scale Nuclear Systems Modeling & Simulation

Newsletter

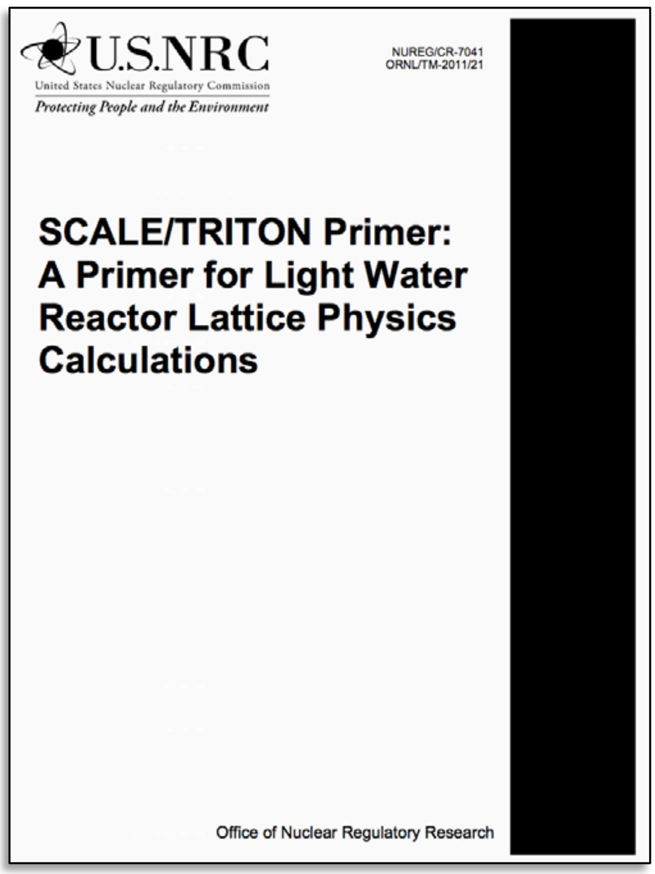
Number 45 Spring 2013

In This Issue

TRITON Primer

A primer for light water reactor lattice physics with SCALE/TRITON is now available. The TRITON Primer provides novice and advanced SCALE users with a guide for using the SCALE graphical user interface GeeWiz to develop lattice physics models for SCALE/TRITON-NEWT. Although the document pertains only to SCALE/TRITON-NEWT, much of the information presented in the primer can also be used for SCALE/TRITON-KENO Monte Carlo depletion models. The document is not meant to replace the SCALE user manuals but rather to provide detailed instructions and examples for application of the information in those manuals to generate lattice physics models.

Novice users should find the step-by-step instructions helpful in getting started using SCALE. After reading this guide and completing the practice problems within, novice users should be able to develop detailed lattice physics models suitable for generating broad-group nodal parameters for reactor core simulators. Advanced users who are familiar with the SCALE code system but unfamiliar with lattice physics analysis should find the advanced features and user-guidance sections of this document helpful in generating appropriate models.


U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment

NUREG/CR-7041
ORNL/TM-2011/21

SCALE/TRITON Primer: A Primer for Light Water Reactor Lattice Physics Calculations

Office of Nuclear Regulatory Research

The primer and sample input files are available at

http://scale.ornl.gov/training_primers.shtml

SCALE 6.1.2 Update

An update is available for SCALE 6.1 to provide enhanced performance in the areas detailed below. This comprehensive update includes enhancements previously released as SCALE 6.1.1 and is recommended for all users of SCALE 6.1 and 6.1.1. Details of the enhancements and instructions for requesting and installing this update are available at

http://scale.ornl.gov/downloads_scale6-1.shtml

ORIGEN ENDF/B-VII.1 Data

The ORIGEN decay data library was updated from ENDF/B-VII.0 to ENDF/B-VII.1 to correct errors introduced in the evaluated ENDF/B-VII.0 decay sublibrary, released by the National Nuclear Data Center (NNDC) in December 2006. The error is observed primarily for simulations of the ^{238}U decay series. The gamma-ray spectrum obtained using ENDF/B-VII.0 data is significantly overestimated, caused primarily by incorrect branching of ^{234}Th beta decay to ground state ^{234}Pa. The NNDC confirmed the problem with their ENDF/B-VII.0 data and released an updated decay library with ENDF/B-VII.1. Additional information on the errors in the ENDF/B-VII.0 decay evaluations, and improvements for ENDF/B-VII.1, is posted on the NNDC website.

http://www.nndc.bnl.gov/exfor/endfb7.1_decay.jsp

Further review of the ENDF/B-VII.0 decay data by ORNL identified systematic errors in decay schemes for the actinides and the recoverable decay energy values. Problems were also observed in many short-lived fission products, although the impact on most typical spent fuel calculations was relatively minor.

In addition to updating the decay library to ENDF/B-VII.1, the fission yield library and the gamma-ray and X-ray library are also updated for compatibility with the new decay library. The fission yield library is still based on ENDF/B-VII.0 (largely unchanged in –VII.1); however, the gamma-ray library is updated using new evaluations in ENDF/B-VII.1.

Impact of update on previous SCALE 6.1 and SCALE 6.1.1 calculations

Uranium-238 decay calculations: Significant reduction in the gamma yield is observed for the decay of ^{238}U with the correction of the ^{234}Th decay scheme in ENDF/B-VII.1. (See Figure 1)

Energy release following fission (decay heat): Total energy release after fission is mostly unchanged for cooling times up to about 4 hours (errors up to several percent), but gamma energy release may be under predicted by up to 15%, with a similar overprediction of the beta energy component.

Gamma spectrum following ^{235}U fission: For times from 1 s to 30 years, the fission product gamma spectra are mostly unchanged, with slightly greater intensities for >2 MeV associated with the adoption of ENDF/B-VII.1 gamma emission data.

Spent fuel isotopic depletion: Generally concentrations agree well within 1% of previous values. Changes larger than 1% generally reflect updates to the values of the nuclide decay half-lives. The update results in an increase in the ^{235}U content in high-burnup spent nuclear fuel of up to about 1% due to improved representation of the production path via ^{235m}U.

ORIGEN Irradiation Calculations

ORIGEN was updated to correct a memory management error in irradiation calculations that would occasionally cause fission products to be produced from non-fissile materials. This error only affects SCALE 6.1 calculations with irradiation time steps of 5–35 days. The error may be encountered when hydrogen or other very light elements exist in the system, producing large masses of fission products with A>162 (~10^8 grams) that are easily identified. When hydrogen does not exist in the system, the error may be more difficult to detect as it only affects the transitions for a small set of fission products with A>162.

All calculations performed with SCALE 6.1 or 6.1.1 using time steps between 5 and 35 days should be reviewed to ensure that additional fission product mass is not generated. Calculations with SCALE versions prior to 6.1 are not affected.

Parallel Branch Calculations with TRITON

An updated RUNNER package has been developed to replace the previous RUNNER deployed with SCALE 6.1 to perform parallel branch calculations with TRITON. The updated RUNNER mitigates instabilities observed by some users when performing calculations on Linux 64-bit platforms. Improved stability is realized through the addition of parallel environment setup, control mechanisms, and feedback enabled in the updated RUNNER and associated codes.

Results from previous calculations that were successfully completed are not affected by this update, but calculations that previously failed due to instability of the parallel framework will now run to completion.

Critical Spectrum Calculations with NEWT

NEWT was updated to correct an error that would cause few-group homogenization calculations to fail in critical spectrum mode when using the user-specified critical buckling value or critical height. Calculations that previously failed will now run to completion. Calculations that do not use these options are not affected by this update.

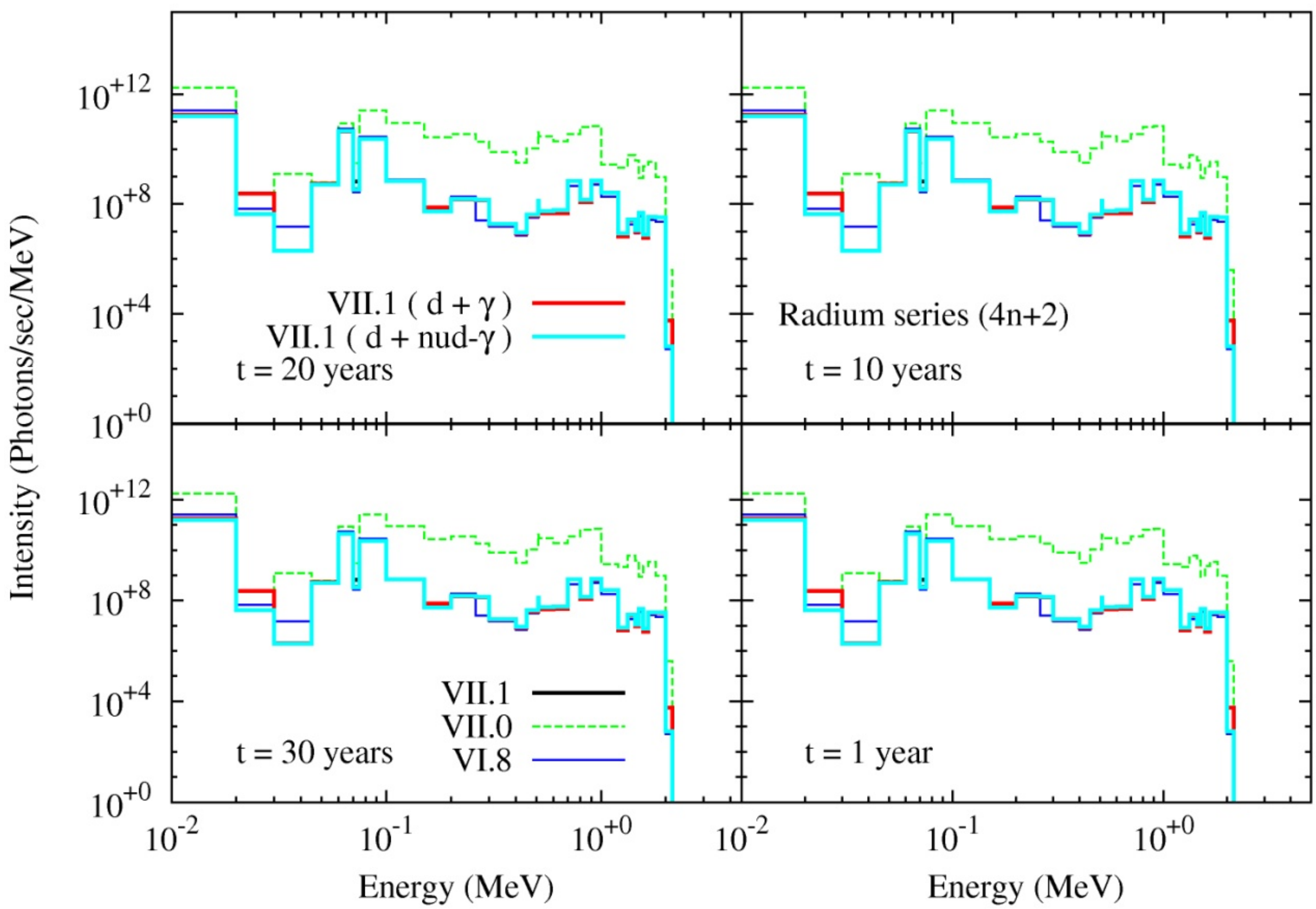


Figure 1. γ-ray spectra calculated for the radium decay series (4n+2) using ENDF/B-VI.8, –VII.0, and –VII.1 and several gamma-ray libraries based on ENDF/B-VII.1 and NuDat.

Implicit Sensitivity Calculations

BONAMIST was updated to prevent an error where implicit sensitivities for some nuclides would occasionally not be written to an internal data file for use in sensitivity calculations in SAMS. The error only impacts a minimal set of test cases, and users who performed recommended direct perturbation calculations would observe the discrepancy. In the past decade of sensitivity work at ORNL, this error only noticeably impacts one set of computational benchmarks.

REORG

REORG, used to post-process ORIGEN data files used with ORIGEN-ARP, was updated to allow increased internal data storage that would occasionally cause calculations to fail. Calculations that previously ran to completion are not affected by this update.

Double Heterogeneous Calculations

CAJUN was updated to correct an issue that occasionally caused double heterogeneous cases with many nuclides to fail. Cases that previously ran to completion are not affected.

Additional Features in Development for SCALE 6.2

The SCALE Team is continuing development activities for SCALE 6.2. Several features were described in the previous SCALE Newsletter, available at http://scale.ornl.gov/newsletters.shtml, and some additional improvements are summarized below.

Improvements in Continuous-Energy Calculations with KENO

A comprehensive review of the continuous-energy treatment in KENO has led to a number of improvements providing reduced biases and reduced memory footprint. In addition to the improvements in the S(α,β) data that provide bias reduction for thermal systems documented in the previous newsletter, the probability tables that provide continuous-energy treatment in the unresolved resonance range have been improved, resulting in reduced biases for systems that are sensitive to the intermediate energy range. Additional improvements in the initialization of the problem-dependent fission spectrum data provide further enhancements in some cases.

Work is continuing to substantially reduce the memory footprint of continuous-energy calculations. An option will be available to disable the use of a unionized energy grid for all materials in a mixture. The new *UUM=no* option can provide an order-of-magnitude reduction in memory requirements, depending on the materials used in the model. The internal storage of continuous-energy cross-section data will be converted from double-precision values to single-precision values, resulting in a further 20–30% reduction in total size with no loss of precision. Several additional opportunities to further reduce the memory requirements for continuous-energy calculations are currently under investigation. The memory required for continuous-energy calculations with KENO could be reduced by 20–95%, depending on the input model, with no loss of accuracy.

Doppler Broadening Rejection Correction

The Doppler Broadening Rejection Correction (DBRC) technique has been implemented in KENO for continuous-energy calculations. DBRC has been demonstrated to improve the accuracy of cases at reactor temperatures by as much as 635 pcm, as shown in Table 1.

Table 1. Impact of DBRC as a Function of Temperature

Temperature (K)	Normal	DBRC	Difference (pcm)
293.6	1.34460	1.34451	-9
600.0	1.33053	1.32932	-121
900.0	1.31941	1.31759	-182
1200.0	1.31029	1.30730	-299
2400.0	1.28113	1.27478	-635

Sampler Uncertainty Assessment

A new *super-sequence* called Sampler is under development to perform uncertainty analysis using any SCALE sequence through a statistical sampling of input nuclear data probability distributions obtained from the SCALE cross-section covariance data library. Data variations for multiple samples of the covariance library are stored on a pre-generated perturbation factor library for subsequent use in Sampler. The perturbations were generated using the XSUSA tool from Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) mbH from Garching, Germany.

Sampler executes any SCALE sequence for the desired number of samples. Uncertainties examined in the development of Sampler include k_{eff}, power distributions, and few group constants, but the uncertainty can be determined for any parameter computed with SCALE multigroup neutron transport. The output distribution of results can be converted into standard deviations and correlation coefficients for any desired responses computed in the sequence.

Sampler operates by perturbing the multigroup cross-section data input to BONAMI and CENTRM with a new module called CLAROL-plus and by perturbing the pointwise (PW) data used by CENTRM with an updated version of CRAWDAD called CRAWDAD-plus. A unique perturbation is performed for each pass through a selected SCALE sequence, resonance self-shielding calculations are performed on the perturbed library, and the transport calculations are performed on the perturbed data. The data flow for the Sampler super-sequence is shown in Figure 2.

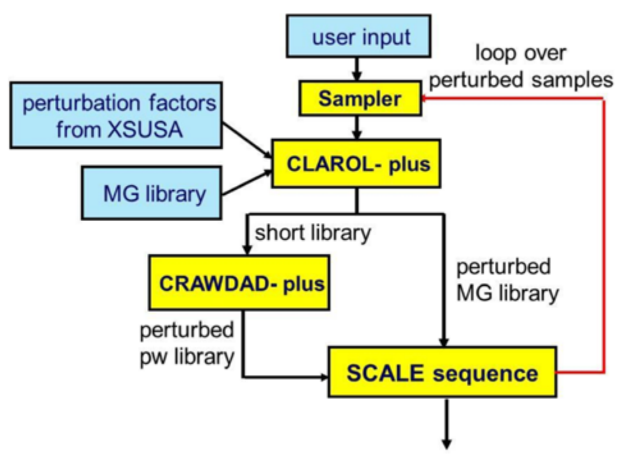


Figure 2. Data flow for Sampler super-sequence.

New ENDF/B-VII.0 Multigroup Library

A new 252-energy-group ENDF/B-VII.0 neutron cross-section library will be available with SCALE 6.2 for reactor physics and criticality safety calculations. The new library is undergoing testing by analyzing a wide variety of critical benchmark experiments and by comparing with continuous-energy Monte Carlo results. Computational benchmark comparisons at elevated temperatures typical of operating light water reactors show agreement of less than 100 pcm in most cases.

A new ENDF/B-VII.0 broad group library, with 50–70 energy groups, is also under development for possible release with SCALE 6.2.

Features Removed from SCALE 6.2

As new features are added to SCALE, it is necessary to remove older features to reduce the overall size of the package and to reduce maintenance and support requirements. Some features that were available in SCALE 6.1 that will not be available in SCALE 6.2 are described below.

- ENDF/B-V nuclear data were deployed from the NNDC in 1978. The 35-year-old data have been replaced with newer evaluations, and the 44- and 238-group ENDF/B-V neutron cross-section libraries will be removed from SCALE.
- ENDF/B-VI nuclear data were deployed from NNDC in 1990. The use of ENDF/B-VII data is recommended over ENDF/B-VI, so the 238-group neutron, 27-neutron/18-gamma group, 200-neutron/47-gamma group, and continuous-energy neutron ENDF/B-VI.8 data will be removed from SCALE. A comprehensive set of ENDF/B-VII.0 data remains available with SCALE.
- The SMORES material optimization tools were designed for criticality safety analysis, but they are used very little and are inconsistent with the TSUNAMI sensitivity and uncertainty analysis tools. SMORES and several associated tools will be removed from SCALE.
- With the removal of ENDF/B-V nuclear data, the NITAWL Nordheim integral treatment resonance self-shielding techniques are no longer applicable to SCALE nuclear data libraries, and they will be removed from SCALE.
- The NITAWLST sensitivity version of NITAWL is also no longer applicable and will be removed from SCALE.
- The QADS (Quick And Dirty Shielding) point kernel analysis module and its associated tools will be removed from SCALE. The MAVRIC tools provide rigorous shielding capabilities in SCALE.
- The WAX module for manipulating *working* formatted multigroup cross-section data libraries has been merged with AJAX, the module for manipulating *master* formatted libraries, so WAX will be removed from SCALE.
- Some seldom used utility modules will also be removed from SCALE.
    - AIM - Module to convert binary cross-section libraries to and from ASCII text
    - REORG – Module for post processing ORIGEN data files used with ARP

Additional Guidance for MCDancoff

Brian J. Ade

The SCALE Newsletter for fall 2010 contained guidance for calculating and using Dancoff factors in SCALE lattice physics calculations (Use of MCDancoff when Modeling Boiling Water Reactors). The guidance contained in the fall 2010 article still holds true – in order to obtain accurate results for highly heterogeneous fuel lattices, users should calculate and utilize MCDancoff-calculated Dancoff factors for fuel pins along assembly edges, in assembly corners, and for any other fuel pin that might experience a neutron flux spectrum that is significantly different from the spectrum for an infinite lattice. While this effect is prominent for high-void boiling water reactor fuel lattices, other heterogeneous fuel designs might need the same treatment in order to generate accurate results.

In addition to the guidance in the fall 2010 newsletter, users should construct 3D lattice models with a significant axial length when using reflective axial boundary conditions. By design, MCDancoff starts particles on all surfaces of a specified shape, including those that are coplanar with a reflective boundary. Particles started on a reflective boundary immediately leave the starting region and are returned to the same region, leading to a nonrealistic increase in the Dancoff factor that approaches 1.0 for models whose pin length is small compared to its diameter. In Figure 3, Dancoff factors for a pin cell have been plotted as a function of the ratio of the length of the fuel pin to the diameter of the fuel pin for varying moderator densities.

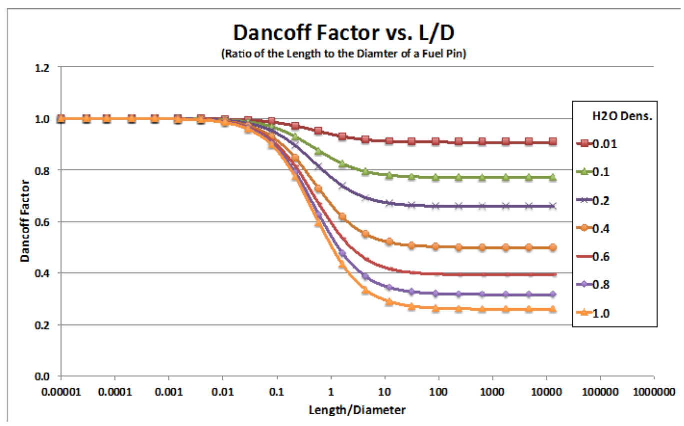


Figure 3. Dancoff factor as a function of the ratio of the length to the diameter of a model for varying moderator densities.

There is little impact on the Dancoff factor for low moderator densities because the free streaming probability between fuel lumps is high, leading to a low sensitivity to the axial effects. However, for higher moderator densities, the sensitivity to the axial length is much greater. Generally, the MCDancoff model should be constructed with a length/diameter ratio of at least 100 in order to obtain accurate results for varying moderator densities.

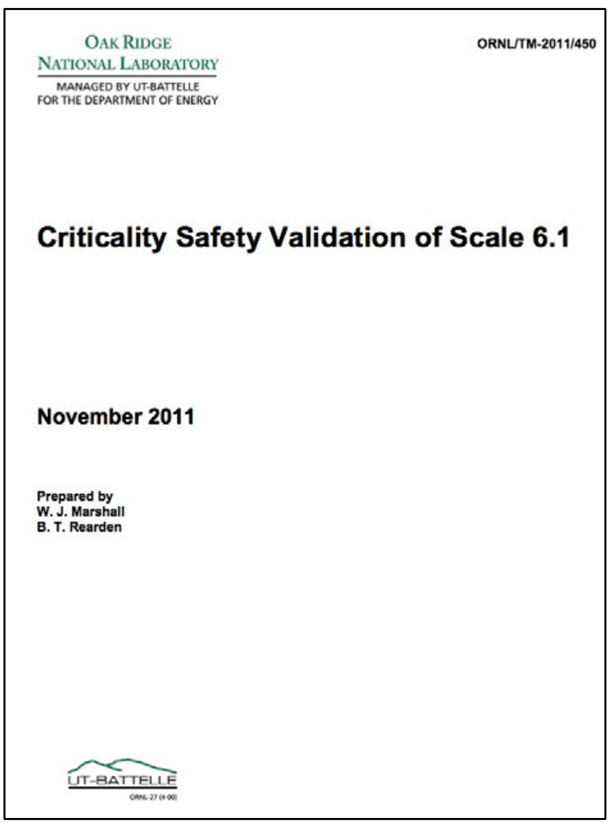
Revision to SCALE 6.1 Criticality Validation Report

William J. Marshall

A revision has been issued for *Criticality Safety Validation of Scale 6.1* (ORNL/TM-2011/450). A minor formatting error impacts the results for KENO-VI multigroup calculations for the HEU-MET-FAST experiments. The report has been revised to correct the error, resulting in changes to Figures 26–28 and Tables 10 and B-1. The revised report is available in the validation report section of the SCALE web site (http://scale.ornl.gov/validation_safety.shtml) or can be accessed directly at

(http://info.ornl.gov/sites/publications/Files/Pub40500.pdf).

Figures 4 and 5 illustrate the correction in the reported calculated k_{eff} value for the multigroup results for HMF-005-004. The average bias for the HMF experiments calculated with multigroup cross sections is slightly more negative in the revised report, with a value of 0.99801 compared to the previously published value of 0.99848.


OAK RIDGE NATIONAL LABORATORY

MANAGED BY UT-BATTELLE FOR THE DEPARTMENT OF ENERGY

ORNL/TM-2011/450

Criticality Safety Validation of Scale 6.1

November 2011

Prepared by
W. J. Marshall
B. T. Rearden

UT-BATTELLE

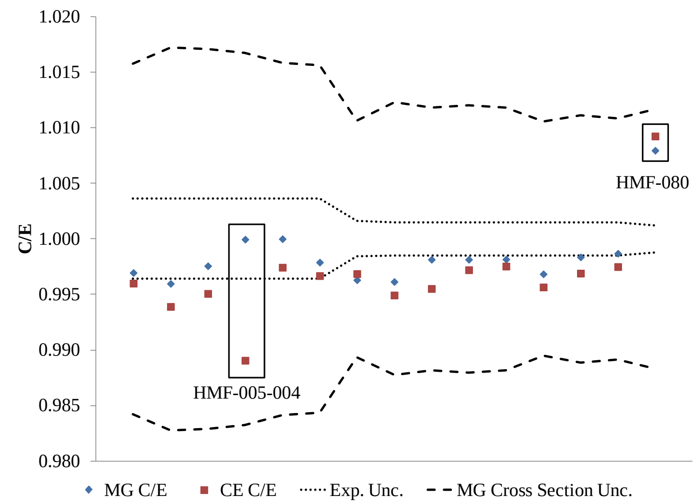


Figure 4. Original Fig. 28 Detailed Results for HMF Systems, KENO-VI.

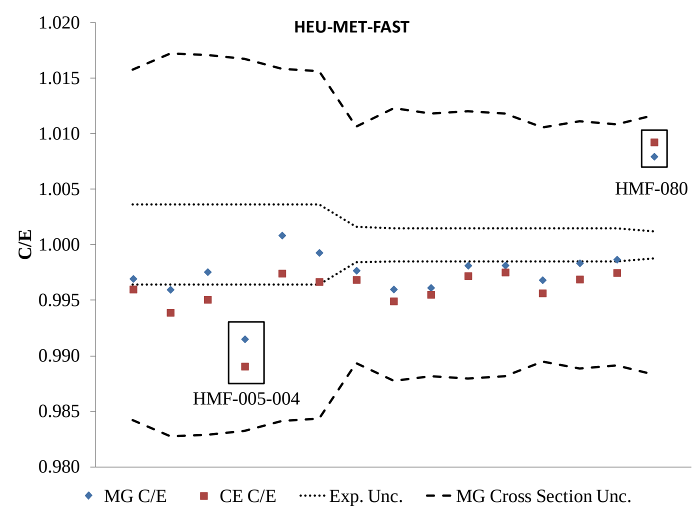


Figure 5. Revised Fig. 28 Detailed Results for HMF Systems, KENO-VI.

Corrected User Guidance for Criticality Accident Alarm System Modeling with SCALE

Thomas M. Miller and Douglas E. Peplow

Since the introduction of the MAVRIC/Monaco three-dimensional shielding sequence with hybrid acceleration in SCALE 6.0, guidance has been provided in the SCALE manual on how to perform analysis of criticality accident alarm systems (CAAS). Through a two-step process, a spatially varying fission source is generated with a KENO-VI criticality calculation in the CSAS6 sequence using a mesh tally, and then detector responses are calculated with MAVRIC from the mesh-based fission source obtained from KENO-VI.

Using the hybrid techniques of MAVRIC, any neutrons and photons included in the source specification will be biased towards the computed response of interest. Thus, adding fission photon data to the source definition from KENO will optimize the convergence of the photon tallies and accelerate the shielding portion of the CAAS analysis. The addition of the fission photons to the neutron mesh source provides accurate modeling of the neutron and photon fission source at each initial source location in the Monaco simulation. However, to avoid producing too many fission neutrons and photons, subsequent fission events must be treated as absorption events producing no fission neutrons or photons because all fission events are accounted for in the KENO-generated mesh source.

The method to avoid producing too many fission photons is based on an assumption that fission photon production can be entirely separated from the production of other photons generated by non-fission neutron interactions. In theory this is a reasonable assumption, but it is only possible if the photon production data are available in an amenable format. In practical application, most evaluated cross-section data, including ENDF/B-VII.1 and all previous releases, are not available in a format where it is possible to entirely separate fission photon production from the production of all other photons. For example, in ENDF/B-VII.1 and earlier versions, fission photons are not separable for neutron reactions above 1.09 MeV for ^{233}U, ^{235}U, and ^{239}P and above 0.1 MeV for ^{241}Pu. For neutron reactions above this cutoff energy, photons due to capture and fission are combined in a single *non-elastic* reaction.

To account for this artifact in the nuclear data, the user guidance provided to account for fission photons during CAAS analysis in all previous SCALE manuals, publications, and training courses can lead to incorrect results for fast-spectrum systems and needs to be revised. The reason this issue only affects fast systems is because each fissionable isotope has an energy cutoff below which the fission photon production is easily separable and above which it is not.

Previous SCALE CAAS Analysis Guidance, Accounting for Fission Photons

The methodology provided with SCALE 6.1 that is recommended to perform CAAS analysis is outlined below.

1. Calculate the spatial and energy-dependent fission neutron distribution using KENO-VI
   a. Set KENO parameter `cds=yes`
   b. Add grid geometry to KENO input
2. Convert the KENO neutron mesh tally to a Monaco mesh source using the MAVRIC utility MT2MSM
3. Calculate the CAAS detector response using Monaco
   a. Input the directory paths to the neutron mesh source and kenoNuBar.txt files, and set the number of fissions
   b. Use the `noFissions` parameter
   c. If no photon CAAS detector response is needed, use the `noSecondaries` parameter
   d. If a photon CAAS detector response is needed
      i. In the `celldata` block set the `moredata` parameter `nFisFot` equal to 1 to remove photon yields from the fission cross sections used by the MAVRIC sequence
      ii. Set the source keyword `fissPhotonZAID` to the primary fission isotope ZAID to add fission photon data to the neutron source

Omit Step 3.d.

A more detailed explanation of each step is available in Appendix C of the MAVRIC manual.

Revised CAAS Analysis Guidance, Accounting for Fission Photons

The problem with the guidance listed above occurs with step 3.d, which is removing fission photons from the (n,γ) transfer matrix (step 3.d.i) and adding fission photons to the Monaco source (step 3.d.ii). *In order to correctly account for fission photons in CAAS analysis, step 3.d should be ignored entirely*. In other words, if a photon CAAS detector response is needed, then the parameters `noSecondaries`, `nFisFot`, and `fissPhotonZAID` should not be used. This is because the data that need to be removed from the (n,γ) transfer matrix cannot be entirely removed when using cross sections based on ENDF/B-VII.1, any previously released version of ENDF, or nearly all other evaluated nuclear data files. If new evaluations separating fission photon production for all energies are included in ENDF, the previous CAAS user guidance, including step 3.d, is the correct procedure to perform CAAS analysis. Otherwise, step 3.d should be ignored. The revised CAAS guidance will be documented in the SCALE user manuals beginning with SCALE 6.2 and can be applied to all calculations including SCALE 6.1 and 6.0 to produce correct results with all ENDF evaluations currently available.

Comparison of Methods

Issues will arise with the previous CAAS user guidance for fast systems. For thermal- and intermediate-energy systems, where the vast majority of fissions occur at energies where fission photon production is separable, the impact is minimal. In order to illustrate the difference between these two methodologies, the results are presented from a simple model that exacerbates the issue. The model consists of a critical sphere of metal ^{239}Pu (radius = 4.946 cm, density = 19.82 g/cm^3). The quantities that are compared are the neutron and photon kerma in air 2 meters from the surface of the critical sphere (ICRP flux-to-dose conversion factors in SCALE), and the calculations used cross-section data based on ENDF/B-VII.0. Additionally, the MAVRIC results are compared with results from XSDRNPM and MCNP5. It should be pointed out that XSDRNPM is the computational tool that was used to develop the Nuclear Criticality Slide Rule. The MCNP calculations used continuous-energy cross sections, while the XSDRNPM and MAVRIC calculations used multigroup cross sections. The results of these calculations are presented in Table 2.

Table 2. Calculated Dose Rates for a Critical ^{239}Pu Sphere[a]

Dose Rates (Air Kerma – Gy/hr/fiss/sec)	XSDRN	MCNP5	MAVRIC CAAS Previous Guidance	MAVRIC CAAS Revised Guidance
Neutron	6.00e-14	5.99e-14	5.99e-14	5.99e-14
Photon	2.23e-14	2.23e-14	3.31e-14	2.23e-14

[a] All Monte Carlo results have a relative uncertainty of less than 0.3%.

The neutron kerma results in Table 2 all agree very well, as was expected. The photon kerma results produced by XSDRNPM, MCNP5, and MAVRIC using the revised CAAS guidance also agree very well. Note in the MCNP5 calculation the thick-target bremsstrahlung model was turned off, as was Doppler energy broadening for photons, and the photon cutoff energy was set at 10 keV to match the SCALE cross-section library. The largest difference in Table 2 is between MAVRIC with previous guidance photon kerma and all the other photon kermas. The previous guidance produces too many fission photons induced by fast neutrons. In this example, the additional fission photons with the previous guidance overestimated the photon kerma by nearly 50%.

The photon kerma being overestimated by nearly 50% with the previous CAAS guidance translates into a 13% underestimation of the minimum accident of concern based on ANSI/ANS-8.3-1997. The minimum accident of concern with the previous CAAS guidance is 1.29×10^{14} fissions per second, but is 1.46×10^{14} fissions per second with the revised CAAS user guidance. The next obvious question is how the underestimation of the minimum accident of concern affects the Monaco CAAS detector response calculation. Therefore, four example calculations were performed. In these calculations the critical Pu sphere described previously was placed in a location shielded from a CAAS detector or a location with direct line of sight to the CAAS detector. With the source in each of these two locations, the CAAS detector response was calculated using the previous and revised CAAS analysis guidance. Below in Figure 6 is a plan view of the geometry. The sphere at the top right is the CAAS detector (filled with air), the sphere at the bottom right is the unshielded Pu sphere location, and the sphere at the bottom left is the shielded Pu sphere location. The blue walls are 15.24 cm thick concrete walls, and the yellow boxes in a 2×2 array are Pb glove boxes with 2.54 cm thick walls. The green material in Figure 6 is air, and the light blue represents 5.08 cm thick stainless steel doors. The distances from the center of the CAAS detector to the center of the shielded and unshielded Pu spheres are 1004 cm and 714 cm, respectively. The calculated photon dose rates for the CAAS detector in Figure 6 for these four calculations are shown in Table 3.

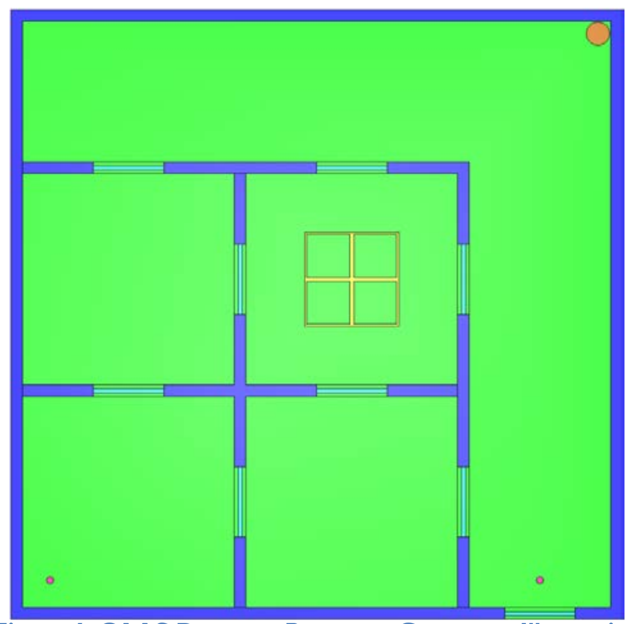


Figure 6. CAAS Detector Response Geometry Illustrating Both Critical Pu Sphere Locations.

Table 3. Calculated Photon CAAS Detector Reponse[a]

Photon Dose Rates (Air Kerma – Gy/min)	Previous Guidance	Revised Guidance
Shielded	2.04e-3	2.35e-3
Unshielded	3.14e-2	3.26e-2

[a] All Monte Carlo results have a relative uncertainty of less than 1%.

For the shielded case in Table 3, the detector response using the revised guidance is about 15% greater than when using the previous guidance. Statistically, this difference is about the same as that observed for the minimum accidents of concern. In other words, the CAAS detector photon response is not affected by the production of fission photons but rather is dominated by photon production near the CAAS detector by neutron scattering. Therefore, the difference in the photon CAAS detector response for the shielded case is primarily driven by the difference in the calculation of the minimum accident of concern and the number of fission neutrons that are produced. So for a full-shielded case, the difference between the calculated CAAS detector responses with the previous and revised CAAS analysis guidance will be approximately equal to the difference in the minimum accidents of concern.

For the unshielded case in Table 3, the detector response using the revised guidance is about 4% greater than when using the previous guidance. Statistically speaking, these two results are the same. The results are the same *because of the error* that led to underestimating the number of fissions in the minimum accident of concern, that is, too many photons per fission. In the CAAS detector response calculation, the previous analysis guidance still produces too many photons per fission, leading to a higher dose rate per fission than the revised analysis guidance. Therefore, in a fully unshielded case, the error in the previous CAAS analysis guidance can compensate for itself because the number of fissions in the minimum credible accident will be low but the dose rate delivered per fission is high.

In summary, the revised CAAS analysis guidance should be used henceforth. However, it is particularly important to use the revised CAAS analysis guidance to ensure no incorrect results are produced for any fast critical system when calculating the minimum accident of concern and when calculating a photon CAAS detector response for fast systems with large amounts of shielding between the system and the CAAS detector. The revised guidance can be used with any version of MAVRIC and ENDF/B cross-section data provided with SCALE.

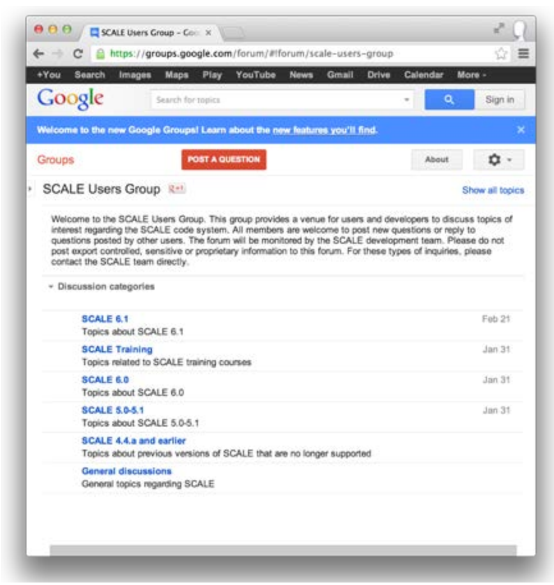


Figure 7. SCALE Users Group Categories.

SCALE Users Group Discussion Forum

A new venue is now available to facilitate interaction between SCALE users and developers. The SCALE Users Group is a new forum hosted by Google and available at https://groups.google.com/forum/#!forum/scale-users-group.

The SCALE Users Group replaces the SCALE User Notebooks, which were hosted by RSICC for many years. All previous Notebook postings related to SCALE 5.0–6.1 have been migrated to the SCALE Users Group, so hundreds of topics are already available for users to browse. The SCALE Users Group is available to everyone, and all licensed SCALE users are welcome to post and respond to topics.

The topics on the SCALE Users Group are organized by categories, as shown in Figure 7. An example of topics currently posted for SCALE 6.1 is shown in Figure 8.

Please note that the SCALE Users Group is an open forum, so please do not post proprietary or export controlled information to the forum.

Group membership is moderated by the SCALE Team, with account and enrollment instructions available at

http://scale.ornl.gov/Readme_SCALE_Users_Group.shtml.

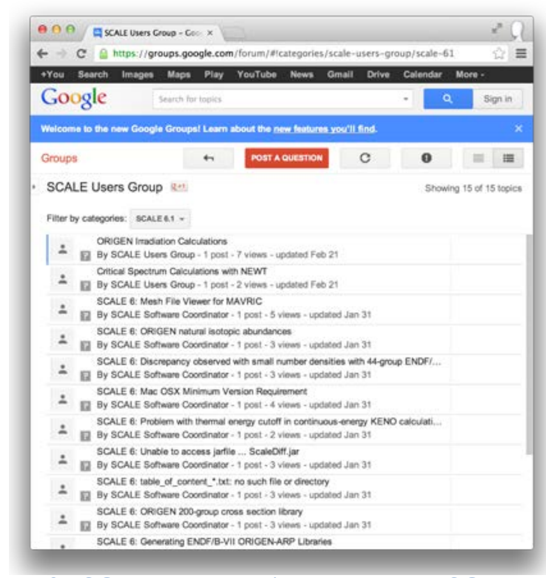


Figure 8. SCALE 6.1 Topics Posted to SCALE Users Group.

SCALE Publications

The SCALE Team provides numerous publications on development and application activities in peer-reviewed journals, technical reports, and conference publications. Often, publications are jointly created with users and developers throughout the community. A summary of some recent and pending publications is provided here.

Peer-Reviewed Journal Articles

J. C. Wagner, D. E. Peplow, and S. W. Mosher, "FW-CADIS Method for Global and Semi-Global Variance Reduction of Monte Carlo Radiation Transport Calculations," *Nuclear Science and Engineering* (accepted for publication)

D. E. Peplow, T. M. Miller, B. W. Patton, and J. C. Wagner, "Hybrid Monte Carlo/Deterministic Methods for Active Interrogation Modeling," *Nuclear Technology* 182(1) (2013)

J. A. Roberts (M.I.T), B. T. Rearden (ORNL), and P. H. Wilson (Univ. of Wisconsin), "Determination and Application of Partial Biases in Criticality Safety Validation," *Nuclear Science and Engineering* 173, 43–57 (2013)

Technical Reports

D. E. Peplow, S. W. Mosher, and T. M. Evans, *Consistent Adjoint Driven Importance Sampling using Space, Energy and Angle*, ORNL/TM-2012/7, August 2012

B. J. Ade, *SCALE/TRITON Primer: A Primer for Light Water Reactor Lattice Physics Calculations*, NUREG/CR-7041 (ORNL/TM-2011/21), November 2012

H.J. Smith, I.C. Gauld, and U. Mertyurek, *Analysis of Experimental Data for High Burnup BWR Spent Fuel Isotopic Validation—SVEA-96 and GE14 Assembly Designs*, NUREG/CR, ORNL/TM-2013/18 (2013)

American Nuclear Society, 2012 Annual Meeting, Chicago, IL, USA, June 2012

"Methods for Detector Placement and Analysis of Criticality Accident Alarm System," D. E. Peplow and L. L. Wetzel

"Examination of Validation Outlier Cases Using the Sensitivity and Uncertainty Analysis Tools of SCALE 6.1," B. T. Rearden and W. J. Marshall

"Criticality Safety Validation of SCALE 6.1 with ENDF/B-VII.0 Libraries," W. J. Marshall and B. T. Rearden

American Nuclear Society, 2012 Winter Meeting, San Diego, CA, USA, November 2012

"Automatic Mesh Adaptivity for Hybrid Monte Carlo/Deterministic Neutronics Modeling of Difficult Shielding Problems," Ahmad M. Ibrahim (ORNL), Paul P. Wilson, Mohamed E. Sawan (Univ. of Wisconsin, Madison), Douglas E. Peplow, John C. Wagner, Scott W. Mosher, Thomas M. Evans (ORNL)

"Comparison of Hybrid Methods for Global Variance Reduction in Shielding Calculations," Douglas E. Peplow

International Conference on Nuclear Data for Science and Technology (ND 2013), New York, NY, March 2013

"Validation and Testing of ENDF/B-VII Decay Data," I. C. Gauld, M. T. Pigni, G. Ilas

"Applications of Nuclear Data Covariances to Criticality Safety and Spent Fuel Characterization," M. L. Williams, G. Ilas, W. J. Marshall, B. T. Rearden

"Inverse Sensitivity/Uncertainty Methods Development for Nuclear Fuel Cycle Applications," G. Arbanas, M. E. Dunn, and M. L. Williams

Mathematics & Computational Methods Applied to Nuclear Science and Engineering, Sun Valley, ID, May 2013

"Comparison of Hybrid Methods for Global Variance Reduction in Shielding Calculations," D. E. Peplow

"Automatic Mesh Adaptivity for CADIS and FW-CADIS Neutronics Modeling of Difficult Shielding Problems," A. M. Ibrahim, D. E. Peplow, S. W. Mosher, J. C. Wagner, and T. M. Evans

"A New Approach for Modeling and Analysis of Molten Salt Reactors Using Scale," J. J. Powers, T. J. Harrison, and J. C. Gehin

American Nuclear Society, 2013 Annual Meeting, Atlanta, GA, USA, June 2013

"Implementation of the Doppler Broadening Rejection Correction in KENO," Shane W. Hart, G. Ivan Maldonado (Univ. of Tennessee), Sedat Goluoglu (Univ. of Florida), Brad Rearden (ORNL)

"Corrected User Guidance to Perform Three-Dimensional Criticality Accident Alarm System Modeling with SCALE," T. M. Miller and D. E. Peplow

"Hybrid Monte Carlo/Deterministic Technique for Shutdown Dose Rate Calculations," A. M. Ibrahim, D. E. Peplow, and R. E. Grove

"Propagation of Uncertainty from a Source Computed with Monte Carlo," D. E. Peplow, A. M. Ibrahim, and R. E. Grove

International Nuclear Fuel Cycle Conference (Global 2013), Salt Lake City, UT, USA, September 2013

"Parametric Analyses of Single-zone Thorium-fueled Molten Salt Reactor Fuel Cycle Options," J. J. Powers, J. C. Gehin, T. J. Harrison, and A. Worrall

SCALE Spotlight

SCALE is developed, tested, documented, and maintained by approximately 40 talented and diverse staff members within the Reactor and Nuclear Systems Division at Oak Ridge National Laboratory. The SCALE Spotlight provides a profile of a team member in each edition.

Dr. Matthew A. Jessee

Dr. Matthew A. Jessee

Position:

Research Staff, Lattice Physics Technical Lead

Working on SCALE since 2008

Focus areas:

Reactor Physics, Sensitivity/Uncertainty Analysis

Most memorable projects:

Since joining the SCALE development team in the spring of 2008, I have been fortunate to be involved in the release of SCALE 6.0, SCALE 6.1, and the upcoming release of SCALE 6.2, as well as several training opportunities for both TRITON and TSUNAMI. The release of SCALE 6.0 stands out to me as the most memorable. We were targeting a December 2008 release, and I was tasked to qualify two new sensitivity and uncertainty (S/U) analysis modules (TSURFER and TSAR), as well additional feature enhancements to TSUNAMI-1D, TSUNAMI-3D, and TSUNAMI-IP. This was an exciting opportunity as a new hire, and I was proud to see these features added to the SCALE S/U toolkit.

Life outside of work:

Outside of work, I still enjoy working on SCALE!

Aside from SCALE, I enjoy spending time with my family. My wife and I have a two-year-old daughter and a four-month-old son, so they keep us busy. I enjoy running when I find the time. I log roughly 450 miles a year, and I try to run one half-marathon every year.

Technical Support and Training

The SCALE Team is dedicated to supporting all SCALE users, but the team can only provide limited complimentary technical support for inquiries submitted to scalehelp@ornl.gov. For basic help in getting started with SCALE, new users are encouraged to attend the public training courses, described on Page 12, where the capabilities of SCALE are presented in detail. Additional community support is available through the SCALE Users Group described on Page 9 of this newsletter.

The Primers distributed with SCALE for KENO V.a, KENO-VI, TSUNAMI, and TRITON provide detailed step-by-step instructions to assist new users in learning how to use these modules for criticality safety, sensitivity/uncertainty, lattice physics, and source term calculations. Direct links to the SCALE Primers are available at http://scale.ornl.gov/training_primers.shtml.

If your team could benefit from customized technical support or training, additional options are available. The SCALE Team can provide direct support or travel to your site to present customized hands-on courses to provide the expertise needed to solve challenging application scenarios. Please contact the scalehelp@ornl.gov for more information.

SCALE Leadership Team

The SCALE Leadership Team consists of the SCALE Project Leader, line managers, program managers, and developers. The Leadership Team meets regularly to discuss the current status and make programmatic and managerial decisions regarding SCALE.

Members of the SCALE Leadership Team are as follows:

Brad Rearden – SCALE Project Leader

Steve Bowman – Group Leader, Reactor Physics

Mike Dunn – Group Leader, Nuclear Data and Criticality Safety

Matt Jessee – Senior Developer, Reactor Physics

Douglas Peplow – Senior Developer, Radiation Transport

Mark Williams – Senior Developer, Nuclear Data and Reactor Physics

Spring 2013 Training Courses

Date	Title	Location	Registration Fee
April 8–12	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
April 15–19	SCALE Sensitivity and Uncertainty Calculations Course *TSUNAMI: 1-D and 3-D sensitivity/uncertainty analysis using TSUNAMI with XSDRNPM and KENO. Advanced S/U methods for code and data validation*	ORNL Oak Ridge, TN, USA	$2000
April 22–26	SCALE 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 non-uniform lattices; generation of libraries for ORIGEN-ARP*	ORNL Oak Ridge, TN, USA	$2000
April 29–May 1	SCALE/ORIGEN Activation and Decay Calculations Course *Isotopic depletion/decay and source term characterization using ORIGEN/ORIGEN-ARP*	ORNL Oak Ridge, TN, USA	$1500
May 27–31	SCALE/Criticality Safety Calculations Course - Paris *Introductory through advanced criticality calculations using KENO V.a and KENO-VI; resonance self-shielding techniques*	NEA Data Bank, Issy-les-Moulineaux, France	€2000
June 3–7	SCALE/Sensitivity and Uncertainty Calculations Course - Paris *TSUNAMI: 1D, 2D, and 3D k_{eff} sensitivity/uncertainty analysis; 2D generalized sensitivity analysis for lattice physics; reactivity sensitivity analysis; advanced S/U methods for code and data validation using trending analysis and data assimilation (data adjustment) techniques; k_{eff} burnup credit validation*	NEA Data Bank, Issy-les-Moulineaux, France	€2000

Please register at least 40 days before the start of the desired course.

For more information and online registration, please visit

http://scale.ornl.gov/training.shtml

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

http://facebook.com/Scale.codes

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, and the U.S. Department of Energy.