}), which provides insight into the reactor's ability to sustain a nuclear chain reaction under various conditions. The radial neutron flux profile was also analyzed to understand neutron distribution across the core, ensuring efficient and safe reactor operation.

Additionally, the study investigated the maximum-to-average flux ratio, a crucial parameter for assessing thermal performance. A high ratio may indicate potential hotspots that must be minimized for safe operation. Reactivity coefficients, including temperature and coolant density coefficients, were characterized to evaluate the reactor's response to operational changes such as variations in temperature and coolant flow.

The simulations showed that the SMR design performs well, with results within expected ranges for a reliable reactor. These were validated by comparison with calculations from the widely used `MCNP` code developed at Missouri S&T, confirming the accuracy of `OpenMC`.

This work demonstrates the effectiveness of open-source tools like `OpenMC` for reactor physics analysis and supports the development of SMR technology to enhance the safety, efficiency, and scalability of nuclear energy.

References

1. B. Dsouza, *Neutronic analysis of light water Small Modular Reactor with flexible fuel configurations*, Missouri University of Science and Technology, 2015.

2. F. Reza, M. H. Sahadath, Y. Akter, *Modeling and neutronic analysis of pin-cell comprising nuclear fuel with different chemical composition and neutron moderator using Monte Carlo code OpenMC*, Ann. Nucl. Energy **151**, 107946 (2021).

3. P. Darnowski et al., *Simulations of the AP1000-based reactor core with SERPENT computer code*, Arch. Mech. Eng. **65**(3), (2018).

4. M. A. Elsawi, A. S. Bin Hraiz, *Benchmarking of the WIMS9/PARCS/TRACE code system for neutronic calculations of the Westinghouse AP1000™ reactor*, Nucl. Eng. Des. **293**, 249–257 (2015).

P11

SUK-M Micro Modular Reactor: A Conceptual Design for Viksit Bharat 2047

Samyak S. Munot[1] and Nitendra Singh[1]

[1] *(IYNS TechSolutions LLP, Pune, Maharashtra, India – 411038)*

India's vision of Viksit Bharat 2047 includes a commitment to 100 GWe of nuclear capacity by mid-century, significantly strengthening energy security and clean-energy goals[1]. Achieving these targets requires low-carbon baseload power to complement rapidly expanding renewables. The SUK-M project proposes a 10 MWe molten-salt microreactor fuelled by thorium as India's first indigenous micro-reactor, contributing to the country's sustainable energy transition and to meet the climate-change mitigation objectives[2]. SUK-M, Swayamchalit Utkrantik Kendrak – Micro, is the abbreviation of Sanskrit words meaning Autonomous Micro Nuclear Reactor in English.

The SUK-M concept is a fluoride-salt fast-spectrum reactor (~30 MWth generating 10 MWe). It uses a proprietary mix of smart uranium and thorium fuel to achieve a long (~15-year) core life[3]. Operating at ~750–800 °C yields high thermal efficiency. Passive heat pipes (alkali-metal thermosyphons) convey fission heat from the core to power-conversion systems without pumps or valves, following the heat-pipe-cooled reactor paradigm[4]. The entire core module is sealed and factory-assembled; the compact unit is sized for standard container transport to enable rapid deployment using rail/road.

SUK-M's molten-salt core runs at near-atmospheric pressure, greatly reducing mechanical stress, and its passive systems "eliminate severe accidents by design". Inherent safety features include drainable freeze-plug systems and strong negative temperature feedback; The low fissile inventory means transients are self-limiting. After ~10–15 years the sealed fuel-and-salt module is removed as a single cartridge, minimizing on-site refuelling and maintenance[2].

SUK-M is promoted as India's first fully indigenous microreactor. Microreactors of this class are intended for remote or distributed applications (e.g. industrial parks, islands, transportation hubs, strategic applications). SUK-M's ~30 MWth output enables cogeneration: in addition to 10 MWe of electricity it can supply process steam, clean hydrogen, or desalinated water[5]. Critically, all components are to be designed and manufactured domestically, aligning with the Make-in-India initiative. By displacing fossil energy, SUK-M's deployment helps India meet its climate and long-term net-zero commitments[2].

References

[1] "Homi Bhabha's pledge stands vindicated by PM Narendra Modi with the launch of 'Nuclear mission' to meet India's increasing requirements through environment friendly clean energy;" Accessed: Jul. 08, 2025. [Online]. Available: https://www.pib.gov.in/www.pib.gov.in/Pressreleaseshare.aspx?PRID=2115857

[2] SUK-M, "SUK-M," SUK-M. Accessed: Jul. 07, 2025. [Online]. Available: https://iynstechsolutions.in/suk-m

[3] "Project Suk-M: Youth Initiative to Develop Microreactor for Energy Transition in India," Nuclear Business Platform. Accessed: Jul. 07, 2025. [Online]. Available: https://www.nuclearbusiness-platform.com/videos/v/project-suk-m

[4] E. Wang, T. Ren, and L. Li, "Review of reactor conceptual design and thermal hydraulic characteristics for heat pipe in nuclear systems," *Front. Energy Res.*, vol. 11, Dec. 2023, doi: 10.3389/fenrg.2023.1264168.

[5] "PHDCCI in association with IYNS, organised the National Technology Day Symposium on Leveraging Nuclear Energy in India - PHD Chamber." Accessed: Jul. 07, 2025. [Online]. Available: https://www.phdcci.in/2024/05/13/phdcci-in-association-with-iyns-organised-the-national-technology-day-symposium-on-leveraging-nuclear-energy-in-india/

IAEA ONCORE Initiative

Panashe Lynn Ndlalambi [1],
Kriventsev, Vladimir [2], **and Nicole Virgili** [2]

The Open-source Nuclear Codes for Reactor Analysis (ONCORE) initiative is an IAEA-facilitated international collaboration framework for the development and application of open-source multi-physics simulation tools to support research, education and training for the analysis of advanced nuclear power reactors. Institutions and individuals participating in ONCORE can collaborate in, and benefit from, the development of open-source software in the field of nuclear science and technology.

An international network of research and academic institutions is creating a common platform in advanced reactor experiments and high-fidelity multi-physics nuclear simulation techniques for open-source code development and validation. The work focuses on three major areas: modelling and simulations, experimental reactor physics and education and training.

The platform is particularly useful to institutions and individual users in nuclear ‘newcomer’ and developing countries as it offers access to knowledge and tools that otherwise may not be easily available. It also facilitates collaboration with recognized experts in the field.

Specific objectives of the ONCORE initiative

1. Build and preserve knowledge in the field of open-source simulation codes and open-access data and facilitate the exchange of information within the nuclear science and technology community.
2. Define best practices for collaborative open-source code development.
3. Assess features, gaps and opportunities for integration of already developed open-source modules and codes.
4. Facilitate sharing of reference solutions, standard benchmark problems and input data for specific applications.
5. Promote the individual tools and platform in education and research environments; and
6. Organize education and training activities.

Catalogues of available codes, data and related training material will be continuously updated to provide an overview of available resources to interested users and developers. These catalogues will help to identify gaps and, in turn, establish development priorities, avoid duplications and stimulate synergies and collaborations.

Benefits of ONCORE

ONCORE promotes collaboration and sharing of resources, materials and tools for research and education. ONCORE members actively contribute to the development of new software, receive community support for the use of available software and participate in the organization of training events and outreach activities.

How to participate in ONCORE

The ONCORE initiative makes use of several tools, including a public Git repository and a public SharePoint [1] site with a members' area to facilitate exchange of information among experts.

In case you wish to contribute to existing open-source codes in terms development, testing and applications, please contact the code developers listed on the SharePoint page. If you have an open-source code you wish to contribute to ONCORE and have it added to the list of available open-access codes, please contact the IAEA [2].

[1] INTERNATIONAL ATOMIC ENERGY AGENCY, Open-Source Nuclear Codes for Reactor Analysis, Available at: https://nucleus.iaea.org/sites/oncore/SitePages/Home.aspx.
[2] oncore@iaea.org

P13

Enviro-Economic Analysis of Refrigeration Cycle Integration into Ground-Source Heat Pump-Supported Space Heating Systems

Elinor Lewis[1], Thao Vy Nguyen-Phuong[1]

[1] *Imperial College London*

This paper examines the integration of waste heat from the refrigeration cycle of a Sainsbury's store into the building heating, ventilation, and air conditioning (HVAC) system to investigate the potential of electricity consumption and carbon footprint reductions. Case studies were proposed with different configurational integration concepts: (0) no integration of the waste heat; (1) indirect integration into the ground; (2A) direct integration via a heat exchanger into the primary HVAC loop after the ground-source heat pump (GSHP); (2B) direct integration via a heat exchanger into the primary HVAC loop before the GSHP. The results then were compared to existing Sainsbury's store performances based on available historical data. All cases considered reduced both the cost required to provide space heating, and the carbon dioxide emissions produced, compared to the base case. The most beneficial case was the indirect integration, case 1, whereby the refrigeration waste heat is directed into the ground near the supermarket, then extracted by the GSHP with an increased theoretical coefficient of performance (COP). This case however, due to the requirement of a GHSP, will be difficult to retrofit to existing stores. The primary integration cases 2A and 2B, do not suffer this drawback, and can readily be implemented into existing stores, including those operating with a gas boiler, providing a reduction in cost and emissions of the space heating systems. Our analysis also emphasises the value of government incentives to make renewable energy solutions and waste heat integration economically competitive to traditional technologies.

P14

Simulation of Reactor Core Material Melting in Nuclear Reactor Accident Using MPS (Moving Particle Semi-Implicit) Method

P. Oktavianto [1,2] **, A. Saputra** [1] **, and A. Pramutadi** [2]

[1] *Research Center for Nuclear Fuel Cycle and Radioactive Waste Technology – BRIN K.S.T. B.J. Habibie Building No. 720, Serpong, South Tangerang, Banten,Indonesia*

[2] *Department of Physic, Institute Technology of Bandung, Jl. Ganesha No. 10, Bandung, Indonesia*

Reactor core material safety throughout operation, shutdown, and accident scenarios is an important consideration that must be maintained. Research on nuclear reactor severe accidents is required in order to understand the accident mechanism that takes place and to develop an appropriate safety and mitigation system in the event of a nuclear reactor accident. Numerous experiments into nuclear reactor accident have been carried out using a range of examples, including the CORA, QUENCH, PHEBUS, LEISAN, LIVE, Ogura, VESTA, and FROMA experiments. A number of computer programs, including MAAP, MELCOR, ASTEC, ICARE/CATHARE, SCDAP/RELAP5, and MPS, have also been validated using some of these experiments. Research on the melting of reactor core material, the first step in the phenomena of nuclear accidents, is still scarce, nevertheless. In order to prepare suitable safety and mitigation measures and avoid unforeseen impacts, this early stage is actually crucial. In this thesis, I thus used the moving particle semi-implicit (MPS) approach to investigate the melting of reactor core material. One software that may be used to model a nuclear accident phenomena is MPS, which is particularly relevant to the melting movement of reactor core material. In this experiment, two samples that is wax and woodmetal were melted in hot liquid medium, which included water and cooking oil. The melting properties are examined by using the heated samples as analogues of reactor core material that would melt in the case of a catastrophic nuclear catastrophe. Because water has a greater thermal conductivity than cooking oil, both experimental observations and MPS simulations indicate that the sample melts more quickly in water medium. According to the study's findings, the MPS approach can accurately simulate the melting of reactor core material.

[1] A. Pramutadi, et al., Annals of Nuclear Energy. **81**, 26 (2015).
[2] H. Anni, et al, Nucl Sci Tech. **32**, 8 (2021).
[3] S. Koshizuka and Y. Oka, Nuclear Science and Engineering. **123**, 421 (1996).
[4] S. Koshizuka and Y. Oka, Comput. Fluid Dyn. Journal. **9**, 366 (2001).
[5] A. Jasmin Sudha, et al, Prog Nucl Energy. **105** (2017).
[6] T. Kawahara and Y. Oka, J. Nucl. Sci. Technol. **49**, 1156 (2012).
[7] Y.U. Kuznetsova, J. Colloids ans Surfaces A: Physiccochem. Eng. Aspects. **520** (2017).
[8] J.R. Lamars and A.J. Barrata, Introduction to Nuclear Engineering, 3rd Ed., Prentice Hall, New Jersey (2001).

P15

Development and Analysis of an OpenMC-Based Model for the Small Modular Heat-Only Reactor TEPLATOR

Abstract:

To address the growing global demand for sustainable and clean energy, a novel Small Modular Reactor (SMR) concept known as TEPLATOR has been developed by a joint team of researchers from the Czech Technical University Prague and University of West Bohemia Pilsen. This concept involves a pressure channel type reactor system that utilize heavy water as both the coolant and moderator. A key feature of this design is its compatibility with irradiated or slightly enriched VVER-440 fuel assemblies. This reactor allows operation under relatively low pressure and low temperature conditions when compared to the conventional nuclear power plants. TEPLATOR primarily aimed at addressing the energy demands of district heating networks and industrial processes, which are currently heavily reliant on fossil fuel-based plants. This leads to a reduction in pollution and other significant environmental impacts, thus addressing the global requirement for clean energy solutions. This approach contributes to lowering pollution and mitigating broader environmental consequences, thereby supporting the worldwide shift toward cleaner and more sustainable energy sources.

In current research work, a three-dimensional reactor core region of TEPLATOR have been developed using OpenMC, an open-source Monte Carlo code for particle transport. Burn-up calculations have been performed and key parameters like control rod worth, excess reactivity, power distribution have been investigated. Calculated results have been compared with the Serpent model. For both the cases, ENDF/B-VIII.0 nuclear data library has been utilized. Results from the OpenMC model exhibit good agreement with the calculated results from Serpent model.

P16

Abstract template for … Poster …

TLANESY: code for the Simulation of Thermalhydrualic systems in Nuclear Reactors

H. Sánchez-Mora[1] , **A. M. Gómez-Torres** [2]

[1] *Department de Ingenieria de Procesos e Hidaulica, Universidad Autonoma Metropolitana-Iztapalapa, Ciudad de Mexico 09310, Mexico*

[2] *Instituto Nacional de Investigaciones Nucleares, Carretera México –Toluca, La Marquesa s/n, Ocoyoacac, Estado de México 52750, México*

Currently, the simulation of nuclear systems requires the use of specialized codes in both the thermohydraulic and neutron fields. In Mexico, the AZTLAN platform has seen considerable development in the neutron field, with the release of codes such as AZKIND, AZTRAN, and AZNHEX [1]. However, thermohydraulic simulation has gained relevance in the last year with the development of a systems code called TLANESY (ThermaLhydrAulic and heat traNsfEr SYstem), capable of two-phase flow simulation, taking into account the three heat transfer mechanisms: conduction, convection, and radiation. This code under development has been validated with different QUENCH [2] experiments and experiments focused on two-phase flow. The objective of TLANESY is to be a reliable, fast-simulating, and open-source code. This first effort has allowed simulations to be performed with acceptable results and with the aim of developing a greater number of simulation capabilities in the future.

[1] http://www.aztlanplatform.mx/ (July, 2025)
[2] https://quench.forschung.kit.edu/ (July, 2025)

P17

Neutronic Evaluation of Reflector Materials for the ALFRED Core

Safius S. Shuddho [1] **, Aqueeb A. Sunny** [1] **, and Abdus S. Mollah**

[1] *(Presenting author underlined) Military Institute of Science and Technology*

This study evaluates the neutronic performance of four reflector materials: Yttria-Stabilized Zirconia (YSZ, baseline), Beryllium Oxide (BeO), Graphite, and Magnesium Oxide (MgO), in the ALFRED reactor core, a demonstrator for the European Lead-cooled Fast Reactor (ELFR), using the open-source Monte Carlo code OpenMC with ENDF/B-VIII.0 nuclear data. Effective multiplication factor k_{eff} and neutron leakage were analyzed. BeO exhibits the highest k_{eff} (1.11189) and lowest neutron leakage (18.51%), demonstrating superior neutron reflection and economy. MgO and Graphite follow closely, while YSZ shows lower k_{eff} (1.07884) and higher leakage (20.18%). The k_{eff} for YSZ aligns closely with literature [1], reporting 1.07767 with an error of 0.108%, validating the simulation model. Performance differences are attributed to neutron cross-section characteristics: BeO, Graphite, and MgO have low (n, γ) capture cross-sections and enhanced (n, 2n) reaction rates, particularly BeO, preserving neutron economy. Burnup analysis of the core using all reflector materials was done which revealed the highest cycle length of 1232 days for BeO, followed by 1152 days for MgO, and 1142 days for Graphite, where the k_{eff} was dropped gradually to 1.0. The lowest cycle length yield was for YSZ of 1055 days. Neutron flux spectra comparisons reveal softer spectra for BeO, Graphite, and MgO, consistent with increased moderation, while YSZ shows intermediate spectral hardness. These findings for BeO align with a previous study [2], which identified BeO as optimal for its softest reflector spectrum, longest core lifetime, and flattest power distribution using RMC simulations. BeO is a promising reflector candidate for ALFRED, potentially improving neutron economy and extending fuel cycle length compared to the baseline YSZ.

[1] Lodi, F., Grasso, G., Cammi, A., Lorenzi, S., Petrovich, C., Mattioli, D. & Sumini, M., 2015. Characterization of the new ALFRED core configuration. Report number: ADPFISS-LP2-085 Rev. 0.
[2] Ji, W. H., Peng, X. J., Cai, Y., Zhou, B. Y., Wang, L. J., & Zhang, B. (2020). Research of reflector optimum design of a lead cooled fast reactor. Nuclear Power Engineering, 41(S2), 16–20. https://doi.org/10.13832/j.jnpe.2020.S2.0016.

P18

Simulation of the cooling channel blockage of the IEA-R1 research reactor using the RELAP5

Oliveira C. F. Eduardo [1] , Castrillo S. Lázara [1,2]

[1] *Universidade Estadual de Pernambuco – UPE-Brazil*

[1,2] *Universidade Estadual do Ceará – UECE-Brazil*

RELAP5 is a highly generic computational code used to simulate the behavior of the cooling system of nuclear reactors and a wide variety of hydraulic and thermonuclear transient regimes, involving mixtures of steam, water, non-condensable gases, and solutes[1]. The MOD 3.4 version, employed in this work, is intended for applications in research reactors, while the MOD 4.0 version is designed for computational development of fuel behavior and specific severe accident cases, such as low-pressure scenarios. The MOD 3.4 version, which will be used in this work, is intended for applications in research reactors . The MOD 4.0 version is available for the computational development of fuel behavior and very specific severe accident cases, such as low-pressure scenarios. Examples of code applications in research reactors can be found in the previously selected articles [2,3].The objective of this study is to use the RELAP5 computational code to analyze a possible severe accident scenario in the core of the Brazilian research reactor IEA-R1[4,5,6]. The simulated accident consists of partial and total blockage at the inlet of the cooling channels of the reactor's hottest fuel element. The transients of the radial and axial temperature and power profiles inside this channel will be analyzed to predict the main safety parameters during the incident, based on the nodalization developed for the model. This initial knowledge of the thermal distribution allows monitoring and preventing the onset of the undesired boiling crisis in the channel, thereby avoiding the occurrence of a severe accident. Furthermore, the presented nodalization can be improved or modified to serve as a basis for modeling more complex and severe accidents. The simulations considered partial and total blockage of the hottest cooling channel, with and without scram (rapid reactor shutdown), using RELAP5 components such as valves, bypasses, pipes, volumes, and time-dependent junctions. Details of the nodalization used will be shown, as well as the dynamics of the main thermo-hydraulic variables throughout the transient, which lasted 250 seconds with scram and 135 seconds without scram.

[1] RELAP5/MOD3 Code Manual, NUREG/CR-5535, (1994).
[2] T. Hamidouche, et al., Annals of Nuclear Energy, 31 (2004).
[3] M. Adorni et al., Annals of Nuclear Energy, 32 (2005).
[4] P.E. Umbehaum, Proceeding of Encit-ABCM (2004).
[5] A.Z. Mesquita. et. al.. Journal of ASTM International, 6,(2011).
[6] P. E. Umbehaum, IPEN (2000).
[7] D. M. Hirata. Aps nível 1. (2009).

P19

An OpenMC framework for automating the modeling and simulation of small modular light water reactors

Artur S. Silva[1], **Júlia de B. B. de Pontes**[1], **Fernando C. da Silva**[1], **Aquilino S. Martinez**[1], **Willian V. de Abreu**[1], **Alessandro da C. Gonçalves**[1] **and Adilson C. da Silva**[1]

[1]*(Presenting author underlined) Department of Nuclear Engineering, COPPE, UFRJ, Rio de Janeiro, RJ, Brazil*

The research and development of small modular reactors is gaining momentum around the world given their potential as a reliable and flexible source of clean energy and their ability to address the problems of traditional nuclear power generation [1]. Light water small modular reactors are particularly suited for near-term deployment due to their technology readiness and operational experience inherited from large nuclear power plants [2].

OpenMC [3] is an open-source neutron and photon Monte Carlo transport code initially developed by MIT's Computational Reactor Physics Group and currently developed collaboratively by researchers all around the world through Github. OpenMC is written in C++ and has a Python application programming interface (API). Therefore, it is possible to write a Python script to automate the creation of input files and employ the Python ecosystem of scientific computing libraries to analyze the simulation results.

This paper describes a framework that leverages OpenMC's Python API to automate the modeling and coupled neutronic-thermal-hydraulic simulation of small modular light water reactors, as well as the analysis of the simulation results. The entire process of defining materials, building the geometry, specifying tallies and adjusting simulation settings has been automated. The user work reduces to filling a file named `inputs.py` with reactor-specific data.

A steady-state subchannel model for the calculation of assembly averaged temperatures has been implemented in Python and coupled with the neutron transport simulation through the `openmc.lib` module. The subchannel model uses the tallied linear power distribution to calculate coolant, clad, gap and fuel temperatures in addition to coolant density for each axial node and each assembly. The model temperatures and densities are updated after each inactive generation and used to adjust the nuclear cross-sections for the next iteration. This coupling scheme allows the user to arbitrarily choose the initial temperatures, since they will converge to values that are consistent with the reactor power profile.

The user has the option to process the simulation output files using a script named `data_analysis.py`, which will automatically generate assembly-wise and pin-wise radial power maps, axial power profiles, radial and axial temperature distributions and evaluate important safety parameters such as the power peak factor, the minimum departure from nucleate boiling ratio and the fuel and coolant maximum temperatures.

The framework described here can help accelerate the research and development in reactor physics by lowering the entry barrier for OpenMC, especially for non-Python users, and by providing a fully open-source code system that automates all tasks related to the high-fidelity modeling and simulation of small modular light water reactors.

[1] Lauren Kiser and Luis Daniel Otero, "Multi-criteria decision model for selection of nuclear power plant type", Progress in Nuclear Energy, **159**, 104647 (2023)

[2] IAEA. Nuclear Reactor Technology Assessment for Near Term Deployment. IAEA Nuclear Energy Series, No. NR-T-1.10, Rev. 1.

[3] Paul K. Romano, Nicholas E. Horelik, Bryan R. Herman, Adam G. Nelson, Benoit Forget and Kord Smith, "OpenMC: A State-of-the-Art Monte Carlo Code for Research and Development", Annals of Nuclear Energy, **82**, 90–97 (2015).

P20

Neutronic Feasibility Study of Dual-Cooled Annular Duplex PuO_2-ThO_2 Fuel for Next-Generation PW-SMRs: Assembly Level Analysis

O. Taibi[1], O. Kabach[1], E. M. Chakir[1], M. Gouighri[1], S. Uzun[2], H. Amsil[3]

[1] *Laboratory of Materials and Subatomic Physics, Faculty of Sciences, Ibn Tofail University, Kenitra, Morocco.*

[2] *Faculty of Engineering and Architecture, Erzincan Binali Yıldırım University, Erzincan, Türkiye.*

[3] *National Center for Energy, Sciences and Nuclear Technique, Rabat, Morocco*

As we face a shortage of uranium and a growing demand for energy, the nuclear industry is stepping up with thorium-based fuels and innovative reactor designs. Nuclear power is a low-carbon and dependable energy source, but traditional fuel cycles have their challenges, especially regarding waste management and proliferation resistance. Small Modular Reactors (SMRs), like NuScale and AP300,are designed to improve safety and scalability, paving the way for next-generation fuels, including dual-cooled annular duplex designs that enhance thermal efficiency and extend burnup [1, 2].

This research investigates the viability of employing dual-cooled annular duplex fuel (ThO_2–PuO_2) [3, 4] in PW-SMR assemblies, utilizing neutronic analysis through DRAGON and OpenMC codes [5, 6].

The research aimed to optimize the inner moderator radius while maintaining the fuel volume, identifying specific geometric configurations that enhance initial reactivity. The burnup analysis showed that PuO_2–ThO_2 fuels have a smoother decline in reactivity, which allows for extended cycles. While they may start with lower reactivity, the gradual decrease actually improves fuel usage. Safety evaluations confirmed negative temperature coefficients, emphasizing the potential of PuO_2–ThO_2 fuel to improve the sustainability and economic aspects of advanced nuclear systems,Fig(1).

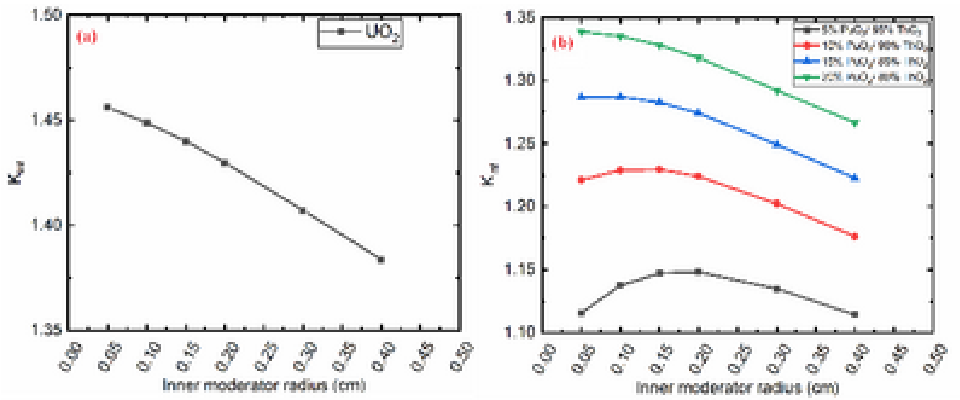

Figure 1: Infinite multiplication factor as a function of inner moderator radius: (a) UO_2 case, (b) PuO_2–ThO_2 cases.

[1] M.D. Carelli, D.T. Ingersoll, Woodhead Publ. Ser. Energy 64, *Handbook of Small Modular Nuclear Reactors*, Woodhead Publishing (2015).

[2] J. Maiorino, F. D'Auria, G. Stefani, R. Akbari, in: *Innov. Des. Technol. Nucl. Power* (2018).

[3] O. El Kheiri, O. Kabach, A. Chetaine, S. Abdelmajid, *Ann. Univ. Craiova, Phys.* **33**, 171 (2023).

[4] O. El Kheiri, O. Kabach, A. Chetaine, *Prog. Nucl. Energy* **160**, 104680 (2023).

[5] N. Shaukat, A. Ali, A. Ahmad, O. Kabach, K. Al-Athel, A. Shams, *Prog. Nucl. Energy* **178**, 105507 (2025).

[6] O. Kabach, E. Mahjoub Chakir, H. Amsil, *Nucl. Eng. Des.* **421**, 113086 (2024).

P21

Stability Mapping of Sodium-Cooled Fast Reactors Using Lyapunov Function and Particle Swarm Optimization

Muthuraj T[1], **John Arul**[2], **and Aditya Bhandari**[2]

[1]*Indian Institute of Technology Madras, Chennai, India*
[2]*Indira Gandhi Centre for Atomic Research, Kalpakkam, India*

Sodium void reactivity feedback is a critical safety parameter in sodium-cooled fast reactors (SFRs), particularly in medium- and large-scale cores where the void coefficient can become positive. Sodium voids may form and propagate due to global transients such as Unprotected Loss of Flow (ULOF) and Unprotected Transient Overpower (UTOP), or local events such as fission gas release, flow blockages, and subassembly boiling. Positive void reactivity can trigger nonlinear power excursions, self-sustained oscillations, or even chaotic reactor behavior, underscoring the need for rigorous stability assessment under voiding scenarios.

Accurate modeling of void feedback requires solving the coupled space-time neutron diffusion and thermal-hydraulic field equations, which is computationally expensive and unsuitable for routine safety evaluations or online monitoring. To reduce complexity, point kinetics models coupled with lumped-parameter thermal-hydraulics are commonly employed. While computationally efficient, these reduced-order models remain nonlinear and can exhibit bifurcation phenomena. Recent studies have shown that persistent voiding can lead to limit cycles, quasiperiodic oscillations, or chaotic dynamics, further motivating the need for nonlinear stability analysis to define safe operating margins[1, 2].

In this work, the Lyapunov function method is applied to analyze the nonlinear stability of a sodium-cooled fast reactor (SFR) core under sodium void propagation[3]. A suitable Lyapunov function is systematically constructed for the coupled neutronic–thermal-hydraulic model, and its level sets are employed to estimate the Domain of Attraction (DOA)—the set of admissible perturbations in neutron density, fuel temperature, and coolant temperature for which system trajectories return to a stable equilibrium. To avoid overly conservative stability estimates, Particle Swarm Optimization (PSO) is used to maximize the size of the invariant set, yielding a more realistic and practically relevant stability margin [4].

The proposed approach provides a computationally efficient framework for quantifying safe operating regions of SFR cores beyond which power oscillations or divergent transients may occur. The results, including DOA maps and optimized Lyapunov surfaces, will be presented and critically analyzed for their relevance to the design of next-generation micro modular and medium-scale SFRs. The insights gained are expected to support the development of robust design guidelines that enhance passive safety and mitigate risks of uncontrolled transients during abnormal operating events.

[1] Ali Reza Armiyoon and Christine Q Wu. A novel method to identify boundaries of basins of attraction in a dynamical system using lyapunov exponents and monte carlo techniques. Nonlinear Dynamics, 79:275–293, 2015.

[2] Sudhansu Sekhar Singh. Nonlinear stability analysis of sodium cooled fast reactors. Nuclear Engineering and Design, 424:113212, 2024.

[3] Steven H Strogatz. Nonlinear dynamics and chaos: with applications to physics, biology, chemistry, and engineering. Chapman and Hall/CRC, 2024.

[4] Jose March-Leuba, Dan G Cacuci, and Rafael B Perez. Nonlinear dynamics and stability of boiling water reactors: part 1—qualitative analysis. Nuclear Science and Engineering, 93(2):111–123, 1986.

P22

FluxGAN: An Open-Source Surrogate Model for Fast, High-Fidelity Neutronics Calculations Using OpenMC with Future Multiphysics Integration

Kavya Wadhwa

Department of Physics, School of Energy Technology
Pandit Deendayal Energy University Gandhinagar, Gujarat, India

Neutronics calculations are fundamental to nuclear reactor analysis, influencing core design, fuel utilization strategies, safety margins, and reactor economics. Monte Carlo (MC) transport codes like OpenMC offer high-fidelity, geometry-flexible simulations for such tasks. However, their high computational cost makes them impractical for large-scale parametric studies, multi-objective optimization, or real-time analysis in digital reactor twins.

To address this, we present FluxGAN, a machine learning–based surrogate model developed using a Generative Adversarial Network (GAN) architecture. FluxGAN is trained on OpenMC-generated data, learning to predict neutron flux and burnup conditioned on fuel enrichment. It achieves a prediction accuracy of 99.68%, while reducing computation time by over 2700×—generating thousands of outputs in under 2 seconds, compared to several hours using OpenMC.

The model is implemented entirely with open-source tools (OpenMC and PyTorch), reinforcing the ONCORE initiative's goal of democratizing access to high-performance nuclear simulation. While the current implementation is limited to a single-cell, one-group neutronics model, the framework is fully scalable to multi-group, multi-pin, and assembly-level configurations. Importantly, ongoing work aims to incorporate physics-informed constraints into the FluxGAN architecture by embedding domain-specific relationships and governing equations directly into the loss function and network structure. Future extensions will include Multiphysics coupling, such as linking flux predictions with temperature feedback, moderator density changes, and thermal-hydraulic variables through surrogate-assisted workflows.

This research supports ONCORE's broader objective of building accessible, modular, and scalable open-source tools for nuclear reactor modelling. By merging the strengths of Monte Carlo physics fidelity with deep learning acceleration, FluxGAN presents a promising pathway toward real-time, high-accuracy simulation tools for research, education, and deployment in emerging nuclear programs worldwide.