Baselining a Spent Nuclear Fuel Cask Shielding Model

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INTRODUCTION

Radiation dose analysis is essential for safe handling, storage, transportation, and disposal of any spent nuclear fuel (SNF) cask. Nuclear facility designers and operators not only must meet US Nuclear Regulatory Commission (NRC) mandated regulatory dose limits for workers and the general public, but they also must perform evaluations to support design and operational decisions to show that radiation doses are as low as reasonably achievable (ALARA). For SNF casks, shielding calculations are performed primarily to determine dose rates outside a given system. These dose rate calculations are typically conservative due to simplified analysis premises, such as modeling a SNF cask with bounding radiation source terms (e.g., maximum burnup and minimum cooling time). Source terms with bounding parameters such as burnup are also typically evaluated using simplified reactor operating histories to maximize the gamma and neutron sources. While this conservative dose analysis approach is acceptable for designing an SNF cask to provide safety to the public, this conservative analysis approach also creates a variety of operational challenges. For instance, this approach may result in a demand for supplementary shielding during loading, overly complex loading procedures to maintain ALARA, and decades of additional cooling time before SNF can be considered transportable. Thus, reliance upon conservative shielding analyses could conceivably limit when a standalone independent spent fuel storage installation (ISFSI) used for dry storage of SNF can ship SNF offsite and land can return to normal use. Although benchmarking of any shielding analysis software code is essential from the code development perspective, shielding code benchmarking is not typically used to support SNF cask licensing due to the conservative nature of the shielding analysis. Plant operators rely on actual dose measurements to ensure that the system's behavior is as per the design basis dose analysis. The measured dose rates should always be much lower than the calculated dose rates, sometimes by orders of magnitude.

More accurate computation of dose rates can provide additional flexibility for both facility designers and operational planners. Detailed analyses producing realistic dose rates can be used to determine the actual (1) earliest time casks are transportable and (2) better estimates of dose to the public during large-scale campaigns for transportation of SNF from ISFSIs to an interim storage location and/or to a disposal facility.

The Used Nuclear Fuel-Storage, Transportation & Disposal Analysis Resource and Data System (UNF-ST&DARDS) is being developed at the Oak Ridge National Laboratory (ORNL) as a foundational resource for the US Department of Energy (DOE) Office of Nuclear Energy to streamline computational analysis, thereby facilitating time-dependent characterization of SNF and related systems [1]. UNF-ST&DARDS automates prediction of the state of SNF far into the future using a range of analyses, including isotopics, shielding, criticality and thermal analyses. The software can be adapted to work with a variety of nuclear analysis packages. UNF-ST&DARDS combines SNF assembly data, cask geometries, and analysis tools such as the SCALE code system to automatically create and evaluate models of SNF casks. UNF-ST&DARDS predicts the activity of each individual assembly in each individual cask. It then uses this information in shielding analysis, yielding more accurate dose rate predictions compared to the those obtained using the traditional bounding analysis.

The realistic dose assessment models in UNF-ST&DARDS must be supported by proper validation to ensure public safety. Validation is necessary to gauge accuracy of shielding models and the general assumptions used to develop input files. Moreover, validation is necessary to understand the accuracy of underlying analysis codes used to support accurate dose estimation by UNF-ST&DARDS. The SCALE code package already uses a substantial benchmarking suite. While the computational methods are validated, sometimes using analytical solutions, the actual inputs and assumptions used to create them should be evaluated to gauge accuracy for the intended application. Numerous radiation transport benchmarks are available for code validation as part of the Shielding Integral Benchmark Archive and Database (SINBAD). Unfortunately, the SINBAD database does not contain SNF cask shielding benchmarks. Additionally, there are only a handful of cask benchmarks published in the open literature. Foremost among these is a 1995 Electric Power Research Institute (EPRI) report listing several benchmark cases with measurements at numerous geometric locations, along with detailed problem specifications [2]. From the same period, work by Jones and Thomas reports dose rates at three locations around a NUHOMS-24P cask [3]. After 1995, a few additional benchmarking cases were developed. These include a TN-12 SNF cask, with dose rates at three locations [4], and

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work by Asami et al. to model cask lifting points, which represent irregularities in the cask surface [5]. In 2008, the Korea Atomic Energy Research Institute published dose rates at six measurement locations along a KN-12 cask [6]. A substantial amount of potential benchmarking information is held by utility sites in the form of radiation surveys, and cask manufacturers in the form of proprietary design drawings, however this is unavailable for general use. To evaluate assumptions in the input, a precise, detailed-problem specification is desirable, and first-hand observation of measurements is even better.

An opportunity to develop additional benchmarking data for SNF casks arose when a NAC International legal weight truck cask (NAC-LWT) containing 25 high burnup (> 45 GWd/MTU) SNF rods arrived at ORNL in 2016. Shortly after its arrival, gamma dose rates were obtained at various locations along the cask's surface using survey equipment on hand. Subsequently, a shielding model was developed to correspond to the cask geometry and contents and was compared to those measurements.

The 25 rods were part of the High Burnup Spent Fuel Data Project [7] to better characterize phenomena occurring during dry storage of high burnup SNF, including the extent and implications of hydride reorientation, a phenomenon that can change cladding mechanical properties. To create baseline measurements for the data project, the 25 rods were delivered to ORNL for post irradiation examination.

A loaded NAC-LWT with impact limiters has a mass of 23.5 metric tons and measures approximately 1.1 m in diameter and 5.1 m in length. The cask can transport a variety of SNF payloads, including an entire boiling water reactor (BWR) assembly, a pressurized water reactor (PWR) assembly, a range of research reactor fuel, and individual fuel rods [8]. As shown in Fig. 1, the side of the cask consists of a lead gamma shield poured into an XM-19 high-strength stainless steel shell and then carefully cooled to avoid creation of voids. A second layer of shielding consists of a borated ethylene glycol solution for neutron absorption enclosed by a type 304 stainless steel shell. A second type 304 stainless steel shell provides overflow space for thermal expansion of the neutron absorber fluid [8]. Stainless steel plates support the shells containing the ethylene glycol solution. The cask is designed to prevent bubbles from entering the neutron shield area during non-accident conditions. The neutron emission rate is substantially lower than the gamma emission rate from SNF, so loss of neutron shields for SNF casks is typically assumed in accident analyses for solid neutron shields as well.

The NAC-LWT received at ORNL was equipped with an aluminum PWR basket that can hold a PWR assembly or a rod canister. In this case, the 25 high burnup rods were transported in a rod canister. An aluminum PWR insert was used to fill the void between the PWR basket and the rod canister [8].

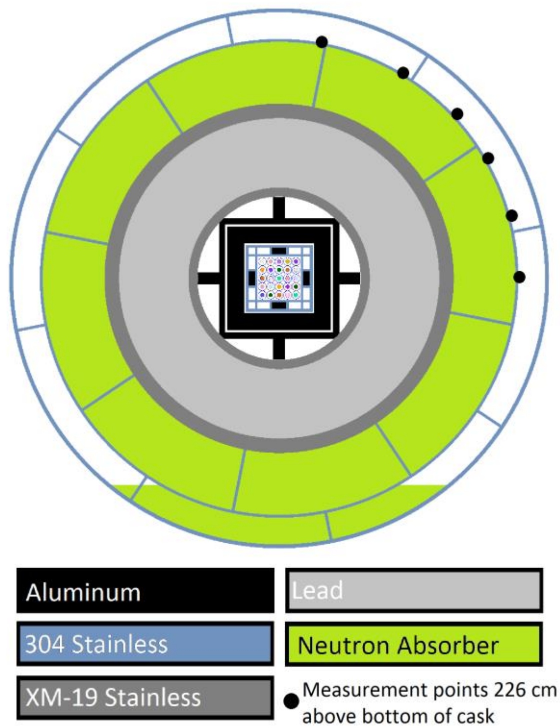

Fig. 1. NAC-LWT radial slice.

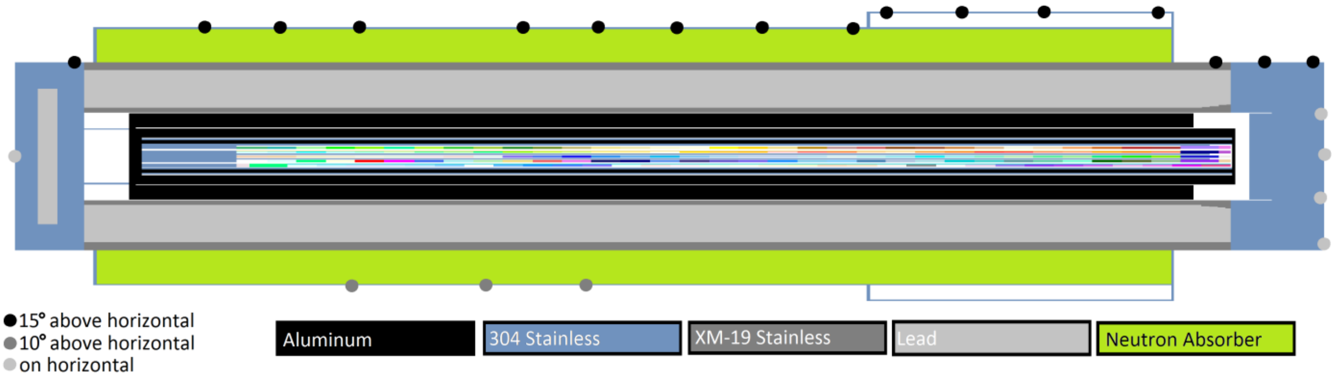

Fig. 2. NAC-LWT axial slice with measurement points.

METHOD

Measurements

Gamma dose rate measurements were taken primarily on the side wall along two axially oriented lines (Fig. 2) and an azimuthally oriented line (Fig. 1) using readily available equipment. The azimuthal line was in a location without the neutron shield overflow tank, as shown in Fig. 1. The first set of measurements was performed with an Eberline RO-20 ion chamber. The second and third sets included additional measurement locations, and they were performed with a Radeye B20-ER Geiger counter. For both instruments, dose rates were close to the detection limit. For the Radeye, this meant that the righthand digits varied with time. Thus, two readings were taken with the Radeye for every measurement location. Unfortunately, neutron dose was not expected to be detectable with the available instrumentation.

Computer Codes and Model

Shielding analysis was performed using the MAVRIC code, that is part of the SCALE code system [9]. Cask shielding using Monte Carlo methods can be a challenging problem because the vast majority of particles are absorbed by the cask. If one in a million particles survives, to obtain a sample of a million particles outside the cask, a trillion particles must be simulated. Modeling casks in any detail can thus become computationally prohibitive. To address this challenge, SCALE uses the FW-CADIS methodology. FW-CADIS uses approximate forward and adjoint discrete ordinates flux calculations to inform Monte Carlo sampling in the shielding model, and thus accelerate solution convergence. The adjoint computation is used to reduce sampling of source particles that contribute almost nothing to the dose rates of interest. For example, photons with an energy in the 1-10 keV range may be a million times less likely to survive through some shield than particles in the 100keV-1MeV range, but the low energy photons are produced ten times as often by the radiation source. An approximate adjoint flux calculation would show that these particles contribute almost nothing to the dose rate, so they are sampled less often. This would result in a 90% reduction in computation time in this example case. The forward computation is used to ensure that every tally region outside the cask has a Monte Carlo uncertainty on the same order of magnitude.

The flux-to-dose conversion factors currently used in the UNF-ST&DARDS shielding models [10] were used to convert the gamma flux at each point to a dose rate.. They are the only flux-to-dose conversion factors specifically deemed acceptable in the standard review plans for storage[11] and transportation [12] of SNF. ENDF/B-VII.0 continuous energy cross sections were used.

Radiation sources for the shielding model were computed using TRITON and ORIGEN, with proprietary data regarding pin dimensions and burnup for assemblies. More generic information about the 25 rods is available in the *High Burnup Spent Fuel Data Project Sister Rod Test Plan Overview* [7]. This process is shown in Fig. 3. For a given assembly geometry, TRITON computes the probability that an isotope will be transmuted as a function of time and burnup. ORIGEN combines these probabilities with assembly power history to compute the isotopes present in the SNF and the amounts and types of radiation that the SNF produces. Both tools are part of the SCALE code package. Radiation sources were computed for 32 axial segments along the active region of each rod, representing assembly-specific axial burnup profiles developed from power maps. The source term for each rod was evaluated using assembly-specific burnup history. Future work will examine the impact of using more approximate data to develop source terms.

Model geometry was based on publicly available licensing drawings [8]. The cask was loaded on a steel frame approximately 1 m above a coated concrete floor. Since measurement points were taken above the horizontal center plane of the cask, the floor and steel frame were not modeled.

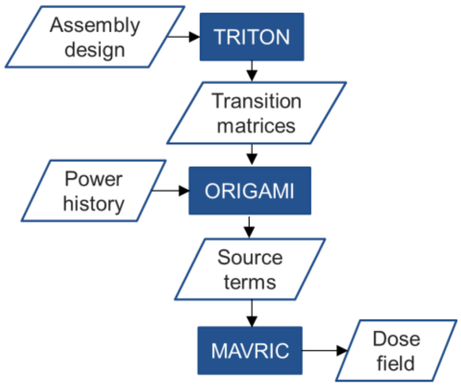

Fig. 3. Computational sequence.

RESULTS

The calculated and measured data are presented in Fig. 4. Calculated dose rates typically fall within 0.1 mR/hr of measured dose rate values. Error bars on calculated values are based on Monte Carlo uncertainty and thus do not reflect all sources of uncertainty. Dose rates were approximately constant along the azimuthal line of measurements taken at 226 cm from the left-most edge of the cask, as shown in Fig. 2.

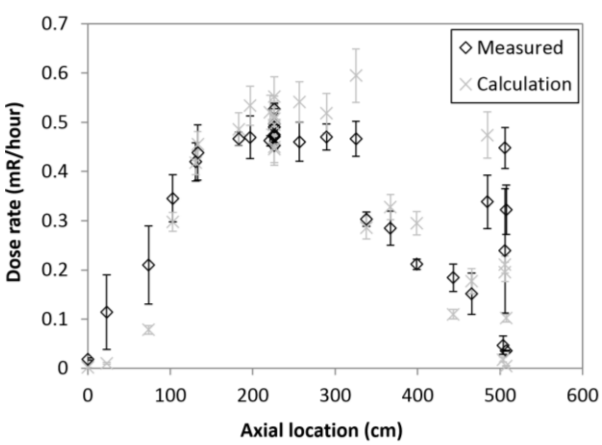

Fig. 4. Measurement vs. calculation; error bars represent one standard deviation; 0 cm is left end of the cask

Calculation vs. measurement ratios are presented in Fig 5. Fifty-eight percent (58%) of the data points fall within 20% of the measured results, with the outliers at the top or bottom of the cask. When considering the possible sources of error in geometry specification, materials data, flux-to-dose conversion function and source term calculation, this is generally deemed acceptable. The cause for the increased discrepancy at the endpoints of the cask is currently unclear. A possibility is under-predicted sources at the extreme ends of the rods. Also, at the endpoints, there is less lead and more iron in the structure, so it could be a material definition issue. In addition, some locations at the top of the cask correspond to bolt recesses which were not modeled in full detail. Regardless, the correspondence between measured and computed values is quite satisfactory at this juncture.

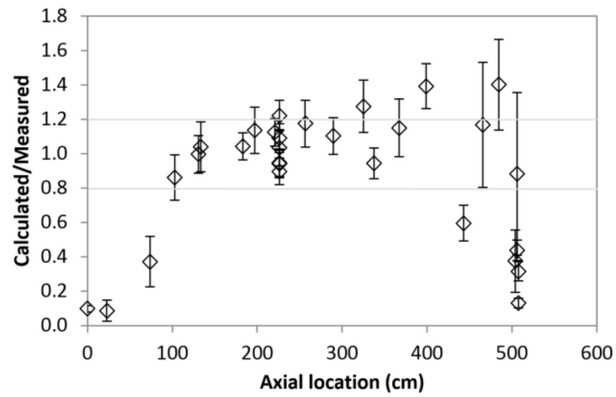

Fig. 5. Calculated to experimentally measured dose rate values with one standard deviation error bars.

CONCLUSION

A benchmarking case for gamma dose rates at the surface of an SNF transportation cask loaded with high burnup SNF has been summarized. Dose rates computed with the MAVRIC radiation transport sequence matched measurements, verifying that the SCALE package employed by UNF-ST&DARDS can produce useful results for cask shielding with high burnup fuel. Additionally, it provides a basis to examine sensitivities, gauge uncertainty, and examine possible simplifying assumptions. Sensitivity information is expected to prove especially useful to guide development of shielding models for cask geometries lacking dose rate datasets, focusing attention on quantities that matter most.

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REFERENCES

1. K. BANERJEE et al., “UNF-ST&DARDS: A Unique Tool for Automated Characterization of Spent Nuclear Fuel and Related Systems,” PATRAM, September 18–23, Kobe, Japan (2016).
2. B. L. BROADHEAD et al., *Evaluation of Shielding Analysis Methods in Spent Fuel Cask Environments*, Electric Power Research Institute, EPRI TR-104329s (1995).
3. K. JONES, B. THOMAS, *Dry fuel storage cask shielding benchmarks*, April 30–May 5, Las Vegas, Nevada (1995).

4. S. KITSOS, B. MARTINOTTI, *Shielding Benchmarks Analysis for Transport/Storage Casks*, *J. Nuclear Science and Technology*, v 37, (2000).
5. M. ASAMI, et al., “Application of Dose Evaluation of the MCNP Code for the Spent Fuel Transport Cask,” *Progress in Nuclear Science and Technology*, v2, (2011).
6. I. CHO et al., “Neutron and Gamma Shielding Estimation for the KN-12 Cask,” *J. Nuclear Science and Technology*, Supplement 5, p 54, (2008).
7. B. HANSON et al., *High Burnup Spent Fuel Data Project Sister Rod Test Plan Overview*, Pacific Northwest National Laboratory, FCRD-UFD-2016-000063 (2016).
8. NUCLEAR ASSURANCE CORPORATION, *NAC-LWT Legal Weight Truck Cask System Safety Analysis Report*, Non-Proprietary Version, Revision 44, Atlanta (2015).
9. B. T. REARDEN, et al., *SCALE Code System* ORNL/TM-2005-39, Oak Ridge National Laboratory (2016).
10. AMERICAN NUCLEAR SOCIETY. WORKING GROUP ANS-6.1.1.; AMERICAN NATIONAL STANDARDS INSTITUTE, *American national standard neutron and gamma-ray flux-to-dose-rate factors*, (1977).
11. NUCLEAR REGULATORY COMMISSIION, *Standard Review Plan for Dry Cask Storage Systems*, NUREG-1536 (2010).
12. NUCLEAR REGULATORY COMMISSIION, *Standard Review Plan for Transportation Packages for Spent Nuclear Fuel*, NUREG-1617 (2000).