NEAMS
Nuclear Energy Advanced Modeling and Simulation
NEWSLETTER
Dec | 2025

A MESSAGE FROM NEAMS LEADERSHIP

Wrapping up a successful FY25 and Looking Towards the Future

The Nuclear Energy Advanced Modeling and Simulation (NEAMS) program had a productive year, completing 23 major (Level 2) milestones and numerous supporting lower-level milestones across five technical areas: Fuel Performance, Structural Materials and Chemistry, Reactor Physics, Thermal Fluids, and Multiphysics Applications. These milestones covered simulation technologies relevant to all major reactor types being pursued by U.S. industry, including the current LWR fleet and the advanced reactors now being rapidly developed for deployment. In addition to the technical achievements, we are also very proud of the NEAMS contributors who have received awards and recognition throughout the year. This includes the following:

- ANS Untermyer & Cisler Reactor Technology Medal: Rui Hu and Elia Merzari
- ANS 40 Under 40: Mauricio Tano and Dillon Shaver
- ANS Landis Young Member Engineering Award: Nathan Capps
- ANS Fellow: Former National Technical Director, Chris Stanek
- ANS THD Best Paper Award: Joshua Hansel and Elia Merzari

The predictive modeling and simulation tools developed by the program enable innovation in the design, development, licensing, deployment, and operation of nuclear reactors. To achieve this vision, it is essential that the program remains well-connected with stakeholders, including other U.S. Department of Energy (DOE) programs, industry partners, regulators, and university research initiatives.

In FY25, we organized several meetings to facilitate these connections, including five virtual reviews by reactor type and meetings with currently funded university projects. Attendance ranged from 70 to just over 110 participants in the program review meetings. This included NEAMS contributors as well as representatives from the U.S. Nuclear Regulatory Commission (NRC), industry, other DOE programs, and academia, with a significant portion representing key industry stakeholders and the NRC.

These meetings proved highly valuable to the program, and we would like to express our appreciation for the broad engagement we received. The slides presented at the review meetings are available on our website at neams.inl.gov under the resources tab. Specific feedback received during the annual review meetings on topics such as validation, software quality assurance, and the role of AI/ML in our research has been incorporated into our updated five-year research plan. We are working on transforming the technical research plan into a document suitable for broad distribution. Please stay tuned for updates in 2026.

Throughout the last year, the NEAMS program devoted significant effort to revising the five-year research plan and aligning research and development efforts to ensure that we are well-positioned to address stakeholder needs and support the nuclear energy goals of the nation, as outlined in the presidential executive orders released over the summer. It is exciting to witness the rapid development and initial deployment of advanced reactors, as well as power uprates, restarts, and new-build announcements for light water reactors. Our research plan is designed to support these efforts in the short and long-term. We welcome feedback on how our research and development efforts can best complement and strengthen these initiatives. Please do not hesitate to reach out to the program leadership or technical points of contact.

The FY26 NEAMS Annual Review Meeting will be held February 10–11, 2026, at Idaho National Laboratory. Both in-person and virtual participation options are available. This year's meeting will feature panel-style discussions with NEAMS experts and other participants. Additional information has been provided in a separate communication. If you did not receive this announcement and would like to participate in the Annual Review Meeting, please reach out to the program leadership. We look forward to your participation.

In the sections below, this newsletter highlights one key FY25 milestone within each NEAMS technical area. Naturally, this represents only a small sample of our work. To provide an idea of the full scope covered, the list at the end of the newsletter summarizes all FY25 major milestones by technical area. For those interested in exploring the technical details further, we will soon provide direct links on the NEAMS website at neams.inl.gov to full journal publications and reports. Please reach out to us if you have any questions or want to learn more.

-David Andersson, David Henderson & Tanju Sofu

Point of contact: David Andersson (andersson@lanl.gov)

TECHNICAL AREA HIGHLIGHTS

Fuel Performance

The Fuel Performance technical area has worked closely with DOE collaborators (e.g., the Advanced Fuel Campaign, AFC) and industry stakeholders to develop simulation capabilities that enable the extension of burnup limits and power uprates of existing light-water reactors. This work covers both UO_2 fuel pellets and cladding materials. The technical area also develops fuel performance models for fuel types considered for advanced reactors, such as TRISO, metallic, and advanced ceramic (e.g., UN) fuels.

In this newsletter, we highlight progress in developing advanced modeling capabilities for zirconium cladding performance under scenarios of interest to our LWR stakeholders. The models developed and applied by NEAMS address critical gaps in areas where few or no predictive tools currently exist and where empirical models carry more uncertainty than desired. The new tools presented below are the first of their kind in being able to predict the transient accident performance of cladding using a lower-length-scale model that has not been fitted to any integral testing data. The material models reviewed here will next be incorporated into multiphysics workflows and analyses that include coupling with reactor physics and thermal-hydraulics capabilities developed by the DOE.

Advancing Predictive Modeling for Zircaloy Cladding

The cladding team in the fuel performance technical area demonstrated progress on micro-structure informed predictive modeling of Zircaloy-4 and its response under transients and accident scenarios in light water reactors. The recent work consolidates prior advances into a unified, high-fidelity modeling framework that links mechanical response, irradiation effects, and temperature-driven property evolution. The outcome is a powerful suite of computational tools poised to accelerate qualification and support high-priority efforts by industry to extend fuel burnup limits and perform power uprates. The application to Zircaloy-4 demonstrates a methodology that could be deployed by industry stakeholders to their latest proprietary cladding materials. The new capability relies on three components outlined below.

Micromechanical modeling: A refined crystal plasticity model provides more accurate predictions of yield strength and creep behavior, while reducing empirical fitting parameters for improved robustness and transferability. Crucially, the model incorporates the effects of irradiation-induced defects, both vacancy- and interstitial-type dislocation loops, offering a more realistic depiction of how reactor conditions alter material strength. New analyses also shed light on time-at-temperature softening, identifying dislocation recovery and precipitate dissolution as key contributors to thermal degradation. These insights highlight the need to model recrystallization for long-term performance prediction.

Data-driven surrogates for reactor-scale simulations: The crystal plasticity models are expensive to evaluate in finite element solvers at the fuel rod scale. Computational gain is accomplished by the creation of a LAROMance surrogate model, which is an efficient, data-driven representation of Zircaloy-4's elasto-viscoplastic behavior. Trained on crystal plasticity simulations and implemented within the MOOSE and BISON frameworks, LAROMance accurately reproduced separate-effects and integral test data from the REBEKA and Halden IFA-650.2 experiments. The surrogate predicted cladding burst timing and deformation under LOCA conditions within seconds of experimental results, see Figure 1 and 2, demonstrating both accuracy and computational efficiency suitable for licensing-relevant applications.

Efficient modeling of irradiation effects through accurate reduced order models: Complementing the surrogate framework, the new LAMIE module (Los Alamos Material Model for Irradiation Extremes) provides a reduced order, spatially resolved cluster dynamics approach. LAMIE captures the evolution of irradiation-induced defects with up to 90% reduction in computational cost compared to high-fidelity models like those in the XOLOTL cluster dynamics code, while maintaining accuracy within 10%. It efficiently represents microstructural features such as vacancies, interstitials, and dislocation loops, enabling realistic predictions of irradiation growth and texture-dependent deformation.

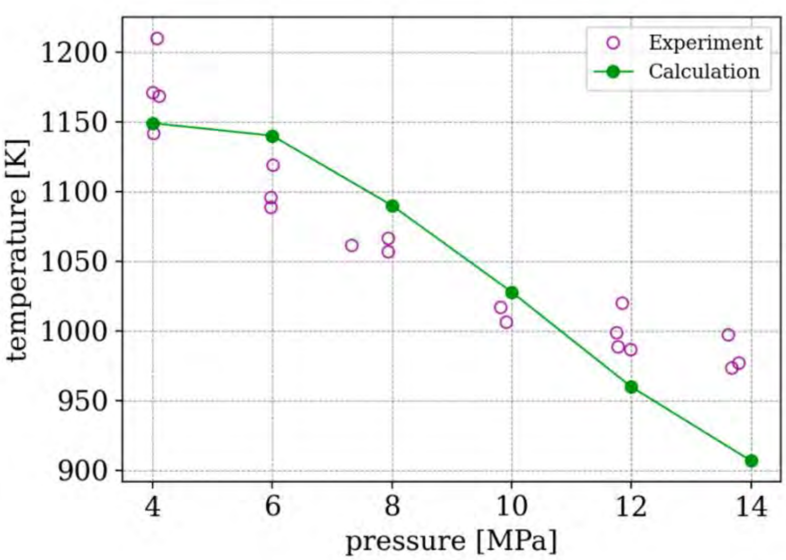

Figure 1: Summary of REBEKA test-cases calculated in BISON using the Zircaloy surrogate model alongside experimental observations [1].

Future efforts will extend the cladding modeling framework to include recrystallization and grain growth, as well as coupling with creep and stress relaxation for service-relevant predictions.

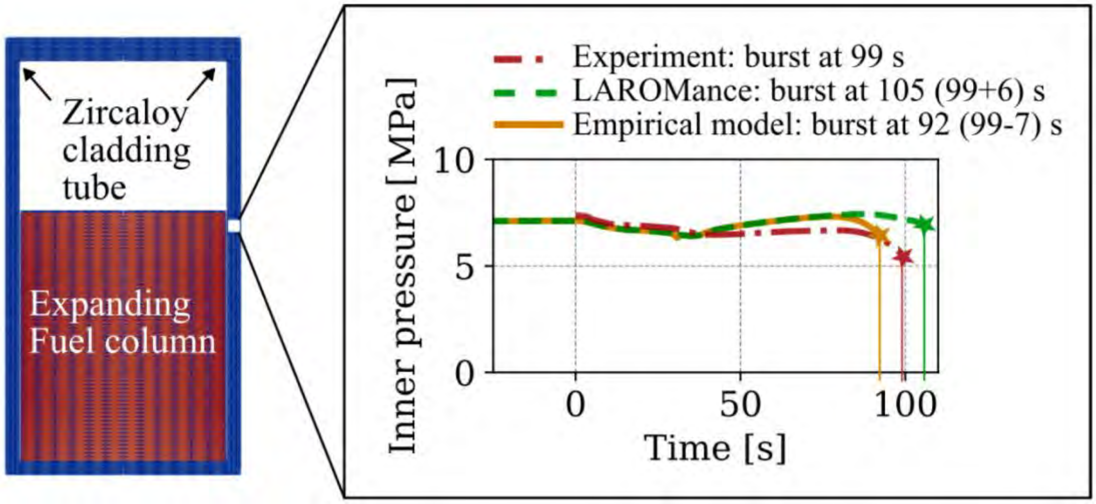

Figure 2: Pressure evolution traces from IFA 650.2 LOCA experiment [2] and BISON simulations with only the plastic instability strain rate threshold used to determine the moment of burst.

Technical point of contact: Laurent Capolungo (laurent@lanl.gov).

Additional reading: 1. A. Rovinelli, A.P. Ruybalid, M.-J. Chen, L. Capolungo, M2MS-25LA0201013: Complete initial integration of constitutive modeling approach into MOOSE for predictions of the effect of temperature transients on mechanical response of Zircaloy under LOCA-like conditions using BISON (2025), LA-UR-25-30454; and **2.** R. T. Sweet, J. I. Espersen, K.A. Gamble, S. R. Novascone, M3MS-25IN0201035: Demonstrate zirconium cladding ROM capability on multiple LWR LOCA test problems (2025), INL/RPT-25-06268.

References

[1]: F.J. Erbacher et al. "Burst criterion of Zircaloy fuel claddings in a loss-of-coolant accident". In: Zirconium in the Nuclear Industry, Fifth Conference, ASTM STP 754, D.G. Franklin Ed. American Society for Testing and Materials. 1982, pp. 271–283.
[2] M. Ek. LOCA Testing at Halden; The Second Experiment IFA-650.2. Tech. rep. HWR-813. OECD Halden Reactor Project, 2005.

Structural Materials and Chemistry

The Structural Materials and Chemistry technical area has completed several studies on modeling alloys, including incorporating irradiation effects in crystal plasticity and surrogate models of Gr91, developing efficient numerical schemes for utilizing advanced materials models and GPU acceleration in MOOSE finite element simulations, and enhancing capabilities for modeling crack evolution, among other topics. In addition, work was conducted on developing molten salt thermophysical property models and modeling alloy corrosion in contact with fuel-bearing molten salts.

The study highlighted below focuses on graphite, an important structural and moderator material in advanced reactors, and demonstrates how uncertainty quantification can improve model parameterization and predictive accuracy. These graphite deformation models will be used in NEAMS multiphysics analyses of advanced reactors.

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Bayesian Calibration Strengthens Graphite Component Modeling for Advanced Reactors

Nuclear-grade graphite plays a critical role in the core designs of advanced reactors such as high-temperature gas-cooled reactors (HTGRs), molten salt reactors (MSRs), fluoride-salt-cooled high-temperature reactors (FHRs), and various microreactor (MR) concepts. As both a moderator and reflector, graphite operates in extreme environments of high temperatures, intense irradiation, and potential chemical interactions with coolant or air. Over years of operation, these conditions can cause nonuniform dimensional changes and stresses that impact component performance and safety.

Understanding and predicting graphite behavior is therefore essential for reliable reactor design and licensing. Uncertainty-quantified models of key graphite thermomechanical properties have been developed using Bayesian calibration methods. These models are implemented in the Grizzly code, based on the MOOSE framework, to simulate component behavior under realistic reactor conditions.

This effort builds on the DOE Advanced Reactor Technologies (ART) program's experimental campaigns, including baseline graphite testing, Advanced Graphite Creep irradiation experiments, and ongoing High Dose Graphite studies. These data provide a foundation for modeling graphite properties such as elastic modulus, thermal expansion, irradiation-induced dimensional change, and irradiation-induced creep for multiple graphite grades including IG-110, NBG-18, NBG-17, PCEA, and 2114.

Using a hierarchical Bayesian framework, multiple data sources were combined and both experimental uncertainty and model-form inadequacy were quantified through the Kennedy–O'Hagan approach. The resulting calibrated models with Gaussian process (GP) correction terms capture property variations across temperature and irradiation levels and enable more robust predictions of graphite component performance.

The newly calibrated models were applied to the thermo-mechanical analysis of a prismatic reflector brick in an HTGR using Grizzly. See Figure 3 for an example result. Simulations revealed how uncertainties in graphite material models significantly influence predicted deformation, stress evolution, and potential failure locations. The behavior of that reflector brick was modeled for multiple graphite grades, and the effect of the GP correction term for model inadequacy was assessed by comparing against a model without correction terms, as shown in Figure 3. For many of the graphite grades considered, the effect of the model-inadequacy correction on the maximum predicted stress was significant, sometimes approaching the magnitude of the stresses in the baseline models. The analysis also demonstrated the ability to identify critical stress regions within components and to separate the effects of elastic, creep, thermal expansion, and irradiation-induced strains.

This study illustrates how Bayesian calibration and uncertainty quantification, integrated within the MOOSE–Grizzly framework, enhance confidence in the predictive models that underpin reactor structural integrity assessments. By combining advanced statistics, experimental data, and multiphysics simulation, NEAMS is providing the nuclear community with tools to better predict component performance, guide operational planning, and support licensing of advanced reactors. This approach and similar ones are not limited to graphite but is also applied to other structural materials, fuels, and cladding.

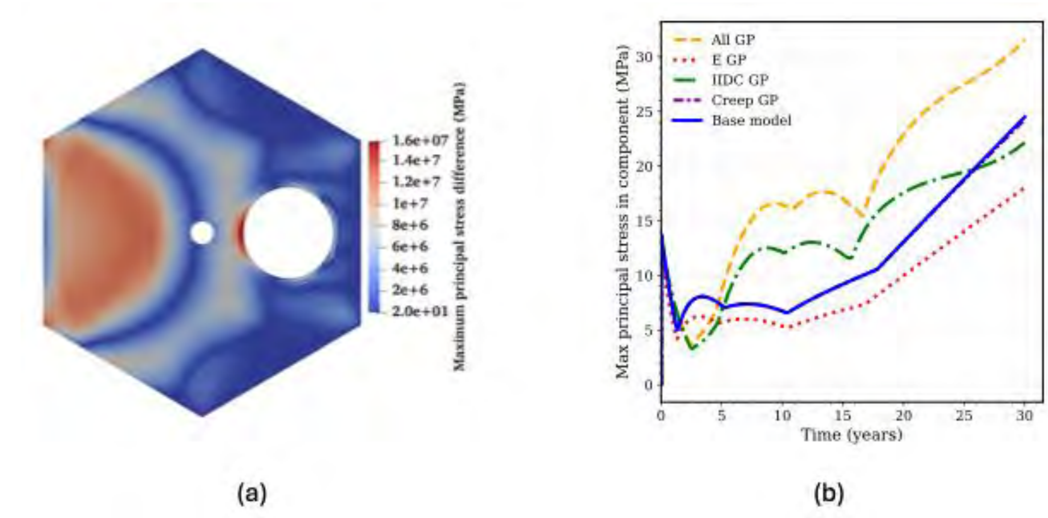

Figure 3: Comparison of the predicted stress with and without GP terms correcting for model inadequacy for an NBG-18 graphite reflector brick. (a) Spatial distribution of the absolute difference in the maximum principal stress at 14.5 years with and without corrections for all four models of considered phenomena. (b) Time history of the maximum principal stress in the component for the base model, with corrections for individual models, and with GP corrections for all models combined.

Technical point of contact: Somayajulu L. N. Dhulipala (Som.Dhulipala@inl.gov).

Additional reading: 1. Somayajulu L. N. Dhulipala, Parikshit Bajpai, Gyanender Singh and Benjamin W. Spencer, Bayesian Calibration of Nuclear Graphite Property Models Accounting for Model Inadequacy and Impacts on Component Performance (2025), INL/RPT-25-86883; and **2.** Somayajulu L. N. Dhulipala, Parikshit Bajpai, Gyanender Singh, and Benjamin W. Spencer, Bayesian Calibration of Irradiated Graphite Property Models Under High Temperatures, npj Materials Degradation, accepted.

Reactor Physics

The Reactor Physics technical area addressed several important topics in FY25, including workflows for Monte Carlo-based running-in simulations of pebble bed reactors using the Shift code, capabilities for performing perturbation theory calculations for uncertainty quantification within Griffin, and improvements to ex-core dose calculations in Griffin through enhanced methods for the neutron deep-penetration problem.

The latter topic is highlighted below due to its relevance to microreactors, which are gaining significant near-term attention for deployment in data center energy systems and national security applications.

Enhancing Neutron Deep-Penetration Reactor Modeling with Griffin

Accurately modeling ex-core neutron and gamma flux is vital for both shielding design optimization and detector signal prediction, which is especially important for transportable microreactors due to their compact design. These capabilities ensure safe operation while minimizing radiation exposure in systems located closer to people and critical equipment.

Ex-core calculations present a deep-penetration challenge for both Monte Carlo and deterministic methods. Monte Carlo simulations suffer from poor statistics because few particles reach the ex-core region, unless variance reduction techniques are used appropriately. Deterministic solvers like Griffin avoid statistical noise but face ray effects from angular discretization, which distort results in narrow streaming paths or small detector regions. Accurately resolving these effects requires many angular directions, greatly increasing computational cost and memory use.

Recent development resulted in major progress on improving the Griffin discrete ordinates (SN) solver for deep-penetration calculations, supporting both steady-state and transient reactor analyses. This work focused on refining Griffin's methodologies, expanding its capabilities, and validating its performance through high-fidelity simulations of the Transient Test Reactor (TREAT), see Figure 4.

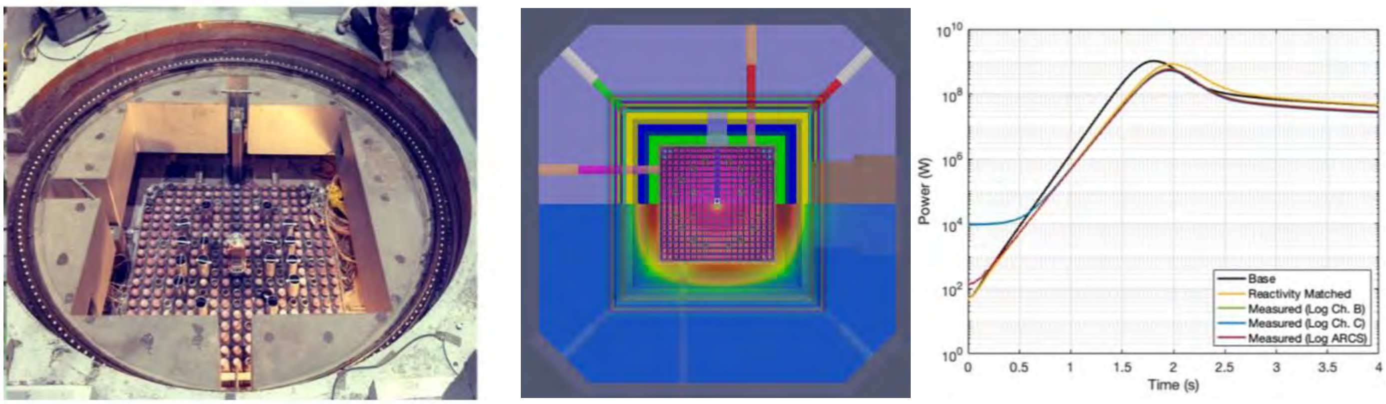

Figure 4: The core view (left) of the TREAT facility at INL, which was modeled using Griffin's new ex-core capabilities. Griffin mesh with detector channels, hodoscope and thermal column in the bio-shield overlaid with the thermal flux distribution (center). Comparison of predicted and measured core power with fully heterogeneous SN transport (right).

Specifically, Griffin's fixed-source calculation framework was upgraded to enhance flexibility and computational efficiency. The uncollided flux solver was refactored and benchmarked against direct discrete ordinates solutions, effectively reducing so-called ray effects. A key improvement, Anderson acceleration applied to the coarse mesh finite difference (CMFD)-accelerated Richardson iteration, significantly improved convergence performance for full ex-core reactor models, where traditional approaches often struggle. Additional optimization through group staging and angular flux reconstruction achieved up to a 20-fold reduction in memory usage for transient calculations, enabling Griffin to handle complex simulations for TREAT.

Using the TREAT model, Griffin successfully predicted ex-core detector signals and core power distributions, demonstrating excellent agreement with experimental results. These analyses revealed subtle calibration inconsistencies in existing detector systems, showing how advanced modeling can directly support operational diagnostics and safety validation.

While Griffin performed exceptionally well, the study identified future areas of refinement, including:

- Enhancing ray tracing for faster uncollided flux calculations.
- Improving cross-section treatments in ex-core regions.
- Implementing energy-dependent angular quadrature optimization to reduce computational waste.
- Conducting mathematical analyses of Anderson acceleration behavior to better characterize convergence properties.

The next phase of development will integrate Griffin's deep-penetration capabilities with the Shift Monte Carlo code, enabling hybrid simulations that combine the strengths of deterministic and Monte Carlo methods. These improvements will particularly benefit microreactor technologies, where compact designs demand precise modeling of radiation transport and control system behavior. Ultimately, expanding the NEAMS

Reactor Physics toolbox for ex-core analyses, including Griffin, Shift and the joint workflows, will greatly benefit the user community.

Technical point of contact: Javier Ortensi (javier.ortensi@inl.gov).

Additional reading: Namjae Choi, Joshua Hanophy, Yaqi Wang, Benjamin Chase, and Javier Ortensi, Griffin Capability Improvements in Support of Ex-core Deep-Penetration Problems, INL/RPT-25-85358 Rev:000.

Thermal Fluids

The Thermal Fluids technical area achieved many accomplishments in FY25, encompassing work on all major reactor types. The Thermal Fluids workflow relies on a multi-scale approach, starting at the CFD scale, where turbulence is accurately resolved, progressing through the engineering scale, and extending to the system scale that captures fluid flow and heat transfer in the entire nuclear power plant. The highlight below summarizes work on the engineering scale for molten salt reactors. However, first we would like to mention NEAMS participation in an important international conference.

NEAMS had a strong presence at the 21st International Topical Meeting on Nuclear Reactor Thermal Hydraulics (NURETH-21) that took place in Busan, Korea from August 31st to September 5th, see Figure 5. NURETH is the premier conference for state-of-the-art research related to the thermal-hydraulics of nuclear reactors. The recently held 21st conference was one of the best attended in the series, attracting over 900 registrants from 37 countries and resulting in 721 papers and numerous panel discussions. NEAMS played a significant role in both organization and participation, with 6 NEAMS researchers attending the conference. NEAMS was affiliated with 26 papers accepted by the conference, with one paper receiving the best paper award. NEAMS attendees also served as session chairs for 11 sessions in the conference, participated in one panel session, and served in Technical Program Committee and Track Member and Reviewer roles.

Figure 5: NEAMS participants at NURETH-21.

Integrated Molten Salt Reactor Modeling Capabilities in NEAMS Thermal Hydraulics Tools

Accurately and efficiently predicting complex thermal-fluid phenomena in molten salt reactors (MSRs) for long transients remains a key challenge for reactor design and safety analysis. Recent significant advancements have aimed to address this need with integrated multi-scale thermal hydraulics modeling through the coupling of the system-level SAM code with the engineering-scale Pronghorn solver and the Saline thermophysical property interface with the Molten Salt Thermal Properties Database (MSTDB), see Figure 6. These developments have notably enhanced the stability, robustness, and physical fidelity of coupled simulations by improving domain-overlapping coupling schemes and reformulating energy and scalar transport to better capture transient and buoyancy-driven phenomena. The expanded verification and validation efforts—including natural convection loops and the Molten Salt Reactor Experiment (MSRE)—demonstrate that the integrated tool can accurately predict complex flow, heat transfer, and fission product transport behaviors critical to MSR operation.

In addition, improvements in Pronghorn's corrosion and noble-metal plating models, supported by refined turbulence closures and buoyancy corrections, provide more realistic predictions of material degradation patterns, directly impacting reactor component longevity and maintenance planning. The implementation of a multiphase Euler–Euler model enables detailed simulation of two-phase flows, such as gas entrainment and bubble dynamics, which are essential for understanding operational safety and performance in MSR components like pump bowls. Updates to SAM's species transport framework, including noble gas migration and reactivity feedback integration, allow for comprehensive modeling of fission product behavior and its influence on reactor kinetics, thereby enhancing predictive capabilities for transient scenarios and shutdown conditions.

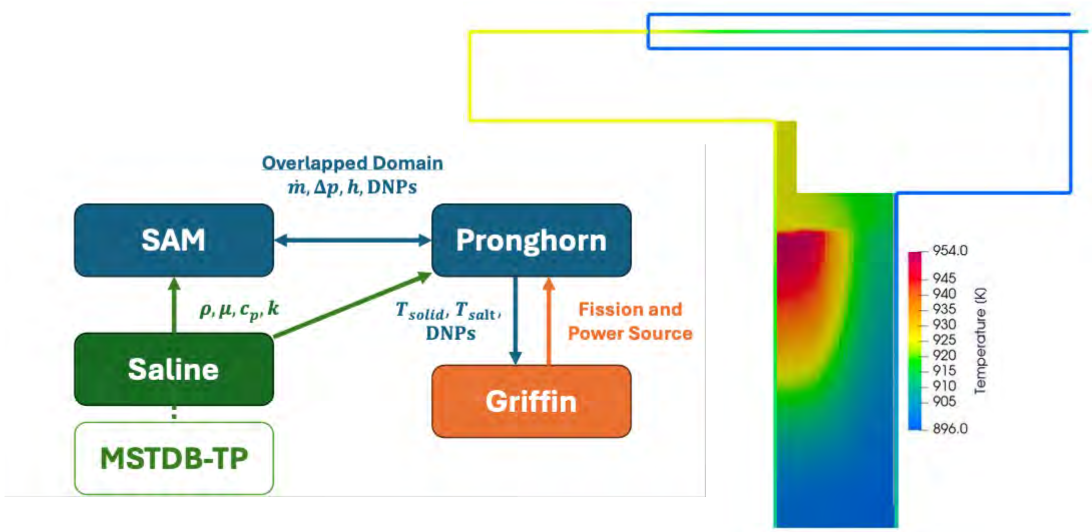

Figure 6: Multi-scale Pronghorn-SAM-Griffin model of the MSRE (fuel salt temperature, right), achieved through an improved domain overlapping coupling scheme (left).

Technical point of contact: Rui Hu (rhu@anl.gov) and Mauricio Tano Retamales (Mauricio.TanoRetamales@inl.gov).

Additional reading: Travis Mui, Mauricio Tano, et al. "Integrated Molten Salt Reactor Modeling Capabilities in NEAMS Thermal Hydraulics Tools." ANL/NSE-25/71, INL/RPT-25-88160, Sep. 2025.

Multiphysics Applications

The NEAMS Multiphysics Applications technical area collaborates closely with the ART program, covering fast reactors, gas-cooled reactors, and molten salt reactors, and the microreactor program to test and mature NEAMS advanced modeling and simulation tools. This work includes validation against historical data collected by DOE and benchmarking against "first-generation" simulation codes already developed for some reactor types, especially sodium-cooled fast reactors. Compared to the first-generation tools, the NEAMS capabilities offer higher fidelity and unique opportunities for multiphysics coupling and analysis. NEAMS greatly values the ART and microreactor program collaborations. They are critical for preparing NEAMS tools for adoption by a broader set of stakeholders. In addition, during FY25 the Multiphysics Applications technical area has made significant strides towards deploying the MOOSE computational framework on GPUs, leveraging AI to improve the user efficiency, and advancing the Workbench user interface.

To provide a flavor of the collaborations with the ART program, below is a summary of progress on benchmarking first-generation tools maintained by the ART Fast Reactor Program (FRP) and NEAMS tools against fast reactor data from the SEFOR experiments.

Validation of Fast Reactor Modeling Tools Using SEFOR Experimental Results

The Southwest Experimental Fast Oxide Reactor (SEFOR) operated from 1969 to 1972, conducting a three-phase program intended to measure Doppler reactivity feedback under varying conditions: zero-power tests, power-ascending tests, and reactivity insertion transients. These experiments produced unique data across a wide temperature range, from about 350 °F to near the 5000 °F melting point of mixed oxide fuel, providing valuable benchmarks for code validation.

ART FRP and NEAMS jointly supported participation in the OECD/NEA SEFOR benchmark study. ART FRP efforts utilized the Argonne Reactor Computation (ARC) suite and the SAS4A/SASSYS-1 system analysis code to successfully model zero-power tests as well as power-ascending tests. NEAMS efforts utilized ARC's MC2-3 cross section generation code and the high-fidelity Griffin reactor physics code to successfully model zero-power tests. These analyses enabled direct comparison with measured data such as reflector worth, kinetics parameters, and reactivity feedback.

The reactivity feedback behavior in SEFOR isothermal tests is dominated by Doppler broadening, core expansion, and sodium density effects. Griffin and ARC predictions of reactivity feedback for SEFOR core configurations I-E and I-I in the isothermal tests showed strong agreement with Monte Carlo reference results (Serpent2, Shift) and experimental data as shown in Figure 7 (within 5%). For power-ascending tests, thermal-hydraulic predictions from the SAS4A/SASSYS-1 model closely aligned with fuel temperature measurements, confirming its reliability for safety analyses.

A major outcome of FY25 was not only the demonstration that Griffin can perform full core heterogeneous transport simulations of the highly detailed SEFOR core configurations with more than 100 fuel assemblies, but also validation of the MC2-3/Griffin models for modeling reactivity feedback effects with direct comparison to experimental data. While Griffin has proven accurate for numerous code-to-code comparisons, this is the first time Griffin has been validated against experimental data for fast reactors. The results begin to establish Griffin's validation basis for important reactivity feedback effects.

Differences among historical SEFOR data sources were observed that introduced challenges in model validation, underscoring the need for consistent, verified benchmark datasets. The use of different modeling approaches in this joint benchmarking study (lower vs. high fidelity) was beneficial for determining impacts of modeling approximations made both in interpretation of the SEFOR benchmark specification as well as in the physics codes themselves. Next, the SEFOR benchmark effort will build on these successes by simulating reactivity insertion transients using the ARC/SAS model and incorporating BISON to automatically account for thermal expansion in the Griffin isothermal test models. This latter integration will enable the first multiphysics simulations of SEFOR's isothermal tests within the MOOSE framework, representing a key step toward integrated multiphysics modeling within the MOOSE ecosystem.

Although the present highlight emphasizes fast reactors, similar activities are ongoing for gas-cooled reactors, molten salt reactors and microreactors.

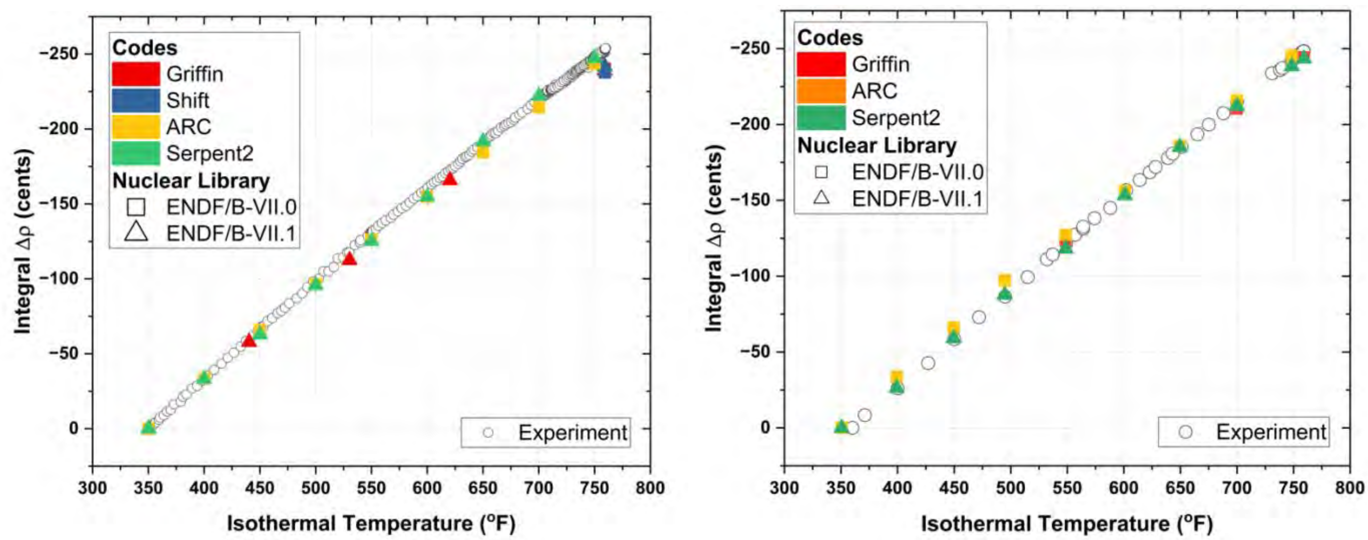

Figure 7: Calculated and measured isothermal reactivity feedback for SEFOR CoreI-E (left) and Core I-I (right).

Technical point of contact: Emily Shemon (eshemon@anl.gov).

Additional reading: Yan Cao, Donny Hartanto, Ahmed A. Abdelhameed, Emily Shemon and Eva Davidson, Validation of Numerical Tools for Calculating Reactivity Feedback in Sodium Fast Reactors Using SEFOR Experimental Data (2025), ANL-ART-312.

LOOKING AHEAD

Upcoming meetings

- NEAMS Annual Review Meeting, February 10–11, Idaho National Laboratory and virtual.

Upcoming training

- MOOSE Framework Fundamentals (2026 January 6-7, Rensselaer Polytechnic Institute).
- BISON Fuels Performance (2026 January 8-9, Rensselaer Polytechnic Institute).
- MOOSE Modeling and Simulation User Training (2026 April 16th, Texas A&M).

See https://mooseframework.inl.gov/training/ for more information.

FY25 MAJOR MILESTONES:

As we reflect on the past year, we are proud to share the major milestones achieved by NEAMS in FY25, see below. The milestone reports will soon be available on OSTI and linked from the NEAMS website (neams.inl.gov).

Fuel Performance

- Completed expansion of the existing multiscale Pd-SiC penetration model in TRISO particle fuel to account for high temperatures and Ag interactions, validated models against experimental data, and quantified uncertainties associated with its predictions
- Completed implementation and assessment of high burnup extension fuel performance models during operation and transient conditions in LWRs
- Completed initial integration of constitutive modeling approach into MOOSE for predictions of the effect of temperature transients on mechanical response of Zircaloy under LOCA-like conditions using BISON
- Demonstrated use of atomic scale data to predict doped UO2 fuel plasticity under reactor operating conditions and generated data for surrogate models in BISON
- Finished development, tested and then demonstrated new baseline fuel performance capability for UN fuel swelling models under steady-state and transient conditions

Structural Materials and Chemistry

- Completed update of MSTDB-TP (Thermo-physical portion of Molten Salt Thermal Properties Database) to include surface tension data and fully accessible predictive modeling of density and viscosity
- Demonstrated multiphysics simulation of graphite component using expanded set of uncertainty-quantified material models
- Developed an updated surrogate model for continuum-scale thermal creep in Grade 91 with an increased range of applicability
- Developed molten salt thermo-physical properties where MSTDB-TP lacks key data using machine-learned potentials within molecular-dynamics simulations

Reactor Physics

- Delivered MOOSE usability improvements: improve 3D meshing capabilities, initiate geometry support for Monte Carlo tools, and enhance MOOSE/Workbench user input interactions using snippets, problem workspaces, and mesh generation workflows
- Developed and demonstrated Shift Monte Carlo reference equilibrium core calculations for Pebble Bed reactors
- Developed and verified perturbation and sensitivity capabilities in Griffin reactor physics code for uncertainty quantification, sensitivity analysis, and potential design optimization
- Implemented performance improvements to Griffin transport solvers for eigenvalue, fixed source, and transient calculations useful for ex-vessel calculations, detector response, and pebble-bed reactor simulations

Thermal Fluids

- Completed advanced solver development for SAM to improve the computational efficiency and robustness for transient system analysis
- Completed Sockeye liquid-conduction and vapor-flow heat pipe model for normal operation
- Demonstrated integrated system and engineering-scale mass transport and material modeling for MSRs

Multiphysics Applications

- Delivered improvements to MOOSE user workflow through support for polyhedral elements, increased automation, and deployment of concise physics syntax
- Delivered MOOSE usability improvements: improved 3D meshing capabilities, initiated geometry support for Monte Carlo tools, and enhanced MOOSE/Workbench user input interactions using snippets, problem workspaces, and mesh generation workflows

- Completed initial steady state core simulation capability for a thermal and pool-type molten salt reactor, coupling reactor physics, thermal hydraulics, and evolving chemistry
- Demonstrated coupled multiphysics capabilities for microreactors including modeling of balance of plant and fission product poisoning, and performed multiphysics code validation using KRUSTY experiment reactivity insertion transient
- Demonstrated NEAMS tool capability improvements for coupled multiphysics simulation of fast reactor core bowing phenomena and summarized verification and validation opportunities for core bowing (joint with ART Fast Reactor Program)
- Identified critical parameters affecting burst susceptibility of BWR fuel during large break loss-of-coolant accident
- Performed gas cooled reactor codes validation using HTTF experimental data and performed code-to-code verification for pebble bed multiphysics model startup calculations (joint with ART Gas-Cooled Reactor program)

The Nuclear Energy Advanced Modeling and Simulation (NEAMS) program is a U.S. Department of Energy-Office of Nuclear Energy (DOE-NE) program developing advanced modeling and simulation tools and capabilities to accelerate the deployment of advanced nuclear energy technologies, including Light Water Reactors (LWRs), non-LWRs, and advanced fuels. We work with DOE-NE, the U.S. Nuclear Regulatory Commission (NRC), and industry to develop, demonstrate, and deploy usable advanced modeling and simulation capabilities across five technical areas: Fuel Performance, Reactor Physics, Structural Materials and Chemistry, Thermal Fluids, and Multiphysics Applications.

The NEAMS program has major work at Argonne National Laboratory, Idaho National Laboratory, Los Alamos National Laboratory, and Oak Ridge National Laboratory.

Learn more about the NEAMS program at neams.inl.gov