Oak Ridge National Laboratory

SCALE Newsletter

Number 42 Summer/Fall 2010

Special Points of Interest:

SCALE Workshops Spring 2011 at NEA DB and ORNL

Inside This Issue:

SCALE 6.1

The SCALE development team is putting the final touches on SCALE 6.1. The features highlighted in the Winter/Spring 2010 Newsletter are nearly a reality. In addition to the numerous new features, I'm particularly proud of the attention to detail the entire SCALE team has dedicated to this release. In this newsletter, you will find descriptions of additional updates and user guidance that have been developed throughout the spring and summer of 2010. A new cylindrical mesh tally is available for shielding calculations to provide better quantification of results for cask geometries. The coupled n-γ nuclear data libraries have been updated with verification through a unit test suite, which identified several anomalies that were subsequently corrected by our nuclear data team. Updates in the depletion capabilities provide for improved results, especially when using restart files, and now allow the use of Dancoff factors generated with the SCALE Monte Carlo Dancoff factor generation code. The application of pin-specific Dancoff factors to a low-moderated boiling water reactor lattice shows a significant improvement in results over the standard method.

We have also transitioned SCALE to a unified set of source code and runtime tools for all supported platforms: Linux 32- and 64-bit, Mac OS X 32- and 64-bit, and Windows 32- and 64-bit. For our users, this transition will ensure a consistent experience no matter where you choose to perform your calculations. For those who have encountered memory limitation issues with the 32-bit Windows version of SCALE 6.0, which could access no more than 2 GB of RAM, the 64-bit version of SCALE 6.1 will enable you to fully utilize your hardware to analyze more complex models. SCALE 6.1 will also ship with a new installer with a graphical user interface for all platforms and a simplified means of running and verifying the sample problems. Our new runtime environment (batch6.1) will allow greater flexibility in running SCALE from a remote file server and will allow multiple inputs to be queued in a single command.

SCALE 6.1 is rapidly nearing completion. In the next few weeks, we plan to submit it to RSICC for final testing. It is expected to be available from RSICC in early 2011.

Brad Rearden
SCALE Project Leader

Monaco/MAVRIC Shielding Sequence

One of the new features added to Monaco/MAVRIC is the use of cylindrical meshes in mesh tallies. For cylindrical spent fuel casks or reactor vessel shielding calculations, tallying information on a cylindrical mesh is more useful than a rectangular mesh. MeshView, the Java utility that comes with SCALE, can be used to display the R-Z views of the cylindrical mesh results, as shown in Figure 1. As part of a recent initiative to integrate advanced visualization tools such as VisIt into SCALE (https://wci.llnl.gov/cadis/visit), multiple cylindrical meshes can be visualized with the outer cask geometry, as shown in Figure 2.

Although the cylindrical mesh tallies will be available in SCALE 6.1, the 3-D visualization capabilities are still under development.

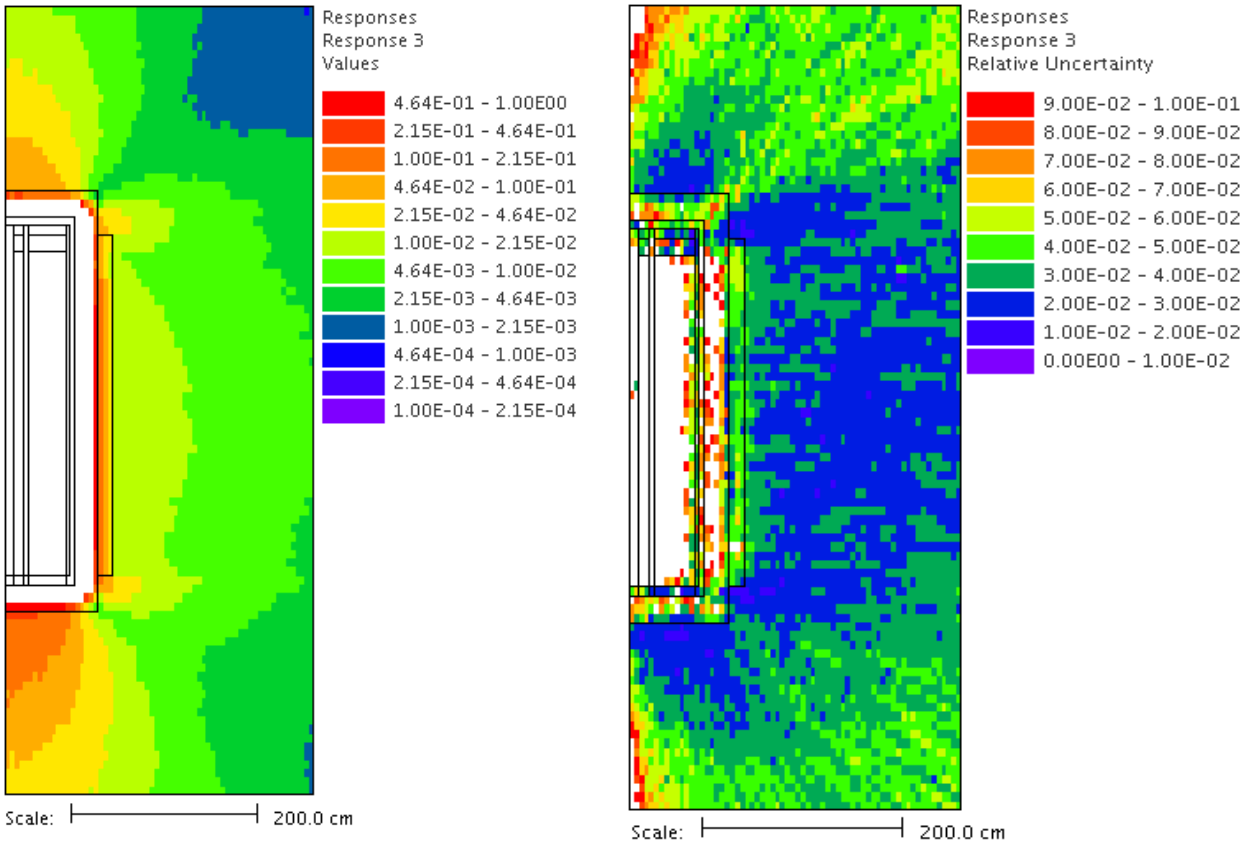


Figure 1. Cylindrical mesh tally of the total dose rate (rem/h) and its relative uncertainty along the θ=0° plane.

Monaco/MAVRIC Shielding Sequence (Cont.)

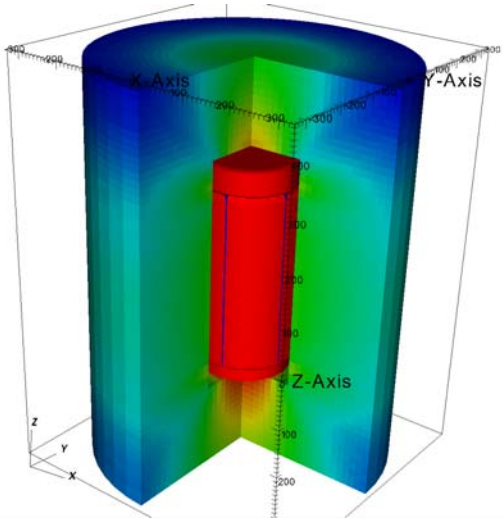


Figure 2. Advanced visualization with VisIt showing the cask geometry (red) and a cut-away view of the total dose rate (rem/h) field surrounding the cask.

Additional TRITON Updates for SCALE 6.1

In addition to the numerous SCALE 6.1 TRITON enhancements documented in the previous newsletter for two new options will be available in TRITON. First, a *length*= keyword specification is available to provide the user more control of the mass normalization used for the depletion calculation. In the case of follow-on depletion analysis, the user can specify the length used to consistently normalize the mass in the initial depletion analysis, which is the underlined value in the TRITON output edit provided below.

```
*******************************************************************************
** System total mass is 6.0017E+00 grams heavy metal per unit length.         **
** Masses will be normalized by a factor of 1.6662E+05 cm to obtain a total   **
** system mass of 1 metric ton of heavy metal                                 **
*******************************************************************************
```

Note that the *length*= keyword is based on TRITON's 2-D depletion sequence, as 2-D problems are normalized "per unit length." The *length*= keyword can also be used in the 3-D KENO depletion sequences or the new 1-D depletion sequences using XSDRN in SCALE 6.1.

Second, the *addnux*= keyword has been extended to accept six different options. The *addnux*= keyword is used to add nuclides to fuel depletion materials in TRITON depletion calculations. Setting *addnux=0* specifies that no addition nuclides are added to fuel depletion materials. For *addnux=1*, 16 nuclides were added to the fuel depletion materials in SCALE 6.0. In SCALE 6.1, only 15 nuclides will be added because the "1/vabsorber" composition is no longer included. In SCALE 6.0, *addnux*=2 added 66 nuclides to the fuel depletion materials. This option will now exist in SCALE 6.1 as *addnux*=-2, where 64 nuclides will be added to fuel depletion materials with "1/vabsorber" and "c-12" no longer included. The *addnux*=2 has been redefined in SCALE 6.1 to include 94 additional nuclides in the fuel depletion materials. The *addnux=3* option will remain the same in that 230 nuclides will be added to the fuel depletion materials. A new *addnux*=4 option will be available in SCALE 6.1, which will add the entire set of nuclides available in the SCALE cross-section library, up to a total of 388 nuclides with ENDF/B-VII.

SCALE 6.0 Updates Available for TRITON and COUPLE

Several SCALE users have recently sent inquiries to SCALE Help scalehelp@ornl.gov regarding slow run-times for the resonance self-shielding calculations of the SCALE criticality safety and reactor physics sequences CSAS and TRITON. The poor performance is due to changes introduced in SCALE 6.0 to the PMC and WORKER modules to use a different internal file format than that used in earlier versions of SCALE. This change in file format led to significantly decreased performance in CSAS and TRITON analyses with large numbers of unit cell calculations. The file format is updated in a patch to SCALE 6.0. For a TRITON example, the runtime for the depletion of a BWR lattice decreases by a factor of 3 with the correction. An update for CSAS and TRITON has been posted on the SCALE website at http://www.ornl.gov/sci/scale/SCALE_Update_July_30_2010.htm. This update significantly improves performance without modifying the computed results and is recommended for all users.

A potential problem was identified when using the ENDF/B-VII cross-section libraries for TRITON depletion calculations. The problem is specifically related to reaction types that can cause the total removal cross section for metastable nuclides to be replaced by incorrect (very small) values. The largest impact on calculated concentrations has been observed for ^{242m}Am and ^{149}S*m* (caused by an error in the ^{148m}Pm precursor cross section). A correction has been made to the COUPLE code that produces the ORIGEN library from cross sections generated by TRITON, and this correction is available from the SCALE Website. Users who have applied ENDF/B-VII libraries with TRITON depletion calculations are advised to check previous calculations to verify this potential problem has not had a significant impact on previous results.

TSURFER User Notice for SCALE 6.0

SCALE developers have recently identified a minor discrepancy in the output of the TSURFER code, which was introduced in SCALE 6.0. TSURFER is used to reduce discrepancies between measured and calculated responses by adjusting the nuclear data and the experimental values such that the overall consistency is maximized. The adjusted data can be compared with the original calculated data to quantify biases and bias uncertainties in calculated values. As part of the TSURFER output, the Adjusted Responses table lists the calculated and adjusted experiment response values, along with their standard deviations. It has been observed that the standard deviations for the adjusted responses have been edited incorrectly. The edited value for the adjusted-response standard deviation is computed as the relative standard deviation multiplied by the calculated response. The correct value is equal to the relative standard deviation multiplied by the adjusted response.

For TSURFER analysis of critical experiments, this output error is fairly minimal because both the calculated and adjusted response values are close to 1.0. Nonetheless, users should be aware of this error when viewing this table. More importantly, the edited values of the adjusted-response standard deviation provided in the Application Bias summary table are edited correctly. The uncertainty values will be corrected in SCALE 6.1.

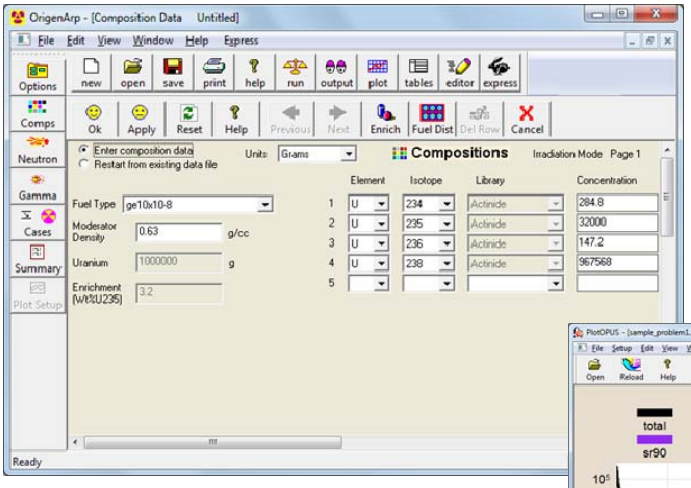
SCALE OrigenArp Primer Now Available

The SCALE OrigenArp Primer is available for download from the SCALE website at http://www.ornl.gov/sci/scale/pubs/OrigenArpPrimer.pdf. This primer is designed to help new users understand and use ORIGEN-ARP with the OrigenArp Windows graphical user interface in SCALE. There is no assumption of familiarity with nuclear depletion codes in general or with SCALE/ORIGEN-ARP in particular. The primer is based on SCALE 6 but should be applicable to earlier or later versions of SCALE as well.

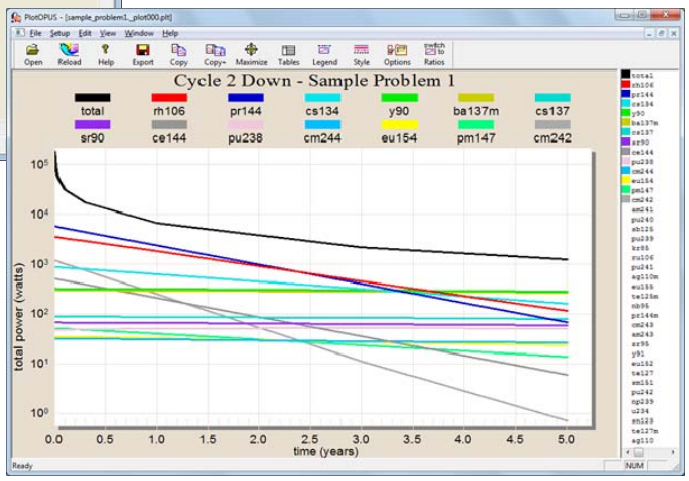
ORIGEN-ARP is a calculational sequence in the SCALE code system used to perform rapid and accurate point-depletion and decay calculations. The Automatic Rapid Processing (ARP) module interpolates pregenerated cross-section libraries to create problem-dependent cross sections for use with the ORIGEN code. Interpolation of cross-section libraries for uranium fuel systems is performed on the following variables: burnup; enrichment; and, optionally, moderator density. For mixed-oxide (MOX) fuels, additional interpolations are performed on the initial plutonium isotopic concentrations. The OrigenArp Windows graphical user interface provides an easy-to-use input processor for the ORIGEN-ARP depletion analysis sequence, which provides menus, toolbars, and forms that help a user set up a SCALE input file to execute ARP and ORIGEN. OrigenArp is coupled with the PlotOPUS Windows plotting program, which plots calculated results from ORIGEN.

Information is included to help new users, along with several sample problems that walk the user through the different input forms and menus and illustrate the basic features. References to related documentation are provided.

The primer provides a starting point for the nuclear analyst who wants to use SCALE/ORIGEN-ARP. Complete descriptions are provided in the SCALE documentation. Although the primer is self-contained, it is intended as a companion volume to the SCALE documentation.



Initial Compositions Input Form



Plot of Decay Heat by Nuclide

Use of MCDANCOFF when Modeling Boiling Water Reactors

The TRITON sequences of SCALE provide powerful and dynamic capabilities for lattice physics analysis and Monte Carlo depletion. Recent studies have quantified that nonuniform lattice effects have a significant impact on the accuracy of the results (eigenvalues, collapsed cross sections, etc.) that can be obtained using standard TRITON techniques for high-void boiling water reactor (BWR) lattices. Fuel pins in the assembly corner, along the assembly edge, and adjacent to liquid water rods in high-void BWR lattices receive significantly more neutron moderation than other fuel pins in the lattice.

The Dancoff factor (DF) provides an established means of quantifying resonance self-shielding effects between fuel pins. A high Dancoff factor (near 1.0) indicates that resonance-energy neutrons can easily stream uncollided between fuel pins. A low Dancoff factor (near 0.0) indicates that most resonance-energy neutrons undergo a collision when passing between lumps. The SCALE *latticecell* treatment assumes that a fuel pin lies in an infinite lattice of identical fuel pins with uniform spacing. With this assumption, an analytical expression for the Dancoff factor is employed and applied uniformly throughout the lattice (see Section M7.2.5.4 of the SCALE Manual). For the aforementioned fuel pins, in an otherwise low-moderated lattice, the uniform Dancoff factor overestimates the actual Dancoff factor. The effect of this overestimation increases as the number of neutron energy groups is decreased. A more accurate location-specific Dancoff factor can be calculated using the SCALE module MCDANCOFF, which computes Dancoff factors using Monte Carlo techniques based on a simplified version of KENO-VI (see Section M24 of the SCALE manual).

With SCALE 6.0, the analytically computed Dancoff factor can be overridden using the *dan2pitch* option in the latticecell input to apply the Monte Carlo-computed Dancoff factor. However, in SCALE 6.0, the *dan2pitch* parameter applies only to static calculations in TRITON, without depletion steps, as well as CSAS and TSUNAMI calculations. SCALE 6.1 will provide the capability for a user to compute and apply Monte Carlo-calculated Dancoff factors to TRITON depletion calculations too.

To generate a MCDANCOFF calculation from a TRITON/NEWT steady-state calculation ~~or~~ the NEWT geometry must first be converted to a KENO-VI model. The fuel pins for which Dancoff factors are desired and the sampling criteria are defined. Users are directed to Section M24 of the SCALE manual for specific instructions and examples.

Although MCDANCOFF runs a Monte Carlo simulation, the calculation is very fast because MCDANCOFF is performing fixed-source simulation (seconds to minutes depending on the number of particles). For each region where a Dancoff factor is requested, a separate output file will be created. The user should open the output files and record Dancoff factors for the materials of interest to be used in subsequent TRITON calculations. Figure 3 shows a Dancoff factor map for an 80%-void BWR lattice. For this study, Dancoff factors for gadolinium-bearing pins were not calculated. Note that the standard *latticecell* Dancoff factor computed by SCALE for this 80%-void lattice is 0.7063, where every fuel pin in this model actually has a lower Dancoff factor because of the non-uniformity of the lattice.

Use of MCDANCOFF when Modeling Boiling Water Reactors (Cont.)

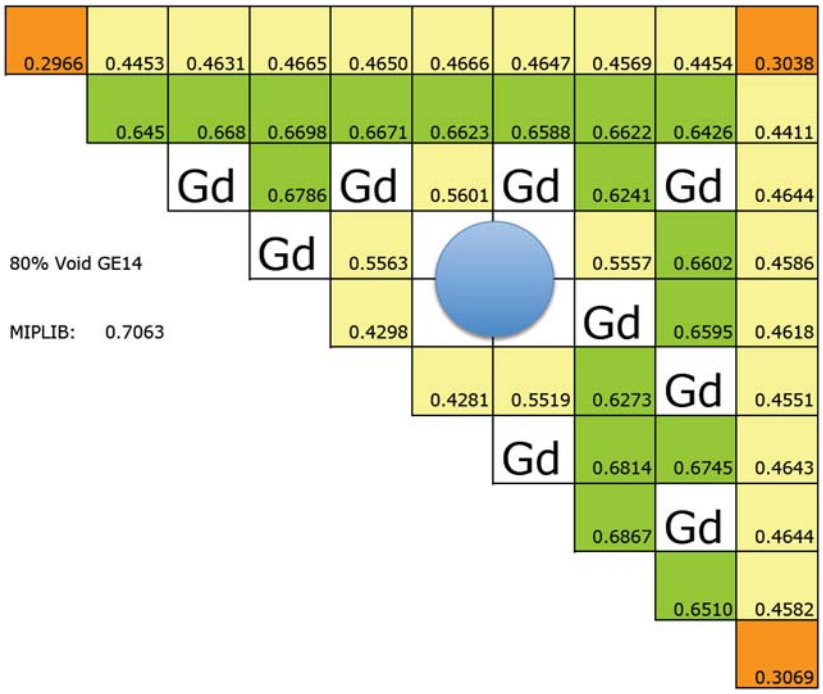


Figure 3: Dancoff factor map for an 80%-Void BWR lattice. The number in each box is the Dancoff factor for that pin; boxes are color-coded into groups of similar Dancoff factors.

After appropriate Dancoff factors are obtained, the user must edit the TRITON calculation and add separate CENTRM cross-section processing blocks for all materials for which special Dancoff factors will be used. Ideally, every fuel pin would have a separate Dancoff factor, but good results have been obtained by grouping similar enrichment and similar Dancoff factor fuel pins along the edge, corner, and near water holes. In each CENTRM cross-section processing block, the CENTRM option *dan2pitch(N)*=DF should be used, where *N* is the fuel mixture number and DF is the MCDANCOFF-calculated Dancoff factor.

After the new DFs are applied to the CENTRM calculation, SCALE/TRITON can be run normally. A BWR lattice test case was run using a void fraction of 80% in the fuel channel and solid saturated water in the bypass and water holes. This test case used the TRITON *PARM=WEIGHT* option to generate a problem-dependent 49-group cross-section library before the calculation. The standard *latticecell* Dancoff factors were used for one simulation, while MCDANCOFF Dancoff factors were used for the other. Throughout most of the depletion range, the difference in k_{inf} is 800-1100 pcm. The results are plotted in Figure 4.

Use of MCDANCOFF when Modeling Boiling Water Reactors (Cont.)

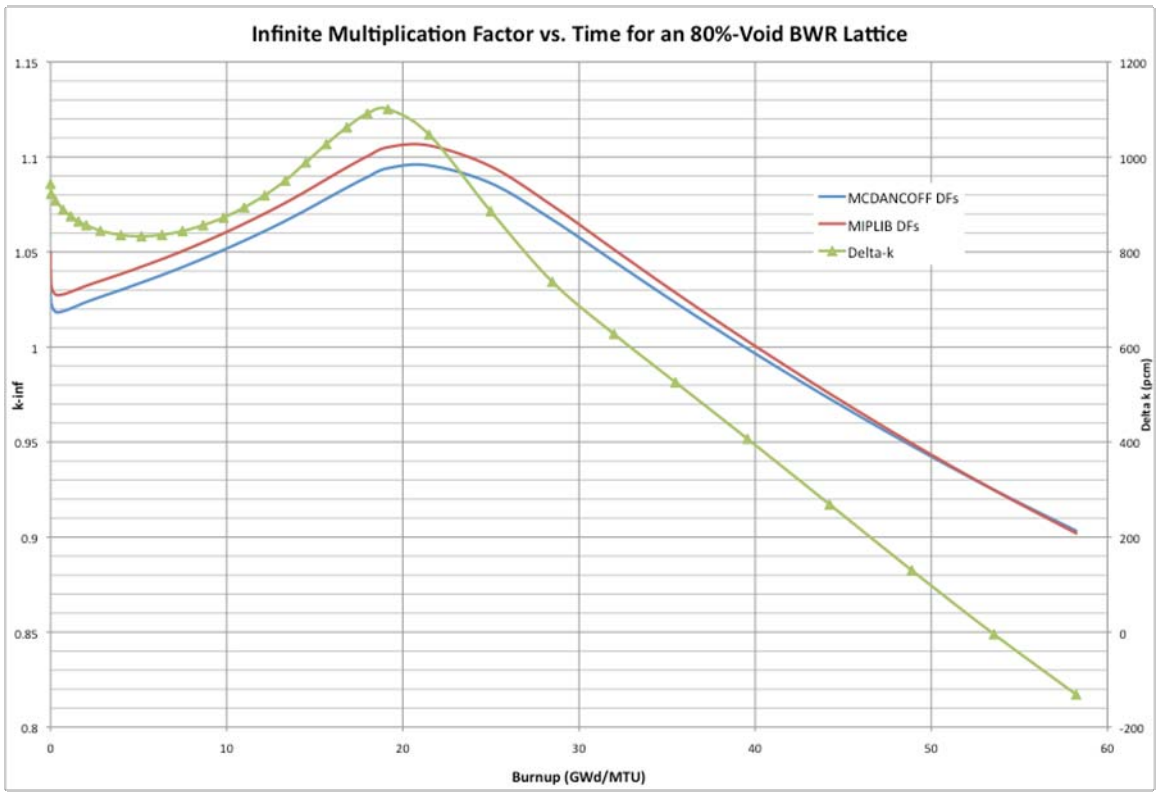


Figure 4: Infinite multiplication factor trajectories for an 80%-void BWR lattice with MIPLIB and MCDANCOFF Dancoff factors. Delta-k is plotted on the secondary axis.

ORNL staff recommends the use of MCDANCOFF Dancoff factors for all high-void BWR lattices where nonuniform lattice effects could be significant. One means of identifying the need for more precise Dancoff factors is to compare the multigroup results from a TRITON/NEWT calculations with those from a continuous-energy KENO-VI calculation of the same system. If a large discrepancy is identified, the KENO-VI model can be modified to compute the needed Dancoff factors for use in the multigroup calculation.

2010 SCALE Testing Team

During the summer of 2010, five students interned at ORNL to form the inaugural SCALE testing team. The testers were provided access to a preliminary version of SCALE 6.1 and were asked to comprehensively test input, functionality, and user interfaces. The team consisted of the following members, each focused on a particular area of SCALE analysis.

Heather Connaway—Massachusetts Institute of Technology
Focus: Criticality safety analysis

David Hartmangruber—Georgia Institute of Technology
Focus: Shielding analysis and hybrid methods

Oscar Lastres—University of Tennessee
Focus: Depletion and lattice physics methods

Chris Perfetti—University of Michigan
Focus: Sensitivity and uncertainty analysis methods

Paul Stauduhar—University of Dallas
Focus: Material input specification and multigroup resonance self-shielding

Figure 5: 2010 SCALE testing team and mentors. First row: Oscar Lastres, Heather Connaway; Second row: David Hartmangruber, Chris Perfetti; Third row: Brad Rearden, Paul Stauduhar; Forth row: Douglas Peplow, Matt Jessee.

SCALE Project Leader Brad Rearden, MAVRIC/Monaco developer Douglas Peplow, and TRITON developer Matt Jessee served as mentors for the testing team. The testers identified approximately 200 issues in SCALE 6.1 and worked directly with the SCALE developers to resolve these issues. Additionally, each tester developed a comprehensive regression test suite for his or her area of SCALE. These test suites will be used to ensure the quality of future SCALE development efforts.

Apart from the testing efforts, the interns participated in the first SCALE 6.1 training course provided to NRC Office of Nuclear Material Safety and Safeguards staff on criticality and shielding methods. The interns received additional targeted training in SCALE and had direct access to SCALE developers throughout the summer. Some interns also identified projects related to SCALE for their academic research.

Figure 6: SCALE interns and friends at Ft. Loudon Lake.

The summer did not consist only of office work. There were many opportunities for socializing, including cookouts and days at the lake. Any students interested in participating in the summer 2011 SCALE testing team should contact scalehelp@ornl.gov for more information.

New Cross-Section "Unit Tests" for SCALE Coupled Data Libraries

SCALE includes three multigroup coupled neutron-gamma cross-section data libraries. The two fine-group libraries, XN200G47V6 and XN200G47V7, are based on ENDF/B-VI.8 and ENDF/B-VII.0, respectively, and contain cross-section data with 200 neutron and 47 gamma groups. The broad-group library, XN27G19V7, is based on ENDF/B-VII.0 and contains cross-section data with 27 neutron and 19 gamma groups. These libraries were recently updated for distribution in SCALE 6.1. The updates corrected errors that were present in some gamma yield data. Because only the gamma-yield matrices for some nuclides were changed, neutron-only and photon-only transport calculations are unaffected by this revision.

While the SCALE data libraries are tested extensively before they are released, there is a limited amount of experimental benchmark data that can be used to validate the photon production data in coupled cross-section libraries. To reduce the potential for errors in library generation with future cross-section updates, a set of "unit tests" has been added to the suite of verification and validation tests that are performed for SCALE cross-section library updates. The unit tests, which are performed for each element in a library, are used to calculate neutron and photon transport through a simple 1-D model geometry. Using an optically thick layer of the material being tested provides a simulation of a deep penetration shielding problem and is particularly useful for checks of total and scattering cross sections. The use of a very thin layer of material in a coupled calculation with a neutron-only source provides a means of focusing on the gamma-production cross sections. Modeling a thin layer of material minimizes the scattering and absorption of the photons within the medium, so the spectrum of emerging photons is dominated largely by the uncollided photons that were produced by neutron reactions.

The unit tests provide several capabilities for evaluating cross-section data. Among those are the following:

- Some processing errors can result in effects that are evident from simple examination of the behavior of the neutron and photon fluxes from the unit test.
- The unit tests provide a convenient means of assessing the impact of changes in cross-section data for individual elements or isotopes. Those types of changes can be difficult to isolate in typical benchmark models, especially for shielding applications.
- The unit tests can be used to compare the results of different transport codes and/or different cross-section processing codes. This is particularly useful as an independent check of cross-section processing techniques with new ENDF evaluations. For example, the formatting of the gamma-yield data changed substantially for many newer evaluations in ENDF/B-VII, with data being moved from Files 12 and 13 to File 6. When formatting changes of this type occur, it is possible that a cross-section processing code that had been used for a previous ENDF evaluation of the same nuclide may fail to properly interpret the new evaluation.

An example of the use of the unit tests in verifying the new coupled libraries is shown in Figure 7. The two histograms in the plot are calculated photon spectra for a unit test with an incident neutron source and a thin layer of nickel. These calculations were performed using XSDRNPM with ENDF/B-VII coupled libraries with 200 neutron groups and 47 gamma groups. The results of the unit test demonstrate an underprediction of the photons produced by neutron interactions in the nickel using the previous version (rev07) of the library and the corrected results with the new coupled library (rev08). The points with error bars are continuous energy MCNP results for an equivalent model. Using MCNP provides an independent comparison of cross-section processing techniques, because the MCNP libraries are created using NJOY.

The inclusion of these unit tests in the verification and validation process for SCALE cross-section libraries provides an additional means of ensuring the quality of the data provided in the SCALE package and demonstrates the commitment of the SCALE team to continual improvement.

New Cross-Section "Unit Tests" for SCALE Coupled n-γ Data Libraries (Cont.)

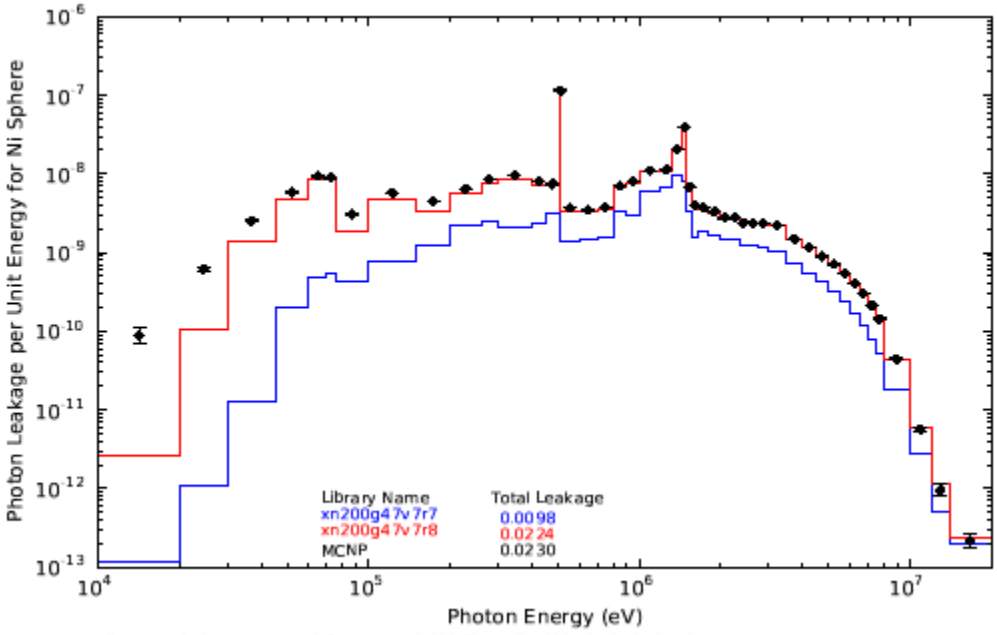


Figure 7. Unit test comparison of SCALE and MCNP photon spectra for nickel.

Words to the Wise . . . TRITON User Notice

Guidelines for Using Standard Composition Restart Files

At the end of TRITON depletion calculations, files are automatically produced for each depletion mixture that list the nuclides and number densities of the mixtures at the end of the final ORIGEN calculation. These files, referred to as standard composition restart files, may be included in a follow-on calculation that requires depleted/decayed number densities from a previous case. Because ORIGEN cross-section libraries contain many more nuclides than are available in SCALE cross-section libraries, only those nuclides for which cross-section data are available in the SCALE libraries are saved in the standard composition restart files.

SCALE/TRITON users should be aware of the following issues regarding the use of standard composition restart files.

Special nuclides in restart file

Within the SCALE cross-section libraries, there are several isotopes for which multiple cross sections are provided, typically isotopes with cross sections prepared with different thermal scattering kernels. These isotopes will have duplicate entries in the standard composition restart file as seen in the following example.

```
' Nuclide concentrations in mixture 1 at t=365.0 days
h-1               1 0  1.20E-8  800.0 end
h-liquid_ch4      1 0  1.20E-8  800.0 end
h-ortho           1 0  1.20E-8  800.0 end
h-para            1 0  1.20E-8  800.0 end
h-solid_ch4       1 0  1.20E-8  800.0 end
```

In this example, the entries for h-liquid_ch4, h-ortho, h-para, and h-solid_ch4 have been incorrectly included in the restart file because h-1 is being tracked in the ORIGEN depletion calculations. *These additional entries in the restart file should be deleted.* Other nuclides that may be incorrectly inserted in the restart file include the following:

```
albound      be-beo       bebound      dfreegas     d-ortho
d-para       febound      hfreegas     h-liquid_ch4 h-ortho
h-para       h-poly       h-solid_ch4  h-zrh2       o-beo
zr90-zr5h8   zr91-zr5h8   zr92-zr5h8   zr93-zr5h8   zr94-zr5h8
zr95-zr5h8   zr96-zr5h8
```

Nuclides that are not depleted are not included in the restart file.

The SCALE cross-section libraries include cross-section evaluations for several naturally occurring elements that are not depleted in ORIGEN. If element compositions are used in the standard composition definition for depletion materials, TRITON uses the same concentration for these compositions throughout the transport calculation. Because the restart files are based on compositions from the ORIGEN depletion calculation, any elemental compositions are not copied to the standard composition restart file. The standard compositions for elements include the following:

```
c            c-12         ca           c-benzene    cd           c-graphite
cl           cr           cr-esigt     crss         cu           eu
fe           fe-esigt     fess         fm-255       ga           h-benzene
hf           in           k            mg           mo           ni
ni-esigt     niss         pb           s            si           ti
v            w            zn           zr           zr-zrh2
```

Words to the Wise … TRITON User Notice (Cont.)

For depletion materials that use these compositions, users should manually insert these compositions in the standard composition restart file before using the file in a follow-on calculation.

Several nuclides are inserted in the standard composition restart file multiple times

Within the depletion calculation, ORIGEN tracks several nuclides as both a fission product and as a light element. The concentrations for these nuclides are written in two different locations on the ORIGEN number density file. For these nuclides, TRITON writes both concentrations to the standard composition restart file: one concentration corresponding to ORIGEN treating the isotope as a light element and a second concentration corresponding to ORIGEN treating the isotope also as a fission product. SCALE will combine duplicate entries, which can result in concentrations that are too large when applied in a follow-on concentration. *SCALE users should identify duplicate entries of isotopes in the standard composition restart file and delete the duplicated entry. Users should be aware that some of the special nuclides listed on the previous page may appear twice in the restart file and that both instances should be deleted.*

For SCALE 6.1, the following apply: (1) the special nuclides will no longer be inserted in the restart file, (2) duplicate entries of isotopes that are tracked in ORIGEN as both a light element and a fission product will appear only once in the restart file; and (3) nuclides and compositions that are not depleted in ORIGEN will appear in the restart file and do not have to be manually inserted. In addition, a warning message will also be printed in the TRITON output to notify the user of nuclides and compositions that are not included in the ORIGEN calculation.

Heavy metal mass normalization using restart files

In calculations that use standard composition restart files, users should be aware that the power used in TRITON is defined as MW per initial metric ton heavy metal (MTHM). Because TRITON recomputes the initial MTHM at the beginning of a TRITON restart calculation, additional user intervention is required to obtain the correct normalization. In restart cases, users need to adjust the power based on the initial heavy metal content by the ratio of the depleted-to-initial heavy metal content.

Users also need to be aware that standard composition restart files contain all the nuclides from the depletion calculation that are on the SCALE library. Nuclides tracked by ORIGEN that are not on the SCALE library are omitted from the restart files. In most cases the omitted nuclides have only a minor impact of the calculations, since the ENDF/B cross sections on the SCALE libraries contain a comprehensive set of nuclides for reactor physics analysis. However, in some cases nuclides that are important to other applications may be omitted, and the user should assess the impact of such omissions. Note that greatest consistency between depletion calculations with and without using the restart file option will be obtained using the *addnux=3* option in SCALE 6.0 and addnux=4 in SCALE 6.1.

Evolving Features of SCALE 6.2

The SCALE development team is working on a number of major initiatives for the future release of SCALE 6.2. More details of these capabilities will be available in the next newsletter.

Continuous-Energy Shielding

The multigroup fixed-source Monte Carlo capabilities of MAVRIC/Monaco are being expanded to support continuous-energy neutron and gamma transport. The use of continuous-energy data offers considerable benefits in terms of accuracy, as compared with multigroup data, for a number of shielding problems. For example, neutron transport through thick steel can be troublesome if the resonance windows are not specifically accounted for in multigroup data. Continuous-energy treatment is required to perform energy-deposition calculations, calculate gamma-ray spectra, or model detector responses as a function of time.

Continuous Energy Depletion

A continuous-energy Monte Carlo depletion capability is under development for TRITON/KENO. This capability will facilitate more accurate depletion calculations than multigroup calculations, especially where resonance self-shielding effects are difficult to treat.

Parallel Capabilities in KENO

The SCALE team has developed an internal version of KENO that operates on parallel computer platforms. The initial version implements MPI parallel algorithms and scales very well on many Linux nodes. This new version of KENO will also support OpenMP threading, especially to improve performance on multicore PCs.

Enhancements and Corrections to SCALE 6

Click http://www.ornl.gov/sci/scale/news/2010-Nov-corrections-updates.pdf to see a summary of code and data modifications that have been made to the configuration-controlled version of SCALE at ORNL since the previous issue of the *SCALE Newsletter*. These will be included in the SCALE 6.1 release.

We are now on Facebook

Follow us on Facebook at http://facebook.com/SCALE.codes.

Tentative Spring 2011 SCALE Training

Date	Title	Location	Registration Fee
Mar. 7–11, 2011	**SCALE Lattice Physics and Depletion Course (ORIGEN-ARP/TRITON)** Isotopic depletion/decay and source term characterization using ORIGEN, 2D reactor physics analysis using NEWT, and 3D Monte Carlo depletion using KENO	NEA Data Bank, Paris, France	€2000
Mar. 28–Apr. 1, 2011	**SCALE Criticality Safety Course (KENO V.a/KENO-VI)** Criticality safety with the KENO Monte Carlo codes for multi-group and continuous-energy calculations using standard and generalized geometry	ORNL Oak Ridge, TN, USA	$2000
Apr 4–8, 2011	**SCALE Criticality Safety and Shielding Course (KENO-VI/MAVRIC)** Introduction to criticality safety using the generalized geometry version of KENO, shielding analysis using automated variance reduction for deep -penetration and complex problems, and criticality accident analysis system analysis.	ORNL Oak Ridge, TN, USA	$2000
Apr. 11–15, 2011	**SCALE Lattice Physics and Depletion Course (ORIGEN-ARP/TRITON)** Isotopic depletion/decay and source term characterization using ORIGEN, 2D reactor physics analysis using NEWT, and 3D Monte Carlo depletion using KENO *ORIGEN-ARP will be offered for 1-day registration of $800*	ORNL Oak Ridge, TN, USA	$2000
Apr. 18–21, 2011	**Special Topics in SCALE Analysis** *Burnup credit analysis with STARBUCS *ORIGEN activation and decay analysis *Generalized perturbation theory analysis for reactor analysis. *These courses will be offered for individual registration for 1 or 2 days each. The schedules are not yet finalized, so please check the SCALE website for updates.*	ORNL Oak Ridge, TN, USA	TBD
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://www.ornl.gov/sci/scale/training.htm

SCALE Newsletter
Oak Ridge National Laboratory
Post Office Box 2008
Bldg. 5700
Oak Ridge, TN 37831-6170

Fax: 865-574-3527
E-mail: scalehelp@ornl.gov

SCALE Web Site:
http://www.ornl.gov/sci/scale

SCALE Electronic Notebook:
http://www.ornl.gov/sci/scale/notebook.htm

Find us on Facebook

SCALE Newsletter* is published twice a year by the Reactor and Nuclear Systems Division of the Oak Ridge National Laboratory

***Managed by UT-Battelle, LLC, for the U.S. Department of Energy under contract DE-AC05-00OR22725**

***SCALE Newsletter* is sponsored by**
U.S. Nuclear Regulatory Commission, Division of Spent Fuel Storage and Transportation
U.S. Department of Energy.

OAK RIDGE National Laboratory