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Development of Criticality Codes and Methods

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New Criticality Safety Analysis Capabilities in SCALE 5.1

S. M. Bowman

Oak Ridge National Laboratory, P.O Box 2008, Oak Ridge, TN 37831-6170, USA*
bowmansm@ornl.gov

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Topic : *Development of Criticality Codes and Methods*

New Criticality Safety Analysis Capabilities in SCALE 5.1

S. M. Bowman

Oak Ridge National Laboratory,[*]
P.O. Box 2008, Oak Ridge, Tennessee 37831-6170
bowmansm@ornl.gov

The SCALE (**S**tandardized **C**omputer **A**nalyses for **L**icensing **E**valuation)[1] computer software system developed at Oak Ridge National Laboratory (ORNL) is widely used and accepted around the world for criticality safety analyses as well as source terms, shielding, reactor physics, and spent fuel characterization. Version 5.1 of the SCALE computer software system developed at ORNL, released in late 2006, contains several significant enhancements of interest to the criticality safety community.

The resolved resonance processor modules CENTRM (**C**ontinuous **En**ergy **Tr**ansport **M**odule) and PMC (**P**ointwise **M**ultigroup **C**onverter) have been enhanced in SCALE 5.1. Discrete-level inelastic cross-section data can now be processed by CENTRM/PMC. The pointwise (PW) thermal flux solution in CENTRM can utilize bound thermal-scattering kernels based on ENDF/B $S(\alpha,\beta)$ data. Down-scattering from inelastic continuum data and thermal up-scattering into the PW range are optionally available via a multigroup treatment. Because CENTRM/PMC calculates a problem-dependent flux profile, it provides a rigorous cross-section treatment that explicitly handles effects from overlapping resonances, fissile material in the fuel and surrounding moderator, anisotropic scattering, and inelastic level scattering. An efficient two-region solution method has also been added to CENTRM. The two-region approximation usually runs much faster than the discrete-ordinates solution and is adequate for a wide range of standard applications such as reactor fuel assembly lattices.

A new resonance self-shielding option in the SCALE criticality safety calculational sequences can address doubly heterogeneous cells (e.g., pebble-bed or prismatic fuel, in which small micrograins of fuel and moderator matrix are contained in larger spheres or rods) by applying sequential CENTRM/PMC resonance calculations for the low- and high-level heterogeneities. This method computes PW disadvantage factors to obtain cell-homogenized PW cross sections from the low-level unit cell to use in the resonance-shielding computation for the high-level unit cell.

SCALE 5.1 includes both continuous-energy and multigroup ENDF/B-VI cross-section libraries. The multigroup library is a 238-group general-purpose criticality safety library with the same energy group structure as the 238-group ENDF/B-V library found in previous versions of SCALE. Complete ENDF/B-V and ENDF/B-VI covariance libraries for sensitivity/uncertainty analyses with the TSUNAMI sequences are also included in SCALE 5.1. Covariance data from JENDL and other sources are used for nuclides that have no covariance data in ENDF.

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An advanced HTML-formatted output interface for KENO V.a has been developed as part of SCALE 5.1. The use of HTML as an alternative output interface for SCALE codes provides the user with a convenient and familiar means of navigating and visualizing data. More advanced technologies such as Java applets and JavaScript are incorporated into the output for advanced navigation and data visualization. In addition to easily navigated and color-coded tables of data, interactive plotting capabilities are available via a new version of Javapeno that executes within the web browser as a Java applet.

The geometry input format for the NEWT (**N**ew **E**SC-based **W**eighting **T**ransport code) flexible mesh discrete-ordinates code has been redesigned in SCALE 5.1, based on the SCALE Generalized Geometry Package (SGGP) combinatorial input format also employed by KENO-VI. The use of a common input format reduces the learning curve in model development for SCALE users. Although NEWT is limited to two dimensions, the use of the SGGP input format provides virtually all the geometric flexibility of Monte Carlo for a deterministic solution. It also allows a means of direct comparison of Monte Carlo and. deterministic methods. Because the same cross-section data and virtually identical geometric configurations are used, differences in results can be attributed to differences in the two transport techniques.

The TRITON control module in SCALE 5.1 is able to perform 3-D depletion calculations using the KENO V.a or KENO-VI Monte Carlo transport codes. TRITON couples these neutron transport codes with ORIGEN-S in a method similar to that used in 2-D depletion calculations with NEWT in SCALE 5. This capability allows the user to perform detailed calculations such as tracking the isotopic depletion by individual fuel rods, axial portions of a fuel rod (in 3-D), or concentric rings in a rod loaded with burnable poison.

SCALE 5.1 has been implemented and tested on Linux, Unix, and Windows computing platforms.

Reference

1. *SCALE: A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation*, ORNL/TM-2005/39, Version 5, Vols. I–III, April 2005. Available from Radiation Safety Information Computational Center at Oak Ridge National Laboratory as CCC-725.