

## Dose Rate Analysis of As-Loaded Spent Nuclear Fuel Casks

Georgeta Radulescu, Douglas E. Peplow, Mark L. Williams, and John M. Scaglione

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

### INTRODUCTION

Used Nuclear Fuel Storage, Transportation & Disposal Analysis Resource and Data System (UNF-ST&DARDS) [1] is a comprehensive data and calculation capability for UNF safety assessments currently being developed at Oak Ridge National Laboratory through a collaborative effort of multiple national laboratories and industry participants. The purpose of UNF-ST&DARDS is to provide a unified domestic UNF system of data and analysis capabilities in support of U.S. Department of Energy waste management and fuel cycle-related objectives. UNF-ST&DARDS is designed to provide safety assessments for UNF storage, transportation, and disposal systems based on actual characteristics of the as-loaded UNF (e.g., fuel assembly burnup, enrichment, decay time, cladding integrity, etc.). The cask-specific safety evaluations allow quantification of realistic safety margins and conditions and provide the technical basis for addressing concerns related to potential fuel failure (e.g., for high-burnup fuel).

This paper<sup>1</sup> describes the UNF-ST&DARDS methodology for calculating radiation dose rates for U.S. Nuclear Regulatory Commission (NRC) licensed casks [2] based on actual UNF canister contents. Realistic dose rate values based on actual canister contents may be used in place of bounding dose rate values to support development of repackaging operations procedures, evaluation of radiation-related transportation risks, and communication with stakeholders.

### UNF-ST&DARDS SOURCE TERM AND DOSE RATE CALCULATION CAPABILITIES

UNF-ST&DARDS has been designed to facilitate automated safety analyses for UNF canisters currently stored at the existing independent spent fuel storage installations (ISFSIs). UNF-ST&DARDS includes a comprehensive UNF database that can interface with various safety and system analysis codes (e.g., SCALE [3]) utilizing code-specific input file templates. Based on the cask loading information from the database, UNF-ST&DARDS uses template files and input

parameters applicable to a safety analysis to assemble and execute a complete input file, collect the output files and relevant information from the output file for subsequent calculations, and illustrate calculation results as a function of decay time and other relevant analysis parameters. Initial focus of UNF-ST&DARDS development has been the criticality and thermal analyses, but UNF-ST&DARDS capabilities are currently being expanded to include dose rate analyses of UNF transportation casks. UNF-ST&DARDS source term generation and dose rate analyses of as-loaded transportation casks are facilitated by new features and enhancements developed in the pre-released version of SCALE 6.2 [4].

### Used Nuclear Fuel Database

The current UNF-ST&DARDS database includes UNF discharge inventory through December 31, 2002, [5] storage data from U.S. commercial reactors, and design characteristics of commercial nuclear fuel assemblies obtained from a variety of open literature sources. Additional UNF discharge inventory data through the year 2013 [6] is currently being assembled to update the UNF discharge inventories. The UNF-ST&DARDS database is expandable and will include new UNF data that becomes available in the future.

### Source Term Calculations with ORIGAMI

ORIGAMI (*ORIGen AsseMbly Isotopics*) is a SCALE 6.2 capability dedicated to calculating nuclide inventories, decay heat, and radiation source terms for UNF assemblies exhibiting axial and radial burnup profiles. This code performs fast ORIGEN irradiation and decay calculations with pre-generated problem-specific ORIGEN-ARP cross-section libraries. UNF-ST&DARDS contains pre-generated ORIGEN-ARP [7] cross-section libraries for representative pressurized and boiling water reactor fuel assemblies within assembly classes. The UNF assemblies are categorized into assembly classes which are further subdivided by assembly type for a total of 134 individual fuel types [5]. The majority of the fuel types within a given class have similar neutronic characteristics that can be represented by a fuel type within that class.

The ORIGEN-ARP libraries have been generated with depletion input parameters that increase actinide nuclide concentrations (e.g., lowest moderator density expected during normal operation and fuel exposure to burnable absorber rods/control blades) [8]. These modeling parameters also increase the concentrations of

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important contributors to the neutron (e.g.,  $^{244}\text{Cm}$ ) and gamma (e.g.,  $^{154}\text{Eu}$ ) dose rate between 5 to 50 years after fuel discharge (i.e., higher radiation source intensities) [9,10].

Assembly-specific radiation source terms are generated with ORIGAMI for the active fuel as well as the assembly hardware regions and other non-fuel radioactive components that may be placed within fuel assemblies. For the fuel source term calculations, the active fuel is represented as 18 axial fuel regions with burnup based on assembly average burnup and axial burnup profiles dependent on assembly average burnup. Thus, a canister holding 24 intact fuel assemblies is characterized by 432 neutron sources, 432 photon sources in the active fuel regions, and 72 cobalt activation sources in assembly upper, plenum, and lower hardware regions.

### Dose Rate Calculations with SCALE/MAVRIC

MAVRIC [11] is the SCALE shielding sequence with automated variance reduction capabilities that is used within the UNF-ST&DARDS to perform cask-specific dose rate analyses. The MAVRIC sequence employs the Denovo [12] discrete ordinates code and the Monaco [11] fixed-source Monte Carlo radiation transport code.

Variance reduction is achieved by 1) sampling more often from the source regions and energy ranges that make important contributions to dose rate within a geometry region of interest and 2) setting energy- and space-dependent weight windows outside the source regions to obtain uniform particle weight within the dose rate region of interest through particle splitting and/or rouletting. The consistent source and particle transport biasing technique is referred to as consistent adjoint driven importance sampling (CADIS) [13]. Good statistical accuracy of the dose rate estimates within every geometry region outside a spent fuel cask is obtained with the variance reduction method referred to as forward-weighted FW-CADIS [14]. This method requires both forward and adjoint discrete ordinates calculations. The forward response is used as a weighting function to the source for the adjoint discrete ordinates calculation to create larger adjoint source strength in the geometry areas of low dose rate and smaller adjoint source strength in the geometry areas of high dose rate.

MAVRIC enhancements in SCALE 6.2 include availability of continuous-energy cross section data for the Monte Carlo transport, simplified source term descriptions for models that contain a large number of radiation sources, and new utilities to post-process mesh tally files. The source energy distribution and total source strength for each axial zone of a fuel assembly are directly obtained from standard ORIGEN f71 binary files produced a priori with ORIGAMI. The new MAVRIC post-processing utilities are useful in identifying the

maximum dose rate values and their locations in the mesh tally within a user-defined geometry region of interest, such as thin radial and axial tally regions located at two meters from the cask surface.

### Transportation Cask Models

The initial development of as-loaded transportation cask models is focused on the casks licensed for transportation of UNF canisters currently on the ISFSIs at shut-down reactor sites (e.g., the UMS<sup>®</sup> universal multi-purpose cask system [15] and the NUHOMS<sup>®</sup>-MP187 multi-purpose cask [16]). Cask geometry models are based on cask design information included in the safety analysis reports for the licensed casks. The geometry models and radiation source characterization for the canister contents are based on canister loading information, fuel assembly characteristics, and other relevant information from the safety analysis reports. A fuel canister may contain standard fuel assemblies, cans with damaged fuel assemblies or consolidated fuel rods, nonfuel components included as part of the UNF assembly, and empty fuel tube locations.

The cask models for dose rate analyses are very detailed including pin by pin representations of the fuel assemblies and assembly-specific radiation source terms in the active fuel as well as in assembly hardware regions and other non-fuel components that may be placed within assembly guide tubes (e.g., control elements or burnable absorber rods). Fuel assembly hardware regions are represented as cuboids with homogeneous material. Consolidated fuel rods are represented as a homogenous mixture within a volume delimited by the fuel rod length and fuel can inner diameter. Damaged fuel contains fuel sources in the assembly hardware regions. The cask models include conservative approximations for input parameters with missing data. For example, the locations of fixed burnable absorber rods within a Combustion Engineering (CE) fuel assembly are not typically available. The absorber rods were neglected in the description of a CE fuel assembly, and fuel pellet density was based on assembly  $\text{UO}_2$  mass.

The method specified for variance reduction is FW-CADIS. The models include air regions outside the cask and specify a cylindrical tally mesh for dose rate calculations. Variation of dose rate with decay time is assessed from multiple MAVRIC calculations using time-dependent radiation source terms.

### MAVRIC Templates

TemplateEngine is a string substitution program employed in UNF-ST&DARDS to construct input files for the safety analysis codes. This method requires development of input file templates specific to each safety analysis code and a JavaScript Object Notation (JSON)

object, which is an unordered collection of name/pair values. The input file templates may contain raw text, attributes to be replaced, and a series of instructions for importing sub templates dedicated to specific calculations (e.g., calculation of fuel composition based on fuel initial enrichment). The data incorporated within UNF-ST&DARDS are organized in relational structured query language (SQL) data tables within a MySQL database. The data tables define unique identifiers for the data within the database, which are consistently used throughout the UNF-ST&DARDS template repository. The Java TemplateEngine executable conducts sub template imports and attribute replacement based on the JSON object, thereby generating a complete input file.

For dose rate calculations, the JSON object provides identifiers for the parameters used in the MAVRIC input templates that characterize assembly type, canister contents, cask design, etc., and the values associated with those parameters. The MAVRIC root template outlines the structure of a MAVRIC input file. Development of each data block is performed by sub templates with specific functions, which offers flexibility in developing cask-specific MAVRIC input files for the variety of canister contents. A sub template is either generally applicable to any MAVRIC input (e.g., the response function for dose rate calculations) or specific to a fuel assembly (e.g., fuel pin array description), cask design (e.g., fuel basket geometry description), and analysis condition (e.g., overpack geometry description for hypothetical accident conditions).

## RESULTS

A sample problem is used to illustrate the types of dose rate calculation results that UNF-ST&DARDS will produce for transportation casks loaded with fuel canisters currently in dry storage at reactor sites. The evaluated cask complies with applicable 10 CFR 71 dose rate limits for exclusive shipment. The analyzed canister contains both high- and low-burnup intact fuel assemblies, 15 assemblies of which have burnup values between 44.0 and 45.5 MWd/MTU. The minimum cool time relative to fuel discharge for the shipment of this canister is 9 years and corresponds to year 2004. This was based on the fuel assembly with the highest burnup within the canister and the characteristics of approved contents provided in the U.S. NRC certificate of compliance issued for the evaluated cask. Calculation results are provided as a function of decay time (e.g., for years 2004, 2015, and 2021).

The MAVRIC calculation results include a detailed dose rate map for the air regions surrounding a cask and summary tables providing the maximum dose rate values within the spatial regions of interest for dose rate assessments. The color-coded dose rate map illustrating dose rate variation as a function of cylindrical mesh

coordinates is presented in Fig. 1. Time-dependent maximum dose rate values at the cask external surfaces and at 2 meters from the outer surfaces are illustrated in Figs. 2 and 3, respectively, for normal conditions of transport. For year 2021, the calculated maximum dose rate at 2 meters from the cask surface was 2.9 mrem/h, which is approximately three times lower than the regulatory dose rate limit of 10 mrem/h for this location.

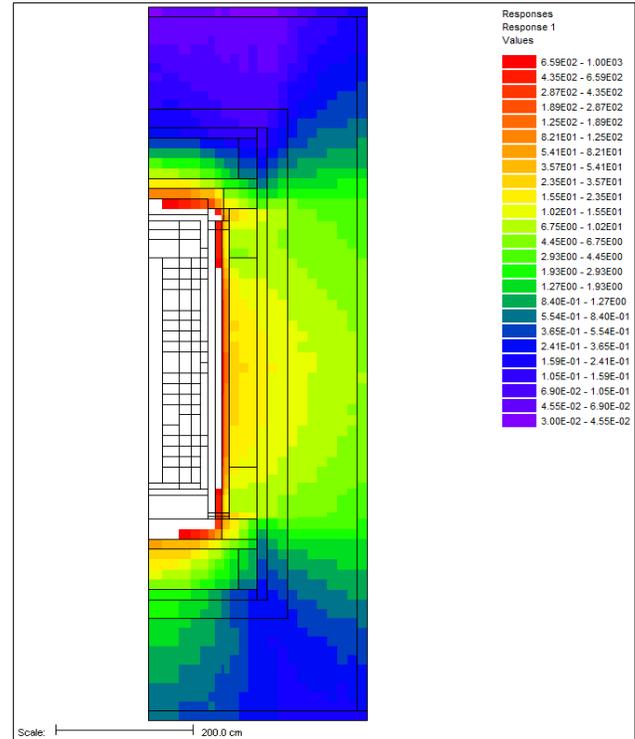


Fig. 1. Example of color-coded dose rate map and dose rate values in mrem/h.

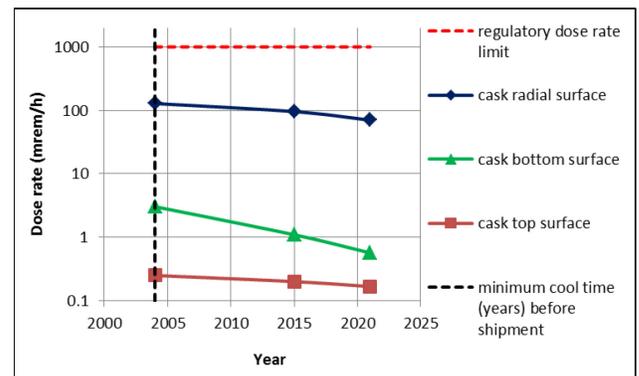


Fig. 2. Variation with time of the maximum dose rate at cask external surfaces.

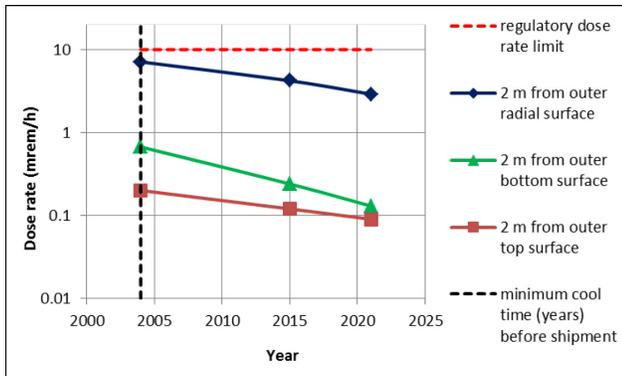


Fig. 3. Variation with time of the dose rate at 2 meters from the cask external surfaces.

## SUMMARY

UNF-ST&DARDS capabilities have been recently expanded to include automated radiation source term calculations and dose rate analyses of as-loaded transportation casks. This unique capability determines realistic dose rate values based on actual canister UNF contents that may be used in radiation dose assessments to demonstrate safe transportation of commercial UNF.

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