Dose Rate Analysis Capability for Actual Spent Fuel Transportation Cask Contents

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ABSTRACT

The approved contents for US Nuclear Regulatory Commission licensed spent nuclear fuel transportation casks are typically based on bounding used nuclear fuel (UNF) characteristics. However, the actual contents of the UNF casks when shipped can be expected to be considerably heterogeneous in terms of fuel assembly burnup, initial enrichment, decay time, cladding integrity, etc. The Used Nuclear Fuel Storage, Transportation & Disposal Analysis Resource and Data System (UNF-ST&DARDS) is an integrated data and analysis system that facilitates automated cask-specific analyses based on actual characteristics of the as-loaded UNF. The UNF-ST&DARDS analysis capabilities have recently been expanded to include dose rate analysis of as-loaded transportation packages. Realistic dose rate values based on actual cask contents can be used to support development of packaging operations procedures, evaluation of radiation-related transportation risks, and communication with stakeholders. This paper[1] describes the UNF-ST&DARDS dose rate analysis methodology based on actual UNF cask contents and presents sample dose rate calculation results. The dose rate was calculated in the air regions external to the transportation package for both normal conditions of transport and hypothetical accident conditions and then compared against the 10 CFR 71 dose rate limits. For the cask system evaluated, the limiting dose rate with respect to the 10 CFR 71 requirements is the dose rate at 2 m from the cask radial surface for normal conditions of transport (i.e., 10 mrem/h). The calculated dose rate values corresponding to year 2025 were between 4 and 12 times lower than the regulatory dose rate limit of 10 mrem/h.

INTRODUCTION

The 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. UNF-ST&DARDS provides a unified domestic UNF system database and associated key analysis capabilities (e.g., SCALE [2]) to support numerous Department of Energy (DOE) waste management and fuel cycle–related objectives, as well as the foundation for tracking UNF from reactor power production through ultimate disposition. UNF-ST&DARDS is designed to provide nuclear safety related 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, etc.). The cask-specific safety evaluations allow quantification of realistic safety margins and conditions and provide the technical basis for addressing aging-related uncertainties that might arise from extended storage and subsequent transportation.

The current UNF-ST&DARDS database includes UNF design characteristics of commercial nuclear fuel assemblies obtained from a variety of open literature sources, discharge inventories through December 31, 2002 [3], and storage data from US commercial nuclear reactors. Additional UNF discharge inventory data through the year 2013 [4] is being assembled to update the UNF discharge inventories. The UNF-ST&DARDS database is expandable and will include new UNF data that become available in the future.

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. The initial focus of UNF-ST&DARDS development has been criticality and thermal analyses, but UNF-ST&DARDS capabilities are being expanded to include dose rate analyses of UNF transportation packages. Realistic dose rate values based on actual contents can be used in place of bounding dose rate values to support development of packaging operations procedures, evaluation of radiation-related transportation risks, and communication with stakeholders. Variation of dose rate with decay time is determined from multiple dose rate calculations using time-dependent radiation source terms.

SCALE 6.2 SOURCE TERM AND DOSE RATE CALCULATION CAPABILITIES

Generation of the radiation source terms associated with individual UNF assemblies and the dose rate analyses of as-loaded transportation packages are facilitated by new code features and enhancements available in the pre-release version 6.2 of the SCALE code system. The specific SCALE codes/calculation sequences employed in radiation source term and dose rate calculations are ORIGAMI and MAVRIC, respectively, which are described in this section.

ORIGAMI Source Term Calculations

Assembly-specific radiation source terms are generated with ORIGAMI (*ORIG*en *A*sse*M*bly *I*sotopics), which 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 variations. This code performs fast ORIGEN irradiation and decay calculations using pre-generated fuel assembly-specific ORIGEN-ARP cross-section libraries [5]. The pressurized water reactor (PWR) and boiling water reactor fuel discharge inventories are categorized into 19 assembly classes that are further subdivided by assembly type for a total of 134 individual fuel types [3]. 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. UNF-ST&DARDS contains pre-generated ORIGEN-ARP cross-section libraries for the representative fuel assemblies within the fuel classes. The ORIGEN-ARP libraries have been generated with depletion input parameters that are conservative from a criticality standpoint (e.g., lowest moderator density expected during normal operation and fuel exposure to burnable absorber rods/control blades) [6]. These modeling parameters are also conservative with respect to dose rate because they increase the concentrations of the nuclides with important contributions to the neutron (e.g., ^{244}Cm) and gamma (e.g., ^{154}Eu and ^{60}Co) dose rates between 5 to 50 years after fuel discharge (i.e., higher radiation source intensities) [7,8].

Assembly-specific radiation source terms are generated for the active fuel as well as the assembly hardware regions. 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 activation sources in assembly upper, gas plenum, and lower hardware regions. Additional activation sources are specified for nonfuel radioactive components that might be placed within fuel assemblies.

MAVRIC Dose Rate Calculations

MAVRIC [9] is the SCALE shielding calculation sequence using Monte Carlo radiation transport and automated variance reduction capabilities. Variance reduction refers to methods and techniques that significantly increase the efficiency of Monte Carlo radiation transport calculations. These methods use information obtained a priori from approximate adjoint calculations, typically performed with a discrete ordinates code. The adjoint calculation solution identifies the source spatial regions and energy ranges that make important contributions to dose rate within a geometry region of interest. This information is used in a non-analog Monte Carlo radiation transport to sample particles more often from the radiation source regions and energy ranges that make important contributions to dose rate within the geometry region of interest. To avoid biasing the dose rate calculation results based on the use of altered radiation sources, biasing parameters are applied to the particles in the Monte Carlo radiation transport in such a way to conserve the actual dose rate values [10].

The MAVRIC sequence employs the Denovo [11] discrete ordinates code and the Monaco [9] fixed-source Monte Carlo radiation transport code. A variance reduction method referred to as consistent adjoint driven importance sampling (CADIS) [12] is used within MAVRIC to obtain dose rate estimates with good statistical accuracy within a specific geometry region (e.g., the air region outside the cask near a trunnion recess). Dose rate estimates with good statistical accuracy within every geometry region outside a transport package are obtained with a variance reduction method referred to as forward-weighted CADIS (FW-CADIS) [13]. 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. FW-CADIS is the variance reduction method used in the MAVRIC calculations for as-loaded transport packages.

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 ft71 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 within geometry regions of interest, such as the 10 CFR 71.47 and 71.51 [14] dose rate locations.

Transportation Package Models

For analysis purposes, the initial development of as-loaded transport package models is focused on the casks licensed for transportation of UNF canisters currently at the independent spent fuel storage installations at shutdown reactor sites (e.g., the UMS® universal multipurpose cask system [15] and the NUHOMS®-MP187 multipurpose cask [16]). A UNF canister can 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 package geometry models use cask design information relevant to dose rate that is provided in the safety analysis reports for the licensed casks. Design details that can cause radiation streaming paths, such as drain tubes and trunnion penetrations through the cask neutron shield, are also described within the calculation models. The geometry models and radiation source characterization of the fuel assemblies and nonfuel components within a UNF canister are based on canister loading information, fuel assembly design characteristics, and other relevant information from the safety analysis reports.

The fuel assembly models describe individual fuel pins and radiation sources within the active fuel as well as assembly hardware regions and other nonfuel components that might be placed within a fuel assembly (e.g., burnable absorber rods or control elements). Fuel assembly hardware regions were represented as cuboids with homogeneous material and radiation sources. Consolidated fuel rods were represented as a homogenous mixture within a volume delimited by the fuel rod length and fuel can inner diameter. The model for a damaged fuel assembly is based on the assumption that fuel fragments and particulates are present within assembly hardware regions, thereby increasing the dose rate at the upper and lower external cask regions. In addition, the package 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 UO_2 mass. The impact limiters are included in the package models for normal conditions of transport. For hypothetical accident conditions, the package model is consistent with the model described in the safety analysis report, which typically does not include the neutron shield, neutron shield casing, and the impact limiters. The package models include air regions outside the cask and specify a cylindrical tally mesh for dose rate calculations.

MAVRIC Input File Templates

UNF-ST&DARDS automatically generates input files for calculating transportation package external dose rates with SCALE/MAVRIC based on specific canister UNF contents. Input file generation for the safety analysis codes within UNF-ST&DARDS is accomplished through a string substitution program. This method requires development of input file templates specific to each safety analysis code and a JavaScript Object Notation (JSON) object. The root template typically contains 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 JSON object is a text file that describes input parameters (e.g., assembly design and irradiation history) and associated values for use within the input file templates. The Java TemplateEngine executable conducts sub-template imports and attribute replacement based on the JSON specifications, thereby generating a complete input file.

The MAVRIC root template outlines the structure of a complete MAVRIC input file. Development of each data block is performed by sub-templates with specific functions, which offers flexibility in developing cask-specific input files for the variety of canister UNF 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), canister design (e.g., fuel basket geometry description), and analysis condition (e.g., package geometry description for normal conditions of transport or hypothetical accident conditions).

Development of the JSON object for each UNF canister is performed automatically using the data within the UNF-ST&DARDS database. 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.

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 for normal conditions of transport (NCT) and hypothetical accident conditions (HAC), respectively. Typically, US Nuclear Regulatory Commission (NRC)-licensed spent fuel casks meet 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 dose rate on the external surface of the package must be < 1,000 mrem/h.
2. The dose rate at any point on the outer surface of the vehicle, including the top and underside of the vehicle, must be < 200 mrem/h.
3. The dose rate at any point 2 m from the outer lateral surfaces of the vehicle (excluding the top and underside of the vehicle) must be < 10 mrem/h.
4. Any normally occupied position of the vehicle must be < 2 mrem/h.

The 10 CFR 71.51(a)(2) dose rate limit at 1 m from a cask undergoing the drop and fire tests specified in 10 CFR 71.73 (HAC) is 1,000 mrem/h.

RESULTS

This paper presents UNF-ST&DARDS dose rate calculation results as a function of decay time and actual UNF canister contents for two NRC-licensed transportation packages containing UNF fuel assemblies and nonfuel components from a decommissioned US commercial nuclear reactor. The transportation package complies with the previously described 10 CFR 71.47(b) and 71.51(a)(2) dose rate limits for exclusive shipment in a closed conveyance. Radiation dose rate varies as a function of many UNF characteristics, the most significant of which are UNF assembly burnup, initial enrichment, and decay time. On the basis of the safety analysis report of a package design, the approved fuel assembly types, initial enrichment, burnup, and decay time are described in the NRC Certificate of Compliance (CoC) issued for the transportation package. In the determination of the package contents, the vendor used a conservative safety analysis approach including (1) a design basis assembly that is bounding with respect to dose rate, (2) both three- and one-dimensional radiation transport calculations, (3) and 2 m from the cask radial surface in place of 2 m from the lateral surface of the vehicle for the 10 mrem/h dose rate limit. The most limiting external dose rate for this transportation package is the dose rate at 2 m from the cask radial surface.

Hence, the radiation dose rate at 2 m from the cask radial surface approaches 10 mrem/h for the combinations of fuel burnup, initial enrichment, and decay time values tabulated in the CoC. With increasing decay time, the cask external dose rate could become significantly lower than the 10 CFR 71 dose rate limits (e.g., more than one order of magnitude), which is demonstrated in this paper for two UNF canisters identified as canisters A and B containing 24 PWR fuel assemblies and nonfuel components from the decommissioned US commercial nuclear reactor.

Canister A contains both low- and high-burnup (i.e., burnup lower and greater than 45 GWd/MTU, respectively) intact PWR assemblies. The radiation dose rate associated with canister A is the highest among the canisters containing UNF assemblies and nonfuel components from the decommissioned reactor. Canister B contains PWR assemblies with average burnup values between approximately 11 and 45 GWd/MTU (i.e., low-burnup UNF), which is representative of the majority of UNF canisters at the decommissioned reactor site. In addition, some of the assemblies in canister B contain nonfuel components with cobalt activation sources that were placed within the assembly guide tubes. The burnup ranges of the individual UNF assemblies within canisters A and B are illustrated in Figure 1. Based on the approved package contents and considering the maximum assembly burnup within the two canisters, the minimum decay times required before the shipment of canisters A and B are 14 and 9 years, respectively, after fuel discharge. This corresponds to years 2004 and 2001 for canisters A and B, respectively. Hence, the radiation dose rate at 2 m from the outer lateral surfaces of the vehicle would have less than 10 mrem/h if canisters B and A were shipped in 2001 and 2004, respectively. Calculation results are further provided as a function of decay time for years 2015, 2025, and 2050, as well as for years 2001 and 2004 for verification purposes. Calculation results are shown for the years 2025 and 2050 because the future management of UNF from civilian nuclear power generation considers the availability of a larger interim storage facility by 2025 and a geologic repository by 2048 [17].

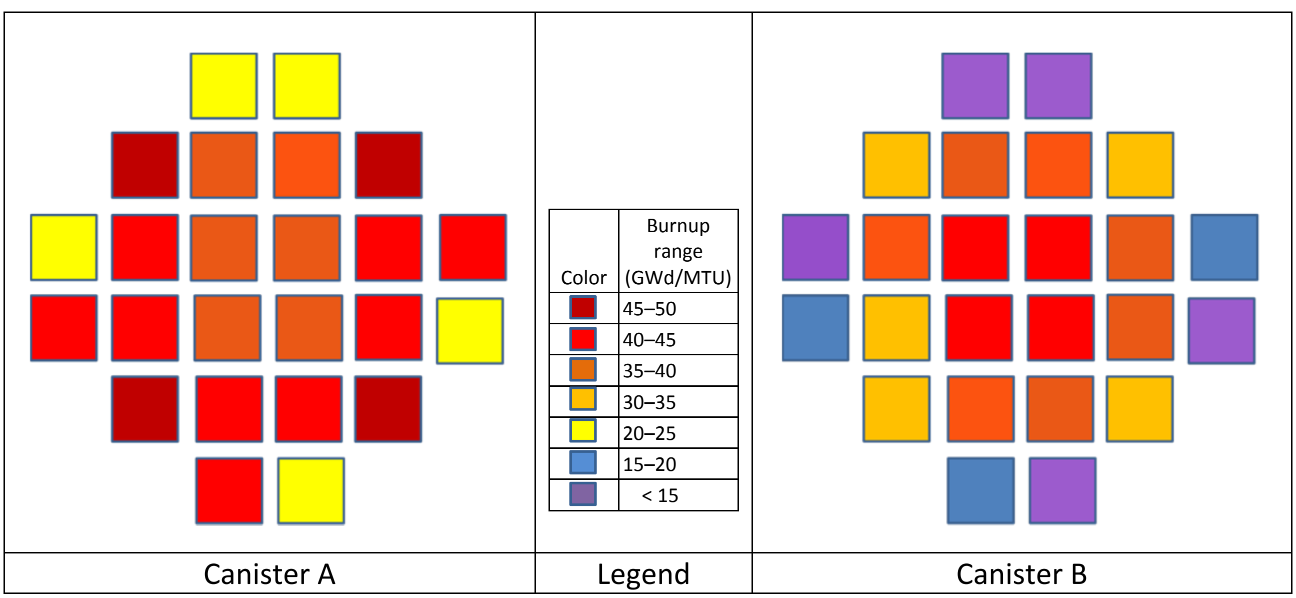
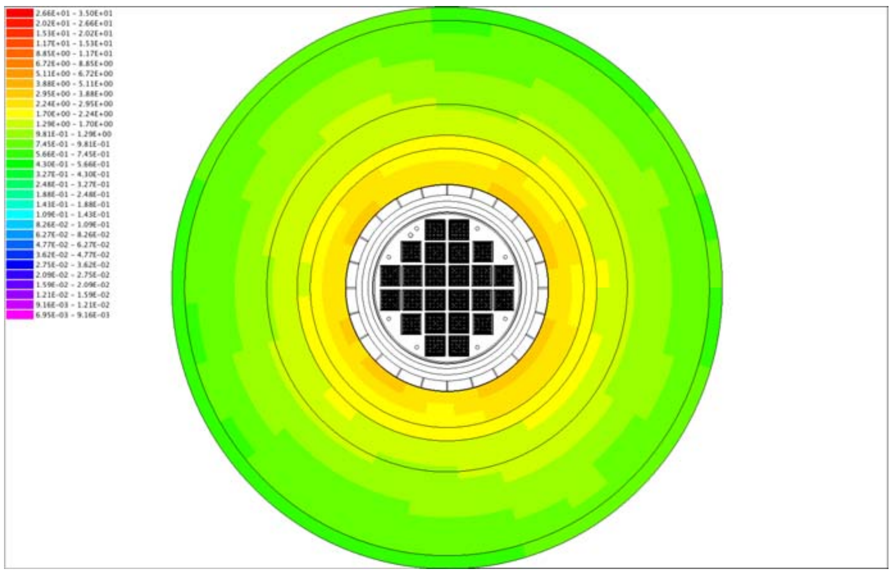
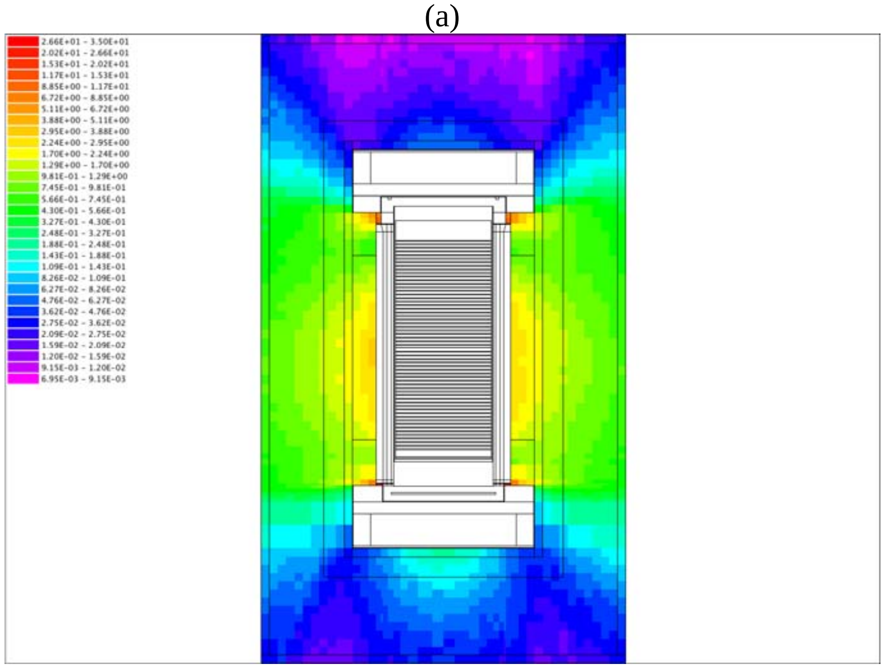


Figure 1. Comparison of UNF assembly average burnup within canisters A and B.

The UNF-ST&DARDS dose rate calculation results include a detailed spatial dose rate distribution based on a specified cylindrical tally mesh and summary tables providing the maximum dose rate values within the spatial regions of interest for dose rate assessments (i.e., the locations specified in 10 CFR 71.47 and 71.51). The statistical uncertainty associated with the dose rate values is below 10% for the vast majority of spatial voxels. Cross sections through the horizontal and vertical mid planes of the transport package illustrating the dose rate within the air regions external to the cask are presented in Figure 2 (a) and (b), respectively. The dose rate is illustrated for the package containing canister B under normal conditions of transport as of January 1, 2025. The two-dimensional dose rate plots show that the dose rate external to the package varies by approximately one order of magnitude in the radial direction and by more than three orders of magnitude in the axial direction.



(a)



(b)

Figure 2. Horizontal (a) and vertical (b) cross sections of the transportation package containing canister B, illustrating the dose rate (mrem/h) within the air regions external to the cask.

The variation with time of the maximum NCT dose rates at the cask external radial surface and at 2 m from the cask radial surface are presented in Figure 3 and Figure 4, respectively, for both canisters. Figure 5 presents the variation with time of the maximum HAC dose rates for both canisters. The maximum dose rates at locations of interest for dose rate assessments vary as a function of canister contents and significantly decrease with increasing decay time. The maximum radial dose rate values for canisters A and B at the locations of interest for dose rate assessments exhibit different variations as a function of time as a result of the decay of the cobalt activation source ($T_{1/2}$ = 5.271 years) associated with the nonfuel components within canister B. The nonfuel activated components significantly contribute to the dose rate at short decay time but have much lower contributions with increasing decay time.

For both canisters, the maximum dose rate at the external surface of the cask is significantly lower than the 1000 mrem/h limit. The most limiting dose rate for this package is the dose rate at 2 m from the cask radial surface, which for both canisters is slightly lower than 10 mrem/h at the minimum required decay time before shipment. For canister A, the 2025 maximum NCT dose rate is 2.6 mrem/h at 2 m from the cask radial surface. For canister B (i.e., typical canister contents), the 2025 maximum NCT dose rate is 0.8 mrem/h at 2 m from the cask radial surface, which is approximately 12 times lower than the regulatory dose rate limit of 10 mrem/h for this location. The 2025 maximum HAC dose rate values for canisters A and B are approximately 6 and 19 times lower, respectively, than the 1000 mrem/h limit. The dose rate values further decrease with increasing decay time. These calculation results demonstrate (1) the utility of the UNF-ST&DARDS methodology for performing realistic dose rate calculations based on the confirmatory calculations for the minimum required decay time before shipment and (2) the dose rates at locations of interest will be significantly lower than the 10 CFR 71 dose rate limits in 2025 and beyond.

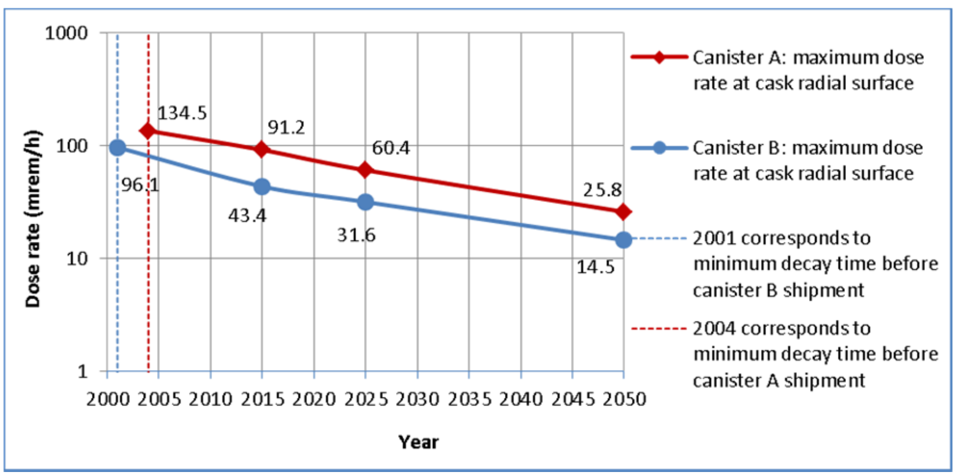


Figure 3. Maximum NCT dose rate at the cask external surface as a function of time.

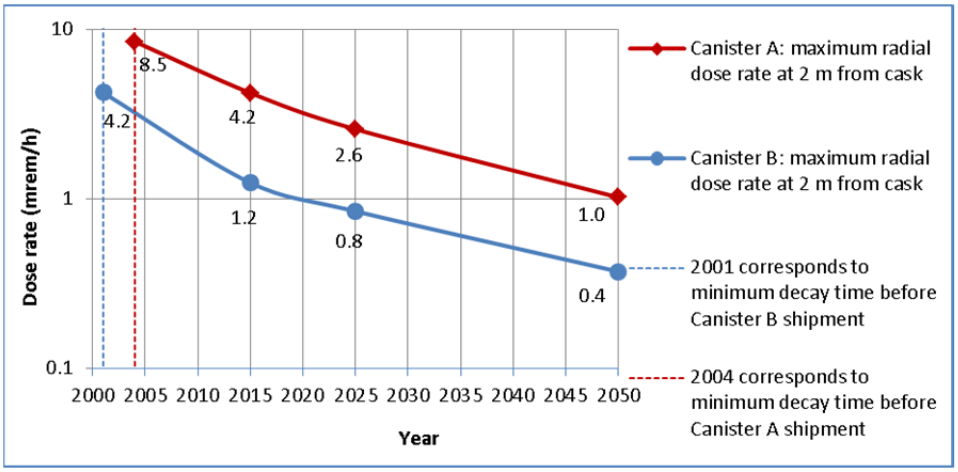


Figure 4. Maximum NCT dose rate at 2 m from the cask radial surface as a function of time.

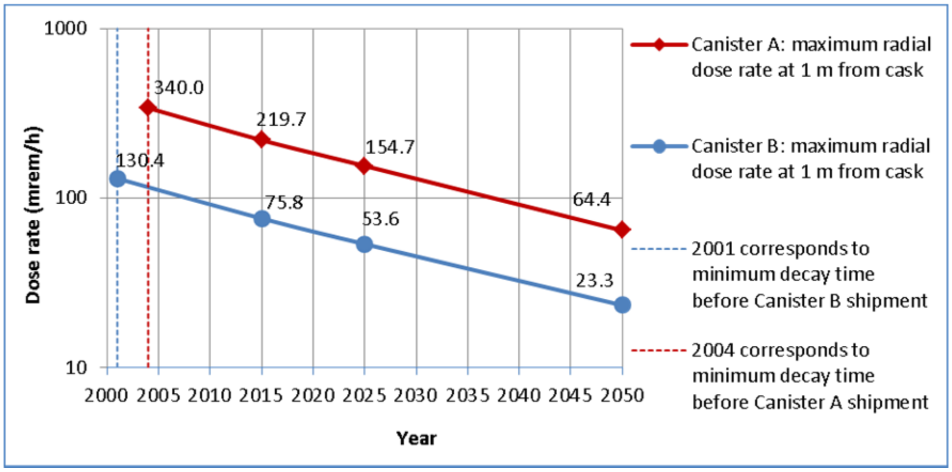


Figure 5. Maximum HAC dose rate as a function of time.

SUMMARY

UNF-ST&DARDS capabilities have recently been expanded to include automated radiation source term calculations and dose rate analyses of as-loaded transport packages using new code features available in the prerelease version 6.2 of the SCALE code system. Depending on the characteristics of the UNF assemblies within a cask (i.e., UNF assembly initial enrichment, burnup, and decay time) at the time of package transportation, the dose rate values within the air regions outside the package could be more than one order of magnitude lower than the 10 CFR 71 regulatory dose rate limits. For the cask system evaluated, the limiting dose rates were at 2 m from the cask radial surface. The dose rate values corresponding to year 2025 were between 4 and 12 times lower than the regulatory dose rate limit of 10 mrem/h.

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