Oak Ridge National Laboratory

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scale

Nuclear Systems Modeling & Simulation

Fiscal Year 2018 Report on SCALE Maintenance and Development

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MISSION STATEMENT

Develop, deploy, and support a quality-assured computational toolkit that advances the state of the art and exemplifies ease of use in a scalable architecture beginning with fundamental physical data and providing research, production, and licensing calculations for current and emerging nuclear modeling and simulation needs.

scale Nuclear Systems Modeling & Simulation

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TABLE OF CONTENTS

» Tables

» Figures

MANAGER'S STATEMENT

The SCALE team is pleased to present this annual report documenting the development, maintenance, distribution, and training accomplishments from fiscal year 2018 (FY18). The SCALE 6.2 release has been very well received. As of September 2018, over 4,500 licenses of SCALE 6.2 have been issued to users in 46 nations, with the cumulative user base for all versions of SCALE exceeding 8,500 users in 59 nations.

SCALE and RSICC leadership personnel from ORNL were honored to receive a 2018 Excellence in Technology Transfer Award from the Federal Laboratory Consortium for Technology Transfer (FLC), Southeast Region, for licensing SCALE 6.2 to such a large community (Figure 1).

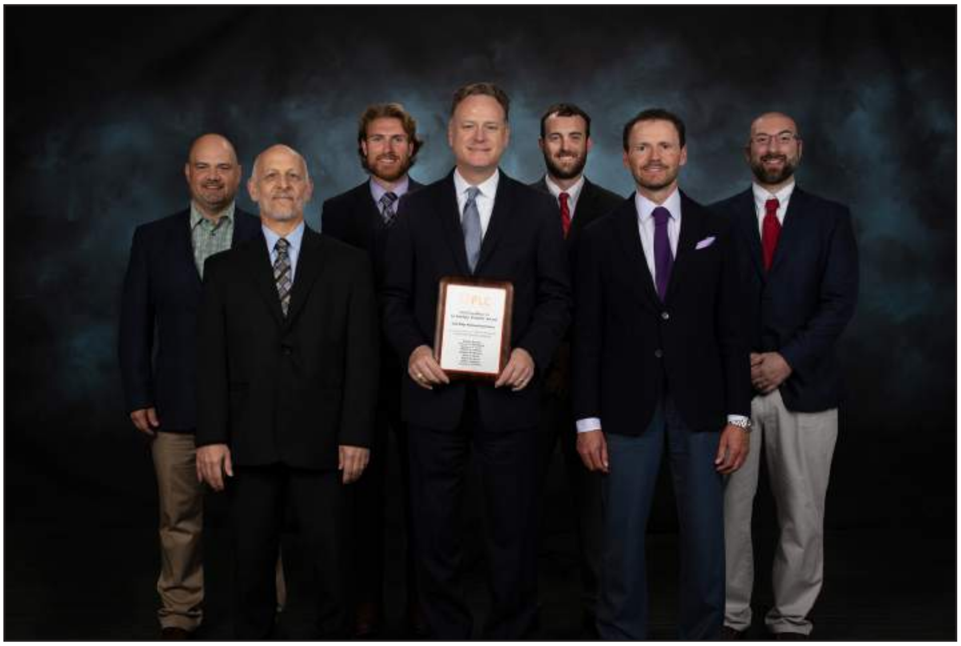

Figure 1. 2018 FLC, Southeast Region, Excellence in Technology Transfer Award for "Licensing SCALE 6.2: A Software Package for Nuclear Safety Analysis and Design" (Left to right) Kevin T. Clarno, Robert E. Grove, Robert A. Lefebvre, Bradley T. Rearden, Matthew A. Jessee, Timothy E. Valentine, and William A. Wieselquist

Organizationally, we are pleased that Dr. William (Will) Wieselquist has assumed the role of Director, SCALE Code System as of August 2018. Will earned a PhD in Nuclear Engineering from North Carolina State University in 2009. From 2009 to 2012, he was a staff member at the Paul Scherrer Institute, where he established an uncertainty quantification platform for reactor core analysis. Will joined Oak Ridge National Laboratory (ORNL) in 2012 and quickly became the lead developer for the ORIGEN depletion/decay tools and the Sampler uncertainty quantification capabilities and a key developer of the Polaris lattice physics code. Will has served as Deputy Director of SCALE since 2017.

In his role as Leader of Modeling and Simulation Integration for the broad portfolio that includes SCALE and many other important projects at ORNL, Brad Rearden will continue to be an integral part of the SCALE team in both advisory and NRC program management capacities.

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Parting Words from Brad Rearden

It has been my great pleasure to lead the SCALE team for the past 9 years. During this time, we have realized tremendous change and growth. I am very proud of the accomplishments the team has realized through a focus on software modernization and quality assurance. This has enabled the development of many new capabilities for SCALE 6.2 that are now emerging as preferred tools for analysis, design, and licensing around the world. I have greatly enjoyed interacting with so many of you in our community, and I look forward to expanded engagements as I take a broader perspective, especially through the ORNL Nuclear Resources – Analysis and Modeling Portfolio (ONRAMP). I encourage you to visit https://www.ornl.gov/onramp to learn more about the tools, laboratories, and expertise available at ORNL.

We take great pride in the many capabilities provided by SCALE and hope you enjoy the contents of this annual report.

William Wieselquist, PhD
Director, SCALE Code System
Reactor Physics Group

Bradley Rearden, Ph.D.
Leader, Modeling and Simulation Integration
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Division of Spent Fuel Management
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Mr. Don R. Algama
Office of Nuclear Regulatory Research
Division of Systems Analysis
Fuel and Source Term Code Development Branch

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National Nuclear Security Administration
Dr. Angela Chambers

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Consortium for the Advanced Simulation of Light Water Reactors

US Department of Energy
Nuclear Energy Advanced Modeling and Simulation Program

Acronyms

- 1D — one-dimensional
- 2D — two-dimensional
- 3D — three-dimensional
- AEC — US Atomic Energy Commission
- API — application programming interface
- ATF — advanced technology fuel
- CADIS — Consistent Adjoint Driven Importance Sampling
- CASL — Consortium for the Advanced Simulation of LWRs
- CE — continuous-energy
- CPU — central processing unit
- CSAS — Criticality Safety Analysis Sequence
- DOE — US Department of Energy
- DSFM — Division of Spent Fuel Management
- ENDF — evaluated nuclear data file
- ENSDF — evaluated nuclear structure data file
- ESSM — Embedded Self-Shielding Methodology
- ExSITE — Extensible SCALE Intelligent Text Editor
- FHR — Fluoride salt-cooled high-temperature reactor
- FW-CADIS — Forward-Weighted CADIS
- GLLS — generalized linear least squares
- GUI — graphical user interface
- HIVE — Hierarchical Input Validation Engine
- HTGR — high-temperature gas-cooled reactor
- I/O — input/output
- JEFF — joint evaluated fission and fusion file
- KMART — KENO Module for Activity-Reaction Rate Tabulation
- LWR — light water reactor
- MACCS — MELCOR Accident Consequence Code System
- MAVRIC — Monaco with Automated Variance Reduction using Importance Calculations
- MCNP — Monte Carlo N-Particle
- MEPhI — Moscow Engineering Physics Institute
- MG — multigroup
- MoC — method of characteristics
- MSR — molten salt reactor
- NCSP — Nuclear Criticality Safety Program
- NE — DOE Office of Nuclear Energy
- NEA — Nuclear Energy Agency
- NEAMS — Nuclear Energy Advanced Modeling and Simulation
- NMSS — Office of Nuclear Material Safety and Safeguards
- NNSA — National Nuclear Safety Administration
- NRC — US Nuclear Regulatory Commission
- OECD — Organisation for Economic Cooperation and Development
- ORIGAMI — ORIGEN Assembly Isotopics
- ORIGEN — Oak Ridge Isotope Generation
- ORNL — Oak Ridge National Laboratory
- QA — quality assurance
- RES — Office of Nuclear Regulatory Research
- RIST — Research Organization for Information Science and Technology
- RNSD — Reactor and Nuclear Systems Division
- RSICC — Radiation Safety Information Computation Center
- SAMS — Sensitivity Analysis Module for SCALE
- SDF — sensitivity data file
- SFR — sodium-cooled fast reactor
- S/U — sensitivity/uncertainty
- TCF — Technology Commercialization Fund
- TRITON — Time-dependent Operation for Neutronic depletion
- TRISO — tristructural-isotropic
- TSAR — Tool for Sensitivity Analysis of Reactivity Responses
- TSUNAMI — Tools for Sensitivity and Uncertainty Analysis Methodology Implementation
- TSUNAMI-IP — TSUNAMI Indices and Parameters
- TSURFER — Tool for S/U Analysis of Response Functions Using Experimental Results
- USLSTATS — Upper Subcritical Limit Statistical Software
- VF — very fine
- VIBE — Validation, Interpretation and Bias Estimation
- WCS — Waste Control Specialists
- XSProc — Cross Section Processing

Introduction

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 the Oak Ridge National Laboratory (ORNL). SCALE provides a comprehensive, verified and validated, user-friendly tool set for criticality safety, reactor physics, radiation shielding, radioactive source term characterization, and sensitivity and uncertainty analysis. Since 1980, regulators, licensees, and research institutions around the world have used SCALE for safety analysis and design. SCALE provides an integrated framework with dozens of computational modules that are selected based on the user's desired solution strategy. SCALE includes current nuclear data libraries and problem-dependent processing tools for continuous-energy (CE), multigroup (MG), 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 (GUIs) assist with accurate system modeling and convenient access to desired results.

This report summarizes the capabilities of SCALE 6.2, the maintenance and development activities performed during FY18, and ongoing development activities. The current public version of SCALE is SCALE 6.2.3, released in April 2018, which followed the releases SCALE 6.2 in April 2016, SCALE 6.2.1 in July 2016, and SCALE 6.2.2 in May 2017.

Background

The SCALE code system dates back to 1969, when ORNL began providing the transportation package certification staff at the US Atomic Energy Commission (AEC) with computational support in the use of the new KENO code. KENO was used to perform criticality safety assessments with the statistical Monte Carlo method. From 1969 to 1976, AEC certification staff members relied on ORNL personnel to assist them in the correct use of codes and data for criticality, shielding, and heat transfer analyses of transportation packages. However, the certification staff learned that occasional users had difficulty in becoming proficient in performing the calculations often needed for an independent safety review. Thus, shortly after the certification staff was moved to the US Nuclear Regulatory Commission (NRC), the NRC proposed development of an easy-to-use analysis system that provided the technical capabilities of the individual modules with which they were familiar. With this proposal, the concept of SCALE as a comprehensive modeling and simulation suite for nuclear safety analysis and design was born. The NRC staff provided ORNL with the general development criteria for SCALE presented here:

1. focus on applications related to nuclear fuel facilities and package designs,
2. use well-established computer codes and data libraries,
3. design an input format for the occasional or novice user,
4. prepare standard analysis sequences (control modules) to automate the use of multiple codes (functional modules) and data to perform a system analysis, and
5. provide complete documentation and public availability.

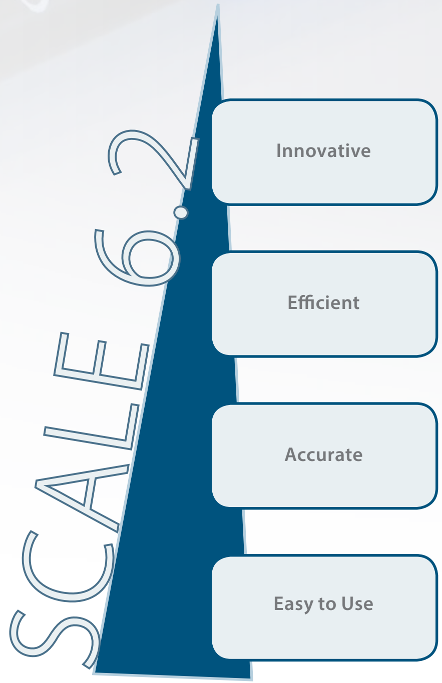
With these criteria, the ORNL staff established the framework for the SCALE system and began development. The initial version of SCALE (Version 0) was distributed in July 1980. Although the system's capabilities continue to evolve, the philosophy established with the initial release still serves as the foundation of this year's SCALE 6.2.3 update, nearly four decades later.

Releases

Year	Version	RSICC ID	Year	Version	RSICC ID
1980	SCALE 0	CCC-288	1998	SCALE 4.4	CCC-545
1981	SCALE 1	CCC-424	2000	SCALE 4.4a	CCC-545
1983	SCALE 2	CCC-450	2004	SCALE 5	CCC-725
1985	SCALE 3	CCC-466	2006	SCALE 5.1	CCC-732
1990	SCALE 4	CCC 545	2009	SCALE 6.0	CCC 750
1992	SCALE 4.1	CCC-545	2011	SCALE 6.1	CCC-785
1994	SCALE 4.2	CCC-545	2016	SCALE 6.2	CCC-834
1996	SCALE 4.3	CCC-545			

Capabilities

A primary goal of SCALE is to provide robust calculations while reducing requirements for user input. The user does not need to have extensive knowledge of the intricacies of the underlying code and data architecture. SCALE provides standardized sequences to integrate many modern and advanced capabilities into a seamless calculation that the user controls from a single input file. Additional utility modules are provided primarily for post-processing data generated from the analysis sequences for advanced studies. The user provides input for SCALE sequences in the form of text files using free-form input, with extensive use of keywords and engineering-type input requirements. SCALE's GUI helps the user create input files, visualize geometry and nuclear data, execute calculations, view output, and visualize results. A diagram showing the key capabilities of SCALE is provided in Figure 2.



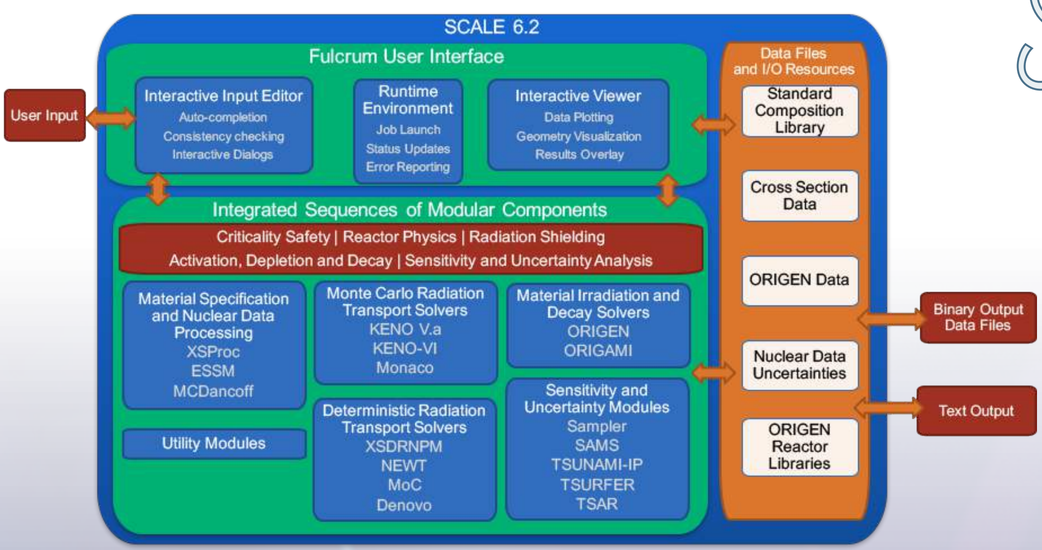


Figure 2. SCALE capabilities

Major SCALE capabilities and analysis areas they serve are provided in Table 1.

Table 1. Summary of major SCALE capabilities

Analysis area	Modules/libraries	Analysis function(s)
Criticality safety	CSAS5/CSAS6	Three-dimensional (3D) MG and CE eigenvalue Monte Carlo analysis and criticality search capability
	STARBUCS	Burnup credit analysis using 3D Monte Carlo
	Sourcerer	Hybrid 3D deterministic/Monte Carlo analysis with optimized fission source distribution
Reactor physics	TRITON	One-dimensional (1D) and two-dimensional (2D) general purpose lattice physics depletion calculations and generation of few-group cross section data for use in nodal core simulators 3D MG and CE Monte Carlo depletion analysis 2D eigenvalue and reaction rate sensitivity analysis
	Polaris	2D streamlined light water reactor (LWR) lattice physics depletion calculations and generation of few-group cross section data for use in nodal core simulators
Radiation shielding	MAVRIC	3D CE and MG fixed-source Monte Carlo analysis with automated variance reduction
Activation, depletion and decay	ORIGEN	General purpose point depletion and decay code to calculate isotopic concentrations, decay heat, radiation source terms, and activity
	ORIGAMI	Simulated 2D and 3D analysis for LWR spent fuel assemblies (isotopic activation, depletion, and decay for LWR fuel assemblies)
	ORIGEN reactor libraries	Pregenerated burnup libraries for a variety of fuel assemblies for commercial and research reactors
Sensitivity and uncertainty analysis	TSUNAMI	1D and 2D MG eigenvalue and reaction rate sensitivity analysis 3D MG and CE eigenvalue and reaction rate sensitivity analysis Determination of experiment applicability and biases for use in code and data validation
	Sampler	Stochastic uncertainty quantification in results based on uncertainties in nuclear data and input parameters

Analysis area	Modules/libraries	Analysis function(s)
Material specification and cross section processing	XSProc	Temperature correction, number density calculation, resonance self-shielding, and flux weighting to provide problem-dependent microscopic and macroscopic MG cross section data integrated with computational sequences; also available for stand-alone analysis
	Standard composition library	Library used throughout SCALE that provides individual nuclides; elements with tabulated natural abundances; compounds, alloys, mixtures, and fissile solutions commonly encountered in engineering practice
	MCDancoff	3D Monte Carlo calculation of Dancoff factors
Monte Carlo transport	KENO V.a/ KENO-VI	Eigenvalue Monte Carlo codes applied in many computational sequences for MG and CE neutronics analysis
	Monaco	Fixed source Monte Carlo code applied in the MAVRIC sequence for MG and CE analysis
Deterministic transport	XSDRNPM	1D discrete ordinates transport applied for neutron, gamma, and coupled neutron/gamma analysis
	NEWT	2D extended step characteristic transport with flexible geometry applied to neutronics analysis, especially within the TRITON sequences
	Denovo	3D Cartesian geometry discrete ordinates transport applied for neutron, gamma, and coupled neutron/gamma analysis, especially to generate biasing parameters within the MAVRIC and Sourcerer sequences (not generally run as stand-alone code in SCALE)
Nuclear data	Cross section data	Recent neutron, gamma and coupled neutron/gamma nuclear data libraries in CE and several MG structures for use in all transport modules
	ORIGEN data	Recent nuclear decay data, neutron reaction cross sections, energy-dependent neutron-induced fission product yields, delayed gamma ray emission data, neutron emission data, and photon yield data
	Covariance data	Recent uncertainties in nuclear data for neutron interaction, fission product yields, and decay data for use in TSUNAMI tools and Sampler
Utilities	Various	Numerous pre- and post-processing utilities for data introspection and format conversion

Criticality Safety

SCALE provides a suite of computational tools for criticality safety analysis that is primarily based on the KENO Monte Carlo codes for eigenvalue neutronics calculations. Two KENO variants provide complementary solution capabilities with different geometry packages. KENO V.a uses a simple, efficient geometry package that is sufficient for modeling many systems of interest to criticality safety and reactor physics analysts. KENO-VI uses the SCALE Generalized Geometry Package, which provides a quadratic-based geometry system with much greater flexibility in problem modeling, but with longer runtimes. Both versions of KENO perform eigenvalue calculations for neutron transport primarily to calculate multiplication factors (k_{eff}) and flux distributions of fissile systems in CE and MG modes. Criticality Safety Analysis Sequence 5 (CSAS5) is typically used to access KENO V.a, and CSAS6 is typically used to access KENO-VI. The CSASs implement XSProc to process material input, and they provide a temperature- and resonance-corrected cross section library based on the physical characteristics of the problem being analyzed. If a CE cross section library is specified, no resonance processing is needed, so the CE cross sections are used directly in KENO, with temperature corrections provided as the cross sections are loaded.

CSAS5 provides search capabilities for finding desired values of k_{eff} as a function of dimensions or densities. The two basic search options within CSAS5 are (1) an optimum search seeking a maximum or minimum value of k_{eff} and (2) a critical search seeking a fixed value of k_{eff}. For CE calculations, reaction rate tallies can be requested within the CSAS input, and for MG calculations, reaction rate calculations are performed using the KENO Module for Activity-Reaction Rate Tabulation (KMART) post-processing tools. A conversion tool is provided to convert KENO V.a input to KENO-VI either as a direct KENO input—K5toK6—or more commonly, as a CSAS sequence—C5toC6.

The Standardized Analysis of Reactivity for Burnup Credit using SCALE (STARBUCS) performs criticality calculations for spent fuel systems employing burnup credit. STARBUCS automates the criticality safety analysis of spent fuel configurations by coupling the depletion and criticality aspects of the analysis, thereby eliminating the need to manually process the spent fuel nuclide compositions into a format compatible with criticality safety codes. For burnup-loading curve-iterative calculations, STARBUCS employs the search algorithm from CSAS5 to determine initial fuel enrichments that satisfy a convergence criterion for the calculated k_{eff} value of the spent fuel configuration.

The Sourcerer sequence applies the Denovo discrete ordinates code to generate the starting fission source distribution in a KENO Monte Carlo calculation. This sequence is mostly applied to burnup credit transportation and storage analysis of as-loaded canisters of used fuel (Figure 3).

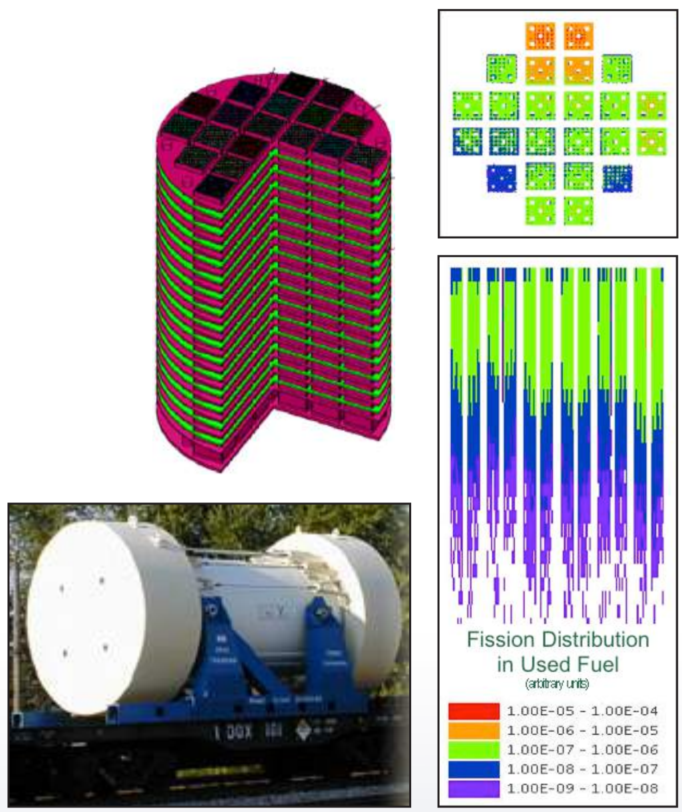


Figure 3. Used fuel storage/ transportation cask

Reactor Physics

SCALE's reactor physics capabilities are integral to the NRC's licensing tools, especially when providing lattice physics data for the PARCS nodal core simulator, as shown in Figure 4.

The TRITON control module provides flexible capabilities to meet the challenges of advanced reactor design analysis. The hallmark feature of TRITON is its support for depletion analysis and nodal data generation through a set of neutronics solvers that provide a wide range of spatial and energy fidelity. TRITON provides 1D multigroup discrete ordinates modeling using XSDRNPM, 2D multigroup, flexible mesh, discrete ordinates modeling using NEWT, or 3D multigroup or CE Monte Carlo transport using KENO, both codes enabled through the KENO-V.a and KENO-VI geometry input format. For MG analysis, TRITON implements XSProc to process material input and to provide a temperature and resonance-corrected cross section library. In all cases, ORIGEN is used for depletion and decay calculations. Additionally, TRITON can produce few group nodal data cross sections for use in core simulators and can simulate time-dependent changes to temperatures and material concentrations.

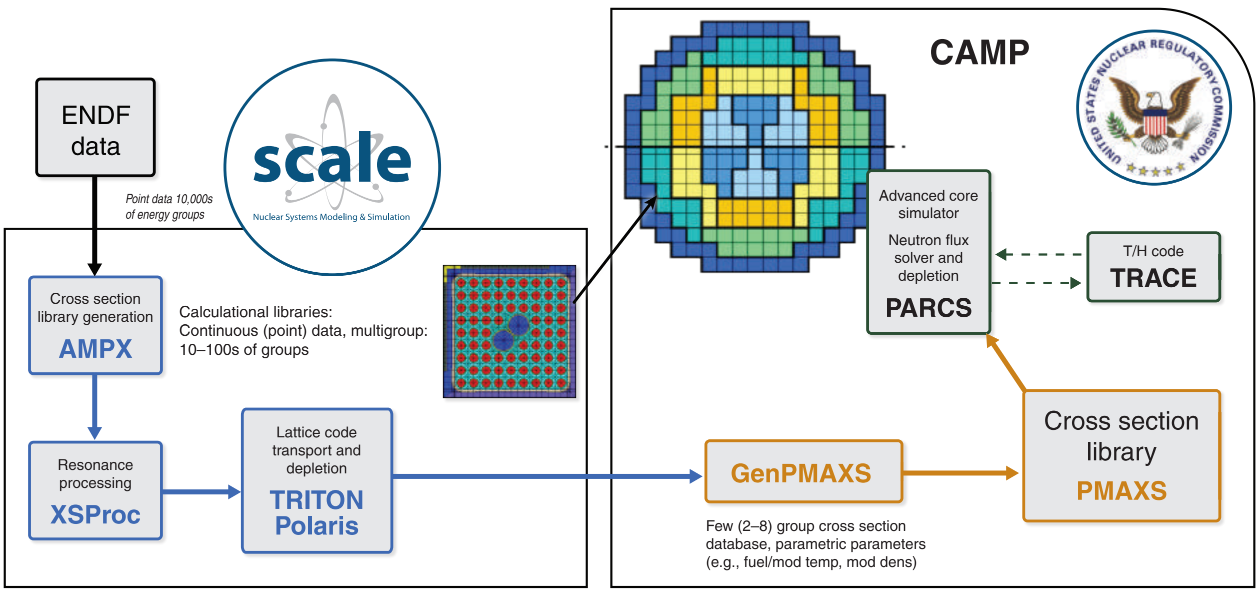


Figure 4. Role of SCALE in NRC reactor licensing calculations

Polaris is a 2D lattice physics tool that produces few group nodal data for LWR analysis with core simulators. Polaris provides simplified input; only a few lines are required to describe the entire model (see Figure 5). Polaris uses an MG self-shielding method called the Embedded Self-Shielding Method (ESSM), a Method-of-Characteristics (MoC) transport solver, ORIGEN for depletion and decay calculations. There are on-going efforts in FY 19 to implement CE Monte Carlo as an option within Polaris as a reference solution using the Shift Monte Carlo package.

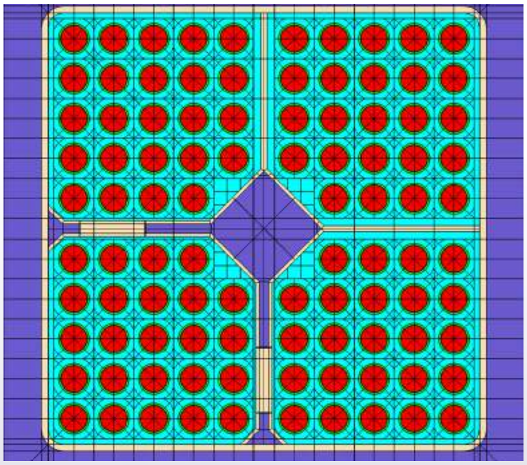


Figure 5. Polaris model of BWR

Radiation Shielding

The Monaco with Automated Variance Reduction using Importance Calculations (MAVRIC) fixed-source radiation transport sequence is designed to apply the MG and CE fixed-source Monte Carlo code, Monaco, to solve problems too challenging for standard, unbiased Monte Carlo methods. The intention of the sequence is to calculate fluxes and dose rates with low uncertainties in reasonable times, even for deep penetration problems. MAVRIC is based on the Consistent Adjoint Driven Importance Sampling (CADIS) methodology, which uses an importance map and a biased source that are derived to work together. MAVRIC generates problem-dependent cross section data, and then it automatically performs a coarse mesh 3D discrete ordinates transport calculation using Denovo to determine the adjoint flux as a function of position and energy. MAVRIC applies this information to optimize the shielding calculation in Monaco. In the Forwarded-Weighted CADIS (FW-CADIS) methodology, an additional Denovo calculation is performed to further optimize the Monaco model to obtain uniform uncertainties for multiple tally locations. Several utility modules are also provided for data introspection and conversion.

The MAVRIC tools were recently applied on behalf of the NRC to assess the site boundary dose rate for the Waste Control Specialists (WCS) Consolidated Interim Storage Facility (Figure 6). This analysis was performed based on the data provided in the publicly available license application, which includes 467 spent fuel canisters of various types.

Detailed design basis models were created for each canister using axially varying source terms from ORIGEN Assembly Isotopics (ORIGAMI) for each fuel assembly. Facility-wide dose rates are depicted in Figure 7, where the concrete pad is 243.84 m × 106.68 m, the air and soil around the concrete pad are 2,713 m × 2,576 m, the air is 959.1 m thick, and the soil is 1 m thick.

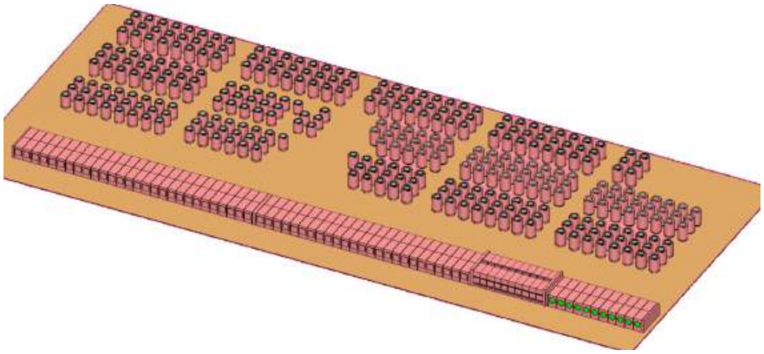


Figure 6. SCALE MAVRIC model of WCS consolidated interim storage facility based on NRC application data

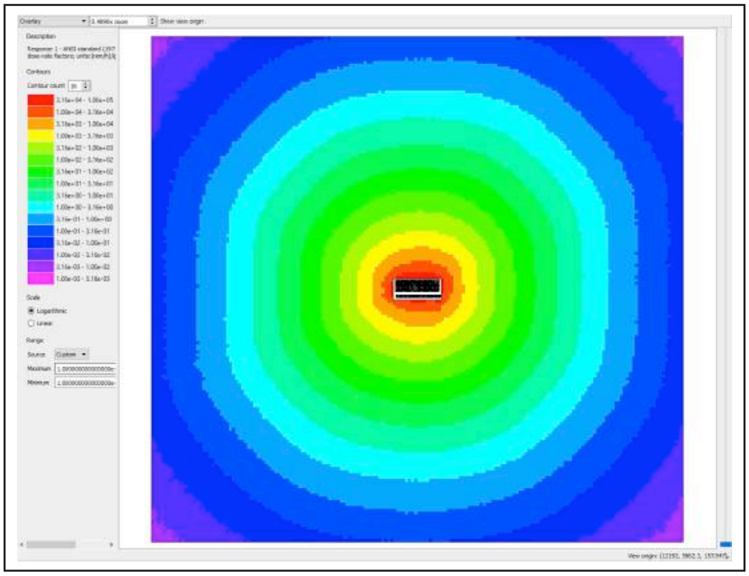


Figure 7. Dose rate for WCS consolidated interim storage facility using detailed design basis models

Activation, Depletion, and Decay

The Oak Ridge Isotope Generation (ORIGEN) code calculates time-dependent concentrations, activities, and radiation source terms for a large number of isotopes that are simultaneously generated or depleted by neutron transmutation, fission, and radioactive decay. Provisions are made to include continuous nuclide feed rates and continuous chemical removal rates that can be described with rate constants for application to reprocessing or other systems that involve nuclide removal or feed. ORIGEN includes the ability to use MG cross sections processed from standard evaluated nuclear data file (ENDF)/B evaluations. Within SCALE, transport codes can be used to model user-defined systems, and the COUPLE code can be applied to calculate problem-dependent neutron-spectrum–weighted cross sections that represent conditions within any given reactor or fuel assembly. These cross sections are converted into a library to be used by ORIGEN. Time-dependent cross section libraries can be produced to reflect fuel composition variations during irradiation. An alternative sequence for depletion/decay calculations is ORIGEN-ARP, which interpolates pre-generated ORIGEN cross section libraries versus enrichment, burnup, and moderator density.

ORIGAMI computes detailed isotopic compositions for LWR assemblies containing UO_2 fuel by using the ORIGEN code with pre-generated ORIGEN libraries for a specified assembly power distribution. The assembly may be represented as (1) a single lumped model with only an axial power distribution, (2) a square array of fuel pins with variable pin powers, or (3) an axial distribution. Multiple cycles with varying burn times and down times may be used. ORIGAMI produces files containing SCALE and Monte Carlo N-Particle (MCNP) composition input for material in the burnup distribution, files containing decay heat for use in thermal analysis, and energy-dependent radioactive source for use in shielding calculations (Figure 8).

A series of 1,470 pre-generated burnup libraries for use in ORIGEN and ORIGAMI is provided with SCALE for 61 fuel assemblies for commercial and research reactors.

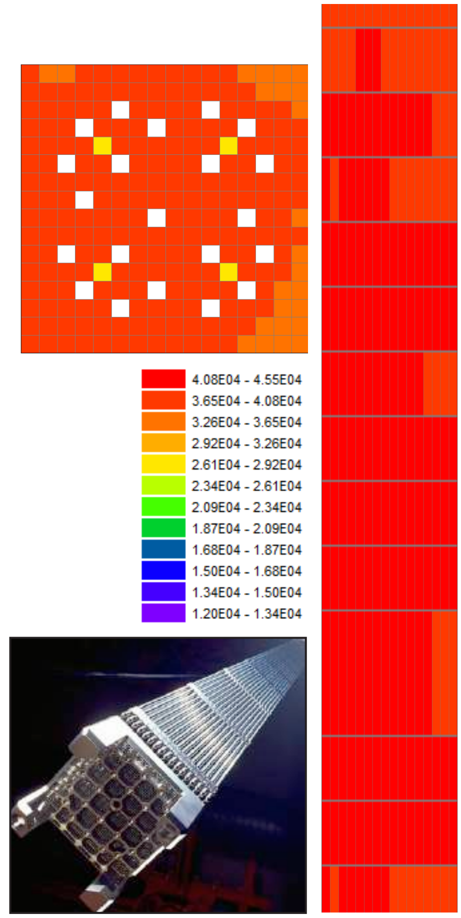


Figure 8. Pin-by-pin burnup and radioactive source terms

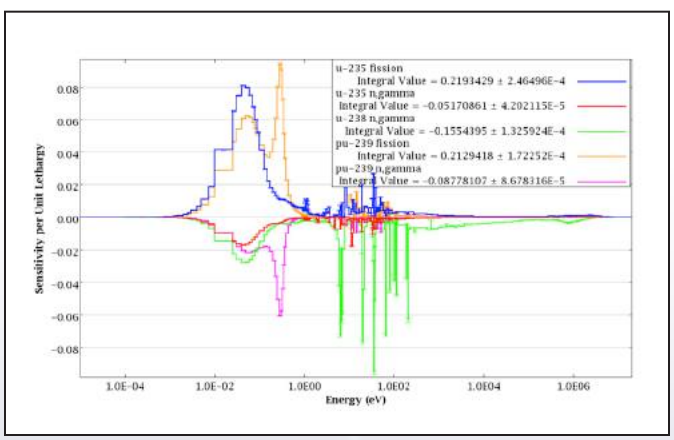


Figure 9. Sensitivity of k_{eff} to cross section data

Sensitivity and Uncertainty Analysis

SCALE provides a suite of computational tools for sensitivity and uncertainty analysis to (1) identify important processes in safety analysis and design, (2) provide a quantifiable basis for neutronics validation for criticality safety and reactor physics analysis based on similarity assessment, and (3) quantify the effects of uncertainties in nuclear data and physical parameters for safety analysis.

The Tools for Sensitivity and Uncertainty Analysis Methodology Implementation (TSUNAMI)-1D, TSUNAMI-2D and TSUNAMI-3D analysis sequences compute the sensitivity of k_{eff} and reaction rates to energy-dependent cross section data for each reaction of each nuclide in a system model (Figure 9).

The 1D transport calculations are performed with XSDRNPM, the 2D transport calculations are performed using NEWT, and the 3D calculations are performed with KENO V.a or KENO-VI. The Monte Carlo capabilities of TSUNAMI-3D provide for sensitivity/uncertainty (S/U) analysis from either CE or MG neutron transport, where the deterministic capabilities of TSUNAMI-1D and TSUNAMI-2D only operate in MG mode. The Sensitivity Analysis Module for SCALE (SAMS) is applied within each analysis sequence to provide the requested S/U data. Whether performing a CE or MG calculation, energy-dependent sensitivity data are stored in group form in a sensitivity data file (SDF) for subsequent analysis. These sequences use the energy-dependent cross section covariance data to compute the uncertainty in the response value due to the cross section covariance data.

The Tool for Sensitivity Analysis of Reactivity Responses (TSAR) computes the sensitivity of the reactivity change between two k_{eff} calculations using SDFs from TSUNAMI-1D, TSUNAMI-2D, and/or TSUNAMI-3D. TSAR also computes the uncertainty in the reactivity difference due to the cross section covariance data.

TSUNAMI Indices and Parameters (TSUNAMI-IP) computes correlation coefficients that determine the amount of shared uncertainty between each target application and each benchmark experiment considered in the analysis. TSUNAMI-IP offers a wide range of options for more detailed assessment of system-to-system similarity. Additionally, TSUNAMI-IP can generate input for the Upper Subcritical Limit Statistical Software (USLSTATS) trending analysis and compute a penalty or additional margin needed for the gap analysis.

The Tool for S/U Analysis of Response Functions Using Experimental Results (TSURFER) predicts bias and bias uncertainty. TSURFER implements the generalized linear least-squares (GLLS) approach to data assimilation and cross section data adjustment that also uses the SDFs generated from TSUNAMI-1D, -2D, -3D, or TSAR. The data adjustments produced by TSURFER are not used to produce adjusted cross section data libraries for subsequent use; rather, they are used only to predict biases in application systems.

The Extensible SCALE Intelligent Text Editor (ExSITE) GUI facilitates analysis with TSUNAMI-IP, TSURFER, TSAR, and USLSTATS. The Validation, Interpretation and Bias Estimation (VIBE) interface is applied to examine SDF files, create sets of benchmark experiments for subsequent analysis, and gather additional information about each benchmark experiment.

Sampler is a super-sequence that performs general uncertainty analysis by stochastically sampling uncertain parameters that can be applied to any type of SCALE calculation, propagating uncertainties throughout a computational sequence. Sampler treats uncertainties from two sources: (1) nuclear data and (2) input parameters. Sampler generates the uncertainty in any result generated by any computational sequence through stochastic means by repeating numerous passes through the computational sequence, each with a randomly perturbed sample of the requested uncertain quantities. Figure 10 shows the Sampler computed probability distribution in neutron detector count rate computed from an active interrogation source and the presence of a randomly oriented fissile source at the bottom of a postulated barrel shipping container. Additional details for this analysis are provided in "Uncertainty Analysis for Neutron Active Interrogation Calculations," Keith C. Bledsoe, Matthew A. Jessee, and Douglas E. Peplow, Advances in Nuclear Nonproliferation Technology and Policy Conference, September 23–27, 2018.

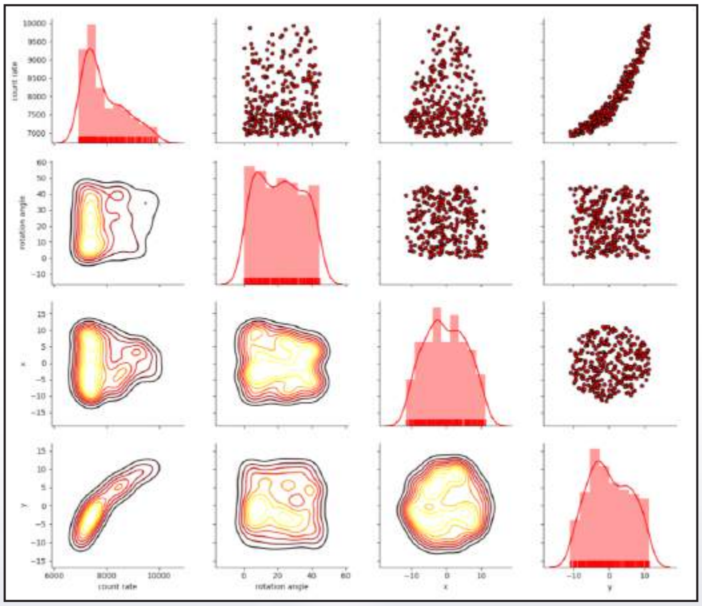


Figure 10. Uncertainty in neutron count rate due to uncertainty in the orientation of a fissile source in a shipping container

Material Specification and Cross Section Processing

Cross Section Processing (XSProc) provides material input and MG cross section preparation for most SCALE sequences. XSProc allows users to specify materials in the model through easily remembered, easily recognizable keywords associated with mixtures, elements, nuclides, and fissile solutions provided in the SCALE Standard Composition Library. For MG calculations, XSProc provides cross section temperature correction and resonance self-shielding, as well as energy group collapse and spatial homogenization for systems that can be represented in unit cell input data as infinite media, finite 1D systems, or repeating structures of 1D systems such as uniform arrays of fuel units. Improved resonance self-shielding treatment for nonuniform lattices can be achieved through the use of the Monte Carlo Dancoff (MCDancoff) code that generates Dancoff factors for generalized 3D geometries for subsequent use in XSProc. Cross sections are generated on a microscopic and/or macroscopic basis as needed. Although XSProc is most often used as part of an integrated sequence, it can be run without subsequent calculations to generate problem-dependent MG data for use in other tools.

Monte Carlo Transport

Monte Carlo transport is discussed throughout the previous sections.

Deterministic Transport

Deterministic transport is discussed throughout the previous sections.

Nuclear Data

The cross section data provided with SCALE include comprehensive CE neutron and coupled neutron-gamma data based on ENDF/B-VII.0 and ENDF/B-VII.1 (Figure 11).

These data have been generated with the AMPX codes. The MG data are provided in several energy-group structures optimized for different application areas, including criticality safety, lattice physics, and shielding analysis. The comprehensive ORIGEN data libraries are based on ENDF/B-VII.1 and recent joint evaluated fission and fusion file (JEFF) evaluations, and they include nuclear decay data, neutron reaction cross sections, neutron-induced fission product yields, delayed gamma ray emission data, and neutron emission data for over 2,200 nuclides. The photon yield data libraries are based on the most recent evaluated nuclear structure data file (ENSDF) evaluations. The libraries used by ORIGEN can be coupled directly with detailed, problem-dependent physics calculations to obtain self-shielded, problem-dependent cross sections based on the most recent evaluations. There are no limitations on compositions or energy spectra. SCALE also contains a comprehensive library of neutron cross section covariance data for neutron interactions and fission product yields, as well as decay data for use in S/U analysis with TSUNAMI codes and Sampler.

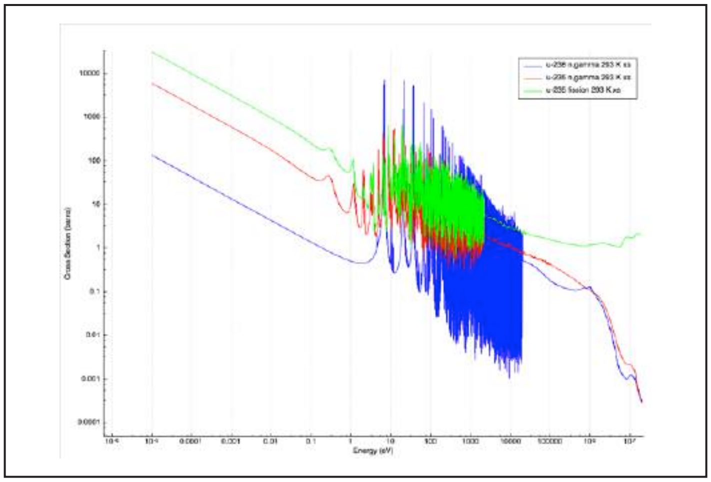


Figure 11. Nuclear data generated with AMPX

The full suite of AMPX codes for generating MG and CE neutron, gamma, and coupled neutron/gamma libraries and covariance data are also included in the SCALE distribution. This allows users to create their own nuclear data libraries, drawing from sources of data and energy group structures other than those provided with SCALE.

Graphical User Interfaces

Fulcrum is a cross platform GUI designed to create, edit, validate and visualize SCALE input, output, and data files (Figure 12).

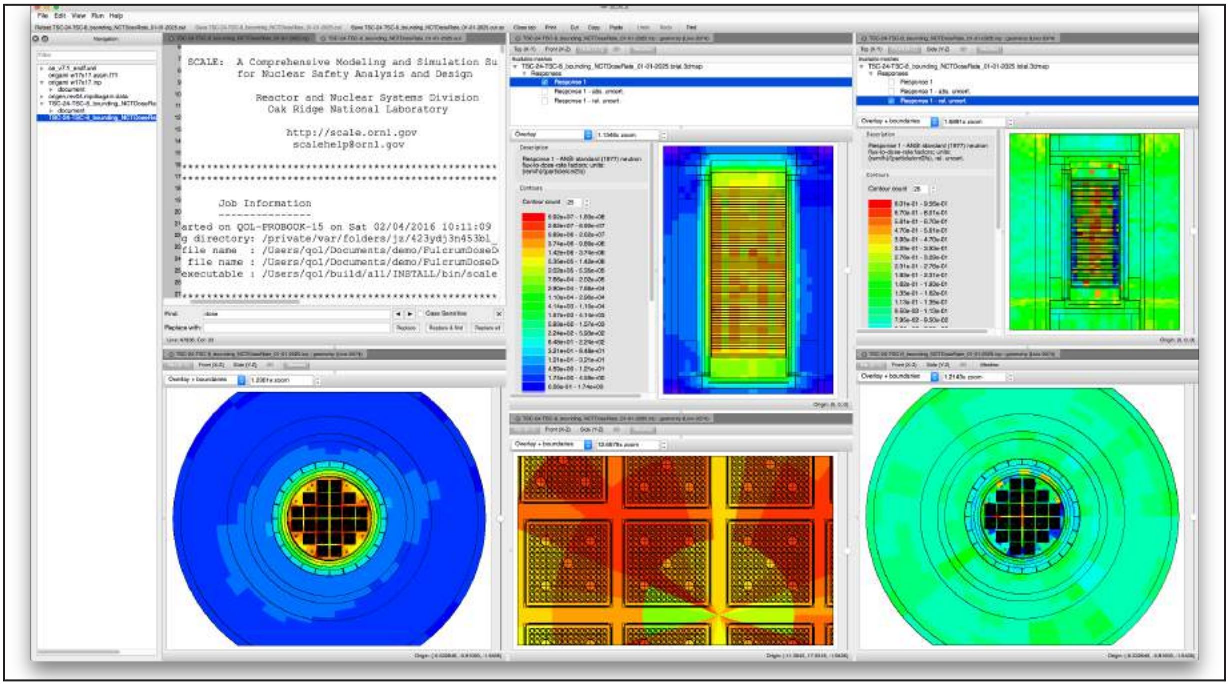


Figure 12. Fulcrum GUI

Fulcrum provides input editing and navigation, interactive geometry visualization for KENO V.a, KENO-VI, and NEWT, job execution, overlay of mesh results within a geometry view, and plotting of data from most SCALE file formats. An error checking parser interactively identifies poorly constructed input with spelling errors or data entry omissions for all SCALE sequences. The Hierarchical Input Validation Engine (HIVE) will identify allowed data ranges and interdependencies in the input and report inconsistencies to the user. Fulcrum will interactively process standard composition data to produce a mixing table, list expanded input aliases for review, provide an internal listing of input as is required for Sampler material and geometry perturbation analysis, and launch the SCALE sample problems. The layout of panels in Fulcrum is highly configurable to accommodate many user preferences.

ORIGAMI Automator, a GUI integrated with Fulcrum, facilitates the quantification of isotopics as a function of time for a large set of fuel assemblies, such as the complete inventory of a spent fuel pool. This tool was developed to support the NRC in severe accident analyses, but it can be adapted to many other uses.

Additional user interfaces include the KENO3D interactive visualization program for Windows for solid-body rendering of KENO geometry models, as well as the previously mentioned ExSITE and VIBE interfaces for S/U analysis. Several SCALE modules provide HTML-formatted output, in addition to the standard text output, to allow for convenient navigation through the computed results using the most common web browsers. Interactive color-coded output and integrated data visualization tools are key features.

Distribution

The SCALE code system continues to provide capabilities for the analysis needs of the multi-agency programs supporting SCALE. The system continues to grow in popularity with domestic and international users. Since 2004, 14,796 copies of SCALE have been distributed to 9,191 unique users in 61 nations (Figure 13).

Since the April 2016 release of SCALE 6.2, distribution centers have issued licenses for 4,864 copies of this latest SCALE version, including 1,875 distributions to new users who had not previously licensed any version of SCALE. The distribution of SCALE licenses over time is shown in Figure 14.

As seen in Figure 14, the growth in the rate of distribution of SCALE is observed as the slope of the distribution plot, with a marked increase after the release of SCALE 6.2.

The distribution of SCALE to end users is subject to US export control regulations, and each user must be individually licensed through an authorized distribution center. SCALE licenses are primarily issued through the Radiation Safety Information Computational Center (RSICC) at ORNL, with mirrors at the Organisation for Economic Cooperation and Development (OECD) Nuclear Energy Agency (NEA) Data Bank in France and the Research Organization for Information Science and Technology (RIST) in Japan. Any license fees collected for the distribution of SCALE are retained by these organizations to offset the costs of background checks and media duplication, and no part of the license revenue is used to support SCALE activities.

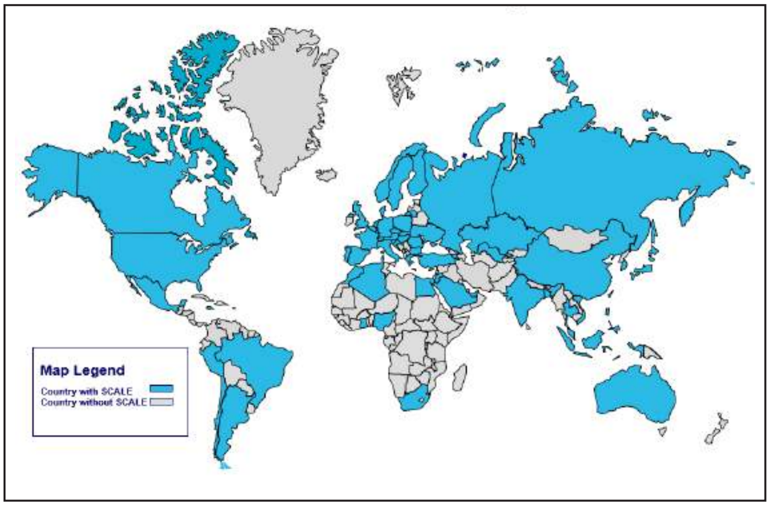


Figure 13. Nations with licensed SCALE users

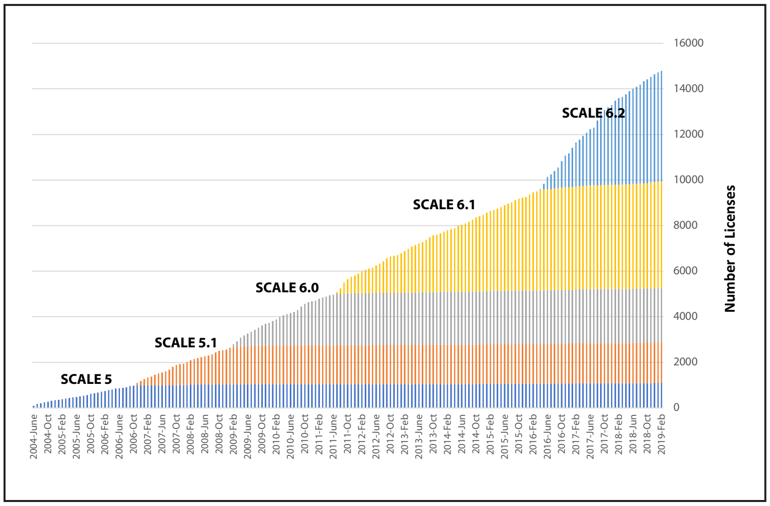


Figure 14. Number of licenses issued for SCALE 5–6.2

Training Courses and Workshops

SCALE training courses and workshops continue to be popular with users. Training is provided by developers and expert users from the SCALE team. Courses provide a review of theory, descriptions of software capabilities and limitations, and hands-on experience running problems of varying levels of complexity. In FY18, 13 weeklong courses were presented at ORNL, the OECD/NEA Data Bank, NRC headquarters, and at user facilities. Additionally, workshops were presented at conferences and universities. In total, SCALE training was presented to more than 150 participants from 14 nations. The training courses are funded through user registration fees and are self sustaining. Site-specific courses can be customized to meet the needs of many teams. Figures 15–17 show attendees from various SCALE training courses.

Figure 15. SCALE Criticality Safety and Radiation Shielding Course
Oak Ridge National Laboratory, Oak Ridge, Tennessee, November 2017

Figure 16. Source Terms and Radiation Shielding for Spent Fuel Transportation and Storage Applications Course
OECD/NEA, Paris, France, March 2018

Figure 17. Site-Specific SCALE/KENO-VI Criticality Calculation Course
Savannah River Nuclear Solutions, Aiken, South Carolina, July 2018

Users’ Group Workshop

The 2nd annual SCALE Users’ Group Workshop was held at Oak Ridge National Laboratory August 27–29, 2018, with 86 registered participants from the NRC, the US Department of Energy (DOE), national laboratories, industry, and academia. The opening plenary session featured keynote speaker Don Algama (NRC), who presented on SCALE’s capabilities that provide essential support to NRC’s licensing review. Some select photos from the workshop are shown in Figures 18–21 below.

Figure 18. Agenda of SCALE Users’ Group Workshop 2018

Figure 19. Participants in the SCALE Users’ Group Workshop 2018

Technical sessions were provided on the following topics:

- LWR reactor modeling with SCALE
- Advanced reactor modeling with SCALE
- Reactor safety analysis with SCALE
- Validation, verification, and uncertainty quantification
- Decommissioning, disposal and spent nuclear fuel burnup credit
- Applications of SCALE for nuclear criticality safety practitioners
- Applications of SCALE to nuclear safeguards and security
- Integration of SCALE capabilities for multiphysics analysis

Figure 20. Will Wieselquist teaching "Sampler applied to source term uncertainty" tutorial

The following two-hour tutorial sessions were presented in the same venue where SCALE training is offered semiannually at ORNL:

- Linking external C++ or Fortran software to the ORIGEN API
- Using ORIGAMI for detailed spent fuel assembly analysis in safeguards applications
- ORIGEN activation analysis of the Silene experiment
- Introduction to the Polaris lattice physics code
- Generation of SCALE multigroup fast reactor libraries with AMPX
- MAVRIC cask analysis with ORIGEN spent fuel sources
- KENO-VI modeling of double heterogeneous reactor systems using TRISO fuel
- Sampler applied to source term uncertainty

Figure 21. Brad Rearden presenting on emerging initiatives in SCALE

The full agenda, including links to the presentations, is available at:
https://www.ornl.gov/scale/scale/2018-scale-users-group-workshop

Photos from the workshop are available at:
https://www.ornl.gov/scale/scale/2018-scale-users-group-workshop-photos

Maintenance and Development Activities

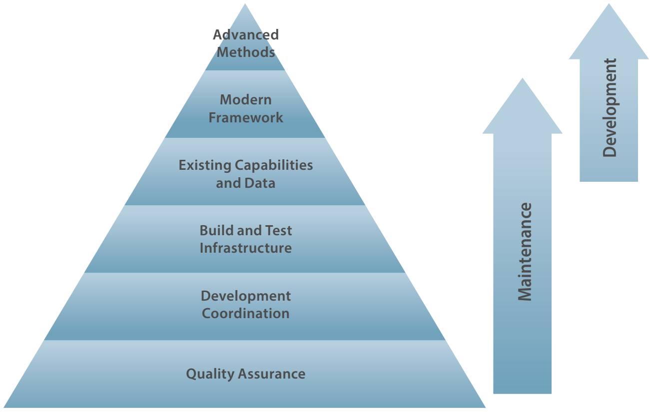


Figure 22. SCALE activities diagram

The primary goal of the SCALE maintenance and development activities is to ensure that the SCALE code system continues to meet the needs of sponsors and users by providing verified, validated results and remaining current with state-of-the-art computing technology.

SCALE maintenance activities provide an essential foundation for all activities related to reliable development and use of SCALE. Maintenance activities include quality assurance (QA), development coordination, build-and-test infrastructure, and support for all existing capabilities and features. Recently, the SCALE team has focused efforts on infrastructure modernization by reviewing and incrementally updating components and procedures which had evolved over a 40-year period, applying modern software development practices and QA standards. An essential component of this ongoing activity is the development of a modern framework for SCALE analysis which enables rapid development of advanced methods, parallel operation, and easy integration of SCALE tools with other analysis packages.

Development activities involve major enhancements and introduction of advanced methods to existing modules, as well as development of new modules, data libraries, and user interfaces. These activities employ current computing and programming techniques, building on the modernized framework of the overall SCALE code system, as illustrated in Figure 22.

Quality Assurance

Activities classified as maintenance begin with the establishment of the QA framework that is applied to all SCALE codes and data. As depicted in Figure 23, the SCALE QA program is kept current with international consensus standards (ISO-9001-2008, ASME NQA-1), DOE orders (DOE 414.1D), NRC guidelines (NUREG/BR-0167) and the ORNL Standards-Based Management System. A review of the SCALE QA plan is performed annually by the ORNL RNSD Software QA Board. The SCALE QA plan continues to be viewed as a model plan both inside and outside ORNL.

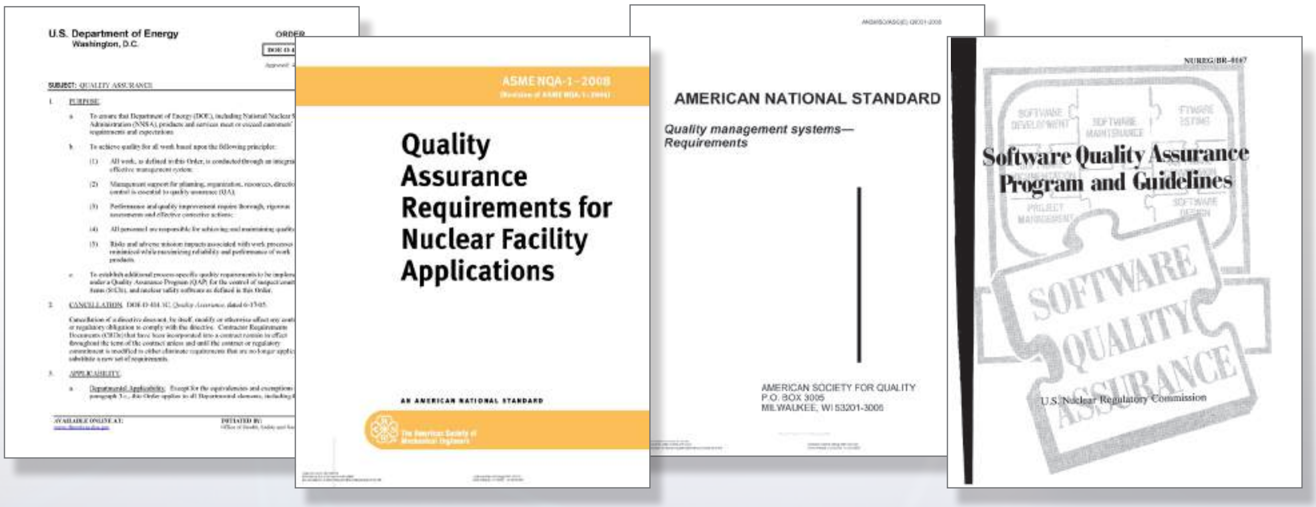


Figure 23. Reference documents for SCALE QA Program

Development Coordination

At ORNL, the SCALE code system is developed, deployed, and supported by dozens of staff members throughout RNSD. All SCALE activities are coordinated to facilitate consistency throughout the project, especially in the application of QA, development practices, and testing strategies. The SCALE Leadership Team consists of the SCALE director, line managers, program managers, and developers as designated by the SCALE director. The Leadership Team meets regularly to discuss the current status and to make programmatic and managerial decisions regarding SCALE.

SCALE teams are organized to coordinate work activities within given areas as shown in Figure 24. Each team meets independently to plan and coordinate work activities. The teams are organized so that members from different work areas are included on multiple teams to improve communication and coordination between work areas. Although the activities of most teams are supported by targeted development tasks, coordination of the teams and review of their work is supported as a maintenance activity. A weekly forum for developers and users is conducted to maintain a productive dialog and collaborative mission among developers, users, and managers throughout ORNL.

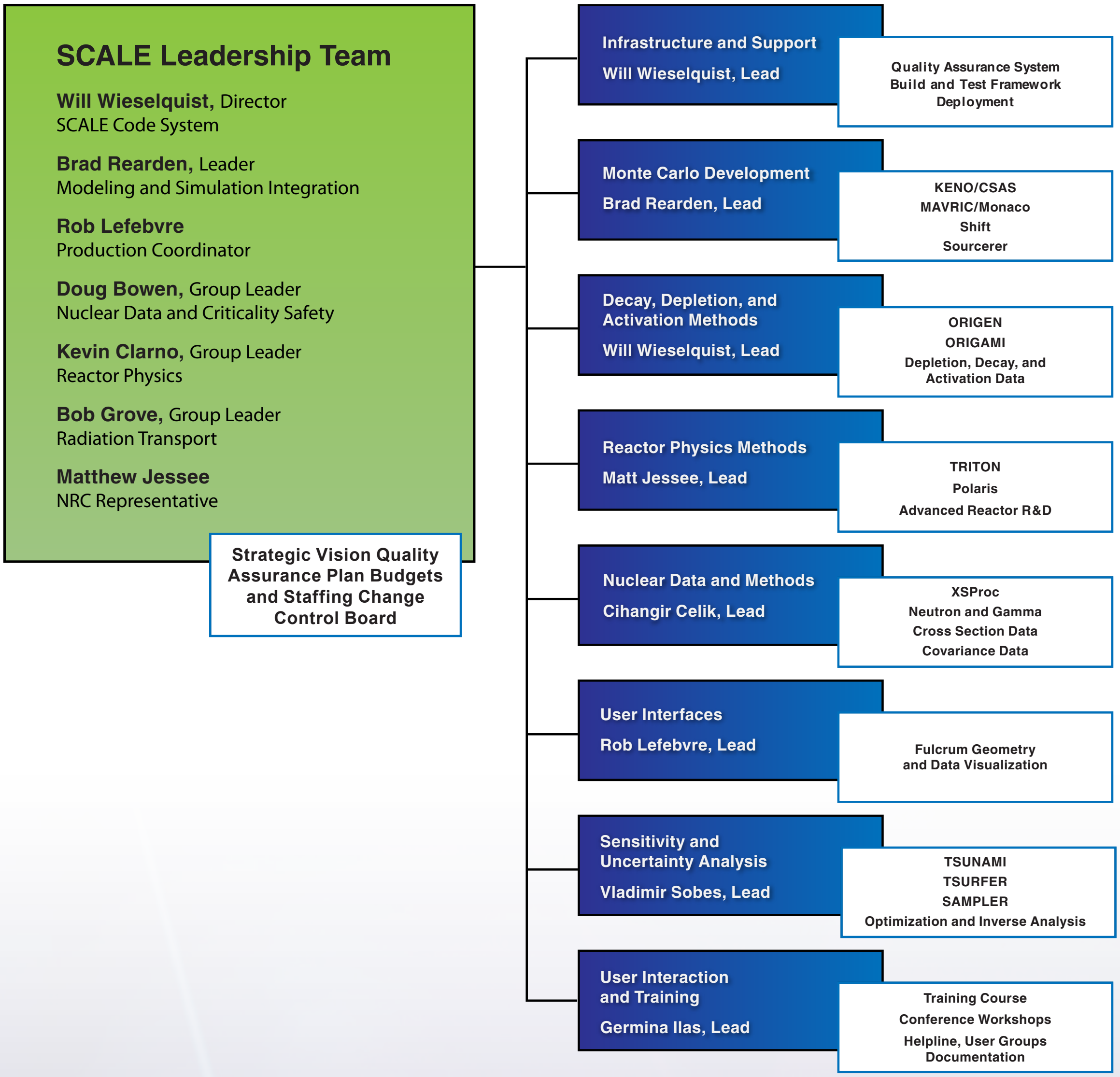


Figure 24. FY18 SCALE team structure (end of FY18)

Figure 25. SCALE 6.2 Team – May 2016
(Left to right) Ahmed Ibrahim, Germina Ilas, Brandon Langley, Andrew Holcomb, Shane Hart, Cihangir Celik, Seth Johnson, Matthew Jessee, Kevin Clarno, Adam Thompson, Bob Grove, Rob Lefebvre, Greg Davidson, Charles Daily, Alan Icenhour, Barbara Snow, Brian Ade, Brad Rearden, Ben Betzler, B. J. Marshall, Kursat Bekar, Will Wieselquist, Mark Baird, Mark Williams, Georgeta Radulescu, Ron Ellis, Thomas Miller, Dan Ilas, Elizabeth Jones, Cecil Parks, Sheila Walker, Teresa Moore, Marsha Henley, Sandra Poarch, Lester Petrie

In addition to the Leadership Team personnel and the team leads shown in Figure 24, almost 50 team members (Table 2, Figure 25) contribute to SCALE on a routine or occasional basis.

Table 2. SCALE team members

- Brian J. Ade
- Lindsey D. Aloisi
- Goran Arbanas
- Mark Baird
- Kaushik Banerjee
- Kursat B. Bekar
- Benjamin R. Betzler
- Elliot Biondo
- Keith C. Bledsoe
- Friederike Bostelmann
- Justin B. Clarity
- Charles R. Daily
- Gregory G. Davidson
- Kevin Dugan
- Colby Earles
- Thomas M. Evans
- Ian C. Gauld
- Cole A. Gentry
- Andrew T. Godfrey
- Steven P. Hamilton
- Shane W. Hart
- Marsha D. Henley
- Andrew M. Holcomb
- Jianwei Hu
- Germina Ilas
- Seth R. Johnson
- Kang Seog Kim
- Brandon R. Langley
- William J. Marshall
- Ugur Mertyurek
- Paul Miller
- Thomas M. Miller
- Tara M. Pandya
- Douglas E. Peplow
- Christopher M. Perfetti
- Joshua L. Peterson-Droogh
- Marco T. Pigni
- Sandra J. Poarch
- Georgeta Radulescu
- Rose B. Raney
- Joel M. Risner
- Katherine Royston
- Diane J. Sams
- Ellen Saylor
- John M. Scaglione
- Steven E. Skutnik
- Vladimir Sobes
- Adam B. Thompson
- Dorothea Wiarda
- Jinan Yang

Build and Test System

The success of any ongoing software project requires routine compilation and testing of software and data, along with providing continual support for the latest hardware and compilers. For SCALE, this foundation is provided as a maintenance activity.

All changes to the SCALE source code are recorded and versioned in a repository system. This system streamlines the development process, facilitates easier collaboration between developers, and provides easier quantification of changes to improve the QA review process.

After each incremental update to the source code, a suite of over 2,000 test cases is run on each of dozens of computer platform configurations, including Linux, Mac, and Windows, with different compilers and compiler options. This rigorous testing is performed on demand and automatically per SQA process each day. The results of the tests and the associated changes are reported to an internal website, the SCALE Dashboard. All developers can review the Dashboard to monitor the performance of numerous SCALE features on different platforms with different compilers using a pass/fail metric, eliminating the need for users to configure and run all tests themselves. In FY18, the SCALE code version and hardware management systems were migrated from Mercurial and Jenkins to Git and an ORNL-hosted GitLab. This migration allows for improved flexibility and agility testing in SCALE. The hardware listed in Table 3, which consists of 436 processors and 1,672 GB RAM, is dedicated to running on-demand and automated SCALE testing 24 hours a day, 7 days a week.

Table 3. SCALE continuous integration hardware

Platform	Hardware
Linux	• 8 cluster nodes each with 8 processors and 32 GB RAM • 1 dedicated computer with 64 cores and 256 GB RAM
Mac	• 3 Mac Pro computers each with 16 processors and 20 GB RAM • 2 Mac Pro computers each with 24 processors and 64 GB RAM
Windows	• 2 Windows 7 computers each with 8 processors and 16 GB RAM • 1 Windows 7 computer with 16 processors and 12 GB RAM • 1 Windows 2012 Server with 32 processors and 128 GB RAM • 2 Windows 2012 R2 Servers with 44 processors and 128 GB RAM • 3 Windows 2016 Servers with 20 processors and 256 GB RAM

Existing Capabilities and Data

SCALE 6.2 consists of approximately 2,000,000 lines of source code for 77 executable modules, 43 GB of nuclear data in approximately 9,000 files, and more than 2,700 pages of user documentation. With 9,191 licensed users in 61 nations, extensive communication is required. The SCALE team provides ongoing support to users. The team addressed approximately 886 inquiries during FY18 through scalehelp@ornl.gov email. Additionally, an online discussion forum is available for SCALE users to post and review issues as a community (https://groups.google.com/forum/#!forum/scale-users-group). User communication in the form of website postings and newsletters is also provided.

Targeted development tasks generate dozens of new capabilities each year, and at the end of each development task, enhancements and user support for these features, additional testing and bug fixes, and integration of new features with existing features are supported as maintenance activities.

Modern Framework

The foundation of modern SCALE is a modular C++ software framework for efficient operation that also enables parallel computations. Individual computational components communicate through an efficient in-memory application programming interface (API) instead of slow file input/output (I/O) to the hard disk used in earlier releases. APIs also enhance communication between components by allowing for clear requirements on the data I/O of each modular component. Each capability that provides an API is referred to as a module. Where internal tests are applied to ensure that data passed through the API meet all requirements of the module, linkages with other modules can be efficiently modified without disrupting any part of the overall system. The concept of individual functional modules as stand-alone executable programs will diminish as individual physics capabilities are consolidated into a unified, executable program capable of performing all SCALE functionality within an efficient parallel infrastructure. Additionally, the modern API-based framework enables the development of a modern GUI that implements the same modules used for computational analysis, eliminating the need to develop and maintain a feature twice, once for computational use and again for the GUI.

Advanced Methods

Advanced methods are developed as targeted tasks unless an incremental advancement is required to correct a discrepancy or enhance an existing feature for compatibility with a new feature. However, once an advanced method is complete, QA and maintenance activities are usually required to continue to provide support for that method. Thus, as new features are integrated into SCALE, the amount of maintenance required is incrementally increased pending removal of deprecated features. While many advanced methods were introduced with SCALE 6.2, the SCALE modernization plan details additional advancements, culminating in the fully modernized SCALE 7. A key aspect of SCALE 7 is the replacement of the KENO and Monaco Monte Carlo codes with the advanced Shift Monte Carlo code.

Ongoing Development

The SCALE team is dedicated to supporting the advanced features provided in SCALE 6.2 and is working to extend these capabilities for additional types of analysis, such as very large, complex interim storage sites for used fuel; analysis of advanced reactors including molten salt reactors (MSRs), fluoride-salt-cooled high-temperature reactors (FHRs), high-temperature gas-cooled reactors (HTGRs) and sodium-cooled fast reactors (SFRs); analysis of advanced technology fuels; and advanced validation approaches for new or challenging systems. Existing capabilities will continue to be improved through additional efficiency and accuracy gains, as well as additional enhancements to the user interface. The development of many of these capabilities is in progress now to be available with the release of SCALE 6.3. The nuclear data generated by the AMPX tools for all SCALE CE, MG, activation/decay, and covariance libraries will continue to be improved through an iterative development cycle that includes increased testing under the QA plan and timely deployment of the most current nuclear data libraries. Modernization plans for SCALE and AMPX include increased synchronization of development activities and shared resources between these two projects. Several specific initiatives are described in more detail below.

Integration of the Shift Monte Carlo Code

It is desirable to position SCALE for the future with an extensive reprogramming of existing capabilities to improve run-time performance and solution fidelity. The most significant changes planned for the future are the ability to execute SCALE in parallel on multiple central processing units (CPUs), whether on desktops, workstation clusters, or high-performance supercomputers. This strategy includes the integration of the Shift Monte Carlo code, which is capable of excellent parallel scaling on leadership class computing architectures such as ORNL's TITAN machine, which includes approximately 300,000 processors. However, the integration of Shift is also important for the desktop and workstation user, as the modern and efficient design of Shift provides single processor calculations that are 2–4 times faster than KENO-VI. The Shift Monte Carlo code leverages many SCALE modernization capabilities such as the input processing, nuclear data resources, and modules for CE and MG physics, modular geometry, sensitivity/uncertainty analysis, and depletion (Figure 26). The staged migration and testing of individual SCALE capabilities in the modern framework ensures robust development, testing, and deployment of this new tool. The long-term modernization plan includes full modularity and parallelization in SCALE 7, including the integration of the Shift Monte Carlo code.

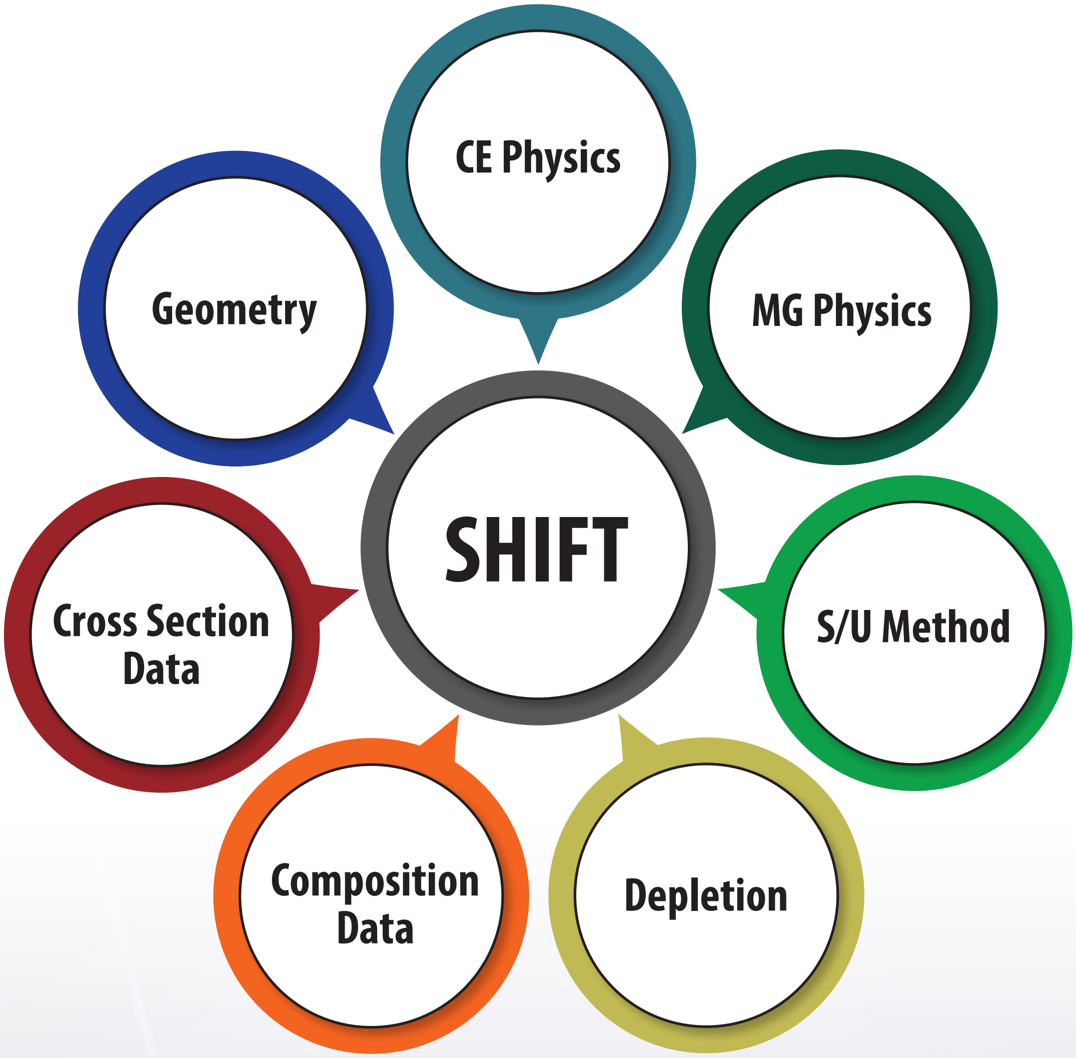


Figure 26. Advanced Monte Carlo methods with Shift

SCALE 6.3 Development for Advanced Reactors and Advanced Technology Fuels

Several projects are under way in FY18 to develop new code and data capabilities in SCALE 6.3 for modeling advanced reactors and advanced technology fuels (ATFs), sometimes referred to as accident-tolerant fuels. Historically, many of SCALE's capabilities have been developed and applied to LWR fuel applications, and the NRC is sponsoring the extension of these capabilities to support the regulatory review of advanced concepts. Most of the recent enhancements focus on the Polaris lattice physics code and the high performance, massively parallel Monte Carlo code Shift. In addition, new MG cross section libraries are being developed for non-LWR applications, and the integration of SCALE with other NRC licensing tools is being improved.

Lattice Physics for Advanced Concepts

For Polaris, the non-LWR capabilities under development include hexagonal geometry support to simulate HTGRs, SFRs, and prismatic assembly designs. Additionally, a double-heterogeneity modeling capability will be added to support HTGR prismatic analysis and ATF based on TRISO-particle fuel forms. For MSRs, a time-dependent chemical processing model and delayed neutron precursor drift model are being integrated into TRITON for SCALE 6.3 (Figure 27). Another new feature is defining the branch and history requirements in Polaris for advanced reactor modeling with PARCS or other nodal core simulators.

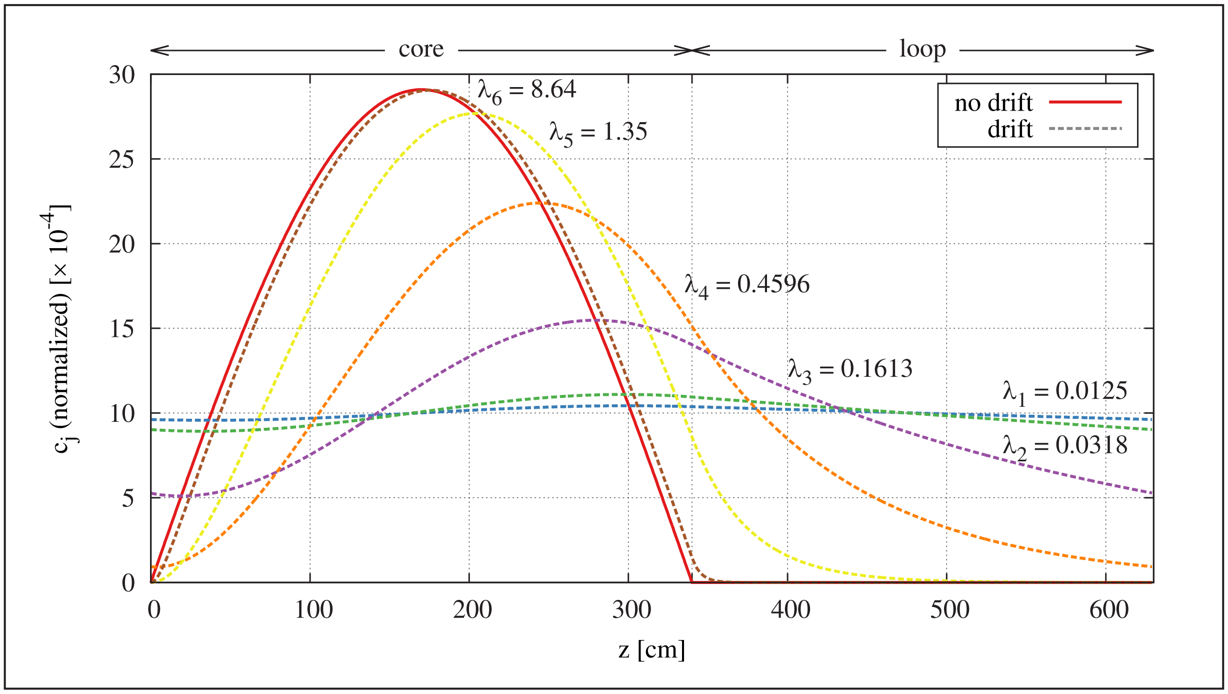


Figure 27. MSR delayed neutron precursor drift modeling

Several ATF and advanced cladding concepts are being considered by industry. Some of these concepts are planned for lead test rods in the next one to two years. The SCALE team is assessing SCALE neutronics capabilities for ATF designs, including the identification of relevant benchmark experiments for validation, and code enhancements to improve SCALE's modeling accuracy. The focus is on lattice level investigations for ATF concepts, such as Cr-doped UO_2, greater-than-5% enriched UO_2, advanced cladding types, and uranium-silicide fuel. Once the assessment is completed, Polaris will be updated for the accurate modeling of ATF fuel concepts. Potential enhancements may include modifications to the energy group structure in the MG library, updates to the nuclear data library such as modified self-shielding factors or scattering data, and updates to the Polaris input interfaces for simple definition of ATF compositions or geometries.

Another capability under development will enable lattice physics calculations with Shift through the established Polaris input and output definitions. This capability will provide reference solutions for non-LWR fuel designs; nodal cross section data will be generated via Shift's CE Monte Carlo solution using the same inputs as the Polaris MG approach. Polaris, which is designed for MG calculations, uses the ESSM to generate problem-dependent cross sections and a MoC transport solver to generate flux solutions that are subsequently used to produce nodal core simulator data for PARCS. The Shift Monte Carlo interface will allow definition of the Monte Carlo sampling parameters and tallies needed for nodal cross section generation. The construction of the lattice geometry will be updated to create Shift native geometry.

Shift Integration for Depletion and Nodal Data Generation

Shift is being integrated into the TRITON sequence to provide high-fidelity CE Monte Carlo depletion capabilities. CE depletion is currently available in the SCALE 6.2 TRITON depletion sequence through use of the KENO Monte Carlo code for neutron transport calculations. However, the KENO reference solution requires significant computational resources because the codes were not designed for large parallel calculations and are inadequate for full core reactor analysis. To support simulation of advanced reactor concepts requiring increasingly complex geometry, a highly parallelizable reference solution is needed. This solution will be provided by Shift.

Additional features are being added to generate few-group nodal cross sections using Shift. Currently, nodal data can only be generated for 2D geometries in SCALE using NEWT or Polaris. Advanced reactors differ significantly from LWRs in geometry and neutron spectra, necessitating different solution methods. The current MG methods are highly optimized for LWRs. Rather than generate a new group structure and cross section processing method for each advanced reactor class, a CE Monte Carlo nodal data generation solution using Shift will be applicable for any solid-fuel reactor design and scalable to high-performance computing platforms. Particle-based fuel designs such as TRISO require significant complexities for the user to model. The geometric placement of individual fuel grains and/or fuel pebbles will be automated so the user may simply specify the number of particles in a fuel volume or the number of pebbles in a core.

Shift Integration for Criticality and Shielding Analysis

As part of the SCALE modernization effort, Shift is also being implemented into SCALE to replace KENO for criticality safety (CSAS) and sensitivity/uncertainty (TSUNAMI) calculations. A new MAVRIC radiation shielding sequence using Shift is under development. It will use Shift instead of Monaco for hybrid deterministic / Monte Carlo radiation shielding applications. For SCALE 6.3, the existing KENO- and Monaco-based sequences, as well as the updated Shift sequences, will be available to facilitate transition to the enhanced capabilities.

Multigroup Nuclear Data for Advanced Concepts

For nuclear data needed to support advanced reactors and ATF, a generic very fine (VF) 1000+ group library is being developed that is applicable to a wide range of reactor spectra, including thermal and fast systems. This VF library will be available to generate collapsed application-specific libraries. Recommended collapsed group structures may be provided for different reactor concepts, but only the generic VF library will be maintained and distributed with SCALE. An automated capability for users to collapse reactor-specific libraries from the generic VF library is also planned for development in 2019, following SCALE 6.3.

Improved SCALE Integration for Fuels Performance and Severe Accident Analysis

The SCALE connectivity to other NRC licensing tools is also being enhanced by improving the interface for SCALE source terms to MELCOR and the MELCOR Accident Consequence Code System (MACCS) for severe accident analysis. Capabilities are also being included for SCALE to provide power distributions and burnup information for the new FAST fuels performance code, which is integrating and extending the capabilities of FRAPCON and FRAPTRAN for current and advanced concepts.

Enhancements in the Fulcrum User Interface

Users of Fulcrum often request the ability to visualize geometry in 3D. In a significant enhancement for SCALE 6.3, advanced 3D capabilities will be available. As demonstrated in Figure 28, the new capabilities allow for custom model cutting, the ability to undo a model cut, transparency layers, and many other desirable features.

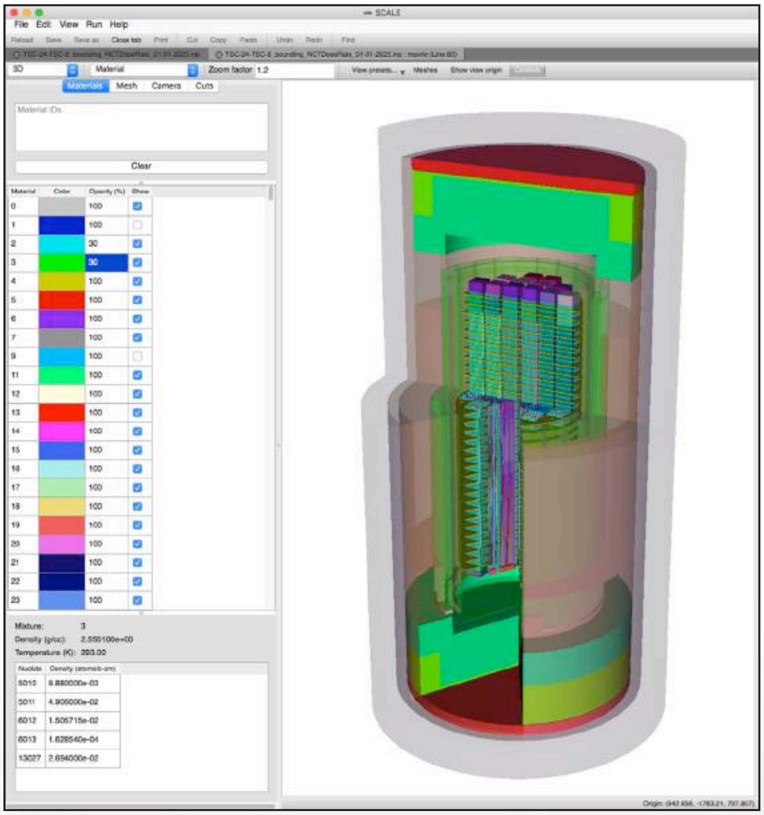


Figure 28 . SCALE 6.3 Fulcrum user interface
3D visualization of a spent fuel canister

Sponsored Activities

The maintenance and development of SCALE and AMPX are supported by several sustaining sponsors who have provided support over many years, as well collaborating sponsors who interact with the SCALE team for particular enhancements important to their missions or integration with their tools. Since 1976, the NRC has been the historical lead sponsor in the development of SCALE, with support provided by both the Office of Nuclear Material Safety and Safeguards (NMSS) and the Office of Nuclear Regulatory Research (RES). Since 1987, SCALE maintenance and development activities have been cosponsored by DOE and the National Nuclear Security Administration (NNSA). Details on sponsors are in Table 4 provided below.

Table 4. Sponsor information

Sponsor		Description
Sustaining Sponsors	NRC/NMSS/Division of Spent Fuel Management (DSFM)	Criticality safety analysis, shielding, source terms, and validation methods for spent nuclear fuel licensing
	NRC/RES/Division of Systems Analysis/Fuel and Source Term Code Development Branch	Nuclear data, lattice physics, criticality safety, depletion, shielding, source terms, and validation for current and advanced reactor licensing
	NNSA/Nuclear Criticality Safety Program (NCSP)	Criticality safety analysis, validation methods, criticality accident alarm system analysis, and nuclear data processing
Collaborating Sponsors	DOE/Office of Nuclear Energy (NE)/ Nuclear Energy Advanced Modeling and Simulation (NEAMS)	Depletion and decay methods, nuclear data uncertainty analysis, user interfaces, and integration with other NEAMS tools
	DOE/NE/Consortium for Advanced Simulation of Light Water Reactors (CASL)	Cross section data and methods integrated with CASL tools
	DOE/Technology Commercialization Fund (TCF)	Enhancements for MSRs
	NNSA/Office of Defense Nuclear Nonproliferation (NA-22)	Enhancements for nonproliferation analysis

scale

Nuclear Systems Modeling & Simulation

http://scale.ornl.gov