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SCALE Enhancements for Detailed Cask Dose Rate Analysis

Douglas E. Peplow, Georgeta Radulescu, Mark L. Williams, and Robert A. Lefebvre

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

INTRODUCTION

The Nuclear Fuels Storage and Transportation Planning Project and the Used Fuel Disposition Campaign, both under the US Department of Energy (DOE) Office of Nuclear Energy, require the ability to predict k_{eff}, heat generation and temperature distributions, and external dose rates from any combination of commercial spent nuclear fuel assemblies loaded into any type of licensed cask. Work in the Reactor and Nuclear Systems Division at Oak Ridge National Laboratory[1] has begun to develop the UNF-ST&DARDS system (Used Nuclear Fuel Storage, Transportation & Disposal Analysis Resource and Data System [1]) that uses many of the nuclear analysis codes from the SCALE package [2], a comprehensive database of used nuclear fuel inventories from utilities [3], and a library of model template files to produce complete input files for the various types of UNF computational analyses. UNF-ST&DARDS uses the template files and input parameters that are applicable to a safety analysis to assemble and execute a complete input file, collect the output files and relevant information for subsequent calculations, and illustrate calculation results as a function of decay time and other relevant analysis parameters. This paper will focus on the shielding aspects and the ability to predict external dose rates from as-loaded casks. Realistic dose rate values based on actual canister fuel contents may be used in developing appropriate plans, procedures, and methods to keep personnel radiation exposures as low as reasonably achievable during cask repackaging and transportation operations, as well as in performing total system analyses and transportation risk evaluations.

The MAVRIC [4] shielding sequence of the SCALE package was created for the US Nuclear Regulatory Commission for analyzing dose rates outside of storage, transfer, or transportation casks loaded with spent nuclear fuel. MAVRIC contains automated variance reduction capabilities based on the CADIS and FW-CADIS methods [5] that are effective for optimizing deep penetration Monte Carlo shielding problems for achieving low relative uncertainties in short computing times. MAVRIC uses the Denovo [6] discrete ordinates code to compute importance maps and biased source terms for the Monaco fixed-source Monte Carlo code, which can use multi-group or continuous-energy cross section data.

ORIGEN (Oak Ridge Isotope Generation and Depletion) [7] is the SCALE computer code for calculating time-dependent isotopic concentrations during irradiation and decay for more than 2,200 nuclides that can be produced by nuclear fuel irradiation and activation.

A series of SCALE enhancements was implemented to enable automatic decay heat and radiation source term generation and shielding safety analysis of as-loaded casks. The enhancements to SCALE made under this project will be a part of the SCALE 6.2 release expected in fall 2014.

SCALE ENHANCEMENTS

Depletion with ORIGAMI

The ORIGAMI (*ORIG*en *A*sse*M*bly *I*sotopics) code was developed to determine assembly isotopics using multiple ORIGEN calculations. The ORIGEN one-group cross sections are interpolated from pre-generated libraries parameterized as functions of burnup and other variables. ORIGAMI can compute one-dimensional axially varying isotopics averaged over all fuel pins, or it can compute a full three-dimensional (3-D) isotopic distribution for all fuel pins, using an input power shape. Nuclide concentrations for each axial depletion zone are output as a text file in the composition format used by KENO for criticality calculations. The decay heat source (gamma and neutron) is computed for use in the COBRA-SFS code to perform thermal analysis of spent fuel containers.

For dose rate analysis, a capability was added to compute energy spectra for gamma and neutron sources in spent fuel. The source spectra are binned into any arbitrary group structure specified in the input, and these may be output either as a text file or contained in the standard binary concentration output file from ORIGEN. The sources may be differentiated based on whether they are produced from decay of fission products or actinides or were produced by activation of light elements found in assembly structural components.

Monaco Monte Carlo Transport

Discrete distributions were added to Monaco for a better representation of gamma line source energy distributions. A new special distribution was added so that

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users can specify an isotope (such as ^{60}Co or ^{238}U), and the code will set up a discrete distribution of the x-rays and gamma rays from that isotope.

MAVRIC Sequence

For problems such as a detailed cask analysis with hundreds of separate sources (e.g., each axial zone of each assembly), the current method of creating voxelized mesh sources for Denovo input was quite slow. The processing time for creating biased mesh sources in the CADIS process was also quite time consuming. Changes were made to these functions to significantly reduce their computational time.

MAVRIC Utilities

Several utilities were added to post-process mesh tally files. The mtMask utility will read a mesh tally file, keep or remove voxels that match certain unit/region/ mixture specifications, and write out a new mesh tally file. The utility mtMinMax will find the top n minimum or maximum values and their locations in the mesh tally file. These two utilities can be used together on a cask dose rate mesh tally to (a) keep only voxels that are in a specific region and (b) find the top 10 maximum dose rates and their locations.

Java Mesh File Viewer

The Java Mesh File Viewer can show a slice through the 3-D mesh files produced by the MAVRIC sequence (mesh tally, Denovo fluxes, mesh sources, importance maps, etc.). Users can select a top view (x-y), a front view (x-z) or a side view (y-z). For cylindrical mesh tallies, the only view available has been a front view (r-z). The viewer has been changed, so cylindrical meshes will show a plot of θ and z for any value of r. This will allow the user to see the dose rate all along the cylindrical outer surface of a cask and be able to identify areas of larger dose rates.

UNF-ST&DARDS MAVRIC Templates

Templates were developed to automate generation of MAVRIC input files for shielding analyses of an explicit model of a Universal Multi-Purpose Canister System [8] transportation cask by NAC International (NAC-UMS) containing used nuclear fuel from a nuclear power plant (name withheld). The MAVRIC root template file contains raw text, attributes to be replaced, and a series of instructions for importing sub-templates dedicated to developing specific input parameters. A MAVRIC input file is created by executing TemplateEngine [1], which is a string substitution program that takes advantage of repeated structures in text files. The Java TemplateEngine executable takes the input parameter data structures represented by a JavaScript Object Notation (JSON) object and the root template file. With these two components, the TemplateEngine conducts attribute replacement and sub-template imports. The MAVRIC root template was used to develop the MAVRIC input file for the normal conditions of transport (NCT) analysis of the NAC-UMS transportation cask; this input is used in the next section to demonstrate application of the new MAVRIC features.

DEMONSTRATION PROBLEM

Problem Description

The example problem that will be used to demonstrate the new capabilities of ORIGAMI and MAVRIC is the NCT dose rate analysis of the NAC-UMS transportation cask containing a Class 1 transportable storage canister (TSC), identified as TSC-8. The NAC-UMS transportation cask features a multiwall body, a bottom forging and a bottom plate with a 1 in. thick neutron shield positioned between them, and a thick stainless steel top lid. The multiwall cask body consists of concentric layers of stainless steel inner shell, lead radial gamma shielding, outer stainless steel shell, and neutron shielding material. The neutron shielding material is a borated hydrogenous material, NS-4-FR. The TSC-8 canister includes 24 irradiated fuel assemblies, 4 of which have been placed into damaged fuel cans. The assembly geometry model includes a pin-by-pin representation of the active fuel region, 18 axial fuel zones with burnup-dependent neutron and gamma sources, and assembly hardware regions represented with homogenous contents. Assemblies are based on 14 × 14 arrays of pins with 5 large water holes, which characterize the Combustion Engineering 14 × 14 assembly type. Figure 1 shows the level of detail included in the cask geometry model. The cask model for NCT analysis also includes upper and lower impact limiters.

Among the fuel assemblies, there are four dates of removal from the core, and the assemblies discharged on a specific date have roughly the same burnup. UNF ST&DARDS assigned an axial burnup profile depending on assembly burnup—one for assembly burnups between 18 and 30 GWd/MTU and another for assembly burnups ≥ 30 GWd/MTU.

Dose Rate Limits

Shielding safety analyses for spent fuel transportation casks must demonstrate that the external radiation levels satisfy the requirements of 10 CFR 71.47 and 71.51 [9] for NCT and hypothetical accident conditions (HAC) of transport, respectively. The cask used in the example problem meets the exclusive-use definition of a closed conveyance during transportation. For exclusive-use shipments with a closed transport vehicle type, the 10 CFR 71.47(b) requirements include the following dose rate limits:

1. the surface of the package must be < 1,000 mrem/h
2. the outer surface of the vehicle must be < 200 mrem/h
3. the top surface of the vehicle must be < 200 mrem/h
4. the underside of the vehicle must be < 200 mrem/h
5. areas 2 m from the outer lateral surfaces of the vehicle must be < 10 mrem/h
6. any normally occupied position of the vehicle must be < 2 mrem/h

The objective of the sample problem is to compute dose rates external to the cask for two different shipping dates—one 15 years after the last discharge and one 25 years after the last discharge. Dose rates at the cask external surface, personnel barrier, and 2 m from the cask will be examined. This sample problem specifies only the radiation sources originating in the assembly active fuel region.

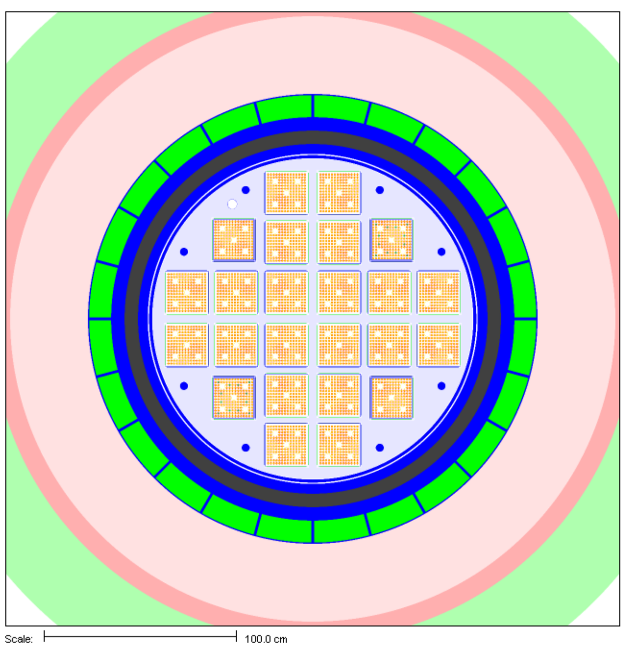

Figure 1. Cask model geometry showing 24 assemblies.

ORIGAMI Calculations

The UNF-ST&DARDS package produced one ORIGAMI input file for each assembly in the cask. The input files rely on ORIGEN-ARP (ORIGEN-Automatic Rapid Processing) libraries that have been made previously for each fuel assembly type. The ORIGAMI input lists the initial enrichment, the amount of fuel (in MTUs), and the 18-zone axial burnup profile. A power history is constructed using the typical operating parameters of the plant and time period to burn the fuel to the listed burnup. This may include one to three cycles with downtime in between. ORIGAMI produces an ORIGEN binary concentration file containing the final isotopic values for each axial zone of the assembly as well as the assembly as a whole.

For this problem, the down time of the last cycle was adjusted to reflect the time between the last discharge and the shipping date (either 15 years or 25 years). Energy group bin boundaries for collecting the neutron and photon emissions from the final isotopes also were added. The neutron boundaries were taken from the SCALE 200-group shielding library, and the photon energy groups used 999 equally spaced bins from 10 keV to 10 MeV.

MAVRIC Calculations

A MAVRIC input file can be generated from the UNF-ST&DARDS program. The ORIGAMI binary concentration files are linked into the run and used as the source distributions. For the example problem, 24 assemblies with 18 axial zones give 432 neutron and 432 photon energy distributions. Each distribution is used by a source definition that includes the corresponding volume of the assembly. Source particles are only started within the fuel material within that axial zone. Cylindrical mesh tallies (one each for neutrons and photons) are used to record the flux inside and outside the cask. The 1977 ANSI standard flux-to-dose rate conversion factors are combined with the flux to compute dose rates in mrem/h.

To converge the total dose rate outside the cask in all areas, the FW-CADIS variance reduction method is used. This creates an importance map that splits and roulettes particles based on how they will contribute to the total dose in all areas surrounding the cask. The importance is generated from the combination of forward and adjoint discrete ordinates calculations using Denovo. This requires a mesh, which is supplied by the UNF templates for the specific cask design. MAVRIC automates the generation of the importance map and the final continuous energy (CE-Monaco) Monte Carlo calculation.

For the two cases (15 years and 25 years after last discharge) of this example problem, the total MAVRIC time was kept to ~100 hours—5 hours for each Denovo calculation and 90 hours for the Monaco calculation.

External Dose Rates

The end result of the MAVRIC calculation is a pair of cylindrical mesh tallies—one for the neutron flux and dose rate, and one for the photon flux and dose rate. The new MAVRIC utilities enable automatic identification of maximum dose rate values within geometrical regions that are of interest for dose rate analyses. The new utilities can be used to add the neutron and photon dose rates into a total mesh tally, extract portions of that mesh tally corresponding to a specific region or material, and then find the maximum dose rate. In the geometry model, different sections of the air surrounding the cask were given different material numbers (all still being "air"). Table I lists the maximum dose rate (neutron + photon) in the 9 areas for both the 15-year and the 25-year cases.

The maximum dose rate values listed in Table I show that the most limiting dose rate with respect to the 10 CFR 71.47 requirements is the dose rate at 2 m from the cask radial surface. The dose rate values on the cask external surface and at the personnel barrier (PB) were significantly below the 10 CFR 71.47 dose rate values for these surfaces, 1,000 mrem/h and 200 mrem/h, respectively. The maximum 2 m radial dose rate of 4.28 mrem/h for the 15-year calculation was the closest to the dose rate limit (10 mrem/h). For both cases, cask external dose rate values were smaller than the dose rate limits of 10 CFR 71.47. The dose rate values used in this analysis do not include contributions from the activation sources in non-fuel regions and are only used to illustrate the MAVRIC enhancements.

Table I. Maximum Dose Rates

Location		15 years after last discharge		25 years after last discharge	
		mrem/hr	rel. unc.	mrem/hr	rel. unc.
Cask	Top	0.23	8%	0.16	8%
	Upper Radial	27.78	3%	18.98	3%
	Middle Radial	19.98	14%	12.45	3%
	Lower Radial	85.27	2%	57.50	2%
	Bottom	1.06	7%	0.34	9%
PB	Radial	11.89	2%	7.94	2%
2 m	Top	0.13	6%	0.08	5%
	Radial	4.28	2%	2.77	3%
	Bottom	0.19	10%	0.10	5%

SUMMARY

Enhancements made to SCALE allow for easier and more efficient external dose rate analysis of detailed cask models. Models can include the same pin-wise geometry used for k_{eff} calculations and source distributions that vary by assembly and assembly axial zone. Future development will include ^{60}Co from activated assembly hardware and studies to determine the optimum calculational parameters to minimize memory and time requirements.

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REFERENCES

1. J. M. SCAGLIONE**,** R. A. LEFEBVRE, K. R. ROBB, J. L. PETERSON, H. ADKINS, T. E. MICHENER, and D. VINSON, "Integrated Data and Analysis System for Commercial Used Nuclear Fuel Safety Assessments," in *Proceedings of PATRAM 2013*, San Francisco, CA, August 18–23, 2013.
2. *SCALE: A Comprehensive Modeling and Simulation Suite for Nuclear Safety Analysis and Design*, ORNL/TM-2005/39, Version 6.1, June 2011. Available from Radiation Safety Information Computational Center at Oak Ridge National Laboratory as CCC-785.
3. *RW-859 Nuclear Fuel Data*, Energy Information Administration, Washington, D.C. (2004).
4. D. E. PEPLOW, "Monte Carlo Shielding Analysis Capabilities with MAVRIC," *Nuclear Technology* **174**(2), 289–313 (2011).
5. J. C. WAGNER, D. E. PEPLOW, and S. W. MOSHER, "FW-CADIS Method for Global and Regional Variance Reduction of Monte Carlo Radiation Transport Calculations," *Nuclear Science and Engineering* **176**(1), 37–57 (2014).
6. T. M. EVANS, A. S. STAFFORD, R. N. SLAYBAUGH, and K. T. CLARNO, "Denovo: A New Three Dimensional Parallel Discrete Ordinates Code in SCALE," *Nuclear Technology* **171**(2), 171–200 (2010).
7. I. C. GAULD, G. RADULESCU, G. ILAS, B. D. MURPHY, M. L. WILLIAMS, and D. WIARDA, "Isotopic Depletion and Decay Methods and Analysis Capabilities in SCALE," *Nuclear Technology* **174**(2), 169–195 (2011).
8. Safety Analysis Report for the UMS® Universal Transport Cask, USNRC Docket No. 71-9270, Revision 2, NAC International, November 2005.
9. Packaging and Transportation of Radioactive Material, 10 CFR 71.