**Modeling and Simulation Capabilities for Advanced Reactor Design and Analysis**

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Short descriptions of the modeling and simulation (M&S) capabilities that are available are summarized in this document, including the associated points of contact. The purpose of this summary is to provide information on the tools that can be used for advanced reactor design and analysis to the potential ARPA-E MEITNER Awardees. The specific M&S capabilities required will be dependent on the Awardees, but will likely include the codes in Table 1. Additionally, the M&S codes that are available for advanced reactor types are summarized in Table 2.

It is planned that this summary will be revised periodically to update information on the M&S capabilities and accommodate the Awardee’s reactor types and requests.

**Table 1. List of potential M&S codes**

|  |  |  |  |
| --- | --- | --- | --- |
| Category | Activities | List of codes | Points of contact |
| Software integration | Physics coupling | SIGMADTKMOOSE | L. Grindernu (ANL)S. Slattery (ORNL)R. Martineau (INL) |
| Usability | VERAin/VERAout/VERAviewWorkbench | R. Lee (ORNL)R. Lefebvre (ORNL) |
| Computational science | Trilinos, Petsc, Libmesh, etc. a) |  |
| Physics Tool | Neutronics | MC2-3DIF3DREBUS-3ORIGENPROTEUSPERSENTMPACTSCALESHIFTOpenMCMCNPSerpent a) | E. Shemon (ANL)M. Smith (ANL)M. Smith (ANL)W. Wieselquist (ORNL)Y. Jung/C. Lee (ANL)M. Smith (ANL)B. Collins (ORNL)B. Rearden (ORNL)T. Evans (ORNL)P. Romano (ANL)F. Brown (LANL) |
| Thermal hydraulics | SE2-ANLSAS4A/SASSYS-1SAMNek5000COBRA-TFSTAR-CCM+ a) | M. Smith (ANL)T. Fanning (ANL)R. Hu (ANL)A. Tentner/E. Merzari (ANL) B. Salko (ORNL) |
| Fuel performance | LIFE-METALBISON | L. Yacout (ANL)Jason Hales (INL) |
| Structural mechanics | NUBOW-3DDIABLOAbacus a)DEER (MOOSE app) b)ANSYS a)WARP3D a)  | J. Grudzinski (ANL)R. Ferencz (LLNL) |
| Chemistry/corrosion | MAMBA | David Andersson (LANL) |
| Systems and Controls | Integrated system modeling | RELAP5-3D | J. Wolf (INL) |
| Safety analysis | SAS4A/SASSYS-1CONTAIN-LMR/MELCORRELAP5-3D | T. Fanning (ANL)L. Humphries (SNL)J. Wolf (INL) |
| Dynamic PRA | ADAPT a) |  |
| Economic and fuel cycle analysis | NE-COSTDYMONDRAVEN | F. Ganda (ANL)B. Feng (ANL)C. Rabiti (INL) |

1. Commercial code or non-DOE code
2. See MOOSE points of contact

**Table 2. M&S Codes for Advanced Nuclear Reactor Analysis**

|  |  |
| --- | --- |
| Tasks | Reactor Types |
| SFR / LFR | MSR | FHR | Other reactor (LWR) |
| Neutronics | Cross section generation | MC2-3OpenMCSERPENTSCALE/TRITON a) | MC2-3OpenMCSERPENTSCALE/TRITON a) | OpenMCDRAGON b)WIMS b)SCALE/TRITON | SCALE/TRITON SCALE/Polaris  |
| Core analysis | DIF3DPROTEUS | DIF3DPROTEUS | DIF3DPROTEUS | MPACT  |
| Fuel cycle analysis | REBUS-3ORIGEN | REBUS-3 a) | REBUS-3 |  |
| Perturbation code | PERSENTSCALE/TSUNAMIMCNP | PERSENT a)SCALE/TSUNAMIMCNP | PERSENTSCALE/TSUNAMIMCNP | SCALE/TSUNAMIMCNP |
| Thermal Hydraulics | Steady state T/H | SE2-ANLNEK5000 | SAMNEK5000CTF | SAMNEK5000CTF | CTF |
| Safety | Transient analysis | SAS4A/SASSYS-1 | SAMCTF | SAMCTF | CTF |
| Structural mechanics  | NUBOW3D | DEER (MOOSE app)ANSYSWARP3D | DEER (MOOSE app)ANSYSWARP3D |  |
| Fuel performance  | LIFE-METALBISON | not applicable | None |  |
| Coolant Chemistry | None | VERA-MSR a) | VERA-MSR a) | MAMBA |
| Shielding Design | MCNPADVANTGSCALE/MAVRIC | MCNPADVANTGSCALE/MAVRIC | MCNPADVANTGSCALE/MAVRIC | MCNPADVANTGSCALE/MAVRIC |
| Economics | NE-COST | NE-COST | NE-COST | NE-COST |

1. Under development for this reactor type and its associated functionalities as of May 2018.
2. Commercial or non-DOE code

**ADVANTG**

The AutomateD VAriaNce reducTion Generator (ADVANTG) software automates the generation of variance reduction (VR) parameters for continuous-energy Monte Carlo simulations of fixed-source neutron, photon, and coupled neutron-photon transport problems using MCNP5. ADVANTG generates space- and energy- dependent mesh-based weight-window bounds and biased source distributions from three- dimensional (3-D) discrete ordinates (SN) calculations that are performed by the Denovo. The deterministic calculations can be performed using multiple cores and/or processors (e.g., on multi-core desktop systems and clusters). The final variance reduction parameters are output in a format that can be used with unmodified versions of MCNP. The primary objective of the development of ADVANTG has been to reduce the computational time required to obtain accurate and precise tally estimates across a broad range of challenging transport application areas.

ADVANTG is currently distributed through the Radiation Safety Information Computational Center (RSICC) at <https://rsicc.ornl.gov>.

Point of Contact: Scott Mosher (moshersw@ornl.gov)

**AMPX**

AMPX is a modular system of computer programs used for nuclear analysis with a primary emphasis on tasks associated with the production and use of multigroup (MG), continuous energy (CE) cross sections, depletion/decay libraries, and covariance data. AMPX accepts basic cross-section data from cross-section evaluations in the international Evaluated Nuclear Data File (ENDF/B) Format. AMPX can be used to generate a variety of MG libraries that can be used with modern transport codes to perform nuclear analyses. CE or point cross section libraries can be produced for use in Monte Carlo codes and other applications. Also, AMPX provides cross-section uncertainty or covariance data for use with sensitivity/uncertainty analysis tools. Furthermore, AMPX can be used to process ENDF/B evaluations to produce depletion and decay libraries needed by depletion codes such as ORIGEN.

AMPX is currently distributed through the Radiation Safety Information Computational Center (RSICC) at <https://rsicc.ornl.gov>.

Point of Contact: Doro Wiarda (wiardada@ornl.gov)

**BISON**

BISON is a finite element-based nuclear fuel performance code applicable to a variety of fuel forms including light water reactor fuel rods, TRISO particle fuel, and metallic rod and plate fuel. It solves the fully-coupled equations of thermomechanics and species diffusion, for 1D spherical, 1D layered, 2D axisymmetric, 2D plane strain, or 3D geometries. Fuel models are included to describe temperature and burnup dependent thermal properties, fission product swelling, densification, thermal and irradiation creep, fracture, and fission gas production and release. Plasticity, irradiation growth, and thermal and irradiation creep models are implemented for clad materials.

More information can be found on the BISON homepage at bison.inl.gov

Point of Contact: Jason Hales (Jason.Hales @inl.gov)

**COBRA-TF**

CTF is a modernized and improved version of COBRA-TF (COlant Boiling in Rod Arrays-Two Fluid) that is used for performing subchannel thermal hydraulic analysis and modeling fuel rod thermo-mechanical behavior in light water reactors.  CTF uses a three-field approach, whereby the vapor, continuous liquid, and droplets can be modeled independently in terms of field velocity.  CTF has a coupling interface that has been used for coupling to other physics tools including neutronics, core chemistry, and fuel performance.

More information can be found at <https://www.ornl.gov/division/rnsd/projects/ctf>.

Point of Contact: Robert K. Salko (salkork@ornl.gov)

**CONTAIN-LMR / MELCOR**

CONTAIN-LMR is a computational tool for safety analysis of containments in Liquid Metal Reactors (LMRs). Included in the phenomena modelled are thermal-hydraulics, radiant and convective heat transfer, aerosol loading and transient response, fission product transport and heating effects, and interactions of sodium and corium with the containment atmosphere and structures. CONTAIN-LMR includes models for sodium-concrete interactions, debris bed phenomena and other LMR-specific models in an integrated manner. CONTAIN-LMR is currently not available for public release. The models from CONTAIN-LMR are currently being incorporated into the MELCOR code. The version of MELCOR with the CONTAIN-LMR models will be released soon by SNL.

To obtain a copy of MELCOR, a user has to request the code through the NRC at the following link: <https://www.nrc.gov/about-nrc/regulatory/research/obtainingcodes.html>

Point of Contact: Larry L. Humphries (llhumph@sandia.gov)

**DIABLO**

DIABLO is a structural mechanics code based on implicit, Lagrangian finite element methods for the simulation of solid mechanics and multi-physics events over moderated to long time frames. For instance, DIABLO can model the core deformations of a sodium-cooled fast reactor arising during excessive thermal environments. The code provides a venue for parallel computation leveraging discretization technologies developed. DIABLO is architected using Fortran 95 data objects and a message-passing programming model. Under the Reactor Product Line simulation system of the DOE NEAMS program, DIABLO has been used to demonstrate taking temperature field from coupled thermal hydraulics (NEK5000) and neutronics (PROTEUS) simulations to determine the resulting structural deformations.

More information can be found at https://e-reports-ext.llnl.gov/pdf/765389.pdf

Point of Contact: R. M. Ferencz (ferencz@llnl.gov)

**DIF3D**

DIF3D solves the steady state, multi-group diffusion and transport equations using the finite difference approximation (one-dimensional slab, sphere, and cylinder, two-dimensional Cartesian, hexagonal, and r-z, and three-dimensional Cartesian, hexagonal prism, and r-θ-φ), the transverse integrated nodal method (two-dimensional Cartesian and hexagonal and three-dimensional Cartesian and hexagonal prism), and a hybrid finite element method (two-dimensional Cartesian and hexagonal and three-dimensional Cartesian and hexagonal prism). It also has a spherical harmonics transport capability built on the hybrid finite element method which provides solutions to the even-parity transport equation. The GAMSOR processing code enables a sequence of DIF3D calculations that allows users to run a coupled neutron-gamma calculation needed for neutronics calculations.

More information can be found at https://www.ne.anl.gov/codes/dif3d

Point of Contact: Micheal A. Smith (masmith@anl.gov)

**DTK**

**DTK**—**D**ata **T**ransfer **K**it–is an open-source software library developed at Oak Ridge National Laboratory. DTK is designed to provide scalable parallel solution transfer services for multiphysics simulations. DTK uses a general operator design to provide scalable parallel services for solution transfer between shared volumes and surfaces.

DTK can be downloaded at <https://github.com/ORNL-CEES/DataTransferKit>

For additional details questions or inquires, please visit the DTK website above.

Point of Contact Stuart Slattery (slatterysr@ornl.gov)

**DYMOND**

DYMOND is a nuclear fuel cycle system dynamics model run within the iThink/Stella software with Microsoft Excel templates for data input/output. The major inputs to the code are the reactor and fuel characteristics, the fuel cycle facility properties, the various pathways of each fuel type (enrichment, recycling, storage, etc.), and the nuclear power demand as a function of time. The code uses externally-calculated fuel cycle recipes for both the charged and discharged fuel compositions. The major output is the reactor fleet composition (installed capacities of each type of reactor) over time. All other system data including mass flows (mining rates, fabrication rates, reprocessing rates, etc.) and inventories (spent fuel inventory, recovered uranium, etc.) can be output via an Excel spreadsheet based on the user’s selection.

Point of Contact: Bo Feng (bofeng@anl.gov)

**LIFE-METAL**

LIFE-METAL is a fuel-performance code to calculate the thermal and mechanical response of fuel elements in a fast-reactor environment. It incorporates a one-dimensional, steady-state heat-transfer analysis and a finite-strain-theory structural analysis based on generalized plane strain and the method of successive elastic solutions. An incremental approach is used so that temperatures, stresses, and strains can be calculated during any specified history of reactor power cycling. Fuel-cladding and sodium-cladding chemical attack is treated by a cladding wastage model. Migration of fabricated and swelling porosities; fuel and cladding thermal expansion, elasticity, swelling, and creep; fission-gas release; and hot pressing are included in the analysis. The code has been developed to perform nuclear, thermal, and structural analyses of U-Pu-Zr fuel elements under steady-state and slow transient operating conditions. Based on input from core-wide neutronics and thermal-hydraulics codes, LIFE-METAL does a detailed analysis of the behavior of fuel and cladding in a single fuel element.

Point of Contact: Latif M. Yacout (yacout@anl.gov)

**MAMBA**

The MAMBA code simulates deposition of corrosion products on fuel pins in PWRs, chemical kinetics and transport inside the porous deposited layer of corrosion products (CRUD) and the effect of the deposits on thermal transport through conduction, convection and boiling. Although the code is currently optimized for simulating CRUD in PWRs in the context of CIPS/CILC, the chemical kinetics and transport capabilities can be applied to other problems captured by the same generic physics and chemistry.

Point of Contact: David Andersson (LANL)

**MCNP**

MCNP is a general-purpose Monte Carlo N-Particle code that can be used for neutron, photon, electron, or coupled neutron/photon/electron transport. Specific areas of application include, but are not limited to, radiation protection and dosimetry, radiation shielding, radiography, medical physics, nuclear criticality safety, Detector Design and analysis, nuclear oil well logging, Accelerator target design, Fission and fusion reactor design, decontamination and decommissioning. Important standard features that make MCNP very versatile and easy to use include a powerful general source, criticality source, and surface source; both geometry and output tally plotters; a rich collection of variance reduction techniques; a flexible tally structure; and an extensive collection of cross-section data.

Point of Contact: Forrest Brown (LANL)

**MC2-3**

MC2-3 is a multigroup cross section generation code for nuclear reactor design and analysis. Various geometries - homogeneous medium, 1D unit cell, 2D or 3D hexagonal or Cartesian assembly - can be solved with the collision probability method (CPM) or the method of characteristics (MOC) using ultrafine or hyperfine energy groups. The resolved resonances are self-shielded using pointwise cross sections reconstructed with analytic Doppler broadening at specified isotopic temperatures. The unresolved resonances are self-shielded using the analytic resonance integral method. For accurately producing broad group cross sections, the code can be coupled with external two-dimensional ultrafine group transport calculation codes such as TWODANT and PARTISN. The code produces broad group neutron and gamma cross sections as well as gamma production matrices and KERMA factors.

More information can be found at https://www.anl.gov/technology/project/mc2-3-multigroup-cross-sections-fast-reactors

Point of Contact: Changho Lee (clee@anl.gov)

**MOOSE**

MOOSE (Multiphysics Object Oriented Simulation Environment) is an [object-oriented](https://en.wikipedia.org/wiki/Object-oriented_programming) [C++](https://en.wikipedia.org/wiki/C%2B%2B) finite element framework for the development of tightly coupled [Multiphysics](https://en.wikipedia.org/wiki/Multiphysics) solvers from [Idaho National Laboratory](https://en.wikipedia.org/wiki/Idaho_National_Laboratory). MOOSE makes use of the [PETSc](https://en.wikipedia.org/wiki/PETSc%22%20%5Co%20%22PETSc) non-linear solver package and libmesh to provide the finite element discretization. Current applications of the MOOSE platform include:

Bison: Fuel performance analysis

Rattlesnake: Multigroup radiation transport

Marmot: Microscopic materials aging and degradation

Mammoth: Rector multiphysics analysis

Grizzly: Component aging

Mastodon: Nonlinear seismic analysis

Additional information can be found on the MOOSE homepage at moose.inl.gov

Point of Contact: Rich Martineau (Richard.Martineau@inl.gov)

**MPACT**

MPACT is based on the Method of Characteristics transport approach for 2D problems with cross section weighting based on the subgroup methodology. The code can be executed in parallel to reduce overall run time. For 3D problems, MPACT uses the 2D/1D method which uses 2D MOC in a radial plane and diffusion or Pn in the axial direction. A 51-group library with subgroup parameters is provided.

The following features are stable:

* Support for Linux OS (32-bit and 64-bit)
* Parallel Spatial Decomposition with MPI
* Parallel Angular Decomposition with MPI
* User defined Macroscopic Cross Sections
* 51-group Macroscopic Cross Section Library Data
* Subgroup Resonance Self-Shielding
* Transport Corrected P0
* Export of Mesh to Legacy VTK file and Parallel VTU for visualization
* 2-D MOC Transport Kernel
* Coarse Mesh Finite Difference (CMFD) Acceleration
* 1-D Nodal Kernels based on NEM-Diffusion and Pn
* 2-D/1-D Full Core Solution
* Multi-State Calculation Capability
* Transient Calculation Capability
* Depletion and Decay
* Critical Boron Search
* Equilibrium Xenon Calculation
* Direct Coupling with COBRA-TF
* Simplified Internal T/H
* General PWR Geometry Modeling
	+ IFBA
	+ Control Rods and Control Rod Banks
	+ Burnable Poison Inserts
	+ Fission Chamber Detectors
	+ Grid spacers, Nozzles, Plenum, Baffle, etc.
	+ Semi-Explicit modeling of grid spacers
* Isotopic Restart File
* Cycle-to-Cycle Fuel Shuffling
* Radial Thermal Expansion

More information on MPACT can be found at <https://www.ornl.gov/division/rnsd/projects/mpact>

Point of Contact: Benjamin S. Collins (collinsbs@ornl.gov)

**NEAMS Workbench**

The Nuclear Energy Advanced Modeling and Simulation (NEAMS) Workbench is a new initiative that will facilitate the transition from conventional tools to high-fidelity tools by providing a common user interface for model creation, review, execution, output review, and visualization for integrated codes. The Workbench can provide a common user input, including engineering-scale specifications that are expanded into code-specific input requirements through the use of customizable templates.

Additional information on the NEAMS Workbench is provided at <https://www.ornl.gov/division/rnsd/projects/neams-workbench>.

Point of Contact: Robert Lefebvre (lefebvrera@ornl.gov)

**Nek5000**

Nek5000 is an open source spectral element method computation fluid dynamics (CFD) code, utilizing Reynolds-averaged Navier-Stokes and large eddy simulation for turbulence formulation. The code contains finite volume method (FVM), finite difference method (FDM), and spectral element method (SEM) discretization techniques on an unstructured grid. Incompressible and weakly-compressible flows can be modeled; two-phase boiling model development is currently underway. Nek5000 is designed to simulate unsteady Stokes, unsteady incompressible Navier-Stokes, low Mach-number flows, heat transfer and species transport, and incompressible magnetohydrodynamics (MHD) phenomena.

More information can be found at https://nek5000.mcs.anl.gov

Point of Contact: Adrian M. Tentner (tentner@anl.gov) / Elia Merzari (emerzari@anl.gov)

**NUBOW-3D**

NUBOW-3D is a special purpose structural analysis code to analyze core deformations using detailed mechanical analyses in support of design of core restraint systems. The code models the core in 3-D space using 1-D beam elements and determines the transverse displacements of the beam (axial displacement is not included). The code accounts for a core restraint system as part of the boundary conditions. Inelastic effects of irradiation creep and swelling are included as is duct-to-duct contact. The code requires thermal power and flux fluence information as input obtained from DIF3D calculations. NUBOW-3D calculates the displacement response over time for specified power to flow ratios. If reactivity displacement worths are available, NUBOW-3D can estimate the reactivity worth of spatial deformation as a post-processing step.

Point of Contact: James J. Grudzinski (jjg@anl.gov)

**ORIGEN**

ORIGEN (**O**ak **R**idge **I**sotope **Gen**eration code) calculates time-dependent concentrations, activities, and radiation source terms for a large number of isotopes simultaneously generated or depleted by neutron transmutation, fission, and radioactive decay. ORIGEN is used internally within several neutronics codes for depletion and decay. As a stand-alone tool, ORIGEN provides unique capabilities to (1) simulate continuous nuclide feed and chemical removal, which can be used to model reprocessing or liquid fuel systems, and (2) generate alpha, beta, neutron and gamma decay emission spectra.

ORIGEN is currently distributed within the SCALE Code System available through the Radiation Safety Information Computational Center (RSICC) at <https://rsicc.ornl.gov>. Additional information on ORIGEN is provided at <https://www.ornl.gov/division/rnsd/projects/origen>.

Point of Contact: William A. Wieselquist (wieselquiswa@ornl.gov)

**PERSENT**

PERSENT is a perturbation and sensitivity calculation code for conventional assembly homogenized diffusion and transport problems. It is built on the VARIANT option of DIF3D. For perturbation, calculations can be done that consider typical material and cross section perturbations. For sensitivity calculations, users can compute eigenvalue, reaction rate, reaction rate ratio, power fraction, reactivity worth, prompt neutron lifetime, and beta effective sensitivities to the microscopic reactions: gamma, alpha, proton, deuteron, tritium, fission, nu, and P0 & P1 scatter cross sections

Point of Contact: Micheal A. Smith (masmith@anl.gov)

**PHISICS**

PHISICS is a neutronics code system whose goal is to provide state of the art simulation capability to reactor designers. The different modules for PHISICS currently under development are a nodal and semi-structured transport core solver (INSTANT), a depletion module (MRTAU) and a cross section interpolation (MIXER) module. Coupling of Phisics to a thermal hydraulics system code RELAP5-3D, will allow full core and system modeling. This will enable the possibility to model coupled (thermal-hydraulics and neutronics) problems with more options for 3D neutron kinetics, compared to the existing diffusion theory neutron kinetics module.

Point of Contact: Cristian Rabiti (Cristian.Rabiti@inl.gov)

**PROTEUS**

**PROTEUS** is a set of high-fidelity-capable advanced neutronics modeling and simulation tools including cross section generation codes, transport solvers (discrete ordinate (SN), method of characteristics (MOC), and NODAL), and mesh generation toolkit. The SN and MOC solvers are based on unstructured finite element meshes to be able to simulate complex geometries, while the NODAL solver can handle Cartesian and hexagonal geometries. All solvers can solve full transient problems. The code uses multigroup cross sections generated from MC2-3 or Monte Carlo codes (Serpent or OpenMC), and can generate the self-shielded multigroup cross sections on-the-fly using the cross section API. Typical reactor geometry meshes for Cartesian or hexagonal assembly based cores can be easily generated using the mesh generation toolkit.

More information can be found at https://www.ne.anl.gov/codes/proteus

Point of Contact: Yeon Sang Jung (yjung@anl.gov) / Changho Lee (clee@anl.gov)

**RAVEN**

RAVEN (Risk Analysis Virtual Environment) can be used to identify and increase the safety margin in nuclear reactor systems. It is a modular or ‘plug-in’ component that can be coupled with other computer modeling systems such as [BISON](https://bison.inl.gov/), [RELAP-7](https://relap7.inl.gov/) and [RELAP5-3D](http://www4vip.inl.gov/relap5/). As a generic software framework, RAVEN is designed to perform parametric and probabilistic analysis based on the response of complex system codes. RAVEN is capable of investigating the system response as well as the input space using Monte Carlo, Grid, or Latin Hyper Cube sampling schemes, but its strength is focused toward system feature discovery, such as limit surfaces, separating regions of the input space leading to system failure, using dynamic supervised learning techniques.

Additional information is available on the RAVEN homepage at raven.inl.gov

Point of Contact: Cristian Rabiti (Cristian.Rabiti@inl.gov)

**REBUS-3**

REBUS is a general purpose fuel cycle analysis code built around DIF3D with features specific to a commercial fast reactor industry. The code can solve problems with 1) the infinite-time, or equilibrium, conditions of a reactor operating under a fixed fuel management scheme or 2) the explicit cycle-by-cycle, or non-equilibrium operation of a reactor under a specified periodic or non-periodic fuel management program. Fast spectrum systems typically require using recycled used fuel as the initial enrichment. The addition of breeding blankets within fast spectrum reactors allows them to progressively eliminate the need for external enrichment feeds as fast spectrum reactors can generate a larger amount of fissile material than they destroy to create power. To model these aspects of the fuel cycle analysis, REBUS-3 contains a fuel fabrication system that can handle multiple feed materials and multiple sources. The fuel fabrication process allows for spent fuel cooling and allows the user to select how effective the reprocessing plant is at separating out the minor actinides and fission products.

More information can be found at https://www.ne.anl.gov/codes/rebus

Point of Contact: Micheal A. Smith (masmith@anl.gov)

**RELAP5-3D**

The RELAP5-3D code has been developed for best-estimate transient simulation of light water reactor coolant systems during postulated accidents. The code models the coupled behavior of the reactor coolant system and the core for loss-of-coolant accidents and operational transients such as anticipated transient without scram, loss of offsite power, loss of feedwater, and loss of flow. A generic modeling approach is used that permits simulating a variety of thermal hydraulic systems. Control system and secondary system components are included to permit modeling of plant controls, turbines, condensers, and secondary feedwater systems.

Additional information can be found at relap53d.inl.gov

Point of contact: Jim Wolf (James.Wolf@inl.gov)

**SAM**

The System Analysis Module (SAM) code is a modern system analysis tool being developed for advanced non-LWR safety analysis. It aims at providing fast-running, whole-plant transient analyses capability with improved-fidelity for SFR, LFR, and MSR/FHR. SAM takes advantage of advances in physical modeling, numerical methods, and software engineering, to enhance its user experience and usability. It utilizes an object-oriented application framework ([MOOSE](http://www.mooseframework.org/)), and its underlying meshing and finite-element library ([libMesh](http://libmesh.github.io/index.html)) and linear and non-linear solvers ([PETSc](http://www.mcs.anl.gov/petsc/index.html)), to leverage the modern advanced software environments and numerical methods.

More information can be found at https://www.ne.anl.gov/codes/sam

Point of Contact: Rai Hu (rhu@anl.gov)

**SAS4A/SASSYS-1**

**SAS4A/SASSYS-1 is a computer code for thermal, hydraulic, and safety analysis of power and flow transients in liquid-metal-cooled nuclear reactors (LMRs). SAS4A can analyze severe core disruption accidents with coolant boiling and fuel melting and relocation, initiated by a very low probability coincidence of an accident precursor and failure of one or more safety systems. SASSYS-1, originally developed to address loss-of-decay-heat-removal accidents, can perform margin assessment in design basis accident (DBA) analysis and for consequence assessment in beyond-design-basis accident (BDBA) analysis.**

More information can be found at https://www.ne.anl.gov/codes/sas4a-sassys-1

Point of Contact: Thomas H. Fanning (fanning@anl.gov)

**SCALE**

The SCALE Code System is a widely-used modeling and simulation suite for nuclear safety analysis and design that is developed, maintained, tested, and managed by the Reactor and Nuclear Systems Division (RNSD) of Oak Ridge National Laboratory (ORNL). SCALE provides a comprehensive, verified and validated, user-friendly tool set for criticality safety, reactor and lattice physics, radiation shielding, spent fuel and radioactive source term characterization, and sensitivity and uncertainty analysis. SCALE provides an integrated framework with dozens of computational modules including three deterministic and three Monte Carlo radiation transport solvers that are selected based on the desired solution strategy. SCALE includes current nuclear data libraries and problem-dependent processing tools for continuous-energy (CE) and multigroup (MG) neutronics and coupled neutron-gamma calculations, as well as activation, depletion, and decay calculations. SCALE includes unique capabilities for automated variance reduction for shielding calculations, as well as sensitivity and uncertainty analysis. SCALE’s graphical user interfaces assist with accurate system modeling, visualization of nuclear data, and convenient access to desired results. The following SCALE sequences are commonly used for neutronics analysis

* SCALE/TRITON – Reactor physics sequence that can couple ORIGEN depletion to Shift Monte Carlo transport, KENO Monte Carlo Transport, NEWT for 2D SN deterministic transport, and XSDRN for 1D deterministic transport
* SCALE/Polaris – Integrated, easy-to-use 2D LWR lattice physics sequence
* SCALE/TSUNAMI – Perturbation-based neutronics sequence for determining cross section sensitivities
* SCALE/MAVRIC – Monte Carlo-based shielding sequence with automated various reduction using adjoint-based SN calculation.

SCALE available through the Radiation Safety Information Computational Center (RSICC) at <https://rsicc.ornl.gov>.

More information can be found at https://www.ornl.gov/scale

Point of Contact: Bradley T. Rearden (reardenb@ornl.gov)

**SERPENT**

Serpent is a multi-purpose three-dimensional continuous-energy Monte Carlo particle transport code, developed at VTT Technical Research Centre of Finland, Ltd. The development started in 2004, and the code has been publicly distributed by the OECD/NEA Data Bank and RSICC since 2009. Serpent started out as a simplified reactor physics code, but the capabilities of the current development version, Serpent 2, extend well beyond reactor modeling. The applications can be roughly divided into three categories: 1) traditional reactor physics applications, including spatial homogenization, criticality calculations, fuel cycle studies, research reactor modeling, validation of deterministic transport codes, etc., 2) multi-physics simulations, i.e. coupled calculations with thermal hydraulics, CFD and fuel performance codes, and 3) neutron and photon transport simulations for radiation dose rate calculations, shielding, fusion research and medical physics.

More information can be found at https:// <http://montecarlo.vtt.fi/>

**SHARP**

SHARP is an advanced multiphysics modeling and simulation toolkit for the analysis of nuclear reactors. It is comprised of several components including physical modeling tools, tools (SIGMA) to integrate the physics codes for multi-physics analyses, and a set of tools to couple the codes within the MOAB framework. Physics modules currently include the neutronics code PROTEUS, the thermal-fluid code Nek5000, and the structural mechanics code Diablo.

More information can be found at https://www.ne.anl.gov/capabilities/sharp

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**SHIFT**

Shift is a massively parallel Monte Carlo radiation transport code.  It features both continuous-energy and multigroup neutron and gamma physics, and supports multiple geometries, including SCALE, MCNP, and CAD geometries.  It scales well from laptops to supercomputers, and is integrated with Denovo for automatic variance reduction and ORIGEN for depletion calculations.

Shift is currently distributed within the SCALE Code system available through the Radiation Safety Information Computational Center (RSICC) at <https://rsicc.ornl.gov>.

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**SIGMA**

SIGMA provides interfaces and tools to access geometry data, create high quality unstructured meshes along with unified data-structures to load and manipulate parallel computational meshes for various applications to enable efficient physics solver implementations. Mesh generation is a complex problem since most problem geometries involve complicated curved surfaces that require physics imposed spatial resolution and optimized elements for good quality. These tools simplify the process of generation and handling of discrete meshes with scalable algorithms to leverage efficient usage from desktop to petascale architectures.

More information can be found at http://sigma.mcs.anl.gov

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**STAR-CCM+**

STAR-CCM+ is a general purpose finite volume formulation continuum mechanics solver for turbulent incompressible/compressible multi-phase fluid flow, heat transfer, structural mechanics and fluid-structure interactions. In nuclear energy applications, STAR-CCM+ is principally used for simulations of turbulent flow and heat transfer, often with multi-phase and/or phase change phenomena, in complex geometries of nuclear reactor primary systems. STAR-CCM+ provides two coupling interfaces that enable integration as a component of multiphysics or multiscale simulations via user coding. STAR-CCM+ is a commercial software package licensed by Siemens PLM; license cost contributions must be considered in planning for use of the code.

**SuperEnergy2-ANL (SE2-ANL)**

SE2-ANL is a multi-assembly, steady-state sub-channel thermal hydraulics code designed to perform core-wide coolant temperature profiles in liquid metal cooled reactors. The user provides coolant mass flow rates and will obtain the radial and axial assembly power profiles from DIF3D coupled neutron/gamma calculations (GAMSOR). It uses a sub-channel model within each assembly with a simplified energy mixing model (porous body model). The version maintained at ANL includes hot spot analysis methods and fuel pin and cladding temperature calculation models that give both average and two-sigma temperatures.

More information can be found at https://www.ne.anl.gov/codes/se2anl

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**TRANSFORM**

The TRANsient Simulation Framework of Reconfigurable Models (TRANSFORM) is a Modelica-based simulation environment developed at ORNL which provides a variety of components for multi-physics, system-level modeling (e.g., lumped fluid flow, heat transfer and control systems). The tool’s primary purpose is to provide a common simulation environment and baseline modeling resources to facilitate rapid development of dynamic advanced reactor models and related systems. Components can either be re-used in a wide variety of applications, as the underlying model descriptions can be made replaceable and modifiable by the user, or new components can be readily created at the level of complexity and detail required of the specific use case.

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**VERA Interface**

**VERAIn** provides simplified, user-friendly ASCII interface for VERA that provides a single source of input and common geometry for all coupled physics codes. VERAIn is publicly available at <https://github.com/CASL/VERAin> .

**VERAOut** is a format specification for HDF5 files that support a common geometric specification for reactor data produced by M&S tools.  This format is used by all VERA codes to ensure consistency and compatibility across different physics and software tools.

**VERAView** is a python-based graphical user interface for reactor analysis of multiphysics data produced by reactor simulators, particularly the VERA tools.  VERAView provides multiple views of very large three-dimensional datasets over the simulation time and provides many analytical tools for processing and outputting the data in common formats used by engineers.  VERAView is publicly available at <https://github.com/CASL/VERAView> .

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