Enhancements in SCALE 6.1

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SCALE CODE SYSTEM

The Standardized Computer Analysis for Licensing Evaluation (SCALE) code system developed at Oak Ridge National Laboratory provides a comprehensive, verified and validated, user-friendly tool set for criticality safety, reactor physics, radiation shielding, and sensitivity and uncertainty analysis.[1] For more than 30 years, regulators, licensees, and research institutions around the world have used SCALE for safety analysis and design. SCALE provides a "plug-and-play" framework with nearly 80 computational modules, including three deterministic and three Monte Carlo radiation transport solvers that are selected based on the desired solution. SCALE's graphical user interfaces assist with accurate system modeling and convenient access to desired results. SCALE 6.1, scheduled for release in the fall of 2010, provides improved reliability and introduces a number of enhanced features, some of which are briefly described here.

CRITICALITY SAFETY

The KENO Monte Carlo neutron transport codes for eigenvalue problems have realized a number of enhancements for SCALE 6.1. The mesh flux and fission source accumulator used in TSUNAMI-3D sensitivity analysis sequences and in criticality accident alarm system analysis [2] has 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 V.a or KENO-VI in multigroup or continuous-energy mode, and the fission distribution can be visualized with the MeshView tool, as shown in Fig. 1. 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.

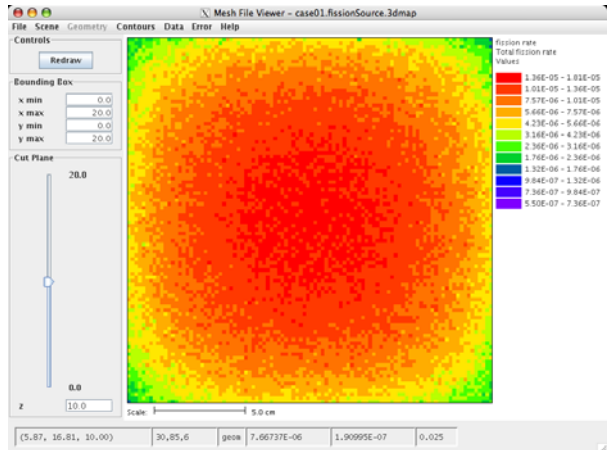

Fig. 1. Mesh fission source from KENO V.a visualized in MeshView.

SHIELDING ANALYSIS

The MAVRIC shielding analysis capabilities with automated variance reduction capabilities, first introduced with SCALE 6.0, have realized a number of enhancements for their second release in 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.[3] Cylindrical mesh grids have been added to more accurately capture spatial effects, as shown in Fig. 2, for a shielding calculation for a spent fuel shipping cask, and a suite of MAVRIC utilities has been developed to postprocess data files.

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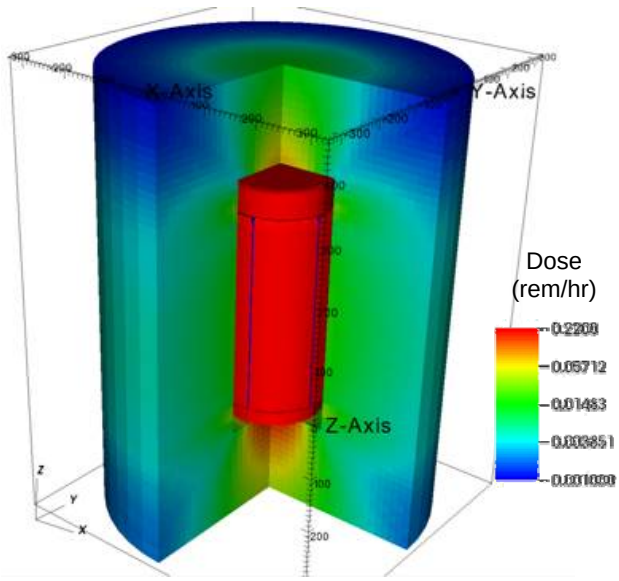

Fig. 2. MAVRIC spent fuel shipping cask model with a cylindrical mesh tally showing dose in rem/hr.

DEPLETION AND DECAY

The ORIGEN and COUPLE codes used for isotopic activation, depletion and decay have been substantially improved 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 PHYSICS

The TRITON reactor physics capabilities have realized 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 clusters with multiple computing nodes, branch calculations can now be run in parallel.

The two-dimensional (2D) generalized geometry lattice physics code NEWT was enhanced with parallel operation, support for inhomogeneous sources for generalized perturbation theory (GPT) calculations,[4] improved support for high temperature gas reactor prismatic geometries, and support for coupled n-gamma calculations. Several corrections were also realized, including improved unstructured course-mesh finite diffusion acceleration, grid generation algorithms, results for few-group homogenized cross sections, and output edits.

SENSITIVITY AND UNCERTAINTY ANALYSIS

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. In addition to generating the sensitivity of k_{eff} and reactivity to the multigroup cross-section data, one-dimensional and 2D capabilities were introduced for GPT calculations, expanding the responses for which sensitivities and uncertainties can be computed to include flux ratios, reaction rate ratios, and collapsed few-group cross sections. The TSURFER data assimilation code for advanced bias and bias uncertainty assessment was updated with improved output edits and plots.

NUCLEAR DATA

The SCALE nuclear data files have also been enhanced. For uranium and plutonium isotopes in ENDF/B-VI.8 and ENDF/B-VII.0 continuous-energy cross sections, the unresolved resonance region probability tables have been improved, providing 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 includes 2227 nuclides, including 174 actinides, 1149 fission products, and 904 structural activation materials.

GRAPHICAL USER INTERFACES

Many of the SCALE graphical user interfaces have been enhanced for SCALE 6.1. Notably, the GeeWiz input interface for Windows 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.

CONCLUSIONS

SCALE 6.1 provides state-of-the-art capabilities for criticality safety, reactor physics, and radiation shielding in a robust yet user-friendly package. The new features and improved reliability of this latest release of SCALE are intended to improve safety and efficiency throughout the nuclear community.

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