

Advanced Modeling & Simulation of Nuclear Systems for Nuclear Engineering, Particle Transport, and Reactor Physics Applications

This document includes a non-exhaustive list of high-level research topics of interest to Dr. Valerio Mascolino and his research group at the Colorado School of Mines. Goals and potential projects and subprojects (tasks) are listed as a way to keep track of current, potential, and future work. Certain subprojects/tasks may already be in progress, in which case the name of the person performing the work and the year they started is listed next to the task in parentheses in italics, e.g., (*V. Mascolino, Spring 2026*). All the projects, subprojects, and tasks are current at the time of writing and may vary and evolve as the projects mature. If interested in working on one or more of the listed projects and/or subprojects/tasks, please contact Dr. Mascolino or visit any of the online resources listed below in case you want to learn more about Dr. Mascolino's research.

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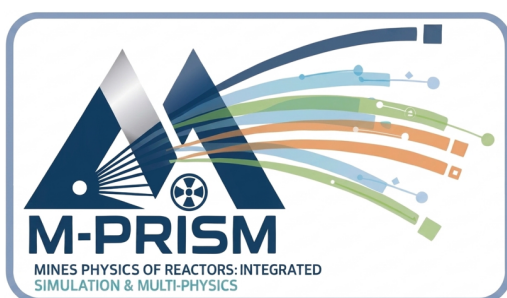
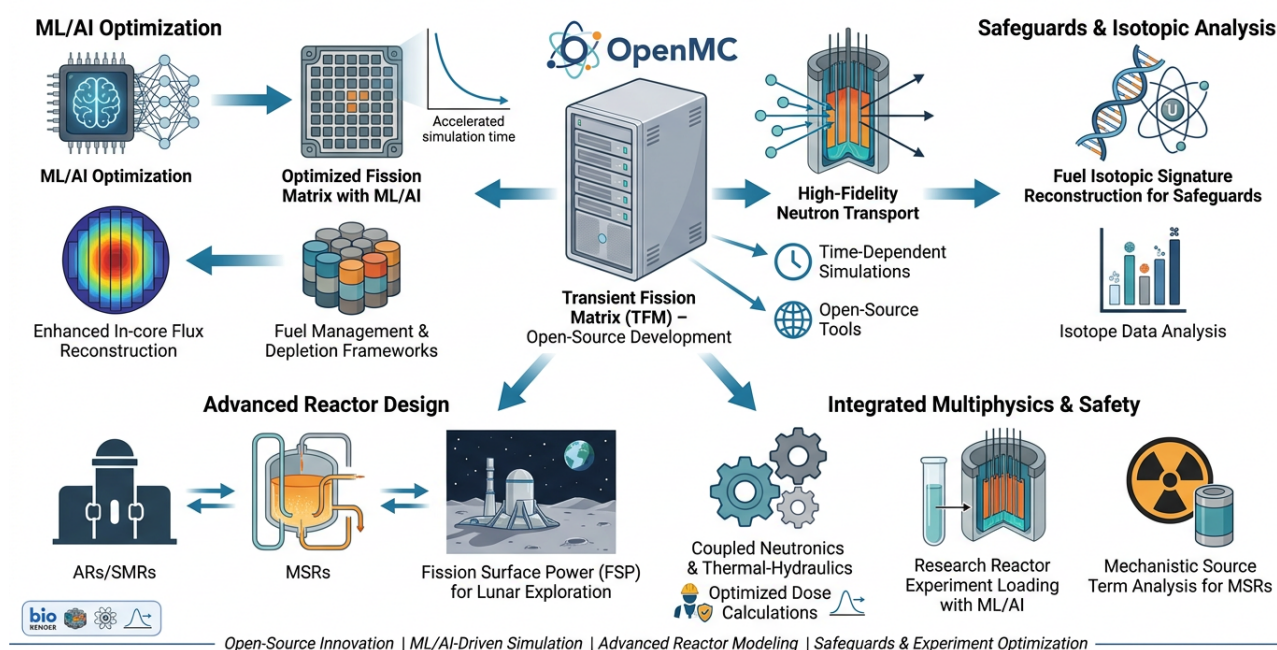
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Advanced Modeling & ML/AI-Driven Simulation for Next-Generation Nuclear Systems



1 Development of an open-source neutron transport kinetic module based on the transient fission matrix (TFM) method into OpenMC

Many Gen IV reactors and advanced reactors (ARs) in general are based on features that differ substantial from current commercial light water reactors (LWRs), for which decades of operational know-how and optimized diffusion codes are satisfactorily and extensively used for design, optimization, and operation. As such, advanced modeling and simulation techniques are required that are capable of modeling physical phenomena and engineering components that may occur in advanced reactors (e.g., molten and TRISO fuels, compact cores, fast reactors).

Traditional modeling approaches fall under two main categories: deterministic and stochastic methods [1.1]. Deterministic methods implement the direct numerical solution of the linear Boltzmann equation (LBE) for neutron transport. Several approximations are required (e.g., multi-group approximation, quadrature set, differencing schemes) to discretize the main independent variables of the phase space (space, direction, and energy). Stochastic method such as the Monte Carlo (MC) method [1.2], on the other hand, do not require any approximation and can provide high-fidelity results, however the statistical convergence of the results can be very slow when detailed 3-D solutions are required.

The above limitations are dramatically exacerbated if high-fidelity time-dependent reactor kinetics calculations are needed. Kinetics calculations are often performed using extremely simplified neutronics models, e.g., based on the point kinetics (PK) approach [1.3], that are often fine-tuned for deployed commercial LWRs. Load-following and accurate yet not overly conservative safety analyses based on high-fidelity time-dependent reactor kinetics are necessary for designing, optimizing, and licensing calculations of the next generation of nuclear reactors.

Hybrid methods have shown to combine the fast execution times of deterministic methods with the accuracy of Monte Carlo methods [1.4]. Of these methods, the transient fission matrix (TFM) method can be used to accurately and effectively simulate time-dependent neutron transport problems [1.5, 1.6]. There is potential for the TFM method to allow for fast and accurate 3-D time-dependent neutron transport calculations to be used for advanced reactor kinetics design, optimization, licensing, and operation, including advanced reactors digital twins (DTs), thanks to their fast execution times.

We will develop a TFM-based algorithm into the open-source Monte Carlo neutron transport software OpenMC [1.7]. Several subproject/tasks are identified to accomplish the overarching project objectives:

- Task 1.1. Perform a comparative analysis of different methods for generating steady-state fission matrix (FM) coefficients, i.e., fixed source and criticality OpenMC calculations.
([S. Balouza](#), started Spring 2026)
- Task 1.2. Implement TFM coefficients calculations directly into OpenMC, suitably modifying the tally filters.
([B. Alfaro](#), started Summer 2026)
- Task 1.3. Implement TFM equations into OpenMC for solution of reactor kinetics problems.
- Task 1.4. Integrated TFM solver into the US DOE MOOSE-based Cardinal solver package for coupling with thermal-hydraulics solvers.
([P. Boccardi](#), started Spring 2027)
- Task 1.5. Perform a systematic sensitivity analysis evaluating how spatial discretization, time-step size (Δt), streaming/advection formulations, and memory-kernel convolution approximations govern numerical stability and fidelity.
- Task 1.6. Develop coefficients interpolation and optimization strategies, for different parameters, e.g., temperatures, fuel enrichment, fuel burnup, control systems position, etc.
- Task 1.7. Develop methodologies to model advanced fuels, such as TRISO, using the TFM method.
- Task 1.8. Model TRIGA pulse experiments to validate the TFM algorithms.
- Task 1.9. Model C5G7 benchmark to validate 3-D time-dependent transport capabilities.
- Task 1.10. Optimize TFM algorithm to run in quasi real-time (~ 10 Hz).
- Task 1.11. Implement TFM algorithm into advanced reactor digital twin.
- Task 1.12. Develop methodologies to tackle traditionally difficult problems with TFM, e.g., geometry variations.

Project 1 References

- [1.1] George I Bell and Samuel Glasstone. *Nuclear reactor theory*. US Atomic Energy Commission, Washington, DC (United States), 1970.
- [1.2] Alireza Haghighat. *Monte Carlo Methods for Particle Transport*. 2nd. Pages: 213. CRC Press Taylor & Francis Group, 2020. ISBN: 0-429-58410-5.

- [1.3] James J Duderstadt and Louis J Hamilton. *Nuclear reactor analysis*. Vol. 650. 1976.
- [1.4] Alireza Haghighat, Katherine Royston, and William 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](https://doi.org/10.1016/j.anucene.2015.06.004). URL: <http://linkinghub.elsevier.com/retrieve/pii/S0306454915003151>.
- [1.5] Valerio Mascolino and Alireza 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](https://doi.org/10.1080/00295639.2023.2197844).
- [1.6] Valerio Mascolino et al. "Experimental Validation of the 3-D Neutron Kinetics Algorithm of RAPID using the JSI TRIGA Mark-II reactor". In: *Progress in Nuclear Energy* 176. November 2024 (2024), pp. 105391–105391. DOI: [10.1016/j.pnucene.2024.105391](https://doi.org/10.1016/j.pnucene.2024.105391).
- [1.7] Paul K Romano et al. "OpenMC: A state-of-the-art Monte Carlo code for research and development". In: *Annals of Nuclear Energy* 82 (2015), pp. 90–97.

2 Development and application of machine learning and artificial intelligence (ML/AI) methodologies for acceleration and optimization of the fission matrix (FM) method

The FM and TFM methods are characterized by extremely large datasets of response functions, in the form of the FM coefficients, that need to be calculated as a function of different system parameters (e.g., fuel and moderator temperature). This makes the FM method inherently "big data". As such, the FM method is a natural candidate for acceleration using machine learning and artificial intelligence (ML/AI) algorithms. Generating, storing, and solving highly dimensional TFM databases for operational and accidental core configurations poses significant computational bottlenecks. This project evaluates AI and ML algorithms to optimize database generation, sampling, interpolation, and time-step convolution integrals towards real-time digital twin execution (10 Hz).

Specifically, it is envisioned that suitable ML/AI algorithms can be used to encode physics that are traditionally lost in the process of interpolation and combination of various FM response functions for varying parameters. For example, the temperature distribution of non-fuel regions and - in general - regions in between the source and destination cells is typically lost in traditional interpolation methods for the FM coefficients [2.1, 2.2]. Recent preliminary work has shown that arbitrary random temperature distributions can be learnt by a neural network that is used to evaluate correction factors for the FM coefficients [2.3].

Three main macro areas are identified that could potentially benefit from the application of ML/AI algorithms within the FM method. In particular, ML/AI algorithms could be used to optimize the parameters range and spacing to optimize the number of precalculations required to populate a system's FM library. Secondly, ML/AI strategies could be used to perform better interpolation and combination of available FM database entries. One such example would be analyzing how to learn and account for arbitrary temperature distributions via machine-learned correction factors.

We will identify and evaluate different ML/AI algorithms, models, and strategies to optimize and accelerate the FM method. The following subprojects/tasks are identified to achieve the objectives listed for this project:

- Task 2.1. Develop a neural network to encode the dependency of fission matrix coefficients on arbitrary temperature distributions, including moderator temperature.
([D. Paine](#), started Fall 2025)
([TJ Baesman](#), started Spring 2026)
- Task 2.2. Evaluate different ML/AI strategies, including neural networks (NNs), convolutional neural networks (CNNs), generative adversarial neural networks (GANs), and physics-informed neural networks (PINNs) to understand which approaches are more suitable to optimize and accelerate the FM method.
([TJ Baesman](#), started Spring 2026)
- Task 2.3. Develop ML/AI-based strategies to optimize the generation of FM coefficients using adaptive learning and MC transport software.
- Task 2.4. Implement autoencoders to reduce system matrix dimensionality.

- Task 2.5. Utilize Recurrent Neural Networks (RNNs) and Long Short-Term Memory (LSTM) to learn time-dependent system evolutions and accelerate time-step convolution integrals during kinetic calculations towards real time execution (10 Hz).
- Task 2.6. Perform data-driven optimization and interpolation using CNNs and PINNs to model complex non-monotonic thermal feedback (e.g., fuel temperature, burnup, control rod movement) and reconstruct TFM configurations from ex-core detector measurements.
- Task 2.7. Develop ML/AI-based algorithms to accelerate the time-dependent solution of the TFM equations, with particular focus on the fast estimation of time-convolution integrals.
- 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).
- Task 2.9. Develop reinforcement and adaptive learning techniques to optimize sampling trajectories and minimize the required number of Monte Carlo calculations for pre-generating TFM databases.

Project 2 References

- [2.1] Maximiliano Dalinger and William Walters. "Analysis of Fission Matrix Databases with Nonuniform Fuel Temperature Profiles for Static and Transient Calculations". In: *Nuclear Science and Engineering* 199 (Apr. 2025). eprint: <https://doi.org/10.1080/00295639.2024.2328944>, S754–S764. ISSN: 0029-5639. DOI: [10.1080/00295639.2024.2328944](https://doi.org/10.1080/00295639.2024.2328944). URL: <https://doi.org/10.1080/00295639.2024.2328944> (visited on 11/20/2025).
- [2.2] Christian Vita et al. "A Correction Ratio Approach to Fission Matrix Interpolation in Systems with Large Temperature Differences". In: *Proceedings of the International Conference on the Physics of Reactors (PHYSOR 2026)*. Turin, Italy (accepted), Apr. 2026.
- [2.3] Daris Paine and Valerio Mascolino. "Neural Network-Enabled Thermal Feedback Modeling in the Fission Matrix Method". In: *Transaction of the American Nuclear Society*. Denver, CO, June 2026.

3 Validation and extension of a **in-core flux reconstruction** algorithm using a **machine-learned Green's function** and **ex-core detectors measurements**

Advanced reactor concepts are based on new designs that tend to be either very compact (e.g., small modular reactors (SMRs)) or have very harsh radiation and temperature environments. These conditions cause designers and developers to limit the number of in-core detectors as much as possible. Many AR designs aim at completely eliminating in-core detectors. This limitation makes monitoring and control of these advanced reactors extremely challenging, since it becomes near impossible to gather data from inside of the core. This is particularly important since, unlike commercial LWR, ARs and SMRs, operational data on these new designs is extremely scarce.

Recently, researchers at Argonne National Laboratory have developed a methodology based on machine-learned Green's functions to reconstruct in-core neutron flux in ARs and SMRs using measurements from ex-core detectors only [3.1, 3.2]. The methodology subverts the traditional "big data" paradigm of machine learning by enforcing a series of mathematical constraints associated with the properties of the Green's function. Instead of trying to fully encode mathematical properties and formulations into a NN, the method enforces the appropriate mathematical constraints and takes advantage of an ML/AI approach to learn a function that would be otherwise impossible to calculate analytically. Additionally, the Green's function is formulated using a Kirkchoff-Helmoltz approach such that all the variable terms are included in a source term and the measurement variations are always accounted for by opportunely lifting the problem's boundary conditions [3.2].

Preliminary validation of the algorithm has been performed at the Purdue Research Reactor (PUR-1), showing very good agreement between the Green's function method and measured detector data obtained utilizing an innovative quadrupole design [3.3]. The preliminary validation infers a single Green's function using a specific control rod configuration in the reactor and is able to reconstruct the in-core flux within a few percents in different configurations that it was not trained on. The preliminary validation did not account for temperature dependencies or nuclear fuel burnup.

The Argonne methodology relies on the use of the mono-energetic neutron diffusion equation [3.4] to derive the mathematical formulation and constraints embedded in the NN. The mono-energetic diffusion equation is an approximation of the linear Boltzmann equation for neutron transport [3.5]. Extensive literature shows that the mono-energetic diffusion approximation breaks down substantially in heterogeneous systems, leading to incorrect

direct estimation of the neutron flux. The validity of using the neutron diffusion in the Green's function inverse problem need to be verified. Additional validation cases should be devised to also study whether the Green's function changes substantially as a function of system parameters (e.g., burnup, fuel enrichment, fuel and moderator temperatures).

We will analyze the validity of the diffusion approximation in Argonne's Green's function method and develop a library of parameter-dependent effective nuclear data to assess the impact on the methodology's accuracy. Eventually, we will analyze the possibility of using more advanced approximations of the transport equation (e.g., P_N , S_N) as an alternative to monoenergetic diffusion. The following subprojects/tasks are identified to achieve the objectives listed for this project:

- Task 3.1. Develop a library of effective nuclear data, including neutron diffusion coefficients and one-group neutron cross-sections, for the JSI TRIGA Mark-II research reactor in Slovenia.
(*J. Lafore, started Spring 2026*)
- Task 3.2. Generate synthetic data based on the JSI TRIGA Mark-II reactor to validate the Green's function algorithm.
- Task 3.3. Develop a micro-reactor model to generate effective nuclear data and synthetic data for performing the algorithm's validation in an environment relevant for ARs and SMRs.
- Task 3.4. Investigate the mathematical formulation of a Green's function based on more advanced approximations of the neutron transport equation.

Project 3 References

- [3.1] Roberto Ponciroli et al. "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 (accepted), Apr. 2026.
- [3.2] Roberto 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](https://doi.org/10.48550/arXiv.2505.09766). URL: <http://arxiv.org/abs/2505.09766> (visited on 01/29/2026).
- [3.3] Haoyu Wang et al. "Measuring Neutron Flux Gradient: Design, Prototyping, and Deployment of a Novel Sensor at the PUR-1 Research Reactor". In: *Proceedings of the International Conference on the Physics of Reactors (PHYSOR 2026)*. Turin, Italy (accepted), Apr. 2026.
- [3.4] James J Duderstadt and Louis J Hamilton. *Nuclear reactor analysis*. Vol. 650. 1976.
- [3.5] George I Bell and Samuel Glasstone. *Nuclear reactor theory*. US Atomic Energy Commission, Washington, DC (United States), 1970.

4 Development of an open-source library of advanced and small modular reactor models using OpenMC

With the renewed interest in nuclear power, with particular focus on ARs and SMRs, substantial research and development efforts are undertaken by universities, national laboratories, and industry players directly to design, optimize, license, and build the next generation of nuclear reactors. Companies are often interested in guarding the specific characteristics of their designs to maintain an edge with respect to their competitors and ensure their technology arrives on the market and generates value. On the other hand, this understandable desire to protect their intellectual property often results in a fragmentation of available computational models as well as an often unnecessary gatekeeping of information from researchers and duplication of work. This effect is further exacerbated by the fact that different companies and laboratories utilize different nuclear software for their modeling. This software is often subject to export control regulations, which make it inherently difficult to access the information and virtually pointless to share the reactor models since these often can't be used by foreign nationals and/or foreign institutions.

As the United States, its allies, and the world in general envision a new renaissance of the pacific use of nuclear energy, scientists should all strive together to fast-track the path to market of the advanced nuclear reactor technologies. One such way is to have access to industry-relevant reactor models that are easily accessible and prevent different groups and organizations to duplicate work by producing redundant reactor models. This way, the precious and limited resources of research institutions and national laboratories can be spent performing novel work and leveraging previous work without any gatekeeping.

We will develop an open-source library of reactor models based on the Monte Carlo neutron transport software OpenMC by collecting existing ones, converting input decks that use non-easily accessible software, and developing those reactor models that are not easily available. The following subprojects/tasks are identified to achieve the objectives listed for this project:

- Task 4.1. Develop an heat-pipe micro-reactor model based on the HP-MR reactor from the Virtual Test Bed [4.1].
(*A. Cubriel, started Spring 2026*)
- Task 4.2. Develop a research reactor model based on the IAEA TECDOC core.
(*T. McCoy, started Spring 2026*)
- Task 4.3. Select a close-to-market molten salt reactor (MSR) design and develop an MSR model with thermal hydraulic coupling.
- Task 4.4. Develop a pebble-bed high temperature gas reactor (HTGR) reactor model.

Project 4 References

- [4.1] Nicolas E. Stauff et al. "High-Fidelity Multiphysics Modeling of a Heat Pipe Microreactor Using BlueCrab". In: *Nuclear Science and Engineering* (Aug. 2024), pp. 1–17. ISSN: 0029-5639. DOI: [10.1080/00295639.2024.2375175](https://doi.org/10.1080/00295639.2024.2375175). URL: <https://doi.org/10.1080/00295639.2024.2375175> (visited on 03/08/2026).

5 Neutronics and thermal-hydraulics modeling and simulation in support of the safety basis for the re-licensing of the USGS TRIGA Reactor (GSTR) at the Federal Center in Denver, CO.

Following an extended shutdown due to the failure of an aluminum-cladded fuel element, the United States Geological Survey TRIGA Reactor (GSTR) requires an updated Safety Analysis Report (SAR) with new fuel configurations for regulatory re-licensing. This project provides the comprehensive modeling, simulation, and computational safety basis required for NRC re-licensing and future experimental operations.

First, a centralized reference safety analysis document will be created to consolidate all geometric, material, operational, and historical data for current and future safety evaluations. Second, existing MCNP computational models will be updated for production regulatory safety basis calculations, while a novel OpenMC model embedded within a modular Python package will be established for agile model generation, research, and experimental campaign planning. Both models will undergo code-to-code cross-validation and benchmarking against experimental data. Finally, the thermal-hydraulics (T-H) safety basis will be re-evaluated, establishing modern T-H computational models to re-evaluate reactor thermal margins and operating limits.

We will establish updated MCNP and open-source OpenMC neutronics models alongside modern thermal-hydraulics safety models to support the USGS TRIGA Reactor re-licensing and safety analysis report. The following subprojects/tasks are identified to achieve the objectives listed for this project:

- Task 5.1. Compile a comprehensive reference safety analysis document detailing GSTR geometry, dimensions, operational parameters, material compositions, fresh/depleted fuel isotopics, and control rod configurations with primary references.
- Task 5.2. Update existing MCNP models of the GSTR core reflecting the updated fuel element loadout for regulatory safety basis submission.
- Task 5.3. Develop a novel, modular Python-packaged OpenMC model of the GSTR for flexible model generation, research applications, and experimental planning.
- Task 5.4. Perform code-to-code cross-validation between MCNP and OpenMC models and benchmark results against available historical and operational experimental data.
- Task 5.5. Review legacy GSTR thermal-hydraulics safety basis calculations, software, and underlying physical assumptions.
- Task 5.6. Develop and execute modern computational thermal-hydraulics models to evaluate core temperature distributions, cooling margins, and safety limits under re-licensing conditions.
- Task 5.7. Document the neutronics and thermal-hydraulics modeling methodologies and safety margins into formal technical reports supporting the GSTR SAR resubmission.

Project 5 References

6 Development of an **open-source fuel management and depletion framework** using the **ADDER** software

Fuel management and depletion modeling capabilities are essential for designing ARs and MSRs as well as existing research reactors, which often have feature cycles with complex fuel management operations (e.g., the Massachusetts Institute of Technology Reactor, MITR [6.1]). Accessible fuel management and depletion software is not widely available and is often dependent on export-controlled neutron transport software.

The Advanced Dimensional Depletion for Engineering of Reactors (ADDER) software, developed at Argonne National Laboratory, provides an advanced fuel management and depletion object-oriented framework that allows for advanced fuel cycle and operation strategies that are not easily achievable with any other state of the art software. [6.2] ADDER's API-like framework was developed to be neutron transport software-agnostic, allowing for the easy development of suitable interfaces with any neutron transport software. Under the umbrella of the U.S. High Performance Research Reactor (USHPRR) conversion project [6.3], MCNP5 [6.4] and MCNP6.2 [6.5] interfaces were developed, verified, and validated under Argonne's Research and Test Reactor group's NQA-1 compliant software quality assurance program. The software has gathered the attention of many stakeholders, among which the MITR and University of Missouri Research Reactor (MURR). However, the MCNP family of software is export-controlled, and not easily available to foreign national. In several cases, international collaborators of NNSA's reactor conversion efforts over the past several decades have expressed the need to adopt and utilize a fuel management and depletion software that does not rely on export controlled software, thus effectively leaving a window of opportunity open for the US's competitors to fill this gap and establish commercial and geopolitical relationships that the US could have filled otherwise.

Additionally, domestic research reactor operators often rely on fuel management and depletion strategies that are not centrally maintained and thus become difficult to adapt to the everchanging experimental and irradiation need of advanced research reactor facilities. Developing an open-source framework based on the neutron transport solver OpenMC to power ADDER would address all of the issues identified above. Additionally, ML/AI algorithm can be embedded into the fuel management software to perform target-based irradiation planning, fuel cycle length and design, and smart critical operation searches.

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

The following subprojects/tasks are identified to achieve the objectives listed for this project:

Task 6.1. Establish and analyze the IAEA TECDOC reactor core model using OpenMC.

(T. McCoy, started Spring 2026)

Task 6.2. Perform "rundown" depletion calculations of the IAEA TECDOC core using OpenMC depletion.

Task 6.3. Compare "rundown" all-OpenMC calculations to ADDER/MCNP calculations for basic code-to-code validation.

Task 6.4. Expand the library of research reactor cores models using OpenMC and converting existing MCNP models for better use within the ADDER framework.

Task 6.5. Extend ADDER fuel management and depletion framework to interface with OpenMC directly.

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

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

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

Project 6 References

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

[6.2] V Mascolino et al. *ADDER v1.1.0 User Guide*. English. Tech. rep. ANL/RTR/TM-24/25. Lemont, IL: Argonne National Laboratory, 2025. DOI: [10.2172/2564957](https://doi.org/10.2172/2564957). URL: <https://www.osti.gov/biblio/2564957>.

[6.3] Erik H Wilson and John G Stevens. "Progress on High-Density LEU Fuel Development and Reactor Conversion-I". In: *Transactions of the American Nuclear Society* 123.ANS Virtual Winter Meeting, November 16-19, 2020 (2020), pp. 311-314.

[6.4] Forrest B Brown et al. "MCNP version 5". In: *Trans. Am. Nucl. Soc* 87.273 (2002), pp. 2-3935.

[6.5] C J Werner. *MCNP Users Manual - Code Version 6.2*. Tech. rep. LANL, 2017.

7 Coupling of the fission matrix method with open-source computational fluid dynamics software NekRS and OpenMC within a multi-physics framework for analysis of molten salt reactors

Among the Gen IV advanced reactor (AR) concepts, one of the most investigated ones are molten salt reactor (MSR), including in SMR form. This innovative design, albeit initially demonstrated decades ago with the MSRE project [7.1], are making a comeback due to their flexibility in using different kinds of fuel, high burnups, continuous refueling, and/or thermal energy storage capabilities.

In liquid-fueled reactors, the fuel and coolant are the same fluid, breaking the spatial stationarity of traditional reactor kinetics. The advection of delayed neutron precursors (DNPs) shifts precursor concentrations downstream, altering kinetics, safety margins, and power distributions. This project extends a fast-running Transient Fission Matrix (TFM) solver to account for DNP transport within a moving fluid and embeds high-fidelity and reduced-order thermal-hydraulic models suitable for digital twin applications.

The transient fission matrix (TFM) method is uniquely positioned to account for the streaming of the delayed neutron precursors (DNPs) due its easy and explicit modeling of the DNP source [7.2, 7.3]. Open-source coupling of a TFM solver to i) CFD, ii) system analysis software, and iii) reduced-order model would enable the detailed kinetic analysis of MSRs with different characteristic computing times.

We will develop a fully open-source framework for the coupled neutronics/thermal-hydraulics steady-state and kinetic analysis of Molten Salt Reactors (MSRs). The following subprojects/tasks are identified to achieve the objectives listed for this project:

- Task 7.1. Development of an MSRE model using the CFD software NekRs [7.4].
(B. Harms, started Spring 2026)
- Task 7.2. Development OpenMC model of MSRE.
- Task 7.3. Coupling of NekRS and OpenMC using Cardinal [7.5].
- Task 7.4. Formulate and program the advection and decay terms for delayed neutron precursors into the TFM solver, accounting for flow profiles and precursor loss outside the active core.
- Task 7.5. Implement a flowing-fuel quasi-static algorithm within a system software framework (e.g., COBRA-TF), utilizing its subchannel T-H solver to generate realistic velocity and temperature fields.
- Task 7.6. Identify, evaluate, and integrate Reduced Order Models (ROM) and Machine Learning accelerators for thermal-hydraulic coupling to alleviate the computational overhead of history-dependent convolution integrals during transients to achieve calculation frequencies suitable for digital twins (10 Hz).
- Task 7.7. Establish low-latency data exchange within the OpenMC–Cardinal–NekRS coupling framework to evaluate streaming terms derived from Monte Carlo transport operators.
(P. Boccardi, started Fall 2026)
- Task 7.8. Perform code-to-code verification and validation between the extended TFM solver and open-source multiphysics codes (e.g., GeN-FOAM, MOOSE) across steady-state shifted precursor profiles and transient flow conditions.

Project 7 References

- [7.1] P. N. Haubenreich et al. *MSRE Design AND Operations Report. Part III. Nuclear Analysis*. English. Tech. rep. ORNL-TM-730. Oak Ridge National Lab., Tenn., Feb. 1964. DOI: [10.2172/4114686](https://www.osti.gov/biblio/4114686). URL: <https://www.osti.gov/biblio/4114686> (visited on 03/08/2026).
- [7.2] A. Laureau et al. "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* April (2015). ISSN: 9781510808041.
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- [7.5] A. J. Novak et al. "Coupled Monte Carlo and thermal-fluid modeling of high temperature gas reactors using Cardinal". In: *Annals of Nuclear Energy* 177 (Nov. 2022), p. 109310. ISSN: 0306-4549. DOI: [10.1016/j.anucene.2022.109310](https://doi.org/10.1016/j.anucene.2022.109310). URL: <https://www.sciencedirect.com/science/article/pii/S0306454922003450> (visited on 01/26/2026).

8 Design, analysis, and neutronics optimization of the Virginia Research and Education Reactor Facility (POLARIS)

The Virginia Innovative Nuclear (VIN) hub has recently obtained funding from the government of Virginia to design a next-generation nuclear research reactor [8.1]. The facility aims at boosting the education of the next generation nuclear workforce in a state with a strong presence of nuclear industry (e.g., Framatome, Dominion). At the same time the proposed Virginia Research and Education Reactor Facility (VA-RERF) [8.2] aims at becoming one of the most accessible and instrumented research reactors in the country, to facilitate nuclear engineering students education as well as development of reactor physics experiments that are paramount to designing, licensing, and operating the next generation nuclear reactors. In particular, VA-RERF is envisioned to be used for benchmarking and experimental validation of the high-fidelity reactor physics software needed to model ARs and SMRs. Finally, VA-RERF is being designed with a large cavity for spectrum-tailored neutron irradiation by means of neutron energy filters and modulators [8.3]. The latter should allow for high-flux irradiation at various neutron energies for maximum experiment flexibility.

We will support the design of POLARIS by designing the flux trap buffer zone and supporting with the criticality analysis of the system. The following subprojects/tasks are identified to achieve the objectives listed for this project:

- Task 8.1. Evaluate different isotopes and materials for designing a buffer zone for spectrum tailoring in the flux trap.
([J. Reitz](#), started Fall 2025)
- Task 8.2. Model different buffer zone configurations using the OpenMC software.
([J. Reitz](#), started Summer 2026)
- Task 8.3. Analyze performance of flux trap with different buffer zones using the Detector Response Function (DRF) methodology.
([J. Reitz](#), started Fall 2026)
- Task 8.4. Optimize flux intensity, spectrum, and irradiation performance as a function of different shield materials.
- Task 8.5. Perform high-fidelity OpenMC Monte Carlo simulations using ENDF/B-VIII.1 cross sections and fine energy bins to calculate 3D volumetric heat generation rates and radiation damage fields across candidate buffer zone materials (including Be-family, D₂O, Graphite, B₄C, Cd, and Hf).
- Task 8.6. Develop custom Abaqus user subroutines (e.g., UTEMP, HETVAL, USDFLD) to map 3D volumetric heating and radiation-damage variables from OpenMC directly onto finite-element meshes.
Collaboration with Politecnico di Milano
- Task 8.7. Perform non-linear coupled thermo-mechanical FEA using Abaqus/Standard and Abaqus/Explicit to evaluate temperature-dependent stress fields, thermal expansion, clearance evolution, dynamic frictional contact, and low-cycle fatigue (LCF) during buffer sheet movement. *Collaboration with Politecnico di Milano*
- Task 8.8. Assess geometric sizing limits, actuation force requirements, and potential thermal jamming mechanisms for both solid plate and liquid-channeled reconfigurable buffer designs. *Collaboration with Politecnico di Milano*

Project 8 References

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9 Optimization and variance reduction of Monte Carlo dose calculations for a fission surface power (FSP) reactor for lunar exploration

Nuclear reactor, thanks to its reliability and large energy density, is one of the top candidates to power space missions, including lunar surface missions. NASA's fission surface power (FSP) initiative aims at powering manned moon operations with fission nuclear reactors. The nuclear industry, e.g., Westinghouse Electric Company, have been tasked by DoE to develop the envisioned reactor technology.

One of the relevant design metrics is the shielding of the FSP to minimize doses to the moon colonies that the reactor is meant to power. Since one of the key characteristic of space nuclear reactors is the need to severely limit their payload, traditional shielding approaches are unfeasible since the high Z materials that are needed to stop gamma radiation also dramatically increase the mass of the device overall. High-fidelity modeling of expected doses at human installation is hence of the utmost importance to reduce overly-conservative safety factor that would make the shielding requirements increase, together with the mission's weight.

The above shielding problem is particularly difficult to solve using deterministic or stochastic neutron transport software since this is a deep penetration problem within the very low-absorbing lunar atmosphere, so much so that most of the dose is expected to be due to scattering and transport through the lunar regolith [9.1]. Under these conditions, both Monte Carlo and deterministic software suffer from either painstakingly slow statistical convergence or extremely pronounced ray effects [9.2].

In order to solve the above modeling issue, Monte Carlo software can be used with variance reduction techniques based on the use of approximate deterministic neutron importance functions, such as the CADIS and FW-CADIS methodologies [9.3, 9.4].

We will develop an FSP model in OpenMC with weight-window variance reduction based on deterministic importance function calculations using PENTRAN [9.5]. The following subprojects/tasks are identified to achieve the objectives listed for this project:

- Task 9.1. Develop FSP model using OpenMC based on existing MCNP model developed by collaborators at the University of Utah.
([S. Olivares](#), started Summer 2026)
- Task 9.2. Evaluate current OpenMC weight-window capabilities and assess main differences with respect to OpenMC.
- Task 9.3. Evaluate current FW-CADIS implementation using deterministic importance function calculation capabilities embedded into OpenMC.
- Task 9.4. Evaluate variance reduction methodologies and dose calculation results across variable parameters (shield thickness, material selection, distance, and regolith composition uncertainties).
([G. A. De Cia](#), started Fall 2026)
- Task 9.5. Develop angular-dependent weight window-based variance reduction into OpenMC to further improve the convergence of the FSP dose calculation problem.
- Task 9.6. Assess the above capabilities and configurations in terms of accuracy of the calculated dose rates and figures of merit.

Project 9 References

- [9.1] Meng-Jen Wang et al. "Solving Large Scale Fission Surface Power Reactor Shielding Problems". In: *Proceedings of the International Conference on the Physics of Reactors (PHYSOR 2026)*. Turin, Italy (accepted), Apr. 2026.
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10 Development of an **agentic AI digital twin and autonomous control framework** for lunar fission surface power microreactors

As NASA's Artemis program advances toward establishing long-term human lunar infrastructure, continuous power generation during the 14-day lunar night necessitates micro-fission surface reactors. Signal transmission latency (2.5 to 3.0 seconds round-trip) and periodic line-of-sight communication blackouts preclude real-time Earth-based human intervention during rapid thermal transients or emergency operational shifts.

To address these operational constraints, this project develops an agentic digital twin and autonomous AI control framework tailored for a nominal 40 kWe Lunar Fission Surface Power (FSP) system. By coupling specialized AI architectures (including LLMs, multi-agent frameworks, and machine learning models) with reduced-order multiphysics engines (point-kinetics, 1-D heat transfer, and power conversion loops), this work establishes a digital benchmark to evaluate autonomous decision-making and reactor control under extreme space operational envelopes.

We will develop an agentic multi-agent AI digital twin coupled with reduced-order physics engines to enable autonomous control and anomaly response for lunar fission surface power systems. The following subprojects/tasks are identified to achieve the objectives listed for this project:

- Task 10.1. Perform a structured review and benchmarking of legacy space reactors (SNAP-10A, Kilopower) and commercial NASA Phase-1 FSP concepts (Lockheed Martin, Westinghouse, IX).
- Task 10.2. Define the system requirements and operational envelope for a 40 kWe FSP microreactor, mapping control mechanisms across the core, thermal-hydraulics, thermo-electric generators, and heat sinks.
- Task 10.3. Formulate the agentic digital twin architecture, determining the optimal distribution of LLM agents, specialized Python-based machine learning models (e.g., PyTorch, TensorFlow), and agentic software frameworks.
- Task 10.4. Develop simplified, fast-running physics engines—including point-kinetics, 1-D heat transfer modules, and power conversion curves—to simulate space-reactor transient behavior.
- Task 10.5. Couple the agentic AI control framework with the reduced-order physics engines to simulate autonomous load-following, transient response, and loss-of-communication operation.
- Task 10.6. Conduct benchmark evaluations of the multi-agent control framework to assess stability, decision-making efficiency, and safety margins during nominal and emergency space power operational shifts.

Project 10 References

11 **Reconstruction of fuel isotopic signatures** using machine learning algorithms and the fission matrix method for **safeguard applications**

The FM and TFM methods (see projects above) are capable of performing fast and high-accuracy neutron transport calculations capable of reconstructing the fission neutron source in a nuclear system, such as a nuclear reactor. The FM method can be altered to evaluate the specific spectrum of the fission source, dependent on the specific fuel isotopics, enrichment, and burnup.

Since the FM equation can be solved in seconds, inference methodologies can then be very effectively used to reconstruct the neutron flux inside of a reactor using a handful of detector data available to inspectors from out-of-core measurements [11.1]. The inference methodologies can be significantly enhanced and sped up by ML/AI surrogate models that can encode a large number of parameters (e.g., specific fuel isotopics, fuel burnup) into a reference configuration based on the fission matrix. Since eventually the neutron physics are captured within the FM coefficients rather than relying on the surrogate model itself, this method lends itself to physics-based and physics-informed ML/AI methodologies. Additionally, since the FM method is inherently reliant on big data, the use of ML/AI algorithm and surrogate models is extremely synergistic for the envisioned safeguard application. In order to evaluate the out-of-core measurements the FM methodology is coupled with the response based detector response function (DRF) method [11.2].

We will develop a novel methodology for fuel isotopic inference based on detector responses using the fission matrix method and ML/AI surrogate models. The following subprojects/tasks are identified to achieve the objectives listed for this project:

- Task 11.1. Select a suitable model for testing the methodology, e.g., the JSI TRIGA Mark-II reactor.
(*Ethan Mohr, started Summer 2026*)
- Task 11.2. Extend the FM methodology to allow for various relevant energy groups in the coefficients, based on the spectra of the main fissile isotopes.

- Task 11.3. Develop inference methodologies to reconstruct case-specific fission matrix based on synthetic detector data.
- Task 11.4. Enhance inference methodologies using ML/AI surrogate models and algorithms.
- Task 11.5. Generate an exhaustive library of FM coefficients for different fuel isotopics, burnups, cooling times, and enrichment.
- Task 11.6. Validate the methodology using arbitrary combinations of database entries.

Project 11 References

- [11.1] Valerio Mascolino, Alireza Haghighat, and Nathan J. Roskoff. "External SNF Cask Dose Calculation Using RAPID". In: *Proceedings of the Advances in Nuclear Nonproliferation Technology & Policy Conference (ANTPC 2016)*. Santa Fe, NM, Sep. 25-30, 2016.
- [11.2] Meng-Jen Wang. "Development of a Novel Detector Response Formulation and Algorithm in RAPID and its Benchmarking". PhD thesis. Virginia Polytechnic Institute and State University, 2019. URL: <http://hdl.handle.net/10919/97247>.

12 Optimization of research reactor experiment loading using machine learning algorithms and the fission matrix method

High-performance research reactors are often required to have very aggressive refueling schedules that do not allow them to use high-fidelity time-consuming calculations to optimize the experiment location and loadout, and can lead to sub-optimal irradiation throughput or mismatched criticality worths for the experiments.

The FM and DRF methods rely on extensive pre-calculations of response functions and coefficients so that an arbitrary system configuration can be modeled in very short turnaround times (seconds to minutes) without the need to recompute time-consuming and expensive state-of-the-art neutron transport calculations, while retaining their level of accuracy and fidelity. Adopting an approach based on response-function methodologies, such as the FM, could enable very accurate irradiation and experiment design and optimization while maintaining the same setup time of the much coarser models that are normally used to support loading at the reactors.

Additionally, this large amount of data can benefit from being encoded into ML/AI algorithms to better extract the implicit dependencies of specific target's metrics on the core loadings's input parameters, e.g., burnup and location of specific fuel elements. These machine learning algorithms could then be used to develop target-based agentic approaches that would devise the optimal reactor core configuration based on available fuel elements to achieve a certain objective, e.g., a certain activation of a sample, a desired cycle length or peak neutron flux.

We will develop FM-based ML/AI-enhanced algorithms for target-based design and optimization of irradiation campaigns and experiment loadings at high performance research reactors. The following subprojects/tasks are identified to achieve the objectives listed for this project:

- Task 12.1. Development of suitable OpenMC research reactor models (e.g., based on MURR or MITR).
- Task 12.2. Development of a library of experiment design, loadouts, and performance metrics.
- Task 12.3. Development of FM-based detector response calculations for evaluating the identified performance metrics.
- Task 12.4. Development of FM coefficients library based on fuel burnup of available fuel inventory at research reactor.
- Task 12.5. Evaluation of ML/AI algorithms for the inference of irradiation performance metrics based on the FM method.
- Task 12.6. Implementation of selected ML/AI algorithms to obtain surrogate model capable of accurately predicting irradiation metrics based on fuel cycle information (e.g., specific fuel elements used for core loading).
- Task 12.7. Validation of the developed algorithms based on generated synthetic data and available experimental data from USHPRR.

Project 12 References

13 Development of **adaptive multiphysics coupling algorithms** for advanced nuclear reactor analysis

Effective deployment of advanced reactor concepts requires bridging high-fidelity modeling tools with fast-running design codes to reduce computational burden. This project integrates three neutronics solvers (OpenMC, FRENETIC, and Fission Matrix) with two thermal-hydraulics (TH) engines (NekRS and FRENETIC's internal subchannel module) into an adaptive multi-fidelity workflow.

We will build an adaptive multi-physics coupling ecosystem leveraging Cardinal, Python wrappers, and f2py interfaces to dynamically optimize calculation speed and precision across solver hierarchies. The following subprojects/tasks are identified to achieve the objectives listed for this project:

- Task 13.1. Develop Python wrappers using f2py to interface legacy Fortran routines in FRENETIC with modern Python/C++ software stacks.
- Task 13.2. Implement Python/C++ wrappers for the Fission Matrix engine to facilitate multi-code integration.
- Task 13.3. Utilize Cardinal to orchestrate spatial mesh data transfers (e.g., power density, fluid temperatures, densities, velocities) between neutronics and TH solvers (including OpenMC and NekRS).
- Task 13.4. Design and test adaptive coupling algorithms (e.g., adaptive relaxation and Picard iterations) that dynamically adjust data exchange frequencies, spatial mappings, and relaxation factors based on local convergence criteria.
- Task 13.5. Conduct multi-fidelity benchmarking on advanced reactor configurations (e.g., NEA LFR small modular fast reactor benchmark), comparing high-fidelity reference solutions (OpenMC-NekRS) with fast-running options (FM-FRENETIC TH, FRENETIC NE-TH).

Project 13 References

14 **Generalized matrix-based eigenvalue methods** for neutron transport applications

The standard fission matrix maps fission neutron production to derive multiplication eigenvalues (k_{eff}). However, alternative transport eigenvalue formulations—such as collision, time (α), and density (γ) eigenvalues—provide valuable insights for core stability, spatial coupling, and transient diagnostics. This research generalizes the response matrix methodology beyond multiplication eigenvalue problems using an integral transport framework.

We will establish a generalized integral transport framework to construct modified source-response matrices for alternative transport eigenvalue formulations. The following subprojects/tasks are identified to achieve the objectives listed for this project:

- Task 14.1. Reformulate collision, time, and density eigenvalue problems within a unified integral-transport operator framework.
- Task 14.2. Formulate modified source-response matrix constructs replacing standard fission-to-fission response with target-specific mechanisms.
- Task 14.3. Analyze spectral consistency and mathematical conditions under which generalized matrix eigenvalues approximate continuous/discretized transport operators.
- Task 14.4. Implement numerical estimators using Monte Carlo and deterministic transport calculations and evaluate them on benchmark problems.
- Task 14.5. Assess physical applications for core design, spatial coupling metrics, transient characterization, and reactor stability monitoring.

Project 14 References

15 **Enhancing techno-economic microreactor optimization via geometry-adaptive fission matrix methods in the MOUSE framework**

Nuclear microreactors, such as Heat Pipe Microreactors (HP-MRs), offer rapidly deployable and resilient power solutions for remote regions but face prominent economic hurdles due to diseconomies of scale. To ensure economic

viability, the open-source, Python-based Microreactor Optimization Using Simulation and Economics (MOUSE) tool couples reactor design calculations (utilizing the OpenMC Monte Carlo code) directly with bottom-up cost estimations. While MOUSE currently utilizes corrected 2D models to balance runtime and accuracy during parametric sweeps, performing 3D transport-based calculations for compact cores makes multi-variable optimizations computationally prohibitive.

Traditional fission matrix (FM) approaches rely on fixed spatial grids, making them notoriously challenging to adapt when physical geometries change (e.g., fuel size, pitch, and cell geometry modifications). This project integrates geometry-adaptive fission matrix methods and physics-informed machine learning algorithms into the MOUSE framework to bypass time-consuming stochastic transport iterations, delivering near-instantaneous evaluations of downstream techno-economic impacts (e.g., levelized cost of energy [LCOE] and total capital investment) while maintaining 3-D transport-level accuracy.

We will develop a geometry-adaptive fission matrix methodology integrated into the MOUSE framework using physics-informed machine learning to enable fast, 3-D, transport-accurate techno-economic optimization of microreactors. The following subprojects/tasks are identified to achieve the objectives listed for this project:

- Task 15.1. Integrate standard fission matrix (FM) response calculation capabilities directly within the MOUSE framework for an HP-MR benchmark concept.
- Task 15.2. Develop mathematical formulations to handle geometric variations (e.g., core fuel dimensions, lattice pitch, cell geometries) within the response-matrix framework.
- Task 15.3. Formulate and train physics-informed machine learning algorithms (e.g., deep neural networks) to dynamically reconstruct, scale, and adjust fission matrices under varying spatial configurations.
- Task 15.4. Link the geometry-adaptive FM solver with MOUSE's bottom-up cost estimation modules to compute near-real-time techno-economic metrics, including levelized cost of energy (LCOE) and capital expenditures.
- Task 15.5. Conduct large-scale design optimization and uncertainty propagation studies to validate the coupled framework against high-fidelity 3-D OpenMC reference calculations.

Project 15 References

16 Integrated neutronic and structural analysis of radiation damage in aluminum fuel cladding of the TRIGA research reactor at LENA

Estimating the residual load-bearing capacity of aged structural components is crucial for aging management and life extension of research reactors. Intense neutron irradiation alters the physical and mechanical performance of fuel cladding over operational history. This project develops an integrated numerical methodology linking reactor operation history to the evolution of structural material degradation in aluminum fuel cladding at the TRIGA Mark II reactor at the Laboratory of Applied Nuclear Energy (LENA) in Pavia.

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. The following subprojects/tasks are identified to achieve the objectives listed for this project:

- Task 16.1. Reconstruct historical reactor irradiation conditions from operational logs (power levels and operation duration) to determine spatial distributions of neutron flux, fluence, and radiation damage metrics (e.g., displacements per atom, DPA).
 - Task 16.2. Develop finite element structural models incorporating irradiation-dependent material properties, including Young's modulus, yield functions, and fracture energy.
 - Task 16.3. Calibrate and validate the coupled neutronic-structural numerical models against the extensive experimental and operational dataset available at the LENA facility.
 - Task 16.4. 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.
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Project 16 References

17 Development of open-source tools for mechanistic source term analysis of molten salt reactors (MSRs) using OpenMC.

The transport of fission products and radioactive species in general in liquid fueled MSRs constitutes a key feature of this system that required the design of suitable chemistry control systems in the primary loop. An example of these systems is off-gas removal tanks, which preliminary results have shown can have significant impact on radioisotopes concentrations in the salt [17.1]. Since these fission products are generated via fission within the flowing salt circulating in the reactor core, it becomes necessary to accurately predict and evaluate the rate of productions in different regions of the reactor based on the distribution of fission events. These change with time and with the composition of the fuel, as well as the distribution of the neutron flux and chemical salt species within the primary loop and reactor core.

In order to evaluate the consequences of MSRs accidents, it is necessary to accurately model the mechanistic radiation source term due to the radioactive isotopes (e.g., fission products) in the fuel. The generation and transport of fission products are governed by the phenomena listed above. The accurate prediction of these phenomena is extremely complex and requires high-fidelity advanced simulation tools in order to enable the design and safety calculations required for commercializations of advanced MSRs. Rigorous tools that can account for the complex multiphysics phenomena necessary to correctly characterize a detailed mechanistic source term analysis are not widely available to developers and researchers. On the other hand, these calculations are needed to ensure the safety of these systems, as well as to evaluate their safeguards.

With our collaborators at the University of Texas-Austin, **we will develop open-source tools directly into the OpenMC Monte Carlo software for accurate mechanistic source term analysis in molten salt reactors**. The following subprojects/tasks are identified to achieve the objectives listed for this project:

- Task 17.1. Develop parallel in-material depletion of nuclear fuel using OpenMC to speed up MSR depletion calculations.
- Task 17.2. Implement source and transfer rates terms for the depletion equations to account for continuous removal and insertion of chemical species within the salt either by concentration difference or at a fixed rate.
- Task 17.3. Track the inventory of MSR loop components, e.g., off-gas systems, used for removal of fission products and radioisotopes from the main salt loop.
- Task 17.4. Use of a multi-region model of the MSRE [17.2] to validate the developed MSR loop capabilities.

Project 17 References

- [17.1] Alexander G Macris et al. "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 (accepted), Apr. 2026.
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