FUEL CYCLE APPLICATIONS WITH THE DATA STORED WITHIN THE UNIFIED DATABASE*

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The Unified Database within the Used Nuclear Fuel Storage, Transportation & Disposal Analysis Resource and Data System (UNF-ST&DARDS) contains information that can support a variety of activities. These activities such as fuel cycle–related evaluations are based on spent nuclear fuel (SNF) characteristics as input. The main datasets stored within the Unified Database include a number of attributes related to existing facilities, federal government radioactive waste, future facilities, transportation infrastructure, SNF assembly–specific data, cask/canisters, and economics. These datasets contain a vast amount of information that can be used for multiple applications, including fuel cycle transitional analysis and cost/benefit analysis for different SNF disposition strategies and can facilitate comparisons of fuel cycle transition options. The SNF inventory data can be applied per assembly and per site for informed decision-making on fuel cycle planning issues such as considering the type and quantity of SNF available from nearby nuclear power plants when siting a reprocessing facility. Examples of factors that could be considered and evaluated include: minimizing the shipping distance of SNF, proliferation risk, and resource utilization. Probably the largest advantage for using the Unified Database for fuel cycle analysis is to ensure that future U.S. fuel cycle analysis that may wish to use the legacy SNF material can do so from the same common set of data.

I. INTRODUCTION

The Nuclear Fuel Storage and Transportation Planning Project (NFST) has been developing the Used Nuclear Fuel Storage, Transportation & Disposal Analysis Resource and Data System (UNF-ST&DARDS).[1] An important part of the UNF-ST&DARDS is the Unified Database, which contains information that can support a variety of activities including fuel cycle–related transition evaluations based on detailed used nuclear fuel (UNF) characteristics. There are potential advantages for transitioning to advanced fuel cycles including improvements in nuclear waste management, proliferation, nuclear material security, safety, environmental impact, resource utilization, development and deployment risks, institutional issues, and financial risk and economics.[2] For any transitional fuel cycle analysis the characteristics of the SNF is required. The legacy material can be calculated using assumptions and simplified calculations. However, the advantage of the Unified Database is that very detailed analyses (assembly by assembly basis) have been performed on SNF and that future U.S. fuel cycle analysis that may wish to use the legacy SNF material can do so from the same common set of data.

II. OVERVIEW OF THE UNIFIED DATABASE

The purpose of UNF-ST&DARDS is to establish a unified, comprehensive SNF database and integrated analysis system to characterize the input to the waste management system; provide a credible, controlled data source for key information; assess issues and uncertainties related to the extended storage and transportability of loaded canisters after extended storage; support R&D prioritization; and preserve SNF related information. UNF-ST&DARDS is intended to serve as a national resource of information for use by the U.S. Department of Energy office of nuclear energy (DOE-NE) or a future Management and Disposal Organization (MDO) in planning for future storage, transport, and disposal of SNF.

There are multiple components that constitute UNF-ST&DARDS as can be seen in Fig. 1. A major component of UNF-ST&DARDS, the Unified Database contains data collected from multiple sources including the EIA-859 database,[3] open literature, vendor data, and utility data. The Unified Database will be periodically updated, as additional data is obtained from these multiple sources especially when new updates from the EIA-859 database become available. In addition to the data obtained from the multiple sources, calculated results from nuclear safety analysis tools such as SCALE[4] and COBRA-SFS[5] are stored within the Unified Database that provide characteristics data on the SNF assemblies and

canister systems. All of the data within the Unified Database has reference traceability. Static references are typically stored on the Centralized Used Fuel Resource for Information Exchange (CURIE)[6] website (www.curie.ornl.gov), which is designed to share and access UNF-related documents and information, and dynamic references (i.e., nuclear safety analysis code outputs) are stored within UNF-ST&DARDS with file paths provided in the Unified Database. Below is a description of the following main datasets stored within the Unified Database [7]:

- Cask/canister attributes: overpack, capacity, physical properties, and unit processing times
- Assembly-specific attributes: both actual and projected assembly discharge information
- Economic attributes: reactor site unit, transportation infrastructure unit, ISF unit, and repackaging facility unit
- Transportation infrastructure attributes: rail, heavy haul, legal weight truck, and barge assets data, in addition to information on transfer times between these transportation modes
- Future facility attributes: an interim storage facility (ISF) and a repackaging facility information
- Federal government attributes: government-owned high-level radioactive waste (HLW) and government-owned SNF data
- Site attributes: facility, reactor, pool, and independent spent fuel storage installation (ISFSI) data

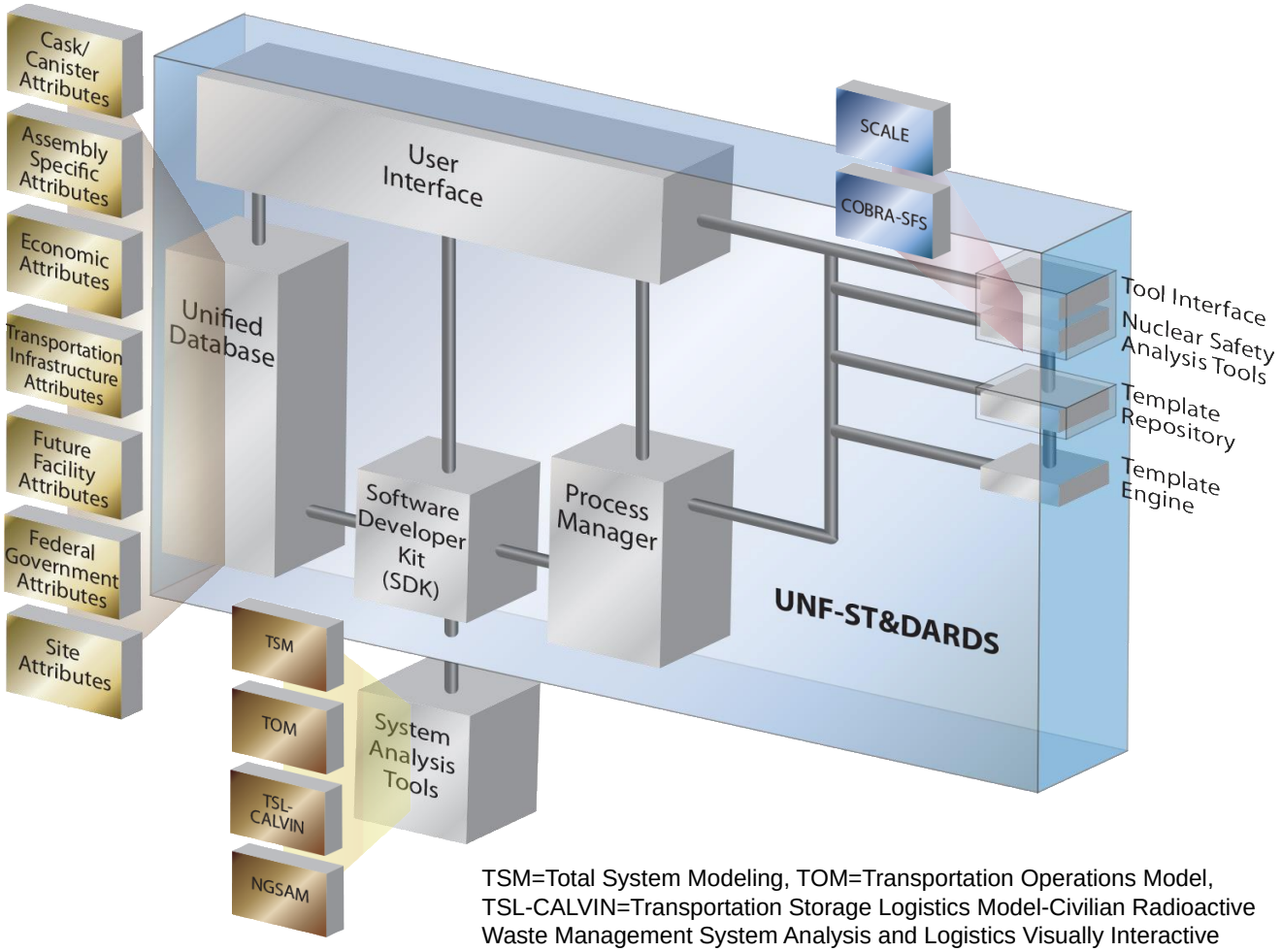

Fig. 1. Components of the UNF-ST&DARDS.

Because of the large amount of data stored within the Unified Database, the MySQL Community edition database was used as the database server. MySQL Community edition is a freely available, well-documented, open source, relational database server that uses a structured query language (SQL). Many tools exist to assist in the development and maintenance of MySQL databases making it a productive environment to streamline the data needs of nuclear safety and systems analysis tools. The relational database can be easily queried using SQL, which allows quick retrieval of desired data easily exported to comma-separated values (CSV), which could then be imported into Microsoft Excel or other tools for additional processing or the generation of plots.

III. FUEL CYCLE RELATED DATA WITHIN THE UNIFIED DATABASE

Though UNF-ST&DARDS has initially been designed primarily to serve as a national resource of information in the planning for future storage, transport, and disposal of commercial reactor–generated UNF, the information within the Unified Database can also be used to support a variety of activities that can utilize SNF characteristics as input, such as fuel cycle evaluations. The main datasets applicable to fuel cycle evaluations are the assembly-specific attributes and the facility attributes.

The assembly-specific attributes contain information, including initial enrichment, burnup, metric tons of heavy metal (MTHM), assembly type, and discharge date. These data are included on an assembly-by-assembly basis. In addition, the Nuclear Safety Analysis Tools (See Fig. 1) has been used to calculate assembly-specific attributes including the isotopic composition, heat, and activity for each fuel assembly over a wide range of times.

Unlike other calculations performed in the past on SNF inventory, the data within the Unified Database have been calculated on an assembly-by-assembly basis using the cross sections generated for the assembly-specific types. For example, as seen in Fig. 2, both the General Electric-14 (G4610G14) and the Advanced Nuclear Fuel for use at LaCrosse (XCL10a) have 10 × 10 lattices, but they have very different assembly designs. Using assembly-specific cross sections provides an added layer of detail for improved assembly-specific radionuclide calculations.

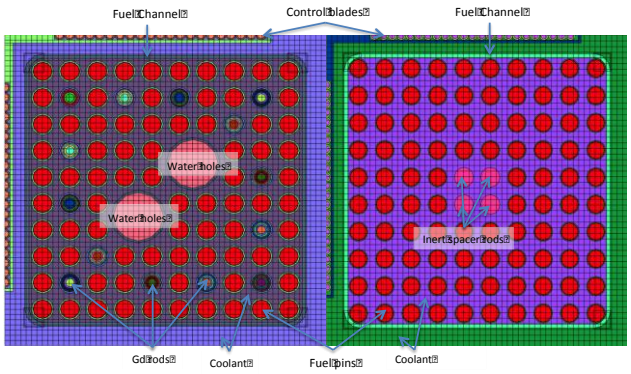

Fig. 2. GE4610G14 assembly type (left), XCL10a assembly type (right).

The individual assembly calculations for specific dates are automated within UNF-ST&DARDS using SCALE, the template repository, the template engine, and the Unified Database (see Fig. 1). This is accomplished within UNF-ST&DARDS by first generating assembly type specific cross sections with the parameters for that assembly type obtained from the Unified Database. Then irradiation and decay calculations are performed for every fuel assembly in the database to obtain the mass, heat, and activity for the individual assemblies at a specified date in time using the assembly type specific cross sections. The process manager (see Fig. 1) contains the business logic (i.e., rules on how data can be created, displayed, stored, and changed) to manage the preparation, execution, and results retrieval of these applications.

The information from the assembly-specific attributes can provide fuel cycle information such as the U-235 content of the uranium after the fuel has been irradiated as a function of burnup and reactor type (see Fig. 3), the quantity of plutonium in a SNF assembly (Fig. 4), and the ratio of the Pu-239 versus Pu-241 within that spent fuel assembly at discharge (Fig. 5). In addition minor actinide and fission product waste streams can also be obtained for each fuel assembly for the entire U.S. fleet. These parameters can then be used in fuel cycle analysis to account for the material flow of plutonium and uranium that is important for transition analysis to couple LWRs to other reactor designs such as mixed oxide fuel (MOX) reactors or fast reactors.

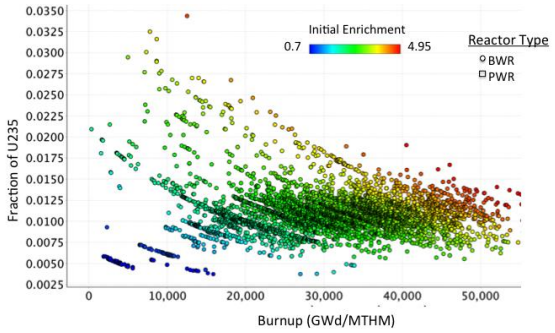

Fig. 3. Enrichment of uranium from a discharged SNF assembly as a function of burnup and reactor type.

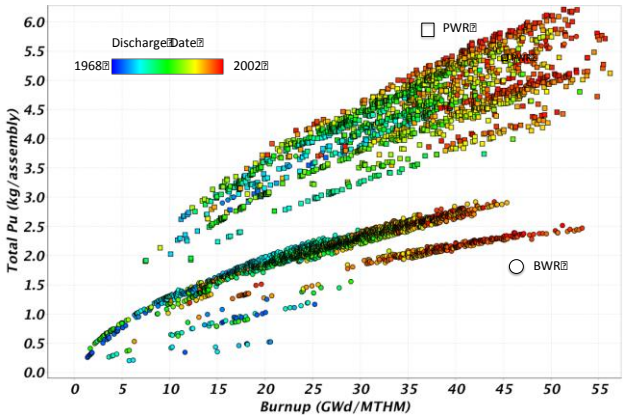

Fig.4. Quantity of plutonium per assembly from a discharged SNF assembly as a function of burnup per assembly and reactor type evaluated at 2025.

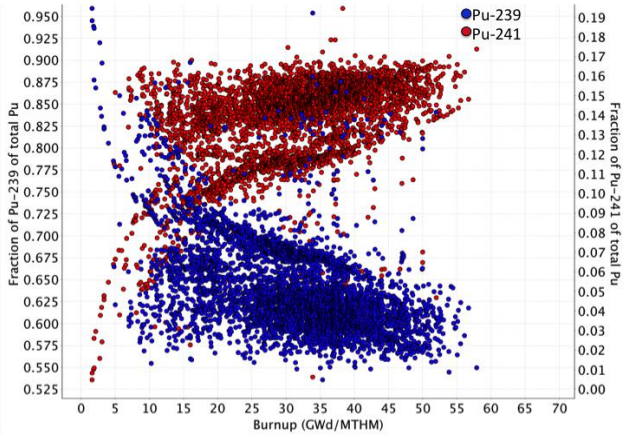

Fig. 5. Percentage of Pu-239 and Pu-241 within a SNF assembly at assembly discharge as a function of burnup.

The facility-specific attributes include information such as reactor types, operating statuses, commercial SNF storage modes (dry/wet), and transportation infrastructures (train, rail, and barge). With the use of SQL and the relational database, the different datasets can be combined to provide more detailed information. For example, when the facility- and assembly-specific attributes are combined, the data can show not only the amount of SNF at each site, but also the operating/shutdown status of the sites, along with the reactor type at these sites (Fig. 6). This information can be used for more detailed fuel cycle analysis.

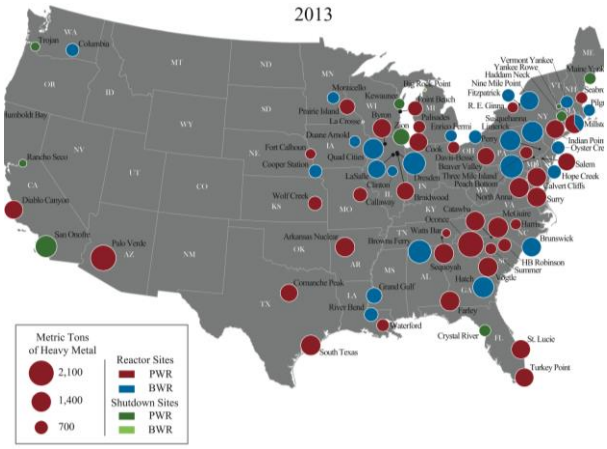

Fig. 6. SNF (MTHM) in the U.S. grouped by boiling water reactor (BWR) and pressurized water reactor (PWR) and operating and shutdown sites at 2013.

Complex questions could also be answered with this database, including barge, rail, or heavy haul truck access, the cost for transporting between different sites, dose to the public along the routes being considered, total distance traveled along those routes, etc.

V. CONCLUSION

The Unified Database within UNF-ST&DARDS contains information that can support a variety of activities that can use SNF characteristics as input. This Unified Database ensures that decisions on centralized storage and disposition of SNF and future U.S. fuel cycles that may wish to use the legacy material, can do so from the same common set of data and assumptions.

REFERENCES

1. J. M. Scaglione, K. Banerjee, K.R. Robb, et. al., "The Used Nuclear Fuel Storage, Transportation, and Disposal Analysis Resource and Data System," Institute of Nuclear Materials Management (INMM), 55th Annual Meeting, Atlanta, Georgia (2014).
2. R. Wigeland, T. Taiwo, H. Ludewig, et al., "Nuclear Fuel Evaluation and Screening – Final Report". FCRD-FCO-2014-000106. U.S. Department of Energy (2014).

3. "RW-859 Nuclear Fuel Data," U.S. Energy Information Administration, Washington, D.C. (2004).
4. *SCALE: A Comprehensive Modeling and Simulation Suite for Nuclear Safety Analysis and Design*, ORNL/TM-2005/39, Version 6.1, Oak Ridge National Laboratory, Oak Ridge, Tennessee, (2011).
5. N. J. LOMBARDO, J. M. CUTA, T. E. MICHENER et al., *COBRA-SFS: A Thermal-Hydraulic Analysis Computer Code, Volume III: Validation Assessments*, PNL-6048 Vol. III, Pacific Northwest Laboratory, Richland, Washington (1986).
6. J. Jarrell, D. White. *"CURIE User Manual,"* FCRD-NFST-2014-000353 Rev. 0, Oak Ridge, Tennessee, (2014)
7. B. van den Akker, J. Jarrell, J. Peterson et. al., "Data Identification and Verification for Waste Management System Analytses" FCRD-NFST-2014-000530 Rev. 0, Oak Ridge National Laboratory, Oak Ridge, Tennessee, (2014).