Streamlining Analysis Capabilities for SNF Management[a] - 15490

Kaushik Banerjee, John M. Scaglione, Robert A. LeFebvre, Georgeta Radulescu, and Kevin R. Robb
ORNL, P.O. Box 2008, MS-6170, Oak Ridge, TN 37831, USA, banerjeek@ornl.gov

ABSTRACT

A comprehensive, integrated data and analysis tool—the Used Nuclear Fuel-Storage, Transportation & Disposal Analysis Resource and Data System (UNF-ST&DARDS) —is being developed for the US DOE Office of Nuclear Energy (DOE-NE) Nuclear Fuels Storage and Transportation Planning Project (NFST). The overarching goal of UNF-ST&DARDS is to provide a comprehensive controlled source of technical data integrated with key analysis capabilities to characterize inputs to the overall US waste management system from reactor power production through ultimate disposition. UNF-ST&DARDS seamlessly integrates a unified SNF system relational database and key analysis capabilities to simplify and automate the performance of accurate and efficient SNF related analyses to support numerous DOE waste management and fuel cycle activities. Hence, UNF-ST&DARDS enables decision making relative to design, safety, and licensing of SNF systems and facilities by providing the best information available. Automated analysis sequences have been deployed to characterize the discharge inventory and perform criticality, thermal, and dose analyses using time-dependent data. UNF-ST&DARDS provides cask-specific as-loaded safety analysis including criticality, thermal and dose evaluations, as well as results visualization.

INTRODUCTION

This paper presents an overview of the nuclear safety analysis capabilities of the Used Nuclear Fuel-Storage, Transportation & Disposal Analysis Resource and Data System (UNF-ST&DARDS) [1]. UNF-ST&DARDS provides a unified SNF system database and seamless integration with key analysis capabilities to support numerous US DOE waste management and fuel cycle–related objectives. UNF-ST&DARDS is the underpinning for tracking SNF from reactor power production through ultimate disposal in a geological repository. The development of UNF-ST&DARDS has been a collaborative effort among multiple national laboratories and several nuclear utilities. Automated inventory, criticality, thermal and shielding analysis capabilities of as-loaded casks have been developed within UNF-ST&DARDS. The SNF system database has been expanded to encompass system analysis data that allow UNF-ST&DARDS to provide waste management system data to various system analysis tools. UNF-ST&DARDS also provides the capability to visualize data and analysis results. Current UNF-ST&DARDS development efforts include as-loaded containment analysis and SNF performance assessment in terms of internal rod pressure.

The overarching motivation of UNF-ST&DARDS is to enable informed waste management system decision making with the most accurate information available. SNF management includes but is not limited to (1) operational safety (e.g., worker dose), (2) storage facility design, (3) transportation planning, (4) management of aging structures, systems, and components important to safety, (5) addressing uncertainties associated with high-burnup fuel while in storage and after transportation, and (6) the transfer of knowledge across multiple generations. For reliable analysis of large, complex systems, a dependable source of information is needed with the capability to dynamically update system

[a]Notice: This manuscript has been authored by UT-Battelle, LLC, under contract DE-AC05-00OR22725 with the US Department of Energy. The United States Government retains and the publisher, by accepting the article for publication, acknowledges that the United States Government retains a nonexclusive, paid-up, irrevocable, worldwide license to publish or reproduce the published form of this manuscript, or allow others to do so, for United States Government purposes.

parameters based on new or changing information. UNF-ST&DARDS is designed to provide an accurate, dynamic SNF data repository and to streamline analyses for the various waste management components.

SNF cask systems are designed and evaluated for approved contents as defined in the Certificate of Compliance (COC). The approved contents specifications provide bounding (enveloping) fuel characteristics such as fuel type, fuel dimension, initial enrichment, and discharge burnup. The bounding fuel characteristics for a system are developed to establish upper limits of the cask's safety parameters (e.g., neutron multiplication factor [k_{eff}], total decay heat, and dose rates). There is a wide range of discharged SNF due to wide variations in SNF assembly burnup values, initial enrichments, and discharge dates. Therefore, cask systems are loaded with assemblies that satisfy the bounding fuel characteristics as defined in the COC with some amount of unquantified and uncredited safety margin[a]. A unique, unprecedented capability within UNF-ST&DARDS is the automated evaluation of actual as-loaded, cask systems. This capability enables quantification of realistic safety margins and conditions of actual as-loaded SNF in existing dry cask storage systems. The previously uncredited margins could potentially be credited in the future to offset uncertainties associated with extended storage and high-burnup fuel issues. Without considering the as-loaded configurations, overly conservative analyses based on bounding generic assumptions could lead to prematurely implementing compensatory measures within the waste management system. Some of these compensatory measures may include repackaging already packaged fuel, canning a significant fraction of SNF before storage, or requiring additional criticality control before transport and disposal (e.g., inserting control rod assemblies). For example, currently used thermal analysis methods based on bounding assumptions typically overpredict the time-dependent temperature profiles by an amount not readily quantifiable in general, and that would vary widely based on the individual characteristics of a given cask. This would hinder efforts to accurately predict the potential for low-temperature phenomena (e.g., when the fuel passes through the ductile-to-brittle transition temperature) and would tend to overpredict the probability of fuel rods that may fail during extended storage and/or transportation because of thermally driven mechanical property changes (e.g., hydride reorientation) [2]. Cask-specific as-loaded analyses performed within UNF-ST&DARDS provide realistic time-dependent temperature profiles that can be used to accurately predict temperature dependent phenomena.

UNF-ST&DARDS ARCHITECTURE

A graphical representation of UNF-ST&DARDS is illustrated in Fig. 1. UNF-ST&DARDS consists of six main elements: (1) a user interface (graphical as well as command line), (2) the Unified Database, (3) a Unified Database software development kit (SDK), (4) a template repository with constructs for specific nuclear safety analysis tools, (5) a template engine for processing and expanding templates to fully developed input files for nuclear analyses, and (6) a process manager that handles interactions between the different elements. The nuclear safety analysis tools and their user interfaces are externally developed software tools that function independently but have been integrated into UNF-ST&DARDS for automating different analyses. Current nuclear safety analysis tools consist of SCALE [3], a comprehensive modeling and simulation suite for nuclear safety analysis and design, and COolant Boiling in Rod Arrays–Spent Fuel Storage (COBRA-SFS) [4], a thermal-hydraulic analysis code.

As depicted in Fig.1, data are collected from available sources (e.g., open literature, vendor-provided data), verified, grouped into various attributes (e.g., cask/canister attributes, assembly specific attributes), and incorporated into the Unified Database. Interfacing with the Unified Database is streamlined through an SDK developed in Java programming language. The UNF-ST&DARDS process manager has been developed to (1) manage the workflow across the Unified Database, (2) integrate nuclear safety analysis

[a] The safety margin is the difference between the licensing basis and as-loaded calculations.

templates with corresponding analysis codes, and (3) retrieve and interrogate results. The templates and template engine are described in further detail below. The user interface is comprised of a graphical user interface (GUI) and a command line interface. The command line interface provides access to launch all UNF-ST&DARDS analysis capabilities. In the GUI, an analysis page allows the user to select an analysis type (e.g., criticality) and either (1) reactor type batch (e.g., all PWRs), or (2) specific analysis. The major two components of UNF-ST&DARDS—the unified database and integrated nuclear safety analysis tools—are described below.

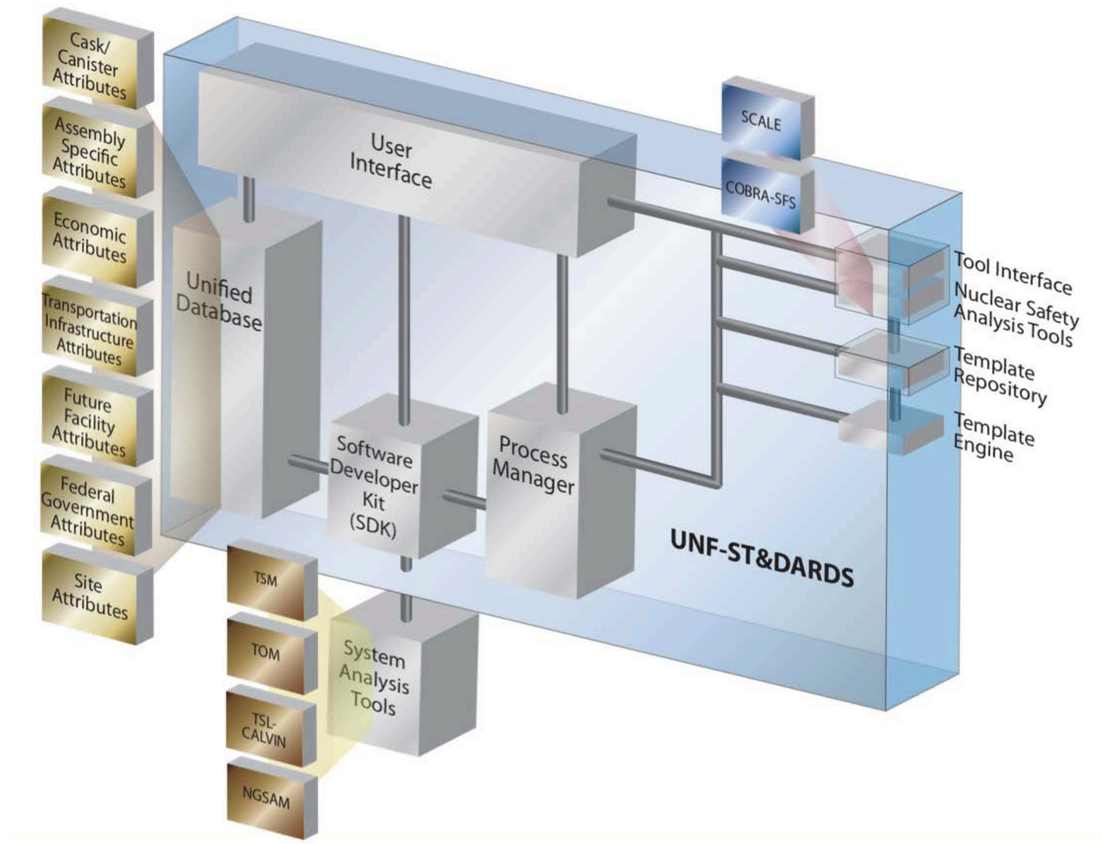

Fig. 1. Information Flow for UNF-ST&DARDS.

Unified Database

Different types and numbers of analysis are required to understand the changing nuclear and mechanical characteristics of the SNF at various stages within the waste management system. Many tools, types of data, and communication between tools and data are essential to perform the volume of detailed calculations necessary to assess the actual condition of as-loaded dry storage systems. Technical data required for these types of analysis include commercial fuel assembly discharge information, fuel assembly design, reactor-specific operation, and cask design and loading data. Hence, various sets of technical data from multiple diverse sources are collected and consolidated into a single controlled Unified Database system.

Currently, the data in the Unified Database can be categorized into seven major attributes as shown in Fig. 1. The fuel assembly discharge data for all US commercial reactors through December 31, 2002 are extracted from the RW-859 database [5]. Other data originate from a variety of open literature sources and from available nonproprietary vendor and utility data. Currently an effort is under way to extract the fuel assembly discharge data for all US commercial reactors through June 30, 2013 from the GC-859 [6] database. The GC-859 is an updated version of the RW-859 SNF discharge data collection form from the US nuclear utilities.

The Unified Database contains a data dictionary with a description for each datum and its associated unit of measure. The data are organized in relational structured query language (SQL) data tables within a MySQL Community Edition database server. MySQL Community Edition is a freely available, well-documented, open source, relational database server that utilizes SQL. To maintain, control, and facilitate database expansion, versions of the Unified Database will be distributed periodically through a controlled release while a live version is maintained to accommodate changes before the next release. While the nuclear analysis tools integrated in UNF-ST&DARDS use the live version of the database for as-loaded safety analysis, external tools, such as the system analysis tools, will use a released version of the database, which will be updated periodically.

Because the quality control of any analysis, both in terms of accuracy and reproducibility, hinges on the pedigree of the input data, the Unified Database supports the primary data with reference information for each record. Every reference can include a note from the data collector and a link to the source document stored on Centralized Used Fuel Resource for Information Exchange (CURIE) at http://curie.ornl.gov/. A companion paper entitled *SNF Data Visualization on CURIE* describes the Unified Database visualization features through the CURIE website.

Computational Analysis Integration and Automation

Different computational analysis packages can be linked to the Unified Database that will receive data, generate results, and write pertinent results back to the Unified Database for other tools or various data interrogations. As mentioned before, UNF-ST&DARDS nuclear analyses are performed by using the SCALE code system [3] and the thermal-hydraulic analysis code COBRA–SFS [4]. SCALE provides the Transport Rigor Implemented with Time-dependent Operation for Neutronic depletion (TRITON) transport and depletion module, the Oak Ridge Isotope Generation (ORIGEN) assembly isotope generation code, the Criticality Safety Analysis Sequence with KENO-VI (CSAS6), and Monaco with Automated Variance Reduction using Importance Calculations (MAVRIC) shielding sequence. SCALE also provides ORIGen AsseMbly Isotopics (ORIGAMI) interface for performing automated ORIGEN calculations. The COBRA-SFS thermal hydraulics application calculates peak and minimum clad temperatures, component temperatures, and cask surface temperatures.

When basic information about the SNF and the cask system is provided, the data relationships defined in the Unified Database allow inputs to the respective codes to be built autonomously. These relationships eliminate the user interaction typically required to build the large number of computer code inputs needed for characterizing each site's SNF. Initial analysis has been focused on inventory, criticality, and thermal analysis. Inventory analysis includes depletion and decay to quantify the SNF inventory in terms of nuclide concentrations, decay heat, and radiation sources. Criticality analysis estimates the k_{eff} of a cask system. Thermal analysis calculates component temperatures (e.g., cladding temperature) and is also important in understanding cask surface degradation mechanisms. A new, recently added capability is dose analysis for transportation. Cask-specific, as-loaded containment analysis and fuel performance analysis based on actual irradiation history are under development. While the containment analysis will provide the cask-specific allowable leakage rate, the initial implementation of fuel performance capability will provide the realistic fuel rod pressure, which is one of the parameters important to clad hydride reorientation.

Although system analysis is not the focus of this paper, the Unified Database, which is the US DOE Office of Nuclear Energy (DOE-NE) resource for SNF management and disposition, provides data support to the systems and logistics analysis codes such as Civilian Radioactive Waste Management System Analysis and Logistics Visually Interactive (CALVIN) [7], Total System Modeling (TSM) [8], and Transportation Operations Model (TOM) [9]. The Unified Database will also support development of the Next Generation System Analysis Model (NGSAM), which will integrate waste management systems and logistics analysis tools over the next several years.

To perform inventory, criticality, shielding, and thermal analyses, a library of model templates for SCALE and COBRA-SFS is employed for different reactor sites and storage and transportation system variants. Templates allow UNF-ST&DARDS to remain application agnostic. Within UNF-ST&DARDS, a template engine (a template processor known as TemplateEngine) is used to combine site-specific input parameters from the Unified Database with the model templates and subtemplates developed for the inventory, criticality, thermal, and shielding calculations to produce complete input files for those types of calculations. TemplateEngine is a string substitution program designed to take advantage of repeated structures in text files. TemplateEngine takes the input parameter data structures represented by a JavaScript Object Notation (JSON) data structure and the root template file. With these two components, TemplateEngine conducts attribute replacement and subtemplate imports.

The model templates contain three basic components: (1) input data blocks that do not vary as a function of fuel assembly characteristics (e.g., description of cask dimensions and construction materials for criticality or thermal calculations), (2) input parameters that vary as a function of assembly characteristics (e.g., fuel pin dimensions in an assembly model for depletion calculations or nuclide concentrations in a cask model for criticality calculations), and (3) subtemplates to be imported (e.g., templates describing fuel pin arrays for depletion or criticality calculations). Model template development, update, and review are conducted using the Mercurial distributed source control management tool [10], which is widely used for version control of files.

An application of UNF-ST&DARDS can be executed through the user interface. Upon execution, UNF-ST&DARDS creates JSON files with values provided by the Unified Database. These files are used with TemplateEngine and previously generated templates to create an input file that is then run by the appropriate code. In general, the process flow managed by the process manager within UNF-ST&DARDS is as follows:

- retrieve the entities required for process execution from the Unified Database,
- retrieve the process input template,
- create a process input by expanding the process template with a JSON parameter set describing the entities,
- execute the process with the expanded process input, and
- upon completion, process and store the results.

APPLICATION

Cask-specific as-loaded analyses for storage and transportation configurations have been performed with UNF-ST&DARDS to assess the realistic state of SNF and the available margin with regard to thermal, criticality, and shielding requirements. Calculations are performed using the design basis licensing conditions documented in the final safety analysis reports (FSARs) to the extent applicable to confirm that the representative model supplies consistent results that provide a basis of comparison against the specific as-loaded fuel to determine the inherent uncredited margins. Note that unlike licensing evaluations, which are designed to cover broad range of SNF characteristics, cask-specific as-loaded analyses are performed without attempting to establish bounding parameters and conditions. Conservatisms from the FSARs are retained for the different analyses where insufficient information is available to justify a change. Estimated as-loaded decay heat load per cask as a function of time is presented in Fig. 2 for seven decommissioned reactor sites referred to as Sites A, B, C, D, E, F, and G that includes both PWR decommissioned reactor sites, and BWR decommissioned reactor sites. Figure 3 illustrates the fuel rod temperature distribution in an as-loaded Site A cask.

Estimated criticality margin results for five decommissioned sites are illustrated in Fig. 4, which shows that the k_{eff} margin typically varies from 0.05 to almost 0.30 Δk_{eff}. Results are clustered into upper and lower bands. The lower band is mainly an attribute of the damaged fuel assemblies in the loaded casks, which are modeled as unirradiated. Because the extent of damage in the fuel assemblies loaded in the casks are not defined, crediting burnup for these damaged fuels may lead to a non-conservative k_{eff} estimation. k_{eff} margin is below 0.05 Δk_{eff} for one of the Site E casks, which contains damaged fuel cans in 38 locations inside the cask with damaged fuel assemblies and/or debris. Figure 4 shows no noticeable change in cask k_{eff} values as a function of time. Perhaps this is because the casks are loaded with assemblies with varying discharge times. The shielding analysis is performed for transportation packages designated for Sites A and C and compared with the regulatory dose rate limit. Figure 5 presents the maximum normal conditions of transport (NCT) 2 m dose rates as a function of decay time and the corresponding regulatory dose rate limit provided in 10 CFR 71.47(b) [11].

Cask-specific analyses are also applied to disposal studies, including an initial assessment of the feasibility of direct disposal of existing dual-purpose casks (DPCs) from a criticality safety perspective [12]. These types of best-estimate analyses enable an understanding of the sensitivity of the overall system to different degradation mechanisms and geometry changes. Using UNF-ST&DARDS, a configuration template can be developed once (e.g., DPC configuration with loss of neutron absorber from the basket when exposed to groundwater over the repository performance period), and then the cask-specific analysis can be executed for all DPCs of a particular variant to understand the sensitivity of the various DPC loadings to the different configuration conditions.

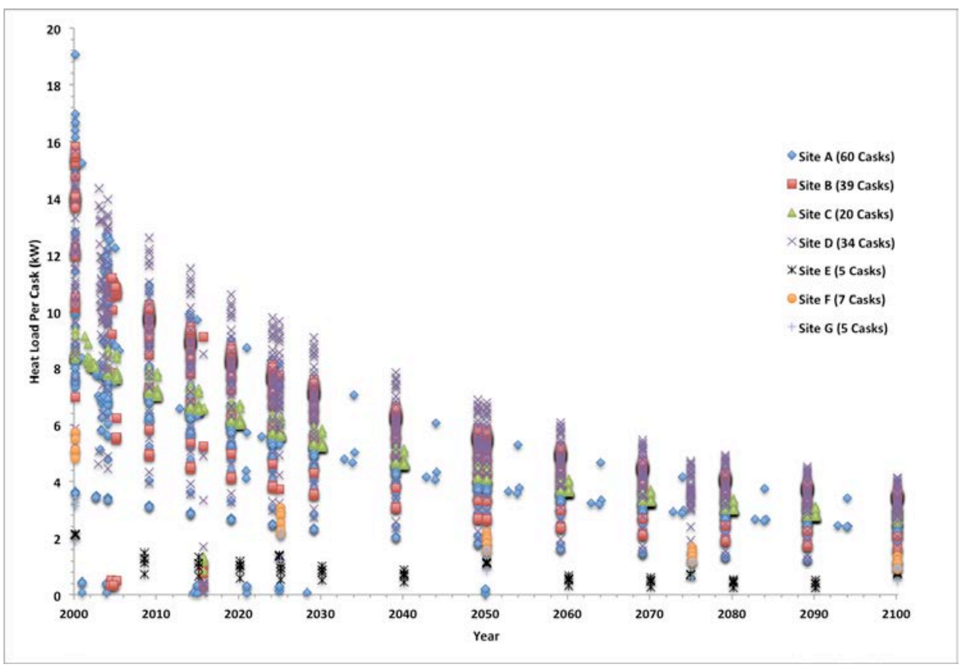

Fig. 2. Total Cask Decay Heat as a Function of Calendar Year for Seven Decommissioned Reactor Sites.

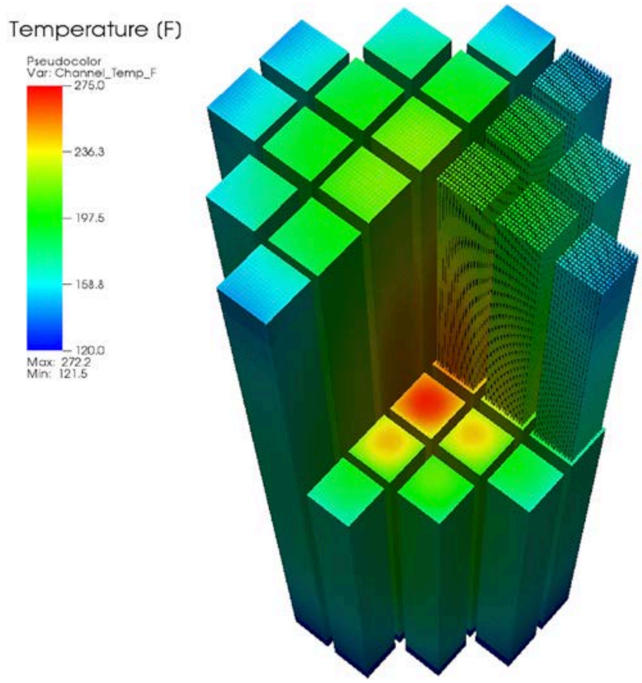

Fig. 3. Example Fuel Rod Temperature Distribution in a Cask at Site A.

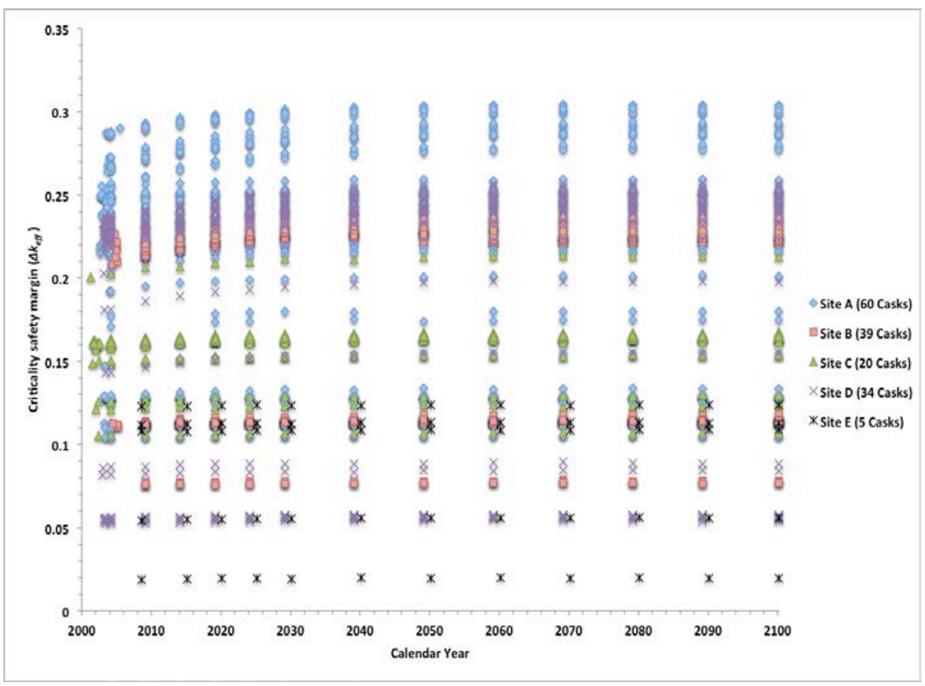

Fig. 4. Estimates of Available Criticality Safety Margin as a Function of Calendar Year at Sites A, B, C, D, and E.

Fig. 5. Maximum NCT Dose Rates at 2 m from Cask Surface as a Function of Year for Sites A and C.

DATA VISUALIZATION

In addition to conducting analysis, the GUI of UNF-ST&DARDS provides (1) an easy way to interact with the Unified database, (2) assembly-specific time-dependent characteristics information (e.g., assembly-specific heat load), (3) as-loaded loading maps and analyses models, and (4) cask-specific as-laded criticality, thermal and shielding results visualization. Some of the default display options through the GUI are illustrated in Fig. 6 and Fig. 7. As shown in Fig. 6, independent spent fuel storage installation (ISFSI) site locations are illustrated on a map of the US. Information about the sites (such as reactor status and wet and dry inventory) are viewable, as well as a satellite photo of the ISFSI from which individual cask-specific characteristics can be observed by selecting and clicking a specific cask.

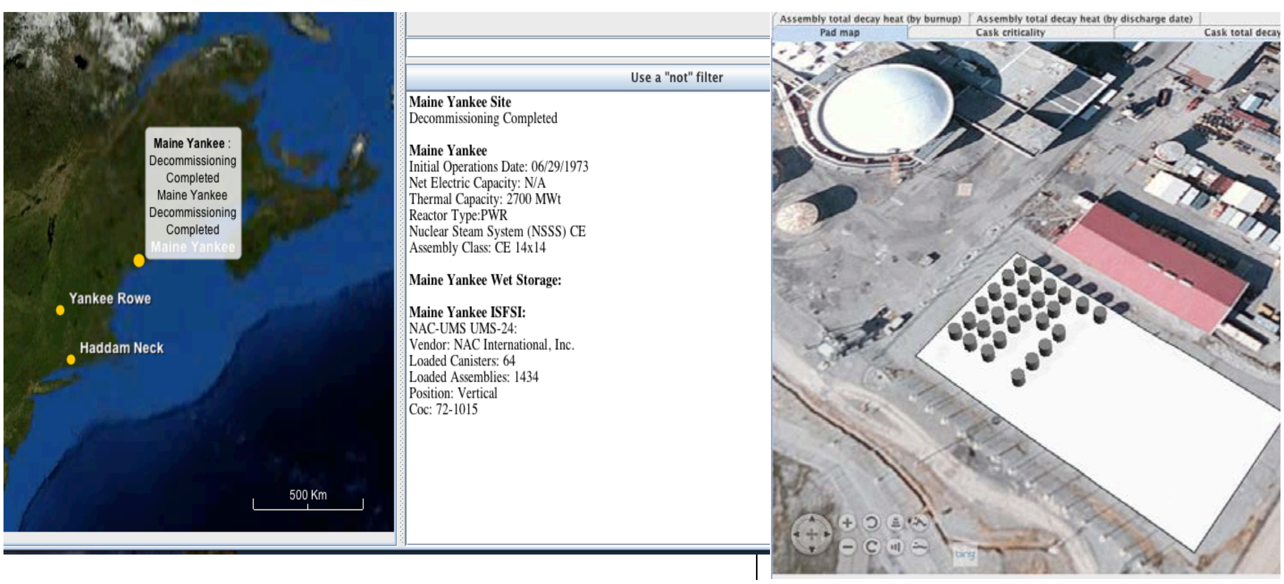

Fig. 6. Example GUI Visualization Capabilities for Site Characteristics.

Cask-specific characteristics include loading maps, fuel assembly initial enrichments, isotopic compositions, burnups, and discharge dates, as well as a thermal map of the fuel rod temperature distribution that can be displayed radially, axially, and as a function of time as depicted in Fig. 7.

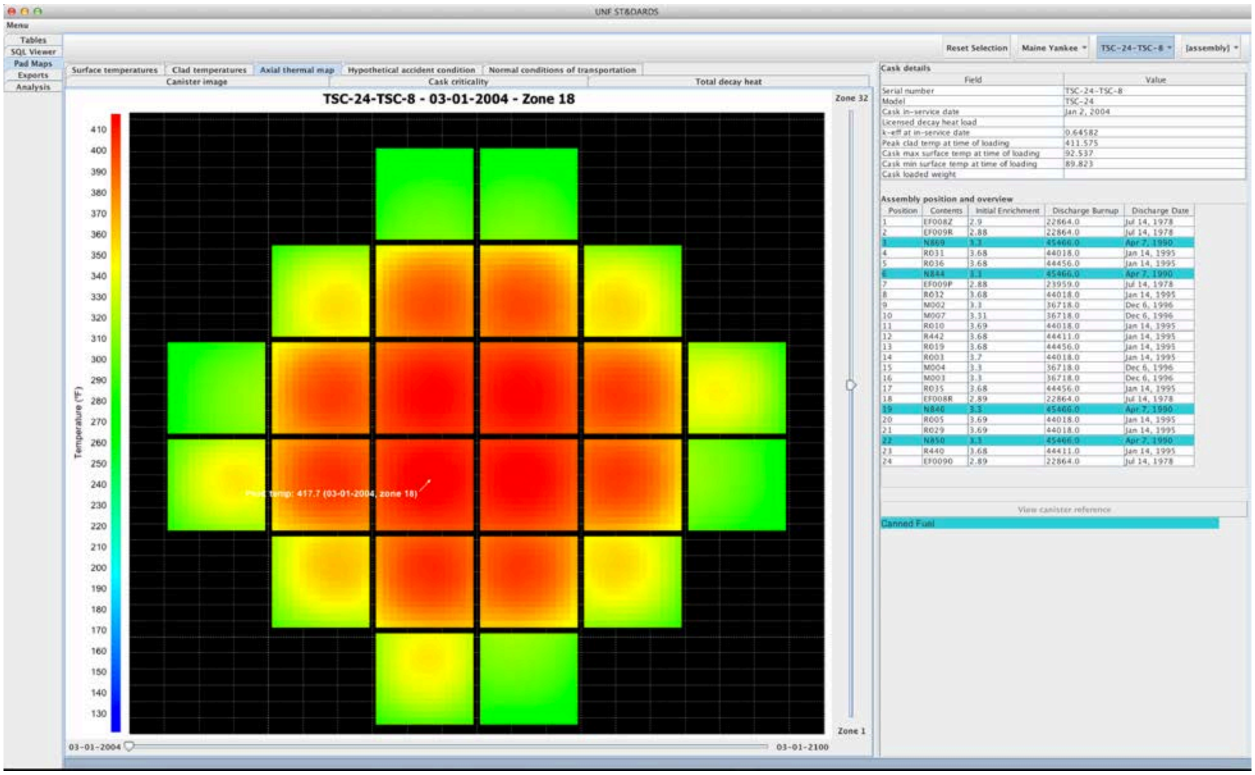

Fig. 7. Maine Yankee Cask-Specific Axial Thermal Temperature Map at a Given Time.

CONCLUSIONS

The automated nuclear analysis capabilities of UNF-ST&DARDS include automated radionuclide inventory, criticality, thermal, and shielding analysis sequences. Integrating the Unified Database with the safety analysis pipelines streamlines analyses for the waste management system by automating input file development. Data and analysis tool integration is a powerful UNF-ST&DARDS feature that enables assembly-specific and cask-specific nuclear safety assessments based on the actual assembly characteristics of the as-loaded SNF (e.g., fuel assembly burnup, enrichment, decay time) to quantify realistic safety margins associated with actual fuel loading compared with the regulatory licensing limits. The results presented herein demonstrate that UNF-ST&DARDS can perform the volume of detailed calculations necessary to assess the actual condition of as-loaded dry storage casks based on its integrated data and analysis capabilities. The information generated by UNF-ST&DARDS allows realistic assessments of the current and future state of the SNF assemblies and canister systems from both a nuclear and mechanical performance perspective. In summary, UNF-ST&DARDS provides a controlled, comprehensive relational Unified Database coupled with nuclear safety analysis capabilities. This system is a new and unprecedented capability/resource that enables automated assembly-specific and cask-specific evaluations, enabling a better understanding of the implications to, and planning for, the public's health and safety related to the large-scale transportation (1000's) of UNF casks and addressing uncertainties related to aging. UNF-ST&DARDS represents the quintessential element of a fully integrated waste management system, and has broad applicability (fuel cycle decisions, safeguards and security, waste management, safety) where the benefits and potential are just starting to be realized.

REFERENCES

1. J. M. SCAGLIONE et al., “The Used Nuclear Fuel Storage, Transport, and Disposal Analysis Resource and Data System,” *Proceedings of Institute of Nuclear Materials Management (INMM) - 55th Annual Meeting*, July 20-24, 2014, Atlanta, Georgia.
2. US NRC, “Interim Staff Guidance-11, Revision 3, Cladding Considerations for the Transportation and Storage of Spent Fuel,” NRC Spent Fuel Project Office, 2003.
3. ORNL, *SCALE: A Comprehensive Modeling and Simulation Suite for Nuclear Safety Analysis and Design*, Oak Ridge: ORNL/TM-2005/39 Version 6.1, Radiation Safety Information Computational Center, ORNL, 2011.
4. T. MICHENER, D. RECTOR, J. CUTA, R. DODGE and C. ENDERLIN, “COBRA-SFS: A Thermal-Hydraulic Code for Spent Fuel Storage and Trasnportation Casks,” PNL-10782, UC-800, PNNL, Richland, WA, September, 1995.
5. Energy Information Administration (EIA), “RW-859 Nuclear Fuel Data,” US DOE, Washington, DC, October, 2004.
6. EIA, “Nuclear Fuel Data Survey Form GC-859,” US DOEWashington, DC, July, 2012.
7. E. KALININA, “CALVIN Database Update in Support of UFD System Architecture Study, FCRD-UFD-2012-000135,” prepared for the US DOE, Used Fuel Disposition Campaign, SNL, Albuquerque, NM, May 2012.
8. E. CORYELL, “User Manual for the Total System Model Version 6.0 Preprocessor,” 50040-UM-02-6.0-00, Bechtel SAIC Company, LLC, Las Vegas, NV, October, 2007.
9. I. BUSCH and R. HOWARD, “Transportation Operations Model (TOM) Technical Manual, FCRD-NFST-2012-000425,” US DOE, Washington DC, October 2012.
10. B. O'SULLIVAN, *Mercurial: The Definitive Guide*, O'Reilly, Sebastopal, CA, 2009.
11. Title 10, *Code of Federal Regulations*, Part 71, 77 FR 34204, Jun. 11, 2012.
12. K. BANERJEE et al., “Feasibility of Direct Disposal of Dual-Purpose Canisters from a Criticality Perspective,” *Proceedings of Institute of Nuclear Materials Management (INMM) - 55th Annual Meeting*, July 20–24, 2014, Atlanta, Georgia