COLORADO SCHOOL OF MINES

M-PRISM

RESEARCH TOPICS

Research topics in advanced modeling and simulation of nuclear systems

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PURPOSE

This document lists the research topics of Dr. Valerio Mascolino and the M-PRISM group (Mines Physics of Reactors: Integrated Simulation and Multi-physics) at Colorado School of Mines. Each topic gives its motivation, its objective, and its tasks. Use it to find a project that fits your interests or to see who works on what.

MAIN POINTS

- Seventeen topics in five areas. Fission matrix methods: Topics 1, 2, 13, and 14. Monitoring and safeguards: Topics 3, 10, and 11. Open models and tools for research and advanced reactors: Topics 4, 5, 6, and 8. Multiphysics and molten salt reactors: Topics 7, 12, 15, 16, and 17. Space reactors: Topic 9.
- Each topic lists the student levels it suits and the main skills it uses.
- A task that is under way names the student who works on it and the term they started. "Planned" marks a student who will start the task soon. A task without a name is open.
- Topics and tasks reflect current, potential, and future work at the time of writing. They will change as the projects mature.

HOW TO GET INVOLVED

Email valerio.mascolino@mines.edu with the topic and task numbers that interest you. The group website (vmascolino.com) lists current projects, papers, and news.

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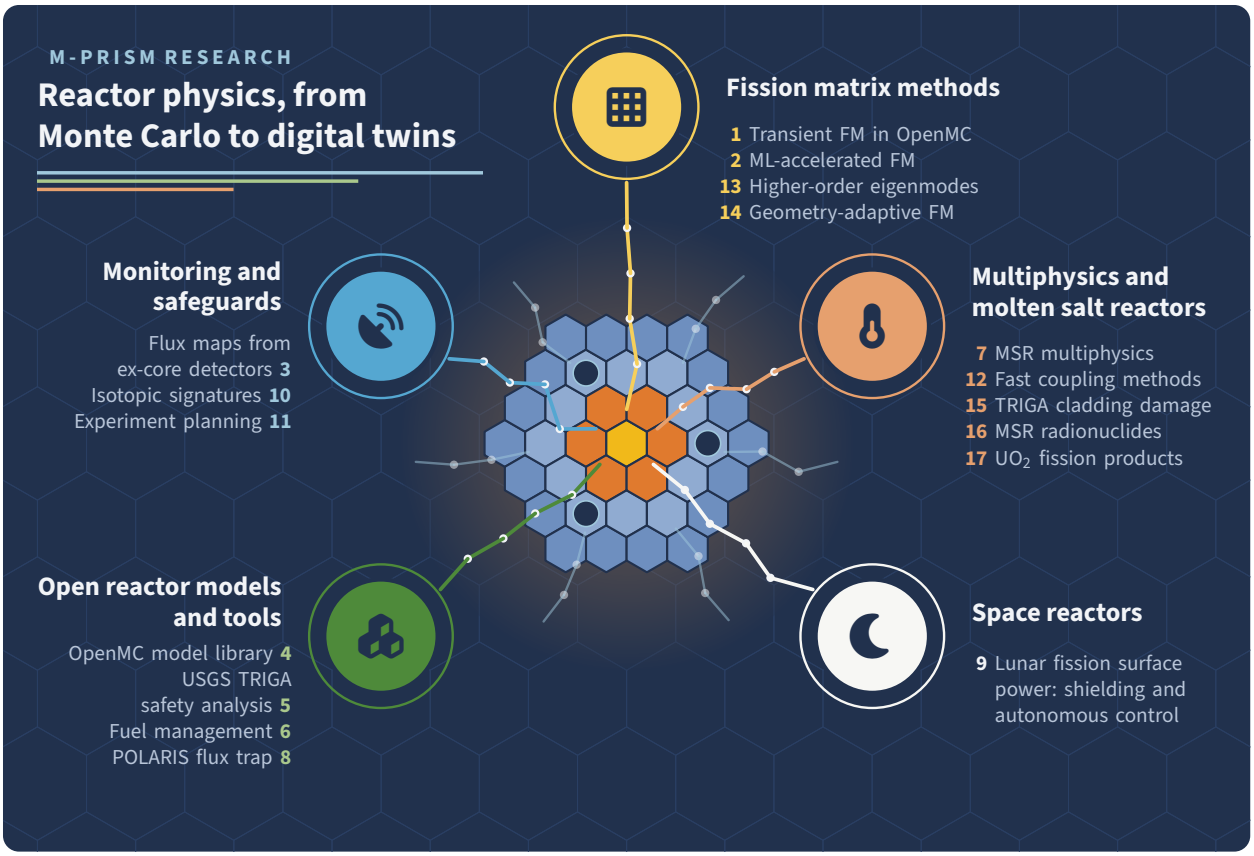

Figure 1. Research areas of the M-PRISM group. A neutron track runs from the core to each area. Numbers refer to the topics in this document.

Table 1. Topics at a glance. Students are those named on each topic's tasks (see the research group on page 25). Open tasks have no student yet.

No.	Topic	Students	Open tasks
1	A transient fission matrix (TFM) kinetics capability for OpenMC	Balouza, Alfaro, Krašovic, Boccardi (planned)	9 of 13
2	Machine learning and AI to accelerate the fission matrix method	Paine, Baesman, Curcio (planned)	6 of 8
3	In-core flux reconstruction from ex-core detectors with machine-learned Green's functions	LaFore	3 of 5
4	An open-source library of research, advanced, and small modular reactor models in OpenMC	Chestnut, Hudak, McCoy, Sauter	3 of 6
5	Neutronics and thermal-hydraulics for re-licensing the USGS TRIGA Reactor (GSTR)	Persons, Chadwick	2 of 7
6	An open-source fuel management and depletion framework with ADDER and OpenMC	McCoy	5 of 8
7	Transient fission matrix, NekRS, and OpenMC multiphysics for molten salt reactors	Boccardi, Harms (planned)	6 of 8

Table 1, continued

No.	Topic	Students	Open tasks
8	Flux-trap buffer zone design for the POLARIS research reactor	Reitz	5 of 9
9	Lunar fission surface power: shielding analysis and an autonomous control digital twin	de Cia	11 of 12
10	Fuel isotopic signatures from the fission matrix and machine learning for safeguards	Mohr	5 of 6
11	Research reactor experiment loading with the fission matrix and machine learning	Mohr	6 of 7
12	Adaptive multiphysics coupling algorithms for advanced reactors	None yet	5 of 5
13	Higher-order and generalized matrix eigenvalue methods for neutron transport	Balouza	7 of 8
14	Geometry-adaptive fission matrix methods for techno-economic microreactor optimization	Cubriel	4 of 5
15	Radiation damage in the aluminum fuel cladding of a TRIGA reactor	None yet	4 of 4
16	Open-source tools for the mechanistic source term of molten salt reactors	None yet	4 of 4
17	Neutronics and thermomechanics of solid metallic fission products in UO_2 fuel	Bellani, Avarino, De Nicolo, Balouza (planned)	2 of 6

1 A transient fission matrix (TFM) kinetics capability for OpenMC

Level: Undergraduate, M.S., and Ph.D.

Skills: Monte Carlo transport with OpenMC, C++ and Python programming, reactor kinetics, and numerical methods.

Many Generation IV (Gen IV) reactors and advanced reactors (ARs) differ substantially from current commercial light water reactors (LWRs), for which decades of operating experience and optimized diffusion codes support design, optimization, and operation. Features such as molten salt fuels, tristructural isotropic (TRISO) particle fuels, compact cores, and fast spectra call for higher-fidelity modeling and simulation.

Neutron transport methods are either deterministic or stochastic [1.1]. Deterministic methods solve a discretized form of the linear Boltzmann equation (LBE) and need approximations in energy, angle, and space. Monte Carlo methods [1.2] avoid these discretizations and give high-fidelity results, but their statistical convergence is slow when detailed 3-D solutions are needed. These limitations grow for time-dependent problems. As a result, kinetics calculations still rely mostly on simplified models, such as point kinetics (PK) [1.3], often tuned for commercial LWRs.

Hybrid methods combine the speed of deterministic methods with the accuracy of Monte Carlo methods [1.4]. A fission matrix gives the number of fission neutrons born in each region of a reactor per fission neutron born in every other region, and its dominant eigenvalue is the multiplication

factor. The transient fission matrix (TFM) method, developed by Laureau et al. [1.5, 1.6], extends this idea in time: it precomputes Monte Carlo response functions and then solves time-dependent problems quickly. Dr. Mascolino and A. Haghighat developed a TFM-based formulation (tRAPID) and validated it against a measured control-rod transient at the Jožef Stefan Institute (JSI) TRIGA Mark-II reactor [1.7, 1.8]. Fast 3-D transport kinetics could support the design, licensing, and operation of ARs, including their digital twins (DTs).

> **Objective.** We will bring TFM-based 3-D kinetics into the open-source Monte Carlo code OpenMC [1.9] through a re-implementation and optimization of the method.

Task 1.1 Compare methods for generating steady-state fission matrix (FM) coefficients in OpenMC (criticality, local fixed-source, and global fixed-source calculations) [1.10].
S. Balouza, since Spring 2026
M. Krašovic, since Spring 2026

Task 1.2 Implement native FM and TFM coefficient tallies in OpenMC, normalized per source neutron, by suitably modifying the tally filters.
B. Alfaro, since Summer 2026

Task 1.3 Use FM source estimates to accelerate fission-source convergence in OpenMC criticality calculations.
B. Alfaro, since Fall 2026

Task 1.4 Implement the TFM equations for the solution of reactor kinetics problems with OpenMC-generated coefficients.

Task 1.5 Integrate the TFM solver into Cardinal, the application based on Idaho National Laboratory's MOOSE framework [1.11] for coupling with thermal-hydraulics solvers.

Task 1.6 Analyze how spatial discretization, time-step size, streaming and advection formulations, and memory-kernel convolution approximations affect numerical stability and fidelity.
P. Boccardi, planned

Task 1.7 Develop coefficient interpolation and optimization strategies for different parameters (e.g., temperatures, fuel enrichment, fuel burnup, and control element positions).

Task 1.8 Develop methodologies to model advanced fuels, such as TRISO, with the TFM method.

Task 1.9 Model TRIGA pulse experiments, in collaboration with JSI, to validate the TFM algorithms.

Task 1.10 Model the C5G7 time-dependent benchmark to verify 3-D time-dependent transport capabilities.

Task 1.11 Optimize the TFM algorithm toward quasi-real-time execution (~10 Hz).

Task 1.12 Implement the TFM algorithm in an advanced reactor digital twin.

Task 1.13 Develop methodologies for problems that are traditionally difficult for the TFM method, such as geometry variations.

References for topic 1

[1.1] G. I. Bell and S. Glasstone. *Nuclear reactor theory*. US Atomic Energy Commission, Washington, DC (United States), 1970.

[1.2] A. Haghighat. *Monte Carlo Methods for Particle Transport*. 2nd ed. CRC Press Taylor & Francis Group, 2020.

[1.3] J. J. Duderstadt and L. J. Hamilton. *Nuclear reactor analysis*. New York: John Wiley & Sons, 1976.

[1.4] A. Haghighat, K. Royston, and W. Walters. "MRT methodologies for real-time simulation of nonproliferation and safeguards problems". In: *Annals of Nuclear Energy* 87 (2016), pp. 61–67. DOI: 10.1016/j.anucene.2015.06.004.

[1.5] A. Laureau, M. Aufiero, P. R. Rubiolo, E. Merle-Lucotte, and D. Heuer. "Transient Fission Matrix: Kinetic calculation and kinetic parameters βeff and Λeff calculation". In: *Annals of Nuclear Energy* 85 (2015), pp. 1035–1044.

[1.6] A. Laureau, M. Aufiero, P. Rubiolo, and D. Heuer. "Coupled Neutronics and Thermal-Hydraulics Transient Calculations Based on a Fission Matrix Approach: Application to the Molten Salt Fast Reactor". In: *Proceedings of the Joint International Conference on Mathematics and Computation (M&C), Supercomputing in Nuclear Applications (SNA) and the Monte Carlo (MC) Method*. Apr. 2015.

[1.7] V. Mascolino and A. Haghighat. "A Novel Hybrid Deterministic and Monte Carlo Neutron Transport Formulation and Algorithm (tRAPID) for Accurate and Fast 3-D Reactor Kinetics". In: *Nuclear Science and Engineering* 198.3 (Mar. 2024), pp. 592–627. DOI: 10.1080/00295639.2023.2197844.

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[1.9] P. K. Romano, N. E. Horelik, B. R. Herman, A. G. Nelson, B. Forget, and K. Smith. "OpenMC: A state-of-the-art Monte Carlo code for research and development". In: *Annals of Nuclear Energy* 82 (2015), pp. 90–97. DOI: 10.1016/j.anucene.2014.07.048.

[1.10] S. Balouza, M. Krašovic, A. Pungerčič, and V. Mascolino. "Comparing Fission Matrix Approaches using a JSI TRIGA Mark-II Reactor Model". In: *Transactions of the American Nuclear Society*. Vol. 135. Phoenix, AZ, Nov. 2026.

[1.11] A. J. Novak, D. Andrs, P. Shriwise, J. Fang, H. Yuan, D. Shaver, E. Merzari, P. K. Romano, and R. C. Martineau. "Coupled Monte Carlo and thermal-fluid modeling of high temperature gas reactors using Cardinal". In: *Annals of Nuclear Energy* 177 (Nov. 2022), p. 109310. DOI: 10.1016/j.anucene.2022.109310.

2 Machine learning and AI to accelerate the fission matrix method

Level: Undergraduate, M.S., and Ph.D.

Skills: Python, machine learning, OpenMC, and linear algebra.

The fission matrix (FM) and transient fission matrix (TFM) methods rely on large databases of response functions, the FM coefficients, computed as functions of system parameters such as fuel and moderator temperatures, burnup, and control element positions. This makes the FM method a natural candidate for machine learning and artificial intelligence (ML/AI) acceleration. Generating, storing, and solving high-dimensional databases for operational and accident configurations is a computational bottleneck on the path to real-time digital twins (~10 Hz). In our approach, ML/AI supports the physics: Monte Carlo transport generates the data, and the FM and TFM equations compute the reactor state.

Interpolation of FM coefficients usually accounts only for the temperature of the destination cell, so the temperatures of the source cell and of the regions in between are lost [2.1, 2.2, 2.3]. A first, preliminary study trained a neural network to learn correction factors for FM coefficients under

arbitrary random temperature distributions [2.4].

We see three areas where ML/AI could help. The first is choosing the parameter ranges and spacing that minimize the number of precalculations needed to populate a system's FM database. The second is better interpolation and combination of existing database entries, for example machine-learned corrections for arbitrary temperature distributions. The third is faster evaluation of the time-convolution integrals in TFM kinetics calculations.

> **Objective.** We will identify and evaluate ML/AI algorithms, models, and strategies to optimize and accelerate the FM method.

Task 2.1 Develop a neural network that encodes the dependence of FM coefficients on arbitrary temperature distributions, including moderator temperature.
D. Paine, since Fall 2025

Task 2.2 Develop interpolation and dimensionality-reduction (encoding and decoding) strategies for control-rod-dependent FM databases, as a baseline for machine-learned encoders [2.5].
TJ Baesman, since Summer 2026

Task 2.3 Evaluate ML/AI strategies, including neural networks (NNs), convolutional neural networks (CNNs), generative adversarial networks (GANs), and physics-informed neural networks (PINNs), to identify the approaches best suited to optimize and accelerate the FM method.

Task 2.4 Develop adaptive strategies (e.g., Bayesian optimization, Gaussian processes, and reinforcement learning) that select the parameter states at which FM coefficients are generated, to minimize the number of Monte Carlo calculations.

Task 2.5 Implement autoencoders to reduce the dimensionality of the system matrix.

Task 2.6 Use recurrent neural networks, such as long short-term memory networks, to accelerate the time-convolution integrals of TFM kinetics calculations toward real-time execution (~10 Hz).

Task 2.7 Use CNNs and PINNs to interpolate combined, non-monotonic dependencies on fuel temperature, burnup, and control element positions, and to reconstruct TFM configurations from ex-core detector measurements.

Task 2.8 Verify and validate ML/AI-accelerated TFM models against experimental and benchmark data from research reactors and simulated advanced reactors (e.g., TRISO-fueled microreactors).

References for topic 2

[2.1] A. J. Rau and W. J. Walters. "Strategies for Fast Fission Matrix Estimation with Fuel Temperature and Control Rod Feedback". In: *Nuclear Science and Engineering* 195.10 (Oct. 2021), pp. 1017–1035. DOI: 10.1080/00295639.2021.1905431.

[2.2] M. Dalinger and W. Walters. "Analysis of Fission Matrix Databases with Nonuniform Fuel Temperature Profiles for Static and Transient Calculations". In: *Nuclear Science and Engineering* 199 (Apr. 2025), S754–S764. DOI: 10.1080/00295639.2024.2328944.

[2.3] C. Vita, W. Walters, V. Mascolino, N. Roskoff, F. Franceschini, N. Abrate, and S. Dulla. "A Correction Ratio Approach to Fission Matrix Interpolation in Systems with Large Temperature Differences". In:

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[2.4] D. Paine and V. Mascolino. "Neural Network-Enabled Thermal Feedback Modeling in the Fission Matrix Method". In: *Transactions of the American Nuclear Society*. Vol. 134. Denver, CO, June 2026, pp. 652–655.

[2.5] V. Mascolino, A. Haghighat, and L. Snoj. "Development and validation of new algorithms for control rods insertion modeling in the RAPID code system using the JSI TRIGA Mark-II reactor". In: *Annals of Nuclear Energy* 166 (Feb. 2022), p. 108711. DOI: 10.1016/J.ANUCENE.2021.108711.

3 In-core flux reconstruction from ex-core detectors with machine-learned Green's functions

Level: Undergraduate, M.S., and Ph.D.

Skills: OpenMC, diffusion and transport theory, Python, and inverse problems.

Many advanced reactors are compact, as microreactors and some small modular reactors (SMRs) are, or operate in harsh radiation and temperature environments. Designers therefore limit in-core detectors as much as possible, and many designs aim to eliminate them. This makes monitoring and control difficult, especially because operating data for these new designs are scarce.

Researchers at Argonne National Laboratory and Purdue University developed GRACE, a method that reconstructs the in-core neutron flux from ex-core measurements only, using a machine-learned Green's function [3.1, 3.2]. Instead of relying on large training data sets, the method enforces the mathematical properties of the Green's function and uses machine learning only for the part that cannot be computed analytically. A Kirchhoff-Helmholtz formulation places the variable terms in a source term and accounts for measurement variations through the boundary conditions [3.3]. On a model of the Purdue University Reactor (PUR-1), the method reconstructed the flux within a few percent for control-rod configurations it was not trained on, and a new quadrupole sensor measured flux gradients at PUR-1 [3.1, 3.4].

The method is built on the one-group neutron diffusion equation [3.5], an approximation of the linear Boltzmann equation [3.6] that can lose accuracy in heterogeneous systems. Its validity for this inverse problem, and the dependence of the effective nuclear data on system parameters, need to be assessed. Our first analysis of the JSI TRIGA Mark-II reactor found that core-averaged one-group data change by a few percent across control-rod configurations [3.7].

> **Objective.** We will analyze the validity of the diffusion approximation in the GRACE method, develop libraries of parameter-dependent effective nuclear data to assess their impact on its accuracy, and investigate more advanced approximations of the transport equation (e.g., P_N and S_N).

Task 3.1 Develop a library of effective nuclear data, including diffusion coefficients and one-group cross sections, for the JSI TRIGA Mark-II reactor as a function of control-rod positions and other parameters.

J. LaFore, since Spring 2026

Task 3.2 Analyze the symmetry of Green's functions computed with Monte Carlo fixed-source

calculations, and test whether a single reference Green's function applies across control-rod configurations.

J. LaFore, since Fall 2026

Task 3.3 Generate synthetic data based on the JSI TRIGA Mark-II reactor to validate the reconstruction algorithm.

Task 3.4 Develop a microreactor model to generate effective nuclear data and synthetic data for validating the algorithm in an environment relevant to advanced reactors (ARs) and SMRs.

Task 3.5 Investigate Green's function formulations based on more advanced approximations of the neutron transport equation.

References for topic 3

[3.1] R. Ponciroli, H. Wang, V. Theos, J. Lau, S. Chatzidakis, and R. B. Vilim. "Ex-Core Neutron-Flux Reconstruction Using Machine-Learned Green's Kernels and Flux-Gradient Measurements". In: *Nuclear Science and Engineering* (2026), pp. 1–27. DOI: 10.1080/00295639.2026.2690565.

[3.2] R. Ponciroli, H. Wang, V. Theos, J. Lau, S. Chatzidakis, and R. B. Vilim. "Who Needs Big Data? Small-Data Flux Reconstruction via Green's Functions and Ex-Core Detectors". In: *Proceedings of the International Conference on the Physics of Reactors (PHYSOR 2026)*. Turin, Italy, Apr. 2026.

[3.3] R. Ponciroli. *On the Well-Posedness of Green's Function Reconstruction via the Kirchhoff-Helmholtz Equation for One-Speed Neutron Diffusion*. arXiv:2505.09766 [math]. May 2025. DOI: 10.48550/arXiv.2505.09766.

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[3.7] J. LaFore, H. Wang, A. Pungerčič, R. Ponciroli, and V. Mascolino. "Evaluation of the Effect of Control Rod Positioning on One-Group Effective Nuclear Data for Ex-Core Flux Reconstruction in the JSI TRIGA Mark-II Reactor". In: *Transactions of the American Nuclear Society*. Vol. 135. Phoenix, AZ, Nov. 2026.

4 An open-source library of research, advanced, and small modular reactor models in OpenMC

Level: Undergraduate and M.S.

Skills: OpenMC, Python, and building models from design documents and benchmark specifications.

Renewed interest in nuclear power, with particular focus on advanced reactors (ARs) and small modular reactors (SMRs), drives large research and development efforts by universities, national laboratories, and industry. Companies understandably protect the details of their designs. This fragments the available computational models, limits researchers' access to information, and leads to duplicated work. Different organizations also use different codes, often export-controlled, which makes models hard to share across institutions and borders.

Open, industry-relevant reference models would let research groups build on each other's work instead of rebuilding similar models. The limited resources of universities and national laboratories could then go to novel work, which would help bring advanced reactor technologies to market faster. OpenMC is open-source and freely available, which makes it a natural common platform.

> **Objective.** We will develop an open-source library of reactor models in the Monte Carlo neutron transport code OpenMC by collecting existing open models and by developing new ones from publicly available documents and benchmark specifications.

Task 4.1 Survey U.S. advanced reactor, SMR, and microreactor designs and select those missing from open model libraries.
K. Chestnut, since Fall 2026
O. Hudak, since Fall 2026

Task 4.2 Develop a model of the heat-pipe microreactor (HP-MR) of the Virtual Test Bed [4.1].

Task 4.3 Develop an open-source, modular OpenMC model of the IAEA-TECDOC-643 research reactor benchmark core (see Topic 6).
T. McCoy, since Spring 2026

Task 4.4 Develop a pebble-bed high-temperature gas-cooled reactor (HTGR) model based on public benchmark specifications.
E. Sauter, since Summer 2026

Task 4.5 Select a close-to-market molten salt reactor (MSR) design and develop an MSR model with thermal-hydraulic coupling (see Topic 7).

Task 4.6 Apply the fission matrix (FM) method to each library model and identify its limitations for each reactor type.

References for topic 4

[4.1] N. E. Stauff, Y. Miao, Y. Cao, K. Mo, A. A. E. Abdelhameed, L. Ibarra, C. Matthews, and E. R. Shemon. "High-Fidelity Multiphysics Modeling of a Heat Pipe Microreactor Using BlueCrab". In: *Nuclear Science and Engineering* (Aug. 2024), pp. 1–17. DOI: 10.1080/00295639.2024.2375175.

5 Neutronics and thermal-hydraulics for re-licensing the USGS TRIGA Reactor (GSTR)

Level: Undergraduate and M.S.

Skills: OpenMC, Python, thermal-hydraulics, and safety analysis.

The U.S. Geological Survey TRIGA Reactor (GSTR) at the Denver Federal Center has been shut down since the failure of an aluminum-clad fuel element. Its Safety Analysis Report (SAR) must be updated for new fuel configurations as part of regulatory re-licensing. In collaboration with USGS and Mines Mechanical Engineering, this project builds a parametric, source-traceable modeling framework, so that the safety-basis analyses can be rerun once the fuel loading and cooling specifications are final.

A reference document will consolidate geometric, material, operational, and historical data, each

tied to its primary source. The facility's existing MCNP models will be updated for the regulatory safety-basis calculations. A new modular OpenMC model, built from facility drawings and published documents, will support research and experiment planning. The two models will be cross-validated and benchmarked against operational data. The thermal-hydraulics safety basis will then be re-evaluated with modern models, to quantify how conservative the original analyses are.

> **Objective.** We will support the GSTR re-licensing and its Safety Analysis Report with updated MCNP models, a new open-source OpenMC model, and modern thermal-hydraulics models.

Task 5.1 Compile a reference document of GSTR geometry, dimensions, operating parameters, material compositions, fresh and depleted fuel isotopics, and control-rod configurations, with primary references.
T. Persons, since Fall 2026

Task 5.2 Update the existing MCNP models of the GSTR core to reflect the new fuel loading, for the regulatory safety-basis submission.

Task 5.3 Develop a modular, Python-packaged OpenMC model of the GSTR for flexible model generation, research applications, and experiment planning.
T. Persons, since Fall 2026

Task 5.4 Cross-validate the MCNP and OpenMC models and benchmark them against historical and operational data.
T. Persons, since Fall 2026

Task 5.5 Review the legacy GSTR thermal-hydraulics safety-basis calculations, software, and physical assumptions.
T. Chadwick, since Fall 2026

Task 5.6 Develop modern thermal-hydraulics models, from the SAR formulas with temperature-dependent properties to coupled neutronics and thermal-hydraulics, to evaluate core temperatures, cooling margins, and safety limits.
T. Chadwick, since Fall 2026

Task 5.7 Document the modeling methodologies and safety margins in technical reports supporting the GSTR SAR.

6 An open-source fuel management and depletion framework with ADDER and OpenMC

Level: Undergraduate, M.S., and Ph.D.

Skills: OpenMC depletion, Python, fuel management, and machine learning.

Fuel management and depletion modeling are essential for designing advanced reactors (ARs) and molten salt reactors (MSRs) and for operating research reactors, which often run cycles with complex fuel management, as at the Massachusetts Institute of Technology Reactor (MITR) [6.1]. Accessible fuel management and depletion software is scarce and often depends on export-controlled neutron transport codes.

The Advanced Dimensional Depletion for Engineering of Reactors (ADDER) software, developed

at Argonne National Laboratory, is an object-oriented fuel management and depletion framework designed to work with any neutron transport code [6.2, 6.3]. Its MCNP5 [6.4] and MCNP6.2 [6.5] interfaces were developed, verified, and validated under Argonne's NQA-1 compliant software quality assurance program, as part of the U.S. High Performance Research Reactor (USHPRR) conversion effort [6.6]. Research reactors such as MITR and the University of Missouri Research Reactor (MURR) are adopting it. MCNP, however, is export-controlled, so foreign national users and international partners cannot easily use these interfaces. An interface to the open-source code OpenMC would remove this barrier. Machine learning and artificial intelligence (ML/AI) algorithms embedded in the framework could also support target-based irradiation planning, fuel cycle design, and criticality searches.

> **Objective.** We will develop and validate an open-source OpenMC/ADDER fuel management and depletion framework, including target-based ML/AI algorithms for the design and operation of research and advanced reactors.

Task 6.1 Establish and analyze an OpenMC model of the IAEA-TECDOC-643 research reactor core, with a code-to-code comparison against MCNP, in collaboration with Argonne National Laboratory.
T. McCoy, since Spring 2026

Task 6.2 Perform rundown depletion calculations of the IAEA-TECDOC-643 core with OpenMC, with fixed control blades and with criticality searches.
T. McCoy, since Summer 2026

Task 6.3 Compare the all-OpenMC rundown calculations with ADDER/MCNP calculations for code-to-code verification.
T. McCoy, since Fall 2026

Task 6.4 Extend the ADDER framework to interface with OpenMC directly.

Task 6.5 Expand the library of open research reactor models in OpenMC (see Topic 4) for use within ADDER.

Task 6.6 Evaluate ML/AI strategies for target-based fuel management and depletion, including irradiation optimization, cycle planning, and criticality searches.

Task 6.7 Develop the most promising ML/AI algorithms for target-based fuel management and depletion.

Task 6.8 Validate the ADDER/OpenMC/ML/AI framework with computational benchmarks and available experimental data from the MITR and MURR reactors.

References for topic 6

[6.1] D. Carpenter, G. Kohse, and L. W. Hu. *MITR User's Guide, Rev. 3*. Tech. rep. 2012.

[6.2] V. Mascolino, K. Anderson, C. Castagna, M. Elsawi, and E. Wilson. *ADDER v1.1.0 User Guide*. Tech. rep. ANL/RTR/TM-24/25. Lemont, IL: Argonne National Laboratory, 2025. DOI: 10.2172/2564957.

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7 Transient fission matrix, NekRS, and OpenMC multiphysics for molten salt reactors

Level: Undergraduate, M.S., and Ph.D.

Skills: OpenMC, computational fluid dynamics (NekRS), reactor kinetics, C++ and Python programming, and high-performance computing.

Molten salt reactors (MSRs), including small modular reactor (SMR) designs, are among the most investigated Gen IV concepts. Demonstrated at Oak Ridge National Laboratory in the 1960s with the Molten Salt Reactor Experiment (MSRE) [7.1], they are making a comeback thanks to their fuel flexibility, high burnup, online refueling, and thermal energy storage capability.

In circulating-fuel MSRs, the fuel is also the coolant. Delayed neutron precursors (DNPs) move with the flow, which shifts their concentrations downstream and changes the kinetics, the safety margins, and the power distribution. The transient fission matrix (TFM) method models the DNP source explicitly, which makes it well suited to account for precursor streaming [7.2, 7.3, 7.4]. Coupling a TFM solver with computational fluid dynamics, system codes, and reduced-order models would enable detailed kinetics analyses of MSRs at different computing costs, down to digital-twin speeds. With colleagues in Mines Mechanical Engineering, we plan to use MSRE experiments, such as the reactivity effect of fuel circulation, as validation data.

> **Objective.** We will develop a fully open-source framework for the coupled neutronics and thermal-hydraulics steady-state and kinetics analysis of molten salt reactors.

Task 7.1 Develop an MSRE model with the computational fluid dynamics code NekRS [7.5].
B. Harms, planned

Task 7.2 Develop an OpenMC model of the MSRE.

Task 7.3 Couple NekRS and OpenMC through Cardinal [7.6].

Task 7.4 Formulate and implement the advection and decay of DNPs in the TFM solver, accounting for flow profiles and for precursor decay outside the active core.

Task 7.5 Implement a flowing-fuel quasi-static algorithm within a subchannel code (e.g., COBRA-TF [7.7]), using its thermal-hydraulics solver to provide velocity and temperature fields.

Task 7.6 Identify, evaluate, and integrate reduced-order models and machine-learning accelerators for the thermal-hydraulics coupling, to reach calculation frequencies suitable for digital twins (~10 Hz).

Task 7.7 Establish low-latency data exchange within the OpenMC, Cardinal, and NekRS frame-

work to evaluate streaming terms derived from Monte Carlo transport operators.

P. Boccardi, since Fall 2026

Task 7.8 Validate the extended TFM solver against MSRE experiments, and verify it against open-source multiphysics codes (e.g., GeN-FOAM and MOOSE) for steady-state shifted precursor profiles and transient flow conditions.

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8 Flux-trap buffer zone design for the POLARIS research reactor

Level: Undergraduate and M.S.

Skills: OpenMC, Python, research reactor design, and radiation damage and heating analysis.

Researchers at Virginia Tech are designing a next-generation research reactor for Virginia, POLARIS (formerly VA-RER) [8.1, 8.2]. The facility aims to educate the next generation of the nuclear workforce in a state with a strong nuclear industry (e.g., Framatome and Dominion). It also aims to become one of the most accessible and instrumented research reactors in the country, supporting both student education and the reactor physics experiments needed to design, license, and operate new reactors. In particular, POLARIS is envisioned for benchmarking and experimental validation of the high-fidelity reactor physics software needed for advanced reactors (ARs) and small modular reactors (SMRs).

A central flux trap, surrounded by a reconfigurable buffer zone of movable material layers, will tailor the neutron spectrum for irradiation experiments [8.3]. The buffer must change the trap spectrum

with minimal feedback on the core. We evaluate buffer configurations efficiently with the fission matrix (FM) method combined with the detector response function (DRF) methodology [8.4].

> **Objective.** We will support the design of POLARIS by designing the flux-trap buffer zone and contributing to the criticality analysis of the system.

Task 8.1 Evaluate isotopes and materials for a buffer zone that tailors the spectrum in the flux trap.
J. Reitz, since Fall 2025

Task 8.2 Model buffer zone configurations in OpenMC.
J. Reitz, since Summer 2026

Task 8.3 Analyze flux-trap performance for different buffer zones with the FM and DRF methodologies.
J. Reitz, since Fall 2026

Task 8.4 Optimize the trap flux intensity and spectrum and the reactivity feedback of the buffer as functions of buffer materials, layer thicknesses, and coolants.
J. Reitz, since Fall 2026

Task 8.5 Develop group-wise flux-trap DRFs that can be combined with sample cross sections to assess the feasibility of irradiation experiments quickly.

Task 8.6 Perform high-fidelity OpenMC simulations with ENDF/B-VIII.1 cross sections and fine energy bins to calculate 3-D volumetric heating and radiation damage in candidate buffer materials (including beryllium-based materials, D_2O, graphite, B_4C, Cd, and Hf).

Task 8.7 Develop custom Abaqus user subroutines (e.g., UTEMP, HETVAL, and USDFLD) to map 3-D heating and radiation-damage fields from OpenMC onto finite-element meshes.
Planned collaboration with Politecnico di Milano

Task 8.8 Perform non-linear coupled thermo-mechanical analyses with Abaqus/Standard and Abaqus/Explicit to evaluate temperature-dependent stresses, thermal expansion, clearance evolution, frictional contact, and low-cycle fatigue during buffer sheet movement.
Planned collaboration with Politecnico di Milano

Task 8.9 Assess sizing limits, actuation forces, and potential thermal jamming for solid-plate and liquid-channel reconfigurable buffer designs.
Planned collaboration with Politecnico di Milano

References for topic 8

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9 Lunar fission surface power: shielding analysis and an autonomous control digital twin

Level: Undergraduate and M.S.

Skills: OpenMC, radiation shielding and variance reduction, control systems, and Python.

Fission reactors are leading candidates to power crewed lunar surface missions, thanks to their reliability and energy density, including through the 14-day lunar night [9.1, 9.2]. NASA and DOE funded concept studies of a lunar fission surface power (FSP) system by industry teams led by Lockheed Martin, Westinghouse, and IX [9.3]. This topic addresses two challenges for such systems: shielding under severe mass limits and autonomous operation.

Shield mass must be kept low, so dose predictions at the lunar outpost must be accurate rather than overly conservative. The problem is one of deep penetration, with near vacuum above the surface. Joint work with collaborators at the University of Utah showed that most of the far-field dose comes from neutrons scattered through the lunar regolith [9.4]. Monte Carlo methods converge slowly for such problems, and deterministic methods suffer from ray effects [9.5]. Monte Carlo with weight windows derived from deterministic importance functions, as in the CADIS and FW-CADIS methods [9.6, 9.7], addresses both issues.

A round-trip signal delay of about 2.5 seconds and periodic communication blackouts rule out real-time intervention from Earth, so a lunar reactor must operate largely on its own. Together with a Mines senior design team, we are developing an AI-driven digital twin for autonomous control that couples reduced-order physics models with an AI-based control framework.

> **Objective.** We will develop a variance-reduced OpenMC shielding model of a lunar FSP reactor, with weight windows based on deterministic importance functions (e.g., from PENTRAN [9.8] or OpenMC's random-ray solver). We will also develop an AI-assisted digital twin for autonomous control and anomaly response.

SHIELDING AND VARIANCE REDUCTION

Task 9.1 Develop an FSP model in OpenMC, based on an existing MCNP model developed by collaborators at the University of Utah.

Task 9.2 Evaluate the current OpenMC weight-window capabilities against established workflows based on deterministic importance functions.

Task 9.3 Evaluate FW-CADIS weight windows generated with the random-ray solver embedded in OpenMC.

Task 9.4 Evaluate variance reduction methods and dose results across shield thickness, material selection, distance, and regolith composition uncertainties.

G. A. de Cia, since Fall 2026

Task 9.5 Develop angular-dependent weight windows in OpenMC to further improve the convergence of the FSP dose calculation.

Task 9.6 Assess these capabilities and configurations in terms of the accuracy of the calculated dose rates and figures of merit.

AUTONOMOUS CONTROL AND DIGITAL TWIN

Task 9.7 Review the SNAP-10A flight reactor, the Kilopower (KRUSTY) ground test, and the NASA Phase 1 FSP concepts.

Task 9.8 Define the system requirements and operating envelope of a lunar FSP microreactor, mapping control mechanisms across the core, heat transport, power conversion, and heat rejection.

Task 9.9 Develop fast-running physics models of the reactor, its heat transport, and its power conversion, with kinetic parameters computed with OpenMC.

Task 9.10 Formulate the digital twin and the autonomous control architecture.

Task 9.11 Couple the control framework with the reduced-order physics models to simulate autonomous load following, transients, and loss-of-communication operation.

Task 9.12 Benchmark the control framework for stability, decision-making efficiency, and safety margins during nominal and emergency operation.

References for topic 9

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10 Fuel isotopic signatures from the fission matrix and machine learning for safeguards

Level: Undergraduate, M.S., and Ph.D.

Skills: OpenMC depletion, Python, machine learning, and inverse problems.

The fission matrix (FM) and transient fission matrix (TFM) methods (see Topics 1 and 2) perform fast and accurate neutron transport calculations that reconstruct the fission neutron source in a nuclear system, such as a nuclear reactor [10.1]. The FM method could be extended to resolve the energy spectrum of the fission source, which depends on the fuel isotopics, enrichment, and burnup.

Since the FM equations can be solved in seconds, inference methodologies could reconstruct the fission source and the neutron flux inside a reactor from a handful of out-of-core detector measurements available to inspectors. Machine learning and artificial intelligence (ML/AI) surrogate models could speed up this inference by encoding a large number of parameters (e.g., fuel isotopics and burnup) relative to a reference configuration based on the FM. Because the neutron physics stays in the FM coefficients rather than in the surrogate model, the approach lends itself to physics-based and physics-informed ML/AI methodologies. To evaluate the out-of-core measurements, the FM methodology will be coupled with the response-based detector response function (DRF) method [10.2].

> **Objective.** We will develop a novel methodology for fuel isotopic inference from detector responses using the FM method and ML/AI surrogate models.

Task 10.1 Select and develop a suitable test model, e.g., a high-flux research reactor such as MURR or the JSI TRIGA Mark-II reactor.
E. Mohr, since Summer 2026

Task 10.2 Extend the FM methodology to multiple energy groups in the coefficients, with a user-selectable group structure based on the spectra of the main fissile isotopes.

Task 10.3 Develop inference methodologies to reconstruct a case-specific fission matrix from synthetic detector data.

Task 10.4 Enhance the inference methodologies using ML/AI surrogate models and algorithms.

Task 10.5 Generate a comprehensive library of FM coefficients for different fuel isotopics, burnups, cooling times, and enrichments.

Task 10.6 Validate the methodology using arbitrary combinations of database entries.

References for topic 10

[10.1] W. J. Walters, N. J. Roskoff, and A. Haghighat. “The RAPID Fission Matrix Approach to Reactor Core Criticality Calculations”. In: *Nuclear Science and Engineering* 192.1 (Oct. 2018), pp. 21–39. DOI: 10.1080/00295639.2018.1497395.

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11 Research reactor experiment loading with the fission matrix and machine learning

Level: Undergraduate and M.S.

Skills: OpenMC, Python, machine learning, and optimization.

High-performance research reactors often follow aggressive refueling schedules. These schedules leave no time for high-fidelity, time-consuming calculations to optimize experiment locations and loadouts. The result can be sub-optimal irradiation throughput or mismatched experiment reactivity worths.

The fission matrix (FM) and detector response function (DRF) methods rely on extensive precalculations of response functions and coefficients. An arbitrary system configuration can then be modeled in seconds to minutes, without repeating expensive transport calculations, while retaining their accuracy [11.1, 11.2]. A response-function approach could therefore enable accurate irradiation and experiment design, with the same setup time as the much coarser models normally used to support core loading at research reactors.

This large amount of data can also be encoded into machine learning and artificial intelligence (ML/AI) algorithms to extract the implicit dependencies of target metrics on the core loading's input parameters, e.g., the burnup and location of specific fuel elements. These algorithms could then support target-based approaches that devise the optimal core configuration from the available fuel elements to reach a given objective, e.g., a certain activation of a sample, a desired cycle length, or a peak neutron flux.

> **Objective.** We will develop FM-based, ML/AI-enhanced algorithms for the target-based design and optimization of irradiation campaigns and experiment loadings at high-performance research reactors.

Task 11.1 Develop suitable OpenMC research reactor models (e.g., based on MURR or MITR) and validate them against facility measurements.
E. Mohr, since Summer 2026

Task 11.2 Develop a library of experiment designs, loadouts, and performance metrics.

Task 11.3 Develop FM-based detector response calculations to evaluate the identified performance metrics, e.g., at flux traps and irradiation positions.

Task 11.4 Develop an FM coefficient library based on the burnup of the available fuel inventory at the research reactor.

Task 11.5 Evaluate ML/AI algorithms for inferring irradiation performance metrics from the FM method, including forward surrogate models and inverse searches of the FM database.

Task 11.6 Implement the selected ML/AI algorithms to predict irradiation metrics from fuel cycle information (e.g., the specific fuel elements used for a core loading).

Task 11.7 Validate the algorithms with synthetic data and available experimental data from U.S. High Performance Research Reactor (USHPRR) facilities, e.g., startup tests, control blade worths, and sample activations.

References for topic 11

[11.1] W. J. Walters, N. J. Roskoff, and A. Haghighat. "The RAPID Fission Matrix Approach to Reactor Core Criticality Calculations". In: *Nuclear Science and Engineering* 192.1 (Oct. 2018), pp. 21–39. DOI: 10.1080/00295639.2018.1497395.

[11.2] V. Mascolino, A. Haghighat, and L. Snoj. "Development and validation of new algorithms for control rods insertion modeling in the RAPID code system using the JSI TRIGA Mark-II reactor". In: *Annals of Nuclear Energy* 166 (Feb. 2022), p. 108711. DOI: 10.1016/J.ANUCENE.2021.108711.

12 Adaptive multiphysics coupling algorithms for advanced reactors

Level: M.S. and Ph.D.

Skills: Multiphysics coupling with Cardinal and MOOSE, C++, Python, Fortran (f2py), and numerical methods.

Effective deployment of advanced reactor concepts requires bridging high-fidelity modeling tools with fast-running design codes to reduce the computational burden. This project will integrate three neutronics solvers (OpenMC [12.1], FRENETIC, and the fission matrix (FM) method) with two thermal-hydraulics (TH) engines (NekRS [12.2] and FRENETIC's internal TH module) into an adaptive multi-fidelity workflow. FRENETIC is a coupled neutronics and TH core simulator developed at Politecnico di Torino, which has been applied to full-core coupled modeling of the EBR-II SHRT-45R transient [12.3]. The MOOSE-based Cardinal application will orchestrate the data transfers between the high-fidelity solvers [12.4].

> **Objective.** We will build an adaptive multiphysics coupling ecosystem, using Cardinal, Python wrappers, and f2py interfaces, to optimize calculation speed and precision dynamically across solver hierarchies.

Task 12.1 Develop Python wrappers using f2py to interface legacy Fortran routines in FRENETIC with modern Python and C++ software stacks.

Task 12.2 Implement Python and C++ wrappers for the FM engine to facilitate multi-code integration.

Task 12.3 Use Cardinal to orchestrate spatial mesh data transfers (e.g., power density, fluid temperatures, densities, and velocities) between neutronics and TH solvers, including OpenMC and NekRS.

Task 12.4 Design and test adaptive coupling algorithms (e.g., adaptive relaxation and Picard iterations) that adjust data exchange frequencies, spatial mappings, and relaxation factors based on local convergence criteria.

Task 12.5 Conduct multi-fidelity benchmarking on advanced reactor configurations (e.g., the Nuclear Energy Agency benchmark of a small modular lead-cooled fast reactor). Compare high-fidelity reference solutions (OpenMC with NekRS) with fast-running options that couple the FM method or FRENETIC neutronics to FRENETIC TH.

References for topic 12

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[12.3] D. Caron, R. Bonifetto, S. Dulla, V. Mascolino, P. Ravetto, L. Savoldi, D. Valerio, and R. Zanino. "Full-core coupled neutronic/thermal-hydraulic modelling of the EBR-II SHRT-45R transient". In: *International Journal of Energy Research* 42.1 (2018), pp. 134–150. DOI: 10.1002/er.3571.

[12.4] A. J. Novak, D. Andrs, P. Shriwise, J. Fang, H. Yuan, D. Shaver, E. Merzari, P. K. Romano, and R. C. Martineau. "Coupled Monte Carlo and thermal-fluid modeling of high temperature gas reactors using Cardinal". In: *Annals of Nuclear Energy* 177 (Nov. 2022), p. 109310. DOI: 10.1016/j.anucene.2022.109310.

13 Higher-order and generalized matrix eigenvalue methods for neutron transport

Level: M.S. and Ph.D.

Skills: Transport theory, numerical linear algebra, Monte Carlo methods, and Python.

The standard fission matrix (FM) maps fission neutron production between regions, and its dominant eigenvalue is the multiplication eigenvalue (k_{eff}). The higher-order eigenvalues and eigenmodes of the same matrix describe spatial harmonics of the fission source [13.1]. They carry information on the dominance ratio, on spatial decoupling, and on the response of a core to local perturbations such as control rod motion. Because the FM is a small explicit matrix, Krylov-subspace (Arnoldi) eigensolvers extract its higher modes directly, although tally noise limits their accuracy. Alternative transport eigenvalue formulations, such as the collision, time (α), and density (γ) eigenvalues, provide further insight into core stability, spatial coupling, and transient behavior. This research analyzes the higher-order spectrum of the FM. With Dr. N. Abrate (Politecnico di Torino), who proposed the generalized-eigenvalue formulation, it also generalizes the response-matrix methodology beyond multiplication eigenvalue problems using an integral transport framework.

> **Objective.** We will analyze the higher-order eigenmodes of the fission matrix and establish a generalized integral transport framework to construct modified source-response matrices for alternative transport eigenvalue formulations.

Task 13.1 Extract and interpret the higher-order FM eigenvalues and eigenmodes of the JSI TRIGA Mark-II reactor as a function of control rod insertion.
S. Balouza, since Summer 2026

Task 13.2 Track near-degenerate eigenmodes across core configurations with eigenvector-overlap methods, and extend the analysis to multi-rod configurations and to localized perturbations (e.g., control rod vibration).

Task 13.3 Compute fundamental and higher-mode prompt α eigenvalues from time-resolved transient fission matrix (TFM) coefficients and compare them with direct Monte Carlo α-

eigenvalue calculations.

Task 13.4 Reformulate collision, time, and density eigenvalue problems within a unified integral-transport operator framework.

Task 13.5 Formulate modified source-response matrices that replace the standard fission-to-fission response with target-specific mechanisms.

Task 13.6 Analyze spectral consistency and the mathematical conditions under which generalized matrix eigenvalues approximate continuous or discretized transport operators.

Task 13.7 Implement numerical estimators using Monte Carlo and deterministic transport calculations, and evaluate them on benchmark problems.

Task 13.8 Assess physical applications for core design, spatial coupling metrics, transient characterization, and reactor stability monitoring.

References for topic 13

[13.1] S. Carney, F. Brown, B. Kiedrowski, and W. Martin. "Theory and applications of the fission matrix method for continuous-energy Monte Carlo". In: *Annals of Nuclear Energy* 73 (Nov. 2014), pp. 423–431. DOI: 10.1016/j.anucene.2014.07.020.

14 Geometry-adaptive fission matrix methods for techno-economic microreactor optimization

Level: M.S. and Ph.D.

Skills: OpenMC, Python, machine learning, and techno-economic analysis.

Nuclear microreactors, such as heat-pipe microreactors (HP-MRs), offer rapidly deployable and resilient power for remote regions, but they face economic hurdles due to diseconomies of scale. The open-source, Python-based Microreactor Optimization Using Simulation and Economics (MOUSE) tool, developed at Idaho National Laboratory, couples reactor design calculations performed with OpenMC directly with bottom-up cost estimates [14.1]. MOUSE uses leakage-corrected 2-D models to balance runtime and accuracy during parametric sweeps, because 3-D transport calculations of compact cores make multi-variable optimization computationally prohibitive.

Traditional fission matrix (FM) approaches rely on fixed spatial grids, which makes them difficult to adapt when the geometry changes (e.g., fuel size, pitch, and cell geometry). This project will integrate geometry-adaptive FM methods and physics-informed machine learning algorithms into the MOUSE framework. The aim is to bypass time-consuming Monte Carlo transport iterations and to deliver near-instantaneous evaluations of techno-economic metrics (e.g., the levelized cost of energy, LCOE, and the total capital investment) with 3-D transport-level accuracy.

> **Objective.** We will develop a geometry-adaptive FM methodology, integrated into the MOUSE framework with physics-informed machine learning, to enable fast, 3-D, transport-accurate techno-economic optimization of microreactors.

Task 14.1 Build fission matrices for an HP-MR OpenMC model, as the first step toward integrating FM response calculations within the MOUSE framework.

A. Cubriel, since Fall 2026

Task 14.2 Develop mathematical formulations to handle geometric variations (e.g., core fuel dimensions, lattice pitch, and cell geometries) within the response-matrix framework.

Task 14.3 Formulate and train physics-informed machine learning algorithms (e.g., deep neural networks) to reconstruct, scale, and adjust fission matrices under varying spatial configurations.

Task 14.4 Link the geometry-adaptive FM solver with the MOUSE cost estimation modules to compute near-real-time techno-economic metrics, including the LCOE and capital expenditures.

Task 14.5 Conduct design optimization and uncertainty propagation studies, and validate the coupled framework against high-fidelity 3-D OpenMC reference calculations.

References for topic 14

[14.1] B. Hanna. "Bridging Microreactor Design and Economics: The Microreactor Optimization Using Simulation and Economics (MOUSE) Tool". In: *Transactions of the American Nuclear Society*. 2026.

15 Radiation damage in the aluminum fuel cladding of a TRIGA reactor

Level: M.S. and Ph.D.

Skills: OpenMC, radiation damage, finite element structural analysis, and materials science.

Estimating the residual load-bearing capacity of aged structural components is crucial for the aging management and life extension of research reactors. Intense neutron irradiation alters the physical and mechanical performance of fuel cladding over the operational history. This project develops an integrated numerical methodology that links the reactor operating history to the evolution of structural material degradation in the aluminum fuel cladding of a TRIGA Mark-II reactor with decades of operating history. The structural side would build on the micromechanical models used in Topic 17.

> **Objective.** We will develop an integrated workflow combining Monte Carlo neutronics, finite element structural analysis, and historical operational data to predict radiation damage and structural degradation in TRIGA fuel cladding.

Task 15.1 Reconstruct historical irradiation conditions from operational logs (power levels and operation duration) to determine the spatial distributions of neutron flux, fluence, and radiation damage metrics (e.g., displacements per atom, dpa).

Task 15.2 Develop finite element structural models incorporating irradiation-dependent material properties, including Young's modulus, yield functions, and fracture energy.

Task 15.3 Calibrate and validate the coupled neutronic-structural models against the facility's experimental and operational data.

Task 15.4 Establish quantitative correlations between neutronic parameters and mechanical property degradation to evaluate the residual load-bearing capacity of aged fuel cladding for long-term reactor management.

16 Open-source tools for the mechanistic source term of molten salt reactors

Level: M.S. and Ph.D.

Skills: OpenMC depletion, C++ and Python programming, and molten salt chemistry.

In liquid-fueled molten salt reactors (MSRs), fission products and other radioactive species are generated in the flowing salt and transported around the primary loop. Chemistry control systems in the primary loop, such as off-gas removal systems, are therefore a key design feature. Preliminary results have shown that the salt loop flow rate and the discretization of the loop affect the predicted off-gas isotopics, especially for short-lived isotopes [16.1]. Since fission products are generated by fission within the circulating salt, their production rates in different regions of the reactor depend on the distribution of fission events. This distribution changes with time, with the fuel composition, and with the distributions of the neutron flux and of the chemical species in the primary loop and reactor core.

Evaluating the consequences of MSR accidents requires an accurate mechanistic radiation source term for the radioactive isotopes in the fuel. Predicting the generation and transport of fission products requires high-fidelity simulation tools that account for the coupled phenomena above. Such tools are not widely available to developers and researchers, yet they are needed for the design, safety, safeguards, and commercialization of advanced MSRs. A fuel depletion benchmark of the MSRE [16.2], compared against measured fuel salt samples, provides a reference for such tools [16.3].

> **Objective.** With our collaborators at the University of Texas at Austin, we will develop open-source tools directly in the OpenMC Monte Carlo code for accurate mechanistic source term analysis of molten salt reactors.

Task 16.1 Develop parallel in-material depletion of nuclear fuel in OpenMC, including the solution of depletion matrices coupled by transfer rates, to speed up MSR depletion calculations.

Task 16.2 Extend the transfer-rate capability of OpenMC depletion so that it combines transfer rates with external source rates, to account for the continuous removal and insertion of chemical species within the salt, either by concentration difference or at a fixed rate.

Task 16.3 Track the inventory of MSR loop components (e.g., off-gas systems) used to remove fission products and radioisotopes from the main salt loop.

Task 16.4 Validate the developed MSR loop capabilities with a whole-core MSRE model coupled to multi-region models of the salt loop and off-gas system, using measured salt, gas, and off-gas data.

References for topic 16

[16.1] A. G. Macris, S. Shahbazi, O. Chvala, V. Mascolino, and K. Clarno. “Modeling the Effects of Salt Loop Flow Rate on Molten Salt Reactor Off-gas Isotopics”. In: *Proceedings of the International Conference on the Physics of Reactors (PHYSOR 2026)*. Turin, Italy, Apr. 2026. DOI: 10.5281/zenodo.20803166.

[16.2] P. N. Haubenreich, J. R. Engel, B. E. Prince, and H. C. Claiborne. *MSRE Design and Operations Report. Part III. Nuclear Analysis*. Tech. rep. ORNL-TM-730. Oak Ridge National Lab., Tenn., Feb. 1964. DOI: 10.2172/4114686.

[16.3] S. Shahbazi, L. Chierici, T. Fei, K. E. Anderson, V. Mascolino, A. G. Nelson, P. K. Romano, and A. Stubsgaard. "Development of a Fuel Depletion Benchmark Model for the Molten Salt Reactor Experiment and Demonstration with MCNP-ADDER". In: *Proceedings of the International Conference on the Physics of Reactors (PHYSOR 2026)*. Turin, Italy, 2026. DOI: 10.5281/zenodo.20803265.

17 Neutronics and thermomechanics of solid metallic fission products in UO_2 fuel

Level: M.S. and Ph.D.

Skills: OpenMC depletion, fuel performance modeling (SCIANTIX), micromechanics, and uncertainty quantification.

Solid metallic fission products (the five-metal ϵ-phase precipitates of Mo, Tc, Ru, Rh, and Pd) form inclusions in irradiated UO_2 fuel and are more concentrated toward the pellet rim. Their mechanical effect on the pellet stress state, which governs radial cracking and pellet-cladding interaction, has received less attention than that of gaseous fission products. Fuel thermomechanics analyses often rely on generic inputs, such as constant fission yields and assumed compositions. Monte Carlo depletion can instead compute the production of these species for the specific fuel, layout, and irradiation history.

In collaboration with Politecnico di Milano, a first multiphysics chain coupled an analytical pellet temperature solution, OpenMC depletion over the first cycle of an AP1000 reactor, and an elastic micromechanical model [17.1]. That study found that a purely elastic model overpredicts the stresses, which motivates inelastic models. Current work couples OpenMC depletion with meso-scale models of precipitate nucleation and growth for SCIANTIX, the open-source fuel behavior code developed at Politecnico di Milano [17.2]. It also assesses how much neutronics fidelity (pin, assembly, or core) the thermomechanical models need.

> **Objective.** We will develop a multiscale neutronics and thermomechanics workflow that provides fuel-specific, radially resolved metallic fission product source terms from Monte Carlo depletion to micromechanical and fuel performance models of UO_2 fuel.

Task 17.1 Calibrate a meso-scale model of the nucleation and growth of metallic precipitates against microscopy data, using neutronics-based source terms.
V. Bellani, since Spring 2026

Task 17.2 Quantify the radial and burnup dependence of the metallic fission product source term with OpenMC pin depletion, including plutonium buildup at the pellet rim and per-ring versus pellet-averaged treatments.
V. Bellani, since Spring 2026
S. Balouza, planned

Task 17.3 Extend the neutronics from pin to assembly and full core, and evaluate how fuel shuffling and power history change the metallic fission product fractions.
D. Avarino, since Fall 2026

Task 17.4 Propagate neutronics input uncertainties (e.g., power, temperature, and enrichment) to the pellet stress state predicted by inelastic micromechanical models.
G. De Nicolo, since Fall 2026

Task 17.5 Assess at each scale (pin, assembly, and core) whether higher neutronics fidelity changes the thermomechanical predictions, to justify simplified inputs where possible.

Task 17.6 Compare model predictions with available post-irradiation examination data.

References for topic 17

[17.1] L. Rizzetto, V. Mascolino, D. Pizzocri, F. Piccagli, and N. Cefis. "Preliminary multi-physics study of the influence of solid metallic fission products in the UO_2 fuel of the AP1000 reactor". In: *Proceedings of the 1st International Conference on Modelling, Testing and Monitoring of Engineering Structures (MOTEMOES 2026)*. Piacenza, Italy: Springer, Sept. 2026.

[17.2] D. Pizzocri, T. Barani, and L. Luzzi. "SCIANTIX: A new open source multi-scale code for fission gas behaviour modelling designed for nuclear fuel performance codes". In: *Journal of Nuclear Materials* 532 (2020), p. 152042. DOI: 10.1016/j.jnucmat.2020.152042.

Research group

Students who work, or have worked, on the topics above. Topic numbers refer to the sections of this document.

Graduate students

Name	Program	Since	Topics
TJ Baesman	Ph.D., Nuclear Engineering	Spring 2026	2
Samah Balouza	Ph.D., Nuclear Engineering	Spring 2026	1, 13, 17
Joe Reitz	M.S., Nuclear Engineering	Fall 2025	8
Andrew Cubriel	M.Eng., Nuclear Engineering	Spring 2026	14

Undergraduate researchers

Name	Program	Since	Topics
Braydon Alfaro	Mechanical Engineering, Nuclear Energy track	Summer 2026	1
Trevor Chadwick	Mechanical Engineering, Nuclear Energy track	Fall 2026	5
Kendall Chestnut	Mechanical Engineering, Nuclear Energy track	Fall 2026	4
Brayden Harms	Mechanical Engineering, Nuclear Energy track	Fall 2025	7
Owen Hudak	Mechanical Engineering, Nuclear Energy track	Fall 2026	4
Jake LaFore	Engineering Physics	Spring 2026	3
Thomas McCoy	Mechanical Engineering, Nuclear Energy track	Spring 2026	4, 6
Ethan Mohr	Mechanical Engineering, Nuclear Energy track	Summer 2026	10, 11

Name	Program	Since	Topics
Daris Paine	Mechanical Engineering	Fall 2025	2
Thomas Persons	Mechanical Engineering, Nuclear Energy track	Summer 2026	5
Ethan Sauter	Engineering Physics	Summer 2026	4

Visiting, joint-thesis, and co-advised students

Name	Program	Since	Topics
Vittoria Bellani POLIMI	Ph.D. student (co-advised M.S. thesis in Materials and Nanotechnology Engineering)	Spring 2026	17
Pietro Boccardi POLITO	M.S., Energy and Nuclear Engineering; joint thesis	Fall 2026	1, 7
Gianantonio Curcio POLITO	M.S., Energy and Nuclear Engineering; joint thesis	Fall 2026	2
Giovanni Antonio de Cia POLITO	M.S., Energy and Nuclear Engineering; joint thesis	Fall 2026	9
Gabriele De Nicolo POLIMI	M.S., Nuclear Engineering; joint thesis	Spring 2026	17
Domenico Avarino POLIMI	M.S., Nuclear Engineering; joint thesis	Fall 2026	17
Matic Krašovic U-LJUBLJANA	M.S. student, University of Ljubljana, working at the Jožef Stefan Institute; co-advised	Spring 2026	1

Senior design and prospective students

- A Mines Capstone Design team of mechanical engineering and computer science students works on the autonomous control digital twin of Topic 9 (Fall 2026 to Spring 2027).
- Two prospective undergraduate researchers are preparing to join the group.

Alumni

Name	Degree and thesis	Year
Lorenzo Rizzetto POLIMI	M.S., Nuclear Engineering, co-advised. *Preliminary neutronics and thermo-micro-mechanical analyses of nuclear fuel from AP1000 reactor*	2025
Luca Piccinini POLIMI	M.S., Nuclear Engineering, co-advised. *Neutronic and structural characterisation of anti-corrosion and anti-permeation coatings for the DEMO WCLL breeding blanket*	2025