scale
Nuclear Systems Modeling & Simulation

Newsletter

Number 43 Summer 2011

In This Issue

Scale 6.1 Released!

The Scale code system, developed at Oak Ridge National Laboratory (ORNL), provides a comprehensive, user-friendly tool set for criticality safety, reactor physics, spent fuel characterization, radiation shielding, and sensitivity and uncertainty analysis. Since its first release in 1980, regulators, licensees, and researchers around the world have used Scale for safety analysis and design. Scale 6.1 has improved reliability and introduces a number of enhanced features in a flexible and robust, yet user-friendly, package that is intended to improve design and safety analysis in the nuclear community.

This newsletter presents an overview of the features of Scale 6.1 and emphasizes important enhancements over the previous release. The Scale user manual, located in scale6.1/scaleman after installation, contains comprehensive documentation of all computational capabilities, nuclear data, input requirements, and output edits. Additional resources are available in the Scale primers, which provide step-by-step instructions for running Scale using the graphical user interfaces. The primers are located in the scale6.1/primers directory after installation and also are available on the Scale website, http://scale.ornl.gov. Numerous additional resources are available on the Scale website or by e-mailing scalehelp@ornl.gov.

Scale 6.1 is available from the Radiation Safety Information Computational Center (RSICC) with source code and as an executable-only edition, package CCC-785, http://riscc.ornl.gov. Scale will also be available from the OECD NEA Data Bank, http://www.oecd-nea.org/dbprog/.

Scale 6 *Nuclear Technology* Special Edition

Scale was featured in a special edition of *Nuclear Technology* in May 2011. The following articles contained therein comprehensively review Scale features as of the Scale 6.0 release.

Foreword: Special Issue on the Scale Nuclear Analysis Code System
John C. Wagner
Scale 6: Comprehensive Nuclear Safety Analysis Code System
Stephen. M. Bowman
Resonance Self-Shielding Methodologies in Scale 6
Mark L. Williams
Isotopic Depletion and Decay Methods and Analysis Capabilities in Scale
Ian C. Gauld, Georgeta Radulescu, Germina Ilas, Brian D. Murphy, Mark L. Williams, Dorothea Wiarda
Reactor Physics Methods and Analysis Capabilities in Scale
Mark D. DeHart, Stephen M. Bowman
Monte Carlo Criticality Methods and Analysis Capabilities in Scale
Sedat Goluoglu, Lester M. Petrie, Jr., Michael E. Dunn, Daniel F. Hollenbach, Bradley T. Rearden
Sensitivity and Uncertainty Analysis Capabilities and Data in Scale
Bradley. T. Rearden, Mark. L. Williams, Matthew. A. Jessee, Donald. E. Mueller, Dorothea. A. Wiarda
Monte Carlo Shielding Analysis Capabilities with MAVRIC
Douglas E. Peplow

Overview of Scale 6.1

The Scale system consists of easy-to-use analytical sequences, which are automated through control modules to perform the necessary data processing and manipulation of well-established computer codes. Computations with Scale are typically characterized by the type of analysis to be performed (e.g., criticality, shielding, or lattice physics) and the geometric complexity of the system being analyzed. The user then prepares a single set of input in terms of easily visualized engineering parameters specified in a simplified, free-form format. The analytical sequence is defined by this single input specification. The Scale control modules use this information to derive additional parameters and prepare input for each of the functional modules necessary to achieve the desired results, especially with the Scale radiation transport codes that employ discrete ordinates, Monte Carlo, or hybrid methods.

Many new features and numerous improvements are incorporated in Scale 6.1. Highlights of these enhancements comprise most of the remainder of this issue.

Material Input and Problem-Dependent Cross-Section Data

A foundation of Scale is the MIPLIB (Material Information Processor Library). The purpose of MIPLIB is to allow users to specify materials using easily remembered and easily recognizable keywords that are associated with mixtures, elements, and nuclides provided in the Scale Standard Composition Library. An example of material input is shown in Fig. 1. MIPLIB also uses other keywords and simple geometry input specifications to prepare input for the modules that perform the problem-dependent cross-section processing. Even when performing multigroup calculations, Scale begins with continuous-energy cross-section data and generates problem-dependent multigroup data based on a pointwise spectrum generated with the CENTRM (Continuous Energy Transport Module) and PMC (Produce Multigroup Cross Sections) modules. An example infinitely dilute and self-shielded multigroup cross section and the pointwise cross section and problem-dependent flux used to produce the shielded data are shown in Fig. 2. A keyword supplied by the user selects the cross-section library from a standard set provided in Scale or designates the reference to a user-supplied library.

For Scale 6.1, the nuclear masses in the Scale Standard Composition Library and all cross-section libraries were updated to reflect the most recent evaluations. Nuclear masses are now available for over 3200 nuclides, nearly an order of magnitude more than are available in ENDF.

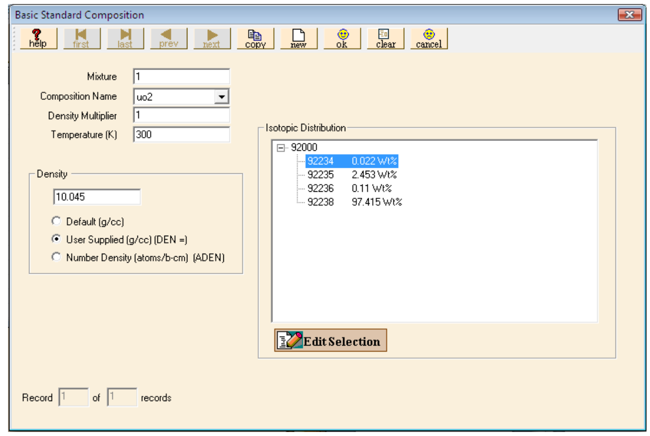

Fig. 1. Scale material input in the GeeWiz graphical user interface

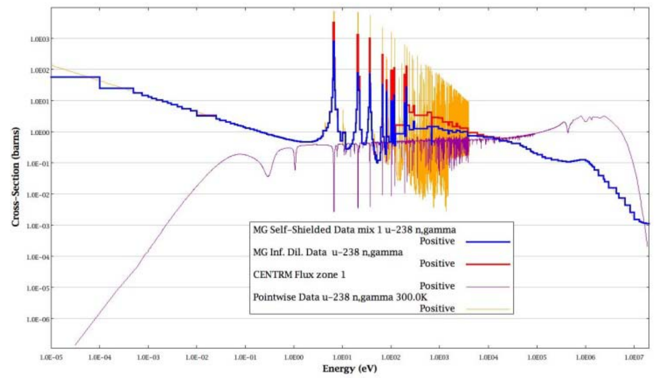

Fig. 2. Resonance self-shielding with CENTRM/PMC

Criticality Safety Analysis

The CSAS (Criticality Safety Analysis Sequence) control module provides for the calculation of the neutron multiplication factor of a system. Computational sequences accessible through CSAS provide automated problem-dependent processing of cross-section data and enable general analysis of a one-dimensional (1D) system model using deterministic transport with XSDRNPM or three-dimensional (3D) Monte Carlo transport solution using KENO V.a. CSAS also provides capabilities to search on geometry spacing or nuclide concentrations, and provides problem-dependent cross-section processing without subsequent transport solutions for use in executing stand-alone functional modules. An example KENO V.a model is shown in Fig. 3. CSAS6 is a separate criticality control module that provides automated problem-dependent cross-section processing and Monte Carlo criticality calculations via the KENO-VI functional module that uses the Scale Generalized Geometry Package (SGGP). The Scale Material Optimization and Replacement Sequence (SMORES) is a Scale control module developed for 1D eigenvalue calculations to perform system criticality optimization. The STARBUCS (Standardized Analysis of Reactivity for Burnup Credit using Scale) control module has been developed to automate the generation of spatially varying nuclide compositions in a spent fuel assembly, and to apply the spent fuel compositions in a 3D Monte Carlo analysis of the system using KENO, primarily to assist in performing criticality safety assessments of transport and storage casks that apply burnup credit. The KMART (Keno Module for Activity-Reaction Rate Tabulation) module produces reaction rates and group collapsed data from KENO. The USLSTATS (Upper Subcritical Limit Statistics) tool provides trending analysis for bias assessment.

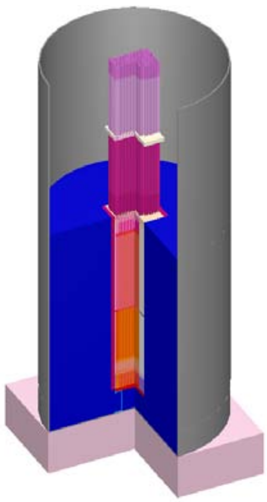

Fig. 3. KENO model of a mixed-oxide critical experiment

For Scale 6.1 a number of enhancements have been implemented in KENO. The mesh flux and fission source accumulator used in TSUNAMI-3D sensitivity analysis sequences and in criticality accident alarm system analysis have been improved with more flexibility in the user definition of mesh intervals and better mesh volume calculations, mesh tracking, and output edits. Mesh fission source data can now be generated using KENO in multigroup or continuous-energy mode, and the fission distribution can be visualized with the MeshView tool, as shown in Fig. 4. Default criticality search parameters have been modified to provide improved convergence to true minimum or maximum values, and region mean free paths can now be computed in continuous-energy mode.

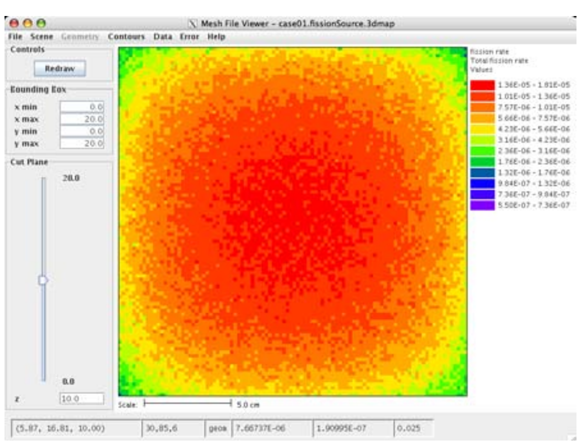

Fig. 4. Mesh fission source from KENO V.a visualized in MeshView

Shielding Analysis

The MAVRIC (Monaco with Automated Variance Reduction Using Importance Calculations) fixed-source radiation transport sequence is designed to apply the multigroup fixed-source Monte Carlo code Monaco to solve problems that are 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 CADIS (Consistent Adjoint Driven Importance Sampling) methodology, which uses an importance map and biased source that are derived to work together. MAVRIC generates problem-dependent cross-section data and then automatically performs a coarse mesh, 3D discrete ordinates transport calculation using Denovo to determine the adjoint flux as a function of position and energy, and to apply the information to optimize the shielding calculation in Monaco. An example MAVRIC cask model and CADIS-based mesh tally are shown in Fig. 5. The SAS1 (Shielding Analysis Sequence No. 1) control module provides general 1D deterministic shielding capabilities, and QADS (Quick and Dirty Shielding) provides for 3D point-kernel shielding analysis.

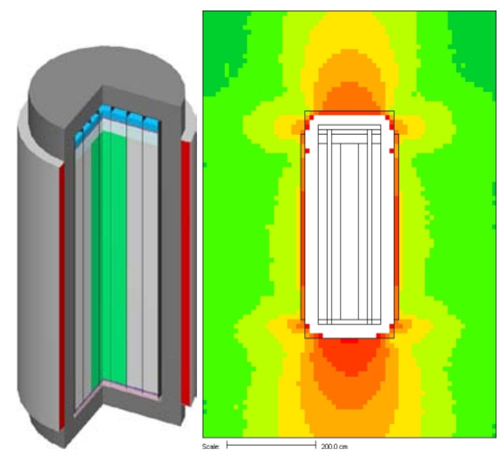

Fig. 5. MAVRIC model of spent fuel shipping cask and mesh tally of dose rate outside the cask

The MAVRIC tools have been updated with a number of enhancements for Scale 6.1. Multiple sources may now be defined with spatial distributions defined within each source. Energy distributions can be imported from an ORIGEN binary concentration file or from response functions read from an AMPX cross-section file. MAVRIC also includes improvements in the advanced variance reduction capabilities such as a macro materials option for improved Denovo deterministic simulations used to generate variance reduction parameters and increased flexibility in forward-weighting strategies. Cylindrical mesh grids have been added to more accurately capture spatial effects, as shown in Fig. 6, for a shielding calculation for a spent fuel shipping cask, and a suite of MAVRIC utilities has been developed to postprocess data files.

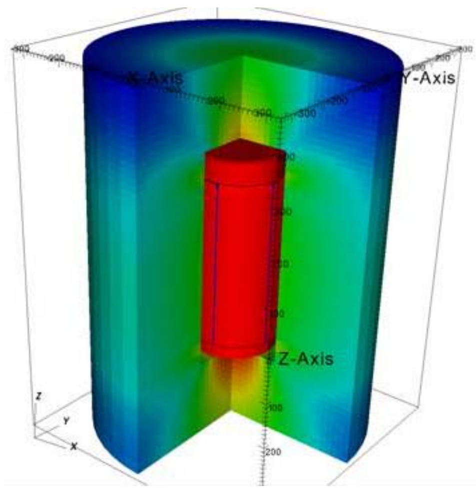

Fig. 6. MAVRIC spent fuel shipping cask model with a cylindrical mesh tally showing dose

Depletion, Decay, and Radioactive Source Term Analysis

The ORIGEN (Oak Ridge Isotope Generation) code applies a matrix exponential expansion model to calculate 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. 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 utilize multigroup cross sections processed from standard 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 are representative of conditions within any given reactor or fuel assembly, and convert these cross sections into a library that can be used by ORIGEN. Time-dependent cross-section libraries may be produced that 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.

The ORIGEN and COUPLE codes have been improved substantially for Scale 6.1. Support is now provided for multigroup cross-section libraries in any group structure, ENDF/B-VII decay libraries, and energy-dependent fission product yields. Cross-section transitions can be included from multiple sources, including JEFF-3.0/A-based AMPX format multigroup cross-section libraries developed for burnup and activation applications, an AMPX library generated by one of the Scale transport codes, and cross sections input manually by the user via the input file.

Reactor Analysis

The TRITON (Transport Rigor Implemented with Time-Dependent Operation for Neutronic Depletion) control module provides flexible capabilities to meet the challenges of modern reactor designs by providing 1D pin-cell depletion capabilities using XSDRNPM, two-dimensional (2D) lattice physics capabilities using the NEWT 2D flexible mesh discrete ordinates code, or 3D Monte Carlo depletion using KENO. With each neutron transport option in TRITON, depletion and decay calculations are conducted with ORIGEN. Additionally, TRITON can produce assembly-averaged few-group cross sections for use in core simulators. Improved resonance self-shielding treatment for nonuniform lattices can be achieved through use of the MCDancoff (Monte Carlo Dancoff) code that generates Dancoff factors for generalized 3D geometries. An example TRITON/NEWT model of a boiling water reactor fuel bundle is shown in Fig. 7.

The TRITON capabilities have incorporated a number of enhancements for Scale 6.1. The KENO-based Monte Carlo depletion capabilities have been substantially improved to more accurately compute power distributions and enable all KENO functionalities such as source specification, region volume input, and geometry plotting. TRITON was updated to use the improved multigroup functionality of ORIGEN and COUPLE, and for computer systems with multiple computing nodes, branch calculations can now be run in parallel.

NEWT was enhanced with parallel operation, support for inhomogeneous sources for generalized perturbation theory (GPT) calculations, improvements for high temperature gas reactor prismatic geometries, and support for coupled n-gamma calculations. Several corrections were also implemented, including improved unstructured coarse-mesh finite diffusion acceleration, grid generation algorithms, results for few-group homogenized cross sections, and output edits.

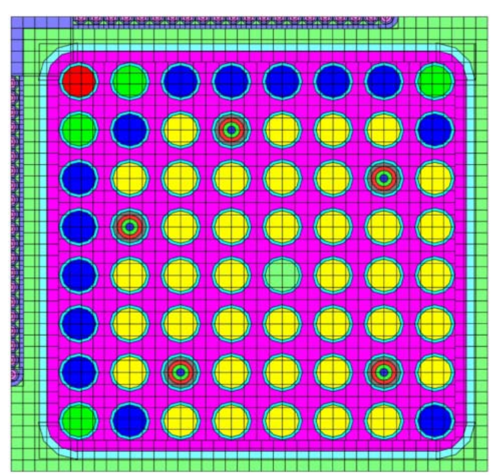

Fig. 7. Lattice physics model of a boiling water reactor fuel bundle

Sensitivity and Uncertainty Analysis

TSUNAMI-1D and -3D (Tools for Sensitivity and Uncertainty Analysis Methodology Implementation) are Scale control modules that facilitate the application of adjoint-based sensitivity and uncertainty analysis theory to criticality safety analysis. Additionally, a TSUNAMI-2D eigenvalue sensitivity analysis capability is available through the TRITON control module. TRITON also provides a generalized perturbation theory capability for 1D and 2D analysis that computes sensitivities and uncertainties for reactor responses such as reaction rate and flux ratios as well as homogenized few-group cross sections. TSAR (Tool for Sensitivity Analysis of Reactivity) provides sensitivity coefficients for reactivity differences, and TSUNAMI-IP (TSUNAMI Indices and Parameters) and TSURFER (Tool for Sensitivity and Uncertainty Analysis of Response Functions Using Experimental Results) provide code and data validation capabilities based on sensitivity and uncertainty data. An example plot of TSUNAMI-3D sensitivity data is shown in Fig. 8, and an example TSUNAMI-IP input in the ExSITE (Extensible Scale Intelligent Text Editor) graphical user interface is shown in Fig. 9.

For Scale 6.1 the TSUNAMI-3D adjoint-based sensitivity and uncertainty analysis capabilities were enhanced through many of the previously described improvements in the KENO mesh capabilities, and a new TSUNAMI-2D capability was introduced using NEWT as the transport solver. Additionally, TSURFER was updated with improved output edits and plots.

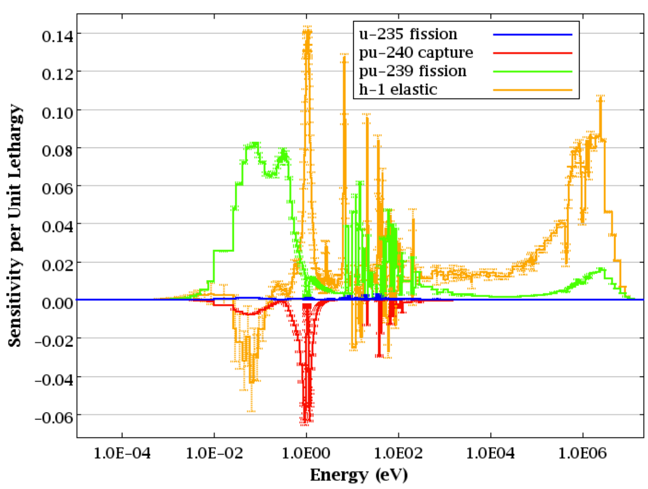

Fig. 8. Energy-dependent sensitivity of k_{eff} to cross-section data for a mixed-oxide critical experiment

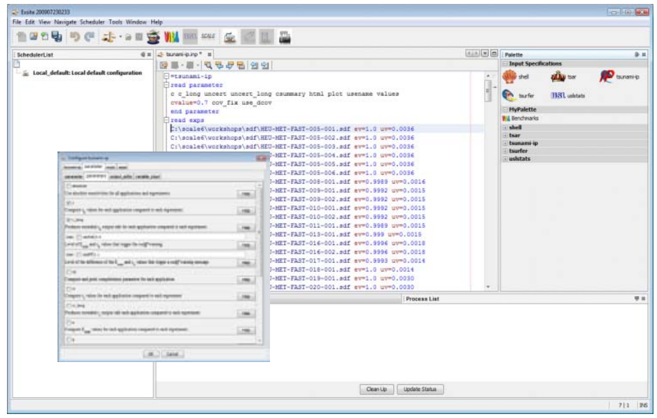

Fig. 9. ExSITE graphical user interface for use with TSUNAMI codes

Nuclear Data

The cross-section data provided with Scale include comprehensive continuous-energy neutron and multigroup neutron and coupled neutron-gamma data based on ENDF/B-VI.8 and ENDF/B-VII.0. Additional ENDF/B-V multigroup neutron libraries are also available. The comprehensive ORIGEN data libraries are based on ENDF/B-VII and JEFF-3.0/A and include nuclear decay data, neutron reaction cross sections, neutron-induced fission product yields, delayed gamma-ray emission data, and neutron emission data. The photon yield data libraries are based on the most recent Evaluated Nuclear Structure Data File (ENSDF) nuclear structure 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 of ENDF/B-VII. Scale also contains a comprehensive library of neutron cross-section-covariance data for use in sensitivity and uncertainty analysis. Example plots of multigroup and continuous-energy cross-section data are shown in Fig. 10, and multigroup cross-section-covariance data are shown in Fig. 11.

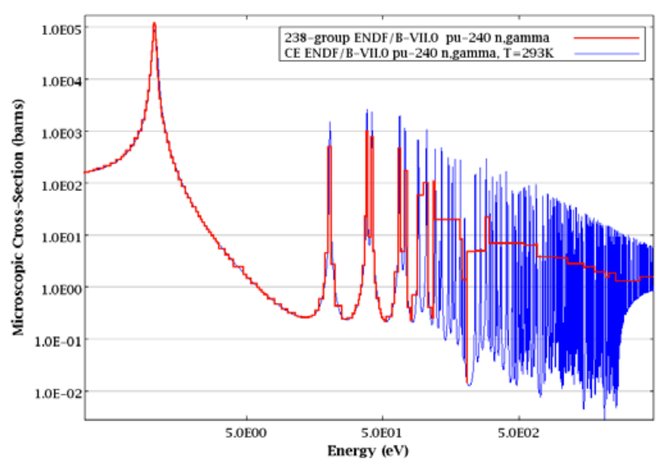

Fig. 10. Continuous-energy and multigroup ENDF/B-VII.0 neutron cross-section data for ^{240}Pu n,gamma

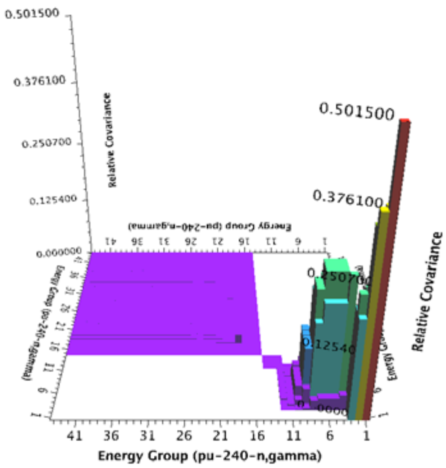

Fig. 11. Cross-section-covariance data for ^{240}Pu n,gamma

Many nuclear data libraries were updated for Scale 6.1. The unresolved resonance region probability tables for continuous-energy ENDF/B-VI.8 and ENDF/B-VII.0 neutron cross sections for uranium and plutonium isotopes have been improved to provide more accurate results, especially for intermediate energy systems.

The 238-group ENDF/B-VI.8 and ENDF/B-VII.0 neutron criticality libraries have been updated with an improved weighting function in which the tie-in for the fission spectrum has been raised from 67.4 to 820.8 keV. This adjustment improved the performance of spectral calculations for very high temperature reactor simulations. In addition, updated versions of AMPX routines using double precision throughout the calculation were used for the library generation.

The ORIGEN data have been updated to include ENDF/B-VII decay and fission yield libraries and JEFF multigroup neutron cross-section libraries in 44-, 47-, 49-, 200-, and 238-group structures. The new decay library contains 2226 nuclides, including 174 actinides, 1149 fission products, and 904 structural activation materials.

The ORIGEN master photon x-ray and gamma-ray library MPDKXGAM was regenerated using the latest evaluations from the ENSDF. This revision corrects missing gamma lines and incorrect intensities for some nuclides. A total of 982 nuclides were processed from ENSDF data, along with gamma data for an additional 51 nuclides, with no ENSDF evaluation adopted from ENDF/B-VI.

Testing was performed for specific nuclides that had previously been noted in the Scale 6 notebook to be missing some lines or to have incorrect intensities. A case for ^{137}Cs was performed to demonstrate that the 662 keV gamma line was assigned correctly to ^{137m}Ba and not ^{137}Cs. Problematic nuclides ^{192}Ir, ^{147}Pm, and ^{241}Pu were checked to ensure that the lines and intensities had been corrected. Comparisons using the previous and revised gamma libraries are shown in Fig. 12.

Several utility modules from AMPX are included to provide users with the capability to edit the cross-section data and reformat user-supplied libraries for use in Scale.

Graphical User Interfaces

Scale includes a number of graphical user interfaces to provide convenient means of generating input, executing Scale, and visualizing models and data. GeeWiz (Graphically Enhanced Editing Wizard), shown in Fig. 13, is a Windows user interface that provides a control center for setup, execution, and viewing results for most of Scale's computational sequences including CSAS, MAVRIC, TRITON, and TSUNAMI. GeeWiz is coupled with the KENO3D interactive visualization program for Windows for solid-body rendering of KENO geometry models. The OrigenArp user interface for Windows provides for rapid problem setup and plotting of results for spent fuel characterization. The Javapeño (Java Plots Especially Nice Output) multiplatform interface, shown in Fig. 14, provides 2D and 3D plotting of cross-section and cross-section-covariance data, multigroup fluxes and reaction rates from KENO and KMART, sensitivity data from TSUNAMI, and pointwise fluxes from CENTRM. The MeshView multiplatform interface produces 2D contour views of mesh data and mesh results from Monaco and KENO, and ChartPlot provides for energy-dependent plots of Monaco results. The ExSITE tool provides a dynamic multiplatform interface for the sensitivity and uncertainty analysis tools TSUNAMI-IP, TSURFER, and TSAR. The USLSTATS multiplatform interface allows for trending analysis with integrated plotting, and VIBE (Validation Interpretation and Bias Estimation) assists with interpretation of sensitivity data and couples with the DICE database from the International Criticality Safety Benchmark Evaluation Program. Additionally, several codes provide HTML-formatted output, in addition to the standard text output, to provide convenient navigation through the computed results using most common Web browsers with interactive color-coded output and integrated data visualization tools, as shown in Fig. 15.

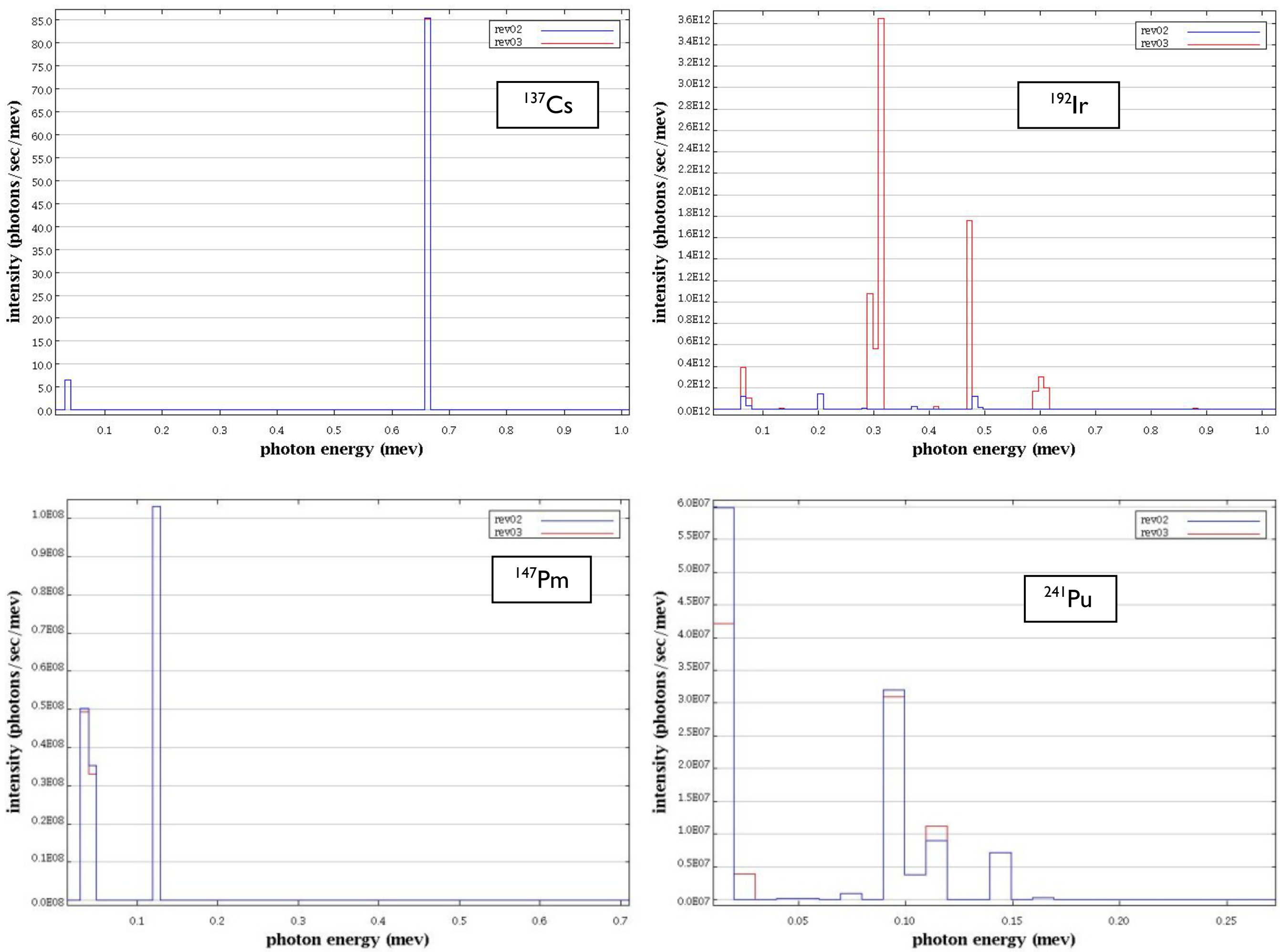

Fig. 12. Improved ORIGEN gamma yield data for Scale 6.0 (rev02) and Scale 6.1 (rev03) for ^{137}Cs, ^{192}Ir, ^{147}Pm, and ^{241}Pu

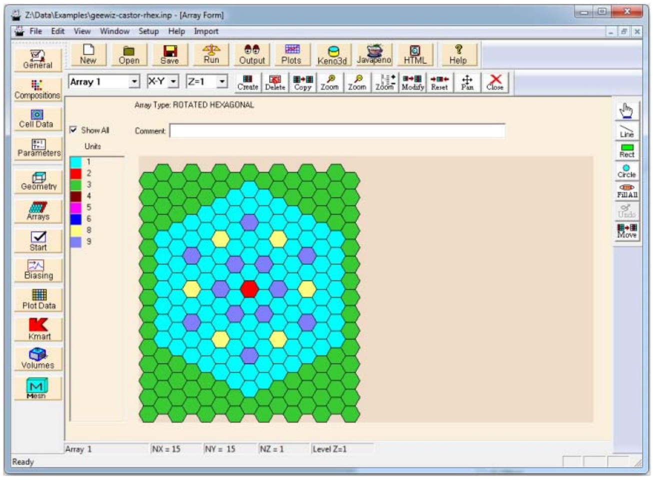

Fig. 13. Simplified array data entry with GeeWiz

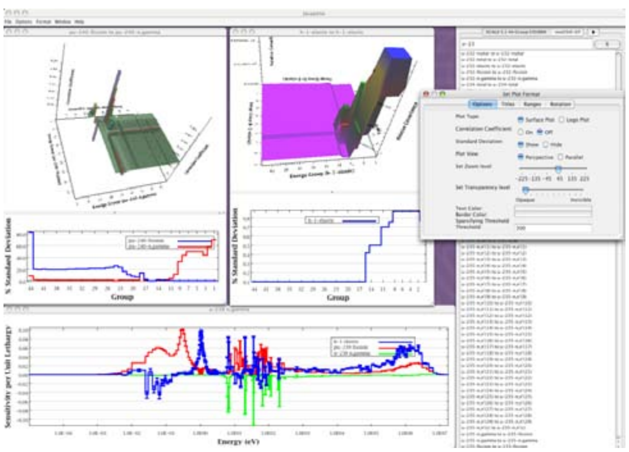

Fig. 14. Data visualization with Javapeño

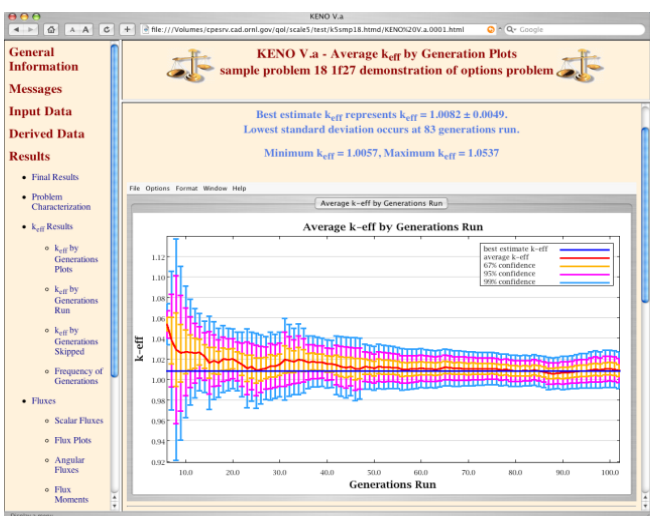

Fig. 15. HTML-formatted output from KENO with embedded Javapeño applet

Many of the Scale user interfaces have been enhanced for Scale 6.1. Notably, GeeWiz now fully supports all major Scale computational sequences and provides a more stable and intuitive work environment. The Javapeño data visualization package now supports plotting Scale continuous-energy data and ORIGEN data output from the OPUS module. The ExSITE interface, developed to support TSUNAMI post-processing calculations, is deployed for the first time, and the VIBE interface for processing and interpreting sensitivity data, shown in Fig. 16, has been updated for compatibility with the latest experimental data from the DICE database distributed with the *International Handbook of Evaluation Criticality Safety Benchmark Evaluations.*

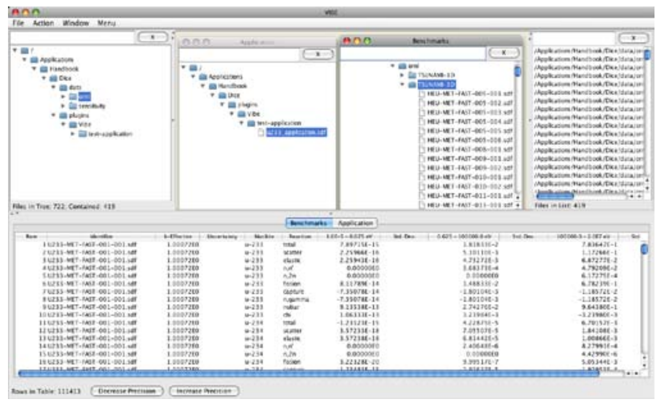

Fig. 16. VIBE sensitivity data graphical user interface

System Requirements

Scale 6.1 is built, tested, and deployed with full support for the following platforms:

- Linux 32- and 64-bit
- Mac OS X (Darwin) 10.6 or newer
- Windows XP, Vista, and 7 in 32- and 64-bit
- Itanium (HPUX) 64-bit

Minimum requirements: 2 GB RAM per CPU, 30 GB of disk space + additional space to store output results

Recommended requirements: 4 GB RAM per CPU, 30 GB of disk space + 100 GB of scratch space + additional space to store output results

Production requirements for large models: 64 GB RAM, 30 GB of disk space + 500 GB of scratch space + additional space to store output results

Scale Team

Scale 6.1 was developed through the dedication of 35 staff members of the Reactor and Nuclear Systems Division (RNSD) at ORNL. Numerous additional staff members are working on emerging features for future Scale releases.

The Scale Leadership Team guides scale development and application work at ORNL.

- John Wagner
  Design, Safety and Simulation Integration Manager
- Brad Rearden
  Scale Project Leader
- Mike Dunn
  Nuclear Data and Criticality Safety Group Leader
- Bob Grove
  Radiation Transport Group Leader
- Steve Bowman
  Reactor Physics Group Leader
- Mark Williams
  Scale Computational Methods Lead

Thirty staff members from RNSD made significant contributions to developing quality-assured software and data, performing quality assurance reviews, testing new features, providing a quality-assurance infrastructure, and ensuring the quality of the documentation.

Developers/Reviewers
- Doro Wiarda
- Douglas Peplow
- Georgeta Radulescu
- Ian Gauld
- Lester Petrie
- Mark Williams
- Matt Jessee
- Rob Lefebvre
- Sedat Goluoglu
- Tom Evans
- Ugur Mertyurek

Reviewers
- Germina Ilas
- Joel Risner
- Thomas Miller

Testers
- Andrew Godfrey
- B.J. Marshall
- Brian Ade
- Dan Ilas
- Davis Reed
- Don Mueller
- Harold Smith
- John Scaglione
- Mark Baird
- Ron Ellis
- Scott Mosher

Infrastructure
- Donnie Newell
- Jordan Lefebvre
- Sheila Walker

Documentation
- Debbie Weaver
- Hannah Turpin

Fig. 17. Many members of the Scale 6.1 Team: From left to right: John Wagner, Thomas Miller, Cecil Parks, Harold Smith, Ian Gauld, Rob Lefebvre, Doro Wiarda, Matt Jessee, Mark Williams, Georgeta Radulescu, Joel Risner, Bob Grove, Don Mueller, Mark Baird, Davis Reed, B.J. Marshall, Lester Petrie, John Scaglione, Douglas Peplow, Debbie Weaver, Ron Ellis, Sheila Walker, Brad Rearden, Hannah Turpin, Steve Bowman, Donnie Newell, and Jordan Lefebvre

Fall 2011 Scale Training

Date	Title	Location	Registration Fee
Sept. 7–9	Scale/ORIGEN Activation and Decay Calculations; Includes ORIGEN-ARP *Isotopic depletion/decay and source term characterization using ORIGEN/ORIGEN-ARP*	ORNL Oak Ridge, TN, USA	$1200
Sept. 12–16	Scale/TRITON Lattice Physics and Depletion Course *2D lattice physics calculations;* *1D, 2D, and 3D depletion calculations;* *Resonance self-shielding techniques including Monte Carlo Dancoff factors for nonuniform lattices;* *Generation of libraries for ORIGEN-ARP*	ORNL Oak Ridge, TN, USA	$2000
Sept. 26–30	Scale/TSUNAMI Sensitivity and Uncertainty Analysis Course *1D, 2D, and 3D eigenvalue sensitivity and uncertainty analysis;* *1D and 2D generalized perturbation theory for reactor analysis;* *Reactivity sensitivity analysis*	ORNL Oak Ridge, TN, USA	$2000
Oct. 3–5	Scale/TSUNAMI Validation Techniques *Advanced validation and bias assessment techniques using sensitivity and uncertainty analysis;* *Burnup credit validation*	ORNL Oak Ridge, TN, USA	$1200
Oct. 10–14	Scale Criticality and Shielding Course *Basic criticality calculations with KENO-VI;* *Shielding analysis with automated variance reduction using MAVRIC;* *Criticality accident alarm system analysis*	ORNL Oak Ridge, TN, USA	$2000
Oct. 17–21	Scale Criticality Safety Calculations Course *Introductory through advanced criticality calculations using KENO V.a and KENO-VI;* *Resonance self-shielding techniques*	ORNL Oak Ridge, TN, USA	$2000
Oct. 17–21	Scale/TSUNAMI Sensitivity and Uncertainty Analysis Course *1D, 2D, and 3D eigenvalue sensitivity and uncertainty analysis;* *1D and 2D generalized perturbation theory for reactor analysis;* *Reactivity sensitivity analysis*	NEA Data Bank, Paris, France	€2000

Foreign National Visitors: You **must** register **at least 40 days** in advance to obtain security clearance.

Payment must be received at least one week prior to training course.

For more information and online registration, please visit http://scale.ornl.gov/training.shtml

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