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Inventory and Waste Characterization Status Report

Spent Fuel and Waste Disposition

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Spent Fuel and Waste
Science and Technology

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ACRONYMS

ANL	Argonne National Laboratory
BRC	Blue Ribbon Commission on America's Nuclear Future
BWR	boiling water reactor
CURIE	centralized used fuel resource for information exchange
DOE	US Department of Energy
DOE-NE	Department of Energy Office of Nuclear Energy
DPC	dual-purpose canister
DWPF	defense waste processing facility
EBR-II	experimental breeder reactor II
EMT	electrometallurgical treatment
FFTF	fast flux test facility
FY	fiscal year
HEU	highly enriched uranium
HIP	hot isostatic pressing
HLW	high-level radioactive waste
INL	Idaho National Laboratory
INTEC	Idaho Nuclear Technology and Engineering Center
ISF	interim storage facility
ISFSI	independent spent fuel storage installation
LEU	low-enriched uranium
LLW	low-level radioactive waste
MCO	multicanister overpack
MEU	medium enriched uranium
MOX	mixed oxide (fuel)
MTU	Metric Tons of Uranium
NRC	Nuclear Regulatory Commission
ORNL	Oak Ridge National Laboratory
PBC	purpose-built canister
PWR	pressurized water reactor
RCRA	Resource Conservation and Recovery Act
SBW	sodium-bearing waste
SMR	small modular reactor

SNF	spent nuclear fuel
SNL	Sandia National Laboratories
SRNL	Savannah River National Laboratory
SRS	Savannah River Site
UNF-ST&DARDS	used nuclear fuel – storage, transportation & disposal analysis resource and data systems
WFDOE	waste form disposal options evaluation
WIPP	Waste Isolation Pilot Plant
WVDP	West Valley Demonstration Project
WTP	waste treatment and immobilization plant

Units

ft	foot
GWd	gigawatt-days
in.	inch
lb	pound
MT	metric ton
MTHM	metric ton of heavy metal
MWd	megawatt-days
MTU	metric ton of uranium
wt %	weight percent
W	watt

SPENT FUEL AND WASTE DISPOSITION/SPENT FUEL AND WASTE SCIENCE AND TECHNOLOGY

1. INTRODUCTION

This report provides an update to Sassani et al. (2016) and includes:

(1) an updated set of inputs (Sections 2.3) on various additional waste forms (WF) covering both DOE-managed spent nuclear fuel (SNF) and DOE-managed (as) high-level waste (HLW) for use in the inventory represented in the geologic disposal safety analyses (GDSA);

(2) summaries of evaluations initiated to refine specific characteristics of particular WF for future use (Section 2.4);

(3) updated development status of the Online Waste Library (OWL) database (Section 3.1.2) and an updated user guide to OWL (Section 3.1.3); and

(4) status updates (Section 3.2) for the OWL inventory content, data entry checking process, and external OWL BETA testing initiated in fiscal year 2017.

As such, this report represents completion of milestone deliverable M2SF-17SN010501014 “Inventory and Waste Characterization Status Report” (SFWD-SFWST-2017-000014), as the final report on FY2017 activities for the work packages SF-17SN01050101 and SF-17SN01050102.

The scope of the inventory and waste form characteristics work in this area covers DOE-managed SNF (DSNF) and DOE-managed (as) HLW (DHLW), with the current intent to dispose of these in a deep geologic repository. It is noted that the DOE-managed (as) HLW include wastes that may be dispositioned in the future with waste classifications different than HLW (and perhaps with a different disposal pathway). In this work, the theoretical geologic repository for these wastes is a deep geologic repository for DOE-managed SNF and (as) HLW (DGRDMSH).

Initial GDSA work (Sevougian et al., 2016) represented the major high-level waste (HLW) groups (Savannah River Site (SRS) and Hanford HLW Glasses) and DOE–managed spent nuclear fuel (DSNF) materials in the inventory to evaluate potential releases from both generic salt and generic crystalline (granitic) repositories. Wilson (2016) provides the preliminary inventory for the GDSA analyses of a DGRDMSH for FY2016 and includes both DHLW and DSNF waste canister counts and thermal information (Tables 2-1, and 2-3 thru 2-6 from Wilson, 2016). The Wilson (2016) report describes each waste form in terms of both average radionuclide content and overage thermal output evolution. The tabulation includes canister counts and thermal characteristics for each DHLW and DSNF waste form considered (Wilson, 2016). For that preliminary DSNF inventory, the detailed list of DSNF types is given in Appendix A to show the specific DSNF groups/items included in the ~2485 canisters (see Table 2-1 from Wilson, 2016). For the waste types/waste forms already incorporated into the GDSA (Sevougian et al., 2016), there are no currently recommended changes from this update to the manner in which their inventories and performance behaviors are represented in the GDSA (Sevougian et al., 2016).

Based on the Sassani et al. (2016) recommendations, the primary FY2017 update to the preliminary DGRDMSH inventory is to include the additional possible DGRDMSH waste forms (DOE, 2014) that were not previously included in GDSA representations (e.g., Sevougian et al. 2016) and are most likely to expand the evaluation range of thermal and/or radionuclide inventory aspects compared to the previous analyses. Specifically, this entailed adding:

- The 340 Hanford Cs/Sr vitrified glass canisters (as detailed in Wilson, 2016, Table 2-6),

- The 34 glass canisters of Hanford Federal Republic of Germany (FRG) glass, which is material that has been managed as HLW (SNL, 2014), and may be disposed in a DGRDMSH,
- The planned waste form for calcine hot isostatically pressed (HIP) into HIP cans that are loaded/stacked into ~320 canisters (~5.5 ft diameter by ~15 ft height, naval canisters/waste packages containing ~10 HIP cans each; SNL 2014), and
- The naval SNF waste packages from the coolest thermal range (~13 naval SNF canisters using the ~1000W per canister thermal threshold for the upper bound—see Figure 3 of DOE, 2014; and SNL, 2014 naval waste package thermal binning listed in Appendix A, p. A-40).

Although most of these updates are relatively small from the standpoint of inventory mass, they may have some implications for analyses of thermal effects. This is because some of these added wastes tend to have higher average thermal loads per canister than the inventory previously evaluated in GDSA. Additionally, some of these waste forms represent larger waste packages, which may expand handling and emplacement considerations (i.e., naval SNF and planned calcine HIP waste form waste packages). Section 2.3 provides the details of these updates.

During FY2017, a number of questions regarding the characteristics of various waste forms led to three ongoing studies on WF characteristics details (Section 2.4). First, in our estimates of HLW glass compositions for postclosure safety analyses, we assume that all the ^{129}I in tank waste becomes part of the vitrified waste form. However, it is not clear if this quantitative assumption is correct, as the ^{129}I activity in the glass waste form is not high enough to warrant direct analysis. Given that the SRS has produced thousands of HLW glass logs, we initiated a study of the detailed documentation for the SRS vitrification process to see if it was possible to trace/quantify the potential sinks for ^{129}I in the various processes that form the HLW glass.

In addition to these uncertainties for SRS HLW glass logs, it was also noted that the inventory for the Cs/Sr capsules did not give the quantity of ^{135}Cs contained in the capsules. Nor did the reported inventory of Cs and Sr for the FRG glass at Hanford provide the quantity of ^{135}Cs contained in those glass logs (SNL, 2014). Because quantities of these two long-lived fission products (half-life of ^{129}I is 1.57×10^7 years, half-life of ^{135}Cs is 2.3×10^6 years) were not readily available, we developed estimated quantities of both radionuclides in SRS glass (Section 2.4.1) and the FRG glass at Hanford (Section 2.4.2).

We began a third study to define characteristic isotopic ratios for various waste forms included in postclosure performance studies (Section 2.4.3). This aspect arose due to questions regarding the relative contributions of radionuclides from disparate waste forms in DGRDMSH GDSA results, particularly, radionuclide contributions of DOE-managed SNF vs HLW glass. Depending on the design of the generic repository evaluated, it may be easy to assess such contributions proximal to the source terms if the waste forms are segregated. However, given the complexity of some geologic systems, isotopic ratios (two or more) that effectively tag their source waste form distinctly would facilitate such assessments at distal points. Using such ratios to define mixing lines may allow quantitative estimates of relative WF contributions to be “mined” from GDSA results, as long as the particular isotopes are tracked.

Throughout FY2017, the OWL database activities have focused on three areas (Section 3.2). First, additional data for waste types (and their potential waste forms) and source documentation have been added to the OWL to flesh out its content covering DOE-managed (as) HLW and SNF (Section 3.2.1). In conjunction with further data entry, a process of checking the data entry into the OWL against the source documentation was launched to search for and rectify any errors in data entry (Section 3.2.2). This checking was performed by technical individuals independent of the data entry process, who documented any issues noted, and resolved the issues with the data entry staff. As the OWL was modified throughout FY2017 in terms of its interface and features, another process to assess the usability of the OWL was recently kicked-off. This process is referred to here as the External OWL BETA test (Section 3.2.3) and involves technical staff from within the DOE (both NE and EM), as well as at other National

Laboratories, using the OWL and providing feedback on its utility and content. Preliminary feedback is summarized herein, with feedback to be continued into the first quarter of FY2018. Each of these three OWL update activities is ongoing into FY2018.

1.1 Purpose and Scope

The *Evaluation of Options for Permanent Geologic Disposal of Used Nuclear Fuel and High-Level Radioactive Waste Inventory in Support of a Comprehensive National Nuclear Fuel Cycle Strategy* contains analysis of the disposal of both Commercial Spend Nuclear Fuel (CSNF) and DOE-managed HLW and Spent Nuclear Fuel (DHLW and DSNF) in the variety of disposal concepts being evaluated within the previous Used Fuel Disposition Campaign (UFDC; SNL, 2014; this report is referred to herein as the Waste Form Disposal Options Evaluation – WFDOE). That UFDC work covered a comprehensive inventory and a wide range of disposal concepts.

The scope of the inventory and waste form characteristics work in this area covers DOE-managed SNF (DSNF) and DOE-managed (as) HLW (DHLW), with the current intent to dispose of these in a deep geologic repository. It is noted that the DOE-managed (as) HLW include wastes that may be dispositioned in the future with waste classifications different than HLW (and perhaps with a different disposal pathway). In this work, the theoretical geologic repository for these wastes is a deep geologic repository for DOE-managed SNF and (as) HLW (DGRDMSH). The primary goal of this work is to evaluate the information needs for analyzing the disposal of a subset of those wastes in a DGRDMSH. Similar to disposal options considered that include CSNF (SNL, 2014), a potential DGRDMSH appears to be safe in the range of geologic mined repository concepts, but may have different design concepts and features because of the different subset of inventory of waste that would be included. This work provides the technical updates, as listed above and detailed below, from FY2017 activities.

Sassani et al. (2016) provided the other technical content of this report including (1) developing a preliminary DGRDMSH included inventory for engineering/design/safety analyses (updated with additions herein as described above); (2) assessing the major differences of this included inventory relative to that in other analyzed repository systems and the potential impacts to disposal concepts (unchanged); (3) designing and developing the prototype on-line waste library (OWL) to manage the information of all those wastes and their waste forms (updated as discussed above); and (4) constraining post-closure waste form degradation performance for safety assessments of a DGRDMSH (unchanged). In addition, Sassani et al. (2016) provided the Sections on Background and Disposal Concepts, as well as work identifying potential candidate waste types/forms to be added to the full list from the WFDOE (SNL, 2014 – see Table C-1), and potential OWL future additions (unchanged).

1.2 Background

The WFDOE (SNL, 2014) provided part of the technical basis for the DOE (2014) assessment of disposal options. The WFDOE (SNL, 2014) work provides the starting point for information consideration of a DGRDMSH. Both the wastes and waste forms considered in the previous work, as well as summaries of disposal concepts evaluated, are given below.

1.2.1 Waste Types and Waste Forms Considered

The scope of the waste in the WFDOE (SNL, 2014) includes all existing SNF from commercial, defense, and research reactors, and SNF from reasonably foreseeable operations of existing reactors (projected to exist in year 2048). That study scope also includes existing HLW forms (e.g., vitrified HLW at Savannah River and West Valley) and waste forms projected to be generated in the future from existing process waste (e.g., projected vitrified HLW from HLW at Hanford, Savannah River and the Idaho National Laboratory). In addition, the WFDOE (SNL, 2014) considers both direct disposal of waste forms that are not currently planned for disposal without further treatment (e.g., calcine waste at the Idaho National Laboratory) and alternatives to planned treatments. The WFDOE (SNL, 2014) acknowledges existing

plans, commitments, and requirements where applicable, but evaluates options for disposal based primarily on technical, rather than programmatic or regulatory constraints.

The WFDOE (SNL, 2014) waste inventory was categorized into 43 different "waste types." For the purposes of that study as well as this one, a "waste type" is defined as the currently existing materials (in whatever form, abundance, and location they occupy) that are to be (or be processed into) some waste form to be disposed in a deep geologic repository (e.g., Hanford tank wastes; commercial spent fuels, HLW glass). In the WFDOE (SNL, 2014), a "waste form" is the end-state material as packaged that is to be disposed of in a deep geologic repository. Some "waste types" may have more than one possible alternative "waste form" depending on the processing needed, whereas "waste types" that require no processing other than packaging may equate to a single "waste form." In this report, the waste form includes its canister, but may not include waste packaging for disposal purposes (i.e., the disposal form).

Considering the alternative treatment options for some of the 43 waste types, the WFDOE (SNL, 2014) defined 50 waste forms, which were aggregated into the ten "waste groups" (Table ES-2; SNL, 2014) with similar disposal characteristics such as expected post-closure degradation behavior, radionuclide inventory, thermal output, physical dimensions, chemical reactivity, packaging of the waste form, and safeguards and security needed for handling, transporting, and disposing of the waste form in the context of the disposal concepts. The aggregation into waste groups allowed a high-level identification of waste forms that have unique qualities in any one of the disposal characteristics listed above. The 10 groupings listed in Table ES-2 of SNL (2014), except those groups consisting solely of CSNF (WG1 and WG2), are utilized below in this study to consider information needs regarding features DGRDMSH concept.

Major assumptions and considerations used in the WFDOE (SNL, 2014- and used here) include the following:

- HLW and SNF considered were restricted to existing materials and those materials that can be reasonably expected to be generated by existing or currently planned facilities and processes.
- The inventory of HLW and SNF was intended to include existing materials in the U.S. requiring deep geologic isolation, and was based on the best available information.
- Technologies under consideration, including both for waste treatments and disposal concepts, are limited to those that can be deployed in the near future.
- Programmatic constraints, including legal, regulatory, and contractual requirements, were acknowledged where applicable, but were not considered in the technical evaluations, consistent with the goal of the study to provide technical input to strategic decisions. For example, the identification of wastes requiring deep geologic isolation was based on consideration of overall risk, rather than on specific U.S. legal and regulatory requirements.
- Evaluations were primarily qualitative, and are based in large part on insights from past experience in waste management and disposal programs in both the U.S. and other nations.

The assumptions above apply also to the present work, which builds off the WFDOE (SNL, 2014) but focusses solely on the disposal of a subset of the DHLW and DSNF. As such, the CSNF aspects assessed previously are not included in the evaluation of a DGRDMSH. As well, only a subset of the DOE-managed naval SNF (the lower thermal load portion of the waste form) would likely be included in this repository concept (DOE, 2015). This work is to assess the inventory included for analyses of a DGRDMSH, delineate any needed changes to repository concepts features relative to concepts that include CSNF, layout the preliminary structure of the on-line waste library (OWL) to manage the waste types/forms information (including any potential additions to the DGRDMSH inventory to be added to the previous list—see Tables C-1 and ES-1, SNL, 2014), and develop constraints on waste form post-closure degradation performance.

The set of disposal concepts used in the WFDOE (SNL, 2014) work is the same as that identified by DOE's UFDC as a primary target for further research and development. These same disposal concepts are presented here as a useful and representative, rather than comprehensive, set of concepts, and are also the concepts being used in this DGRDMSH work.

1.3 Disposal Concepts Considered

The WFDOE (SNL, 2014) considers the four representative disposal concepts selected for further research and development activities by the DOE Office of Nuclear Energy's (DOE-NE) UFDC (Rechard et al., 2011). These four concepts are mined repositories in three geologic media—salt, clay/shale rocks, and crystalline (e.g., granitic) rocks—and deep borehole disposal in crystalline rocks. As summarized by Rechard et al. (2011), selection of these four concepts begins with the observation that options for disposal of SNF and HLW have been evaluated in multiple nations for decades, and deep geologic disposal was recognized as early as the late 1950s to be the most promising approach (National Academy of Sciences Committee on Waste Disposal 1957). By the 1980s, the U.S. waste management program concluded that multiple geologic media had the potential to provide robust isolation, and that conclusion remains valid today. Experience gained in waste management programs in other nations reinforces that conclusion (NWTRB, 2009). For example, Finland has been granted a construction license and Sweden has a license application pending for proposed mined repositories for SNF in crystalline rock. The U.S. has an operating repository in salt for transuranic (TRU) waste at the Waste Isolation Pilot Plant (WIPP), and Germany has extensive experience with the design of a mined repository for SNF and HLW in salt (e.g., BMWi, 2008). France, Switzerland, and Belgium have completed detailed safety assessments for proposed SNF and HLW repositories in clay and shale media. Although no nations are currently planning deep borehole repositories, the concept has been evaluated in multiple programs since the 1970s, and remains viable for waste forms small enough for emplacement in boreholes (e.g, Brady et al., 2009).

Variants of the four primary geologic disposal concepts are also considered where appropriate. For example, as described by Hardin et al. (2012a), some mined repository concepts can be implemented in an open mode. That is, an open mode that includes active ventilation during the operational period. Choice between an open mode implementation versus a closed mode with early emplacement of backfill would depend, in part, on thermal load management needs.

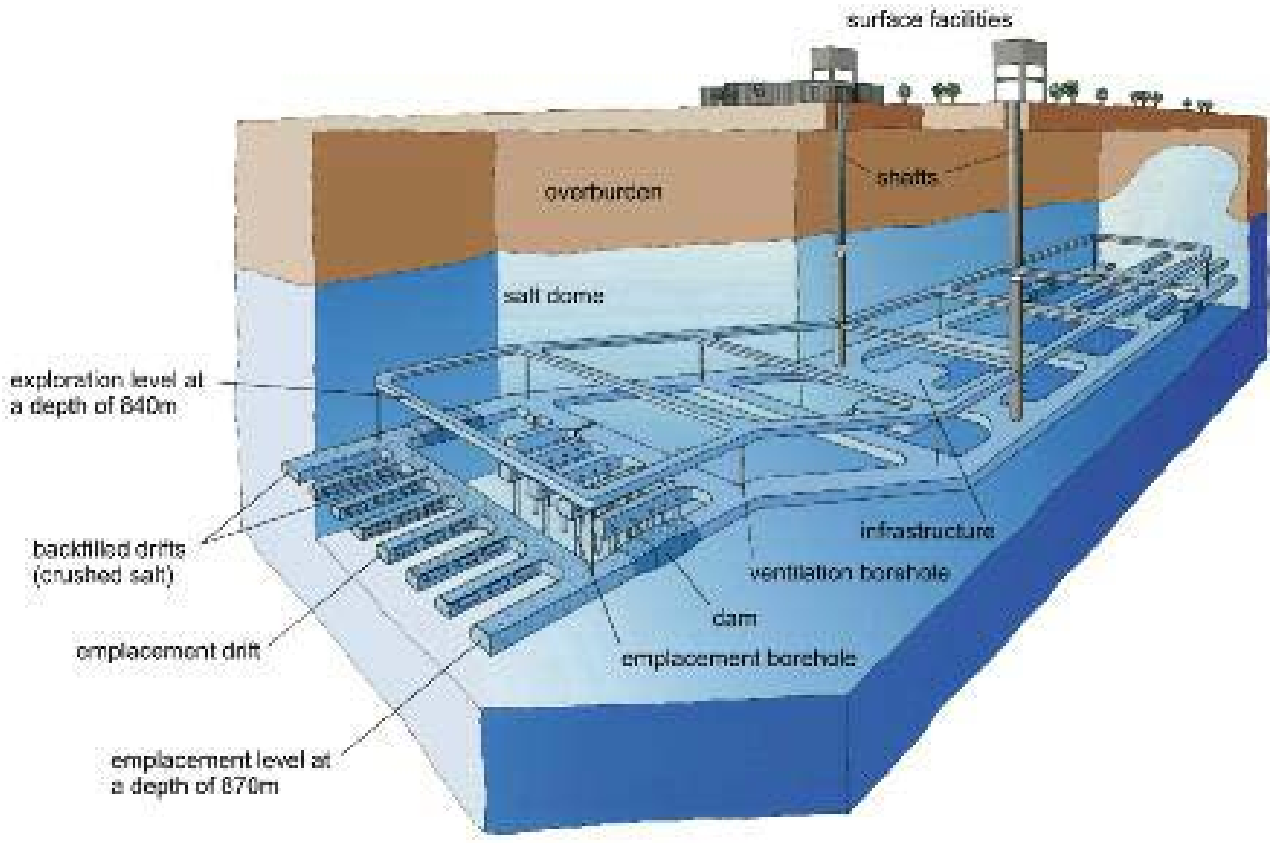
Other geologic disposal concepts have been proposed and are potentially viable. For example, Canada is currently evaluating a mined repository for intermediate-level radioactive waste in carbonate rocks (NWMO, 2011) and the U.S. has evaluated a potential mined repository concept in volcanic tuff (DOE, 2008).

1.3.1 Mined Repositories in Salt

The primary information sources for mined repositories in salt come from the U.S. WIPP program (DOE 1996b; DOE, 2009) which is an operating repository accepting and emplacing defense-related transuranic waste, and the proposed German repository at Gorleben (e.g., BMWi 2008). Figure 1-1 shows a representative design for a salt repository. Emplacement of waste would occur in horizontal tunnels (referred to as "drifts" in mining terminology), or in sub-horizontal boreholes drilled along the drifts, at depths between 500 and 1000 meters below the land surface. As proposed, access to the emplacement areas would be by hoists in vertical shafts. Primary isolation would be provided by the essentially impermeable nature of intact salt.

Other attributes of salt relevant to repository design and waste disposal include relatively high thermal conductivity, which allows conductive transfer of heat away from the waste, relatively low water content, and visco-plastic mechanical response. This visco-plastic behavior of salt allows creep behavior under differential stress that causes salt to slowly flow. This slow flow leads to closing and healing of fractures and open spaces, allowing for the use of access shaft seal systems that will compact under lithostatic load to achieve extremely low permeability. The salt creep will tend to close emplacement regions relatively

rapidly on geologic time scales (perhaps within decades) after waste emplacement, potentially complicating the implementation of extended periods of ventilation without significant drift support. However, the relatively high thermal conductivity of salt significantly reduces the need for remove heat with ventilation and allows more flexibility of thermal loading to meet temperature limits, which are generally higher than for concepts that include an in-drift clay backfill/barrier.



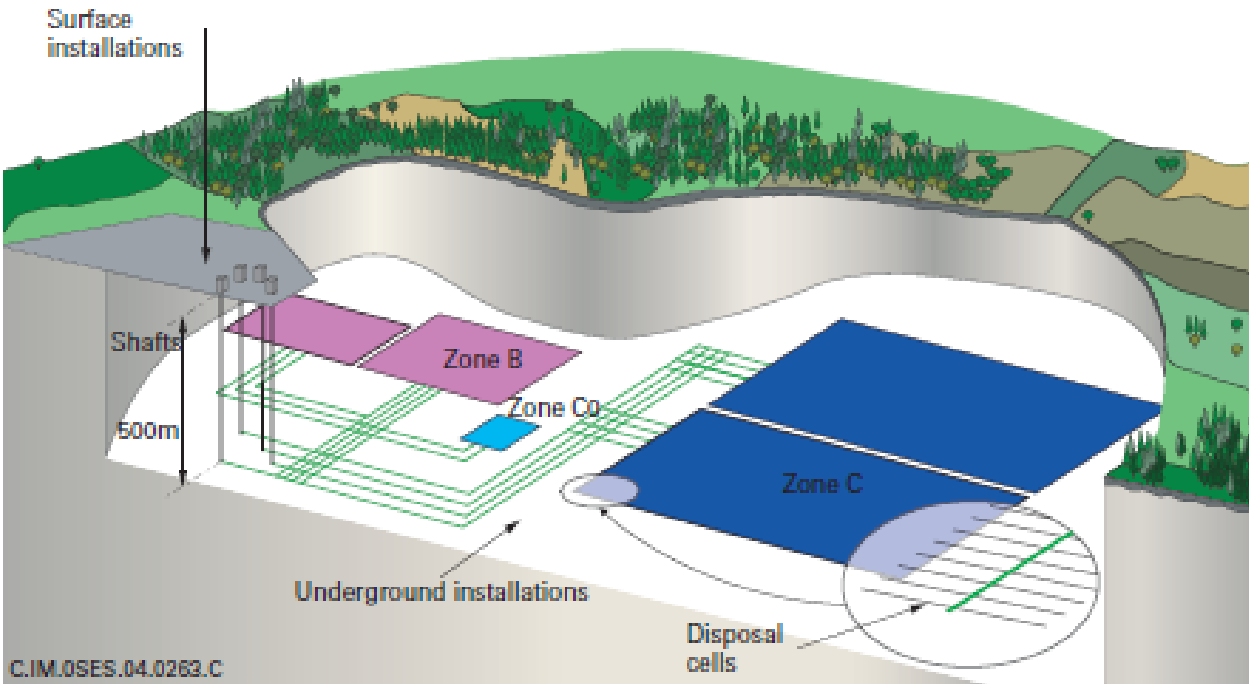
Source: BMWi, 2008, Figure 15.
Figure 1-1. Schematic representation of a mined repository in salt

There are a couple major end-member types of salt systems being examined. One system is bedded salt, which occurs in horizontal layers of nearly pure sodium chloride originally deposited from shallow, evaporating salt-saturated seawater. Bedded salt can contain both small quantities of trapped brine and interbedded layers of clays and other evaporite minerals such as anhydrite (calcium sulfate). The second system is domal salt, which has moved from its original bedded form into dome-shaped structures due to visco-plastic flow over geologic time. Domal salt tends to have less water, and fewer impurities and intact interbeds than bedded salt, but domal salt is more restricted geographically. To the extent that sufficient water may be present to saturate a repository waste emplacement region in either bedded or domal salt, it will form salt-saturated brine and chemical conditions will be reducing. Any free oxygen introduced would be consumed by corrosion of metal in the waste packages or other engineered systems. Because of the essentially impermeable nature of the salt host rock and the very low potential for advective transport of radionuclides away from the disposal region, little or no reliance for the long-term performance is given to the waste form or the waste packaging.

1.3.2 Mined Repositories in Clay and Shale Rocks

The primary information sources for mined repositories in clay and shale rocks come from the French, Swiss, and Belgian national programs, each of which is evaluating disposal in argillaceous host rocks

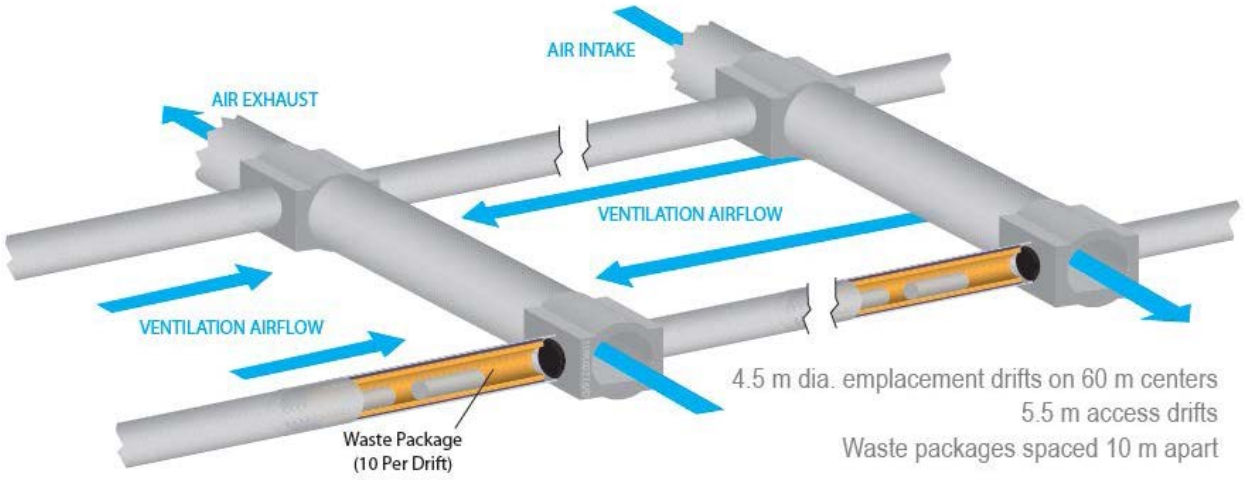
(ANDRA, 2005a, 2005b; NAGRA, 2002; ONDRAF/NIRAS, 2011). Figure1-2 shows a representative design for a mined repository in clay or shale. Emplacement of waste would occur in horizontal holes bored laterally from access drifts at a nominal depth of 500 m below the land surface. As proposed, access to the underground emplacement region would be by hoists in vertical shafts. Isolation would be provided by long-lived waste packages, waste forms that are long-lived in the chemically reducing environment, and by the extremely slow rate of diffusion through the low-permeability host rock. Sorption of radionuclides on clay minerals within the backfill and the host rock would effectively prevent long-term releases of all but the most mobile radionuclides, such as ^{129}I and ^{36}Cl, and long-term releases of these species would remain very low because transport is diffusion dominated.



Source: ANDRA, 2005b.

Figure 1-2. Schematic representation of a mined repository with various waste form zones in argillaceous rock

Argillaceous rocks display a broad range of physical properties from weakly indurated clays capable of visco-plastic flow (e.g., the formation being evaluated for a repository in Belgium), to strongly indurated and massive argillites such as that being evaluated for disposal in France, to laminated shales common in many sedimentary basins, especially in the U.S. All these lithologies are characteristically extremely low permeability, which will lead to diffusion-dominated release pathways and contain an abundance of clay minerals that contribute to radionuclide sorption. All argillite varieties have lower thermal conductivity than salt. Mined repository concepts in clay and shale rocks must be designed accordingly to accommodate thermal loads. The most widely adopted approach to manage decay heat in clay/shale rocks is to use relatively small waste packages (up to 4 spent fuel assemblies per package) and to space the emplacement drifts relatively far apart. Hardin et al. (2012a) evaluated the potential for increasing the thermal loading capacity of a mined repository in shale by considering an “open-emplacement” design concept in which emplacement drifts remain completely open to allow extended ventilation to remove decay heat, as illustrated in Figure 1-3. Backfilling and sealing of access drifts would occur at repository closure, with the option of leaving the emplacement drifts open permanently, without backfill, if the operational constraints so dictate. Some argillites would require ground support for maintaining the openings for long durations.

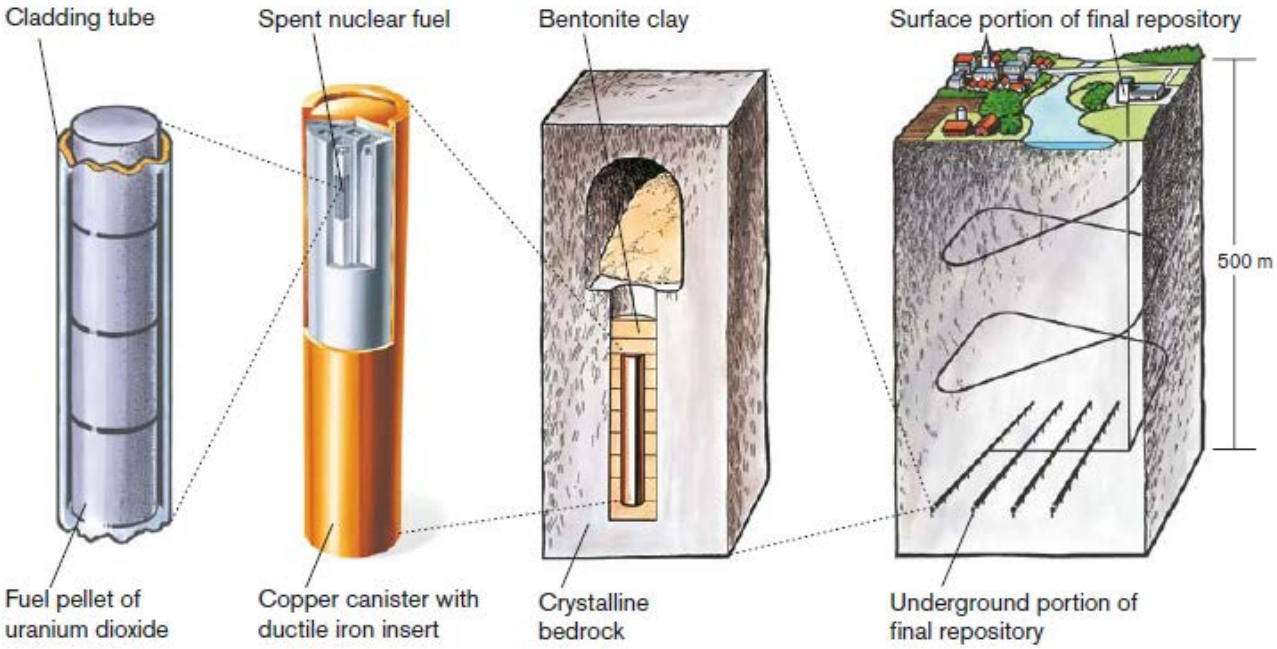


Source: Hardin et al., 2012a, Figure 1.5-3.
Figure 1-3. Schematic of shale open (i.e., no backfill) emplacement repository concept

1.3.3 Mined Repositories in Crystalline Rock

The primary sources of information for mined repositories in crystalline rock come from the Swedish and Finnish programs (SKB, 2011; Posiva Oy, 2013), which are in the process of seeking licenses to construct and operate facilities for the permanent disposal of SNF. Other nations are also conducting research on mined repositories in crystalline rock, including Canada, Japan, Korea, China, and the Czech Republic. Figure 1-4 shows a representative disposal concept developed for the Swedish program. Wastes (SNF in this example) are emplaced in vertical boreholes drilled in the floor of horizontal drifts at a nominal depth of 500 m below the land surface. Alternative design options call for emplacing waste in horizontal tunnels drilled into the sides of the access drifts. In either case, access to the waste disposal region is by an inclined ramp in this concept, rather than vertical shafts and hoists.

Generally, crystalline rock repository systems provide isolation by long-lived corrosion-resistant copper waste packages, by the durability of the uranium oxide SNF waste form, and by the high sorption capability of the bentonite clay buffer that would surround the waste packages in the Swedish repository concept (SKB, 2011). Both the copper waste package and SNF planned for disposal in the Swedish repository are more durable under chemically reducing conditions. Other reduced waste forms (e.g., metallic fuels) would be closer to their equilibrium conditions and would corrode more slowly than in oxidizing environments. Still other waste forms (e.g., HLW glass) may not benefit from the reducing environment as much in terms of waste form lifetimes in such a disposal concept, but many radionuclide solubility limits would be very low and substantial performance would be expected based on the waste package lifetime and the bentonite backfill capabilities. Open and interconnected fractures, which can occur in crystalline rocks at these depths, have the potential to provide pathways for advective transport of radionuclides from the repository to the near-surface environment if the near-field barriers were to be breached. Design concepts therefore avoid emplacement in areas intersected by fractures and surround waste packages with a low-permeability bentonite clay buffer (SKB, 2011; Posiva Oy, 2013).



Source: SKB, 2011, Figure S-1.
Figure 1-4. Schematic representation of a mined repository in crystalline rock

Because bentonite undergoes durable physical changes at elevated temperatures, crystalline repository concepts generally have defined a peak temperature constraint at the waste package surface of approximately 100°C. Existing design concepts meet this constraint with relatively small waste packages, accommodating four spent fuel assemblies per package.

As discussed by Hardin et al. (2012a; 2013), alternative design concepts for mined repositories in crystalline (or other hard) rocks can address thermal load management issues by emplacing waste in large tunnels or vaults that remain open, without backfill, for extended periods of ventilation prior to permanent closure. In unsaturated rocks, above the water table, the limited availability of water for advective transport has the potential to allow permanent disposal without backfill emplacement, although the oxidizing conditions in an unsaturated environment will require alternative robust designs for waste packaging and could allow for more rapid degradation of UO_2 waste forms once exposed. The same would be true for other reduced waste forms, especially metallic waste forms, which would also have higher potential for exothermic oxidation phenomena. Additionally, the HLW glass waste form may undergo different degradation mechanisms in a humid environment versus saturated conditions (Cunnane et al., 1994). In saturated environments, emplacement of a clay backfill will be desirable after extended ventilation, to reduce the potential for advective transport away from the waste packages.

1.3.4 Deep Borehole Disposal in Crystalline Rock

Deep borehole repositories for permanent isolation of radioactive materials has been proposed and investigated intermittently for decades in the U.S. and other nations (e.g., O'Brien et al., 1979; Halsey et al., 1995; MIT, 2003; Nirex, 2004; Åhäll, 2006; Brady et al., 2009). The earliest proposals for deep borehole disposal considered direct disposal of liquid HLW from reprocessing (National Academy of Sciences Committee on Waste Disposal 1957; Hess, 1957). Subsequent analyses have considered disposal of solid wastes of various types, including glass HLW forms and surplus weapons-grade plutonium (e.g., Halsey et al., 1995). Published analyses to date have concluded that the overall concept has the potential to offer excellent isolation, but deep borehole disposal of solid wastes has not been implemented in any nation. This is due in part to the lack of applicable mining technologies at the time that national policy

decisions were made, and in part because of questions on the feasibility of retrieving waste from deep boreholes. Advances in drilling technologies over the last several decades (Beswick, 2008) suggest that the construction of deep boreholes should not be viewed as a greater technical challenge than deep mines. The technical advances also suggest that retrieval, if required, should not be viewed *a priori* as unachievable. Retrieval of wastes is likely, however, to remain more difficult from deep boreholes than from most mined repository concepts. If permanent disposal is not intended, deep boreholes should not be a preferred option.

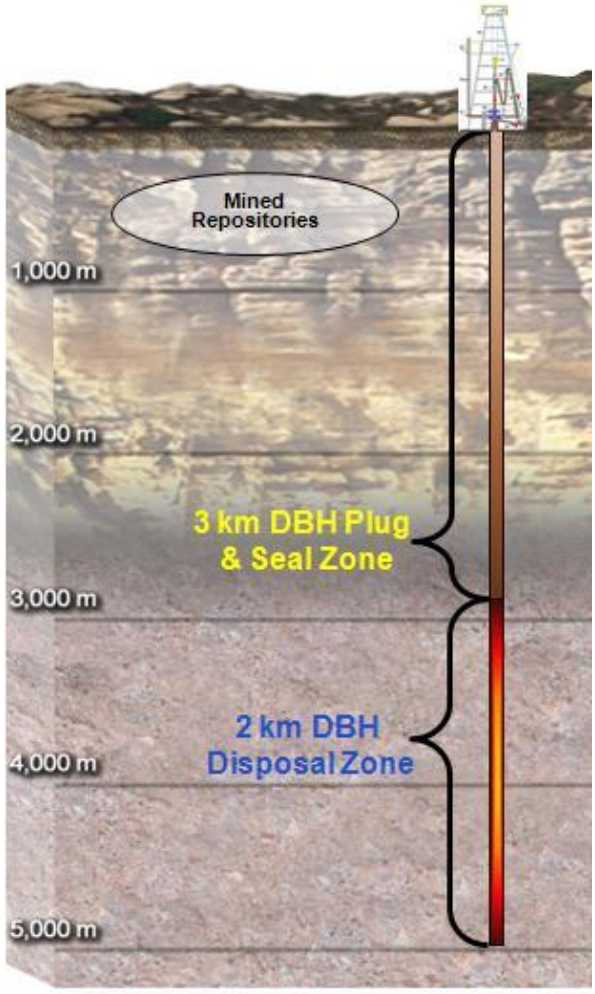


Figure 1-5. Schematic representation of a deep borehole repository

As described by Arnold et al. (2011; 2012) and illustrated in Figure 1-5, a representative reference design for borehole disposal calls for drilling a borehole to a total depth of approximately 5 km, with at least 3 km of the lowest portion of the hole penetrating crystalline rock. The hole would have a nominal diameter of 0.43 m at depth (requiring larger hole diameters at shallower depth), to accommodate emplacement of waste canisters with maximum external diameters of 0.30 m. Packages would be up to 4.2 m in length. The borehole would be lined with steel casing after drilling, to facilitate emplacement of waste packages vertically in the lower 2 km of the borehole. Following emplacement, the casing would be removed from the upper portion of the hole, and seals of alternating sections of concrete and compacted bentonite would be emplaced in the hole.

The deep borehole disposal (DBD) reference design in Arnold et al. (2011) is based on a maximum borehole diameter of 0.43 m (17 in.) at a depth of 5 km because it is expected to be reliably achievable in crystalline basement rocks with currently available, commercial drilling technology. There are no known technical issues that present unreasonable barriers to drilling to this diameter at depth. Land-based drill rigs with the necessary capacity to drill and complete a 17-in. borehole to 5 km depth are commercially available with seven companies in the U.S. operating such rigs at the time of the report. Confidence in the ability to drill and complete a borehole decreases with increasing depth and increasing borehole diameter. Future developments in technology may increase capabilities at such depths.

Similar to mined repositories, the safety of DBD isolation comes from the natural system properties (crystalline basement in this case), but with a disposal zone significantly deeper than the depths proposed for mined repositories. The waste isolation would be provided primarily by the extremely low permeability of crystalline rocks at these depths, by the long pathway for diffusive transport upward through the long borehole seal system, and by the saline, rock-dominated, isolated fluids at depth. Low permeability of the host rock and the absence of open fractures would need to be verified through borehole testing before waste was emplaced; testing would also confirm the absence of low-salinity or young groundwater.

Because of the primary reliance on the geologic barriers and secondary reliance on a long seal system, little long-term performance would be required from the waste packages, which could be constructed of standard drilling-industry steel pipe. The strongly reducing environment in the deep portion of the hole would stabilize reduced redox-sensitive species in the waste and would greatly limit the mobility of many radionuclides because of low radionuclide solubility limits under these geochemical conditions. Other reduced waste forms (e.g., metallic) would be closer to their equilibrium conditions and would corrode more slowly than in oxidizing environments. Still other waste forms (e.g., HLW glass) may not benefit from the reducing environment as much in terms of waste form lifetimes in such a disposal concept, but many radionuclide solubility limits would be very low and substantial performance would be expected from the bentonite backfill capabilities.

For the purposes of evaluating a DGRDMSH disposal concept, only the three mined geologic repository concepts in crystalline rocks, argillite, and salt are considered in detail for the inventory. This is primarily because the DBD concept is currently being considered only for a subset of small waste forms, which could be disposed in deep boreholes with diameters much less than 17-in. The DBD concept provides a potential disposal pathway for some alternate waste forms, which allow flexibility for the disposal mission implementing any mined disposal concept. The work described below assesses aspects of mined repository concepts that may need to be modified in a DGRDMSH repository relative to each counterpart concept for a repository that includes CSNF.

2. INVENTORY INCLUDED IN A DGRDMSH AND CONSIDERATIONS OF RESULTANT DISPOSAL CONCEPTUAL VARIATIONS

The overarching conclusions of the WFDOE (SNL, 2014) are that:

- the full inventory of DOE-managed and commercial HLW and SNF is diverse, and DOE has a broad range of viable options for disposing of it; and
- the selection of preferred options will involve policy and programmatic considerations outside the scope of the report, and will be influenced by, and may help inform decisions about, multiple factors that could include future storage and packaging of commercial SNF, treatment and packaging of existing DOE wastes, and progress in repository siting.

All of the disposal concepts evaluated in that study have the potential to provide robust long-term isolation for specific wastes. In addition, each of the three mined repository concepts could accommodate essentially all of the identified waste groups (the only exception was for direct disposal of untreated sodium-bonded SNF, for which information is insufficient to support evaluation for disposal in any geologic disposal concept). It was also concluded that deep boreholes are feasible for disposal of small waste packages and provide flexibility to any disposal strategy. Additional generic and site-specific R&D is needed before any disposal options can be implemented, although no recommendations were made with respect to specific R&D activities.

The results of the WFDOE (SNL, 2014) study indicate that some disposal options for mined repository concepts may provide greater flexibility or fewer challenges than others. Specifically:

a) Salt provides greater flexibility for disposal of heat generating wastes because of the high thermal conductivity and high temperature limit. Disposal in this media provides greater confidence in estimates of long-term performance because it limits radionuclide transport (low permeability) and reduces the reliance on the waste form and waste package lifetimes. The relative lack of water and the high cross-section of chlorine for capture of thermal neutrons make it easier to address criticality concerns. In some cases, it may be appropriate to directly dispose of some untreated waste types, potentially reducing cost and risks associated with waste treatment. The operational experience at the WIPP provides additional confidence in this disposal concept.

b) Clay/Shale is a disposal media with a significant amount of world-wide experience and it showed strong results as a disposal option for most waste groups with respect to most metrics. It is an attractive disposal option because it limits far-field radionuclide transport (low permeability and high sorption) and, therefore, reduces the reliance on the waste form and waste package lifetimes, compared to a crystalline disposal concept. However, compared to salt, there is more reliance on source-term performance and thermal constraints are more stringent.

c) Mined repositories in crystalline rocks may offer operational advantages because of the rock strength, which allows easy maintenance of the robust openings for long periods without substantial support. This provides the potential flexibility of possible ramp access. However, for fractured crystalline systems, high reliance on clay barriers immediately surrounding the waste package poses additional challenges for high thermal loads that may degrade such barriers. Because of the need for robust performance of the source-term, confidence in system performance may be directly dependent on very conservative thermal management.

In addition to the WFDOE (SNL, 2014), a number of previous studies have evaluated the full inventory for storage and transportation purposes (Carter and Leduc, 2013; Carter and Vinson, 2014) and the more restricted inventory, smaller volume of generally cooler waste forms, for a DGRDMSH (Carter et al., 2012, 2013). These studies also inform the analyses done in FY2016 for a DGRDMSH within the UFD

Campaign (Sevougian et al., 2016). The previous DGRDMSH inventory estimates (Carter et al., 2012, 2013) were synthesized and integrated by Wilson (2016) to provide a preliminary inventory for use those UFD Campaign scoping analyses. It should be noted that the DGRDMSH inventory defined in Wilson (2016) was a preliminary one for use in the prototype analyses (Sevougian, 2016). The preliminary DGRDMSH inventory (covered in Section 2.1) is updated/expanded in this report (Section 2.3). Given the major characteristics of the DGRDMSH inventory (e.g., cooler, smaller total volume versus one that includes hotter waste forms like commercial SNF), the broad generalities for disposal concepts defined above are assessed for differences to system performance reliance on natural features and for variations on design concepts. Discussion is given also for potential additional wastes/waste forms for future inventories considered for deep geologic disposal (Section 2.2).

2.1 DOE-Managed HLW and SNF Inventory Included in Inventory for DGRDMSH Analyses

This section provides an overview of the inventory included for DGRDMSH engineering and system performance analyses for FY2016, and the FY2017 updates to that (Section 2.3). The included set of materials for the DGRDMSH inventory may change in the future based on the designation decisions made by DOE and/or updated technical information about the wastes/waste forms. The included inventory in this report is only for use in analyses of a potential generic DGRDMSH, and there is no intent to indicate how to classify any of the wastes/waste forms included in this.

Wilson (2016) provides the preliminary inventory for the analyses of a DGRDMSH and includes both DHLW and DSNF waste canister counts and thermal information (Tables 2-1, and 2-3 thru 2-6 from Wilson, 2016). Wilson (2016) describes each waste form in terms of both average radionuclide content and average thermal output evolution for each thermal category. That tabulation includes canister counts and ranges of thermal characteristics for each DHLW and DSNF waste form considered (Wilson, 2016). For the preliminary DGRDMSH inventory assembled in this report, the various specific DSNF types contained in the ~2485 DSNF canisters (see Table 2-1 from Wilson, 2016) are listed in Appendix A. The included DHLW canister counts are given in Wilson (2016) in Tables 2-3 thru 2-6, respectively, for Savannah River glass (7824 canisters), Hanford glass (11,800 canisters), INL hot isostatic pressed (HIP) calcine (4391 canisters), and Hanford vitrified Cs/Sr capsules (340 canisters—see SNL, 2014 also).

The major updates in FY2017 (Section 2.3) to the preliminary DGRDMSH inventory include (a) added the cooler naval SNF waste packages (~13 naval SNF canisters based on ~1000W per canister as thermal threshold—see Figure 3 of DOE, 2014), (b) added the 34 glass canisters of “German” (generated for FRG testing) glasses (SNL, 2014), (c) added the planned waste form for HIP calcine in ~320 canisters (~5.5 ft diameter by ~15 ft height, naval canisters; SNL, 2014); and (d) revised the list of DSNF materials included in the inventory based on any applicable DOE decisions and/or new technical data. Though most of these updates are relatively small from the standpoint of inventory mass, they may have some implications for thermal analyses (naval SNF and FRG glasses) and handling considerations (naval SNF and planned calcine waste forms).

2.1.1 Discussion of Current Included Inventory for DGRDMSH Analyses versus Previous Inventory Data Sets

Major variations in the inventory considered for any particular repository concept may influence more than just the total radionuclide content of that repository. The inventory also affects the total thermal input, the temporal thermal distribution, the numbers of packages to be handled, and the ranges of size/mass of packages to be handled in a repository concept. A comprehensive compilation and analysis of waste form information was conducted for the Safety Assessment Report (SAR) prepared in support of the Yucca Mountain (YM) Project (DOE, 2008). The SAR inventory slated for a repository at YM included a large portion of CSNF, only ~46% of the SAR projected DHLW canisters, and nearly all of the DSNF (including all of the projected ~400 naval DSNF canisters).

The waste inventory for DGRDMSH analyses listed at the end of Section 2.1, differs from the SAR inventory in primarily three ways.

- No commercial SNF (CSNF);
- Larger quantities of the various DHLW Glass included; and
- Smaller quantities of DSNF included (this has been updated in FY2017, but still does not include all DSNF and includes only the ~13 coolest naval spent fuel waste packages)

A DGRDMSH may not include any CSNF (DOE, 2014; 2015). If this is the case, it results in a very large reduction in the total radionuclide content and the thermal mass relative to a repository that includes CSNF (e.g., the YM SAR inventory).

The DGRDMSH inventory from Wilson (2016) includes about 2.5 times as many DHLW canisters as was planned for the YM repository[a]. (The Nuclear Waste Policy Act placed a legal limit on the amount of radioactive waste (in metric tons heavy metal – MTHM) that could be disposed of in a YM repository. A portion (4,667 MTHM) of this limit was allocated to DHLW glass (DOE, 2008, Table 1.5.1-1). The SAR projected a total of 21,228 DHLW canisters to be delivered to the YM site from Hanford, Savannah River Site, and Idaho National Laboratory (DOE, 2008, Section 1.5.1.2.1.2). Of this total, the SAR projected that only ~9,300 DHLW canisters would be included in the YM inventory (DOE, 2008, Table 1.5.1-1; see also SAR section 1.5.1.2.1.1). At the time that the SAR was completed this amount of DHLW represented less than half of the projected DHLW inventory. The current DGRDMSH inventory includes a higher number of total projected DHLW canisters (~24,400) than the SAR projections, with the specific differences between those two inventory projections discussed below.

The SAR waste inventory included all projected DSNF at that time (DOE, 2008, Section 1.5.1.3). Those projections included 65 MTHM for naval SNF in ~400 naval SNF canisters, and 2,268 MTHM for other DOE-managed SNF (SAR Table 1.5.1-1). It is noted that naval fuel that may be generated after 2035 was not included in that 65 MTHM specification from the SAR, which is still our working estimate. The current estimate of the DSNF inventory is a total of 2,336 MTHM (DOE 2014, Section 2.2.1). The current DGRDMSH inventory includes most, but not all of the DSNF, and has been updated in FY2017 to include the coolest naval SNF canisters (<~1000W). A more detailed discussion is presented below on the types of DSNF included in the DGRDMSH inventory and the differences from the SAR inventory.

2.1.1.1 DGRDMSH Included DHLW Inventory Compared to SAR Inventory

High-Level Waste (HLW) has been generated as a by-product of reprocessing SNF. Currently these wastes are stored primarily as liquid tank wastes at DOE facilities at Hanford, Savannah River, and INL (SNL, 2014). Processing of the various DHLW wastes into their final planned waste forms has not been uniform at the various sites. As a result, the wastes currently have different physical characteristics depending on the details of the processes used, or planned to be used, for a given waste. These characteristics may be quite different for the existing waste versus the planned waste forms (SNL, 2014).

[a] Note that this means the SAR projected inventory of total DHLW canisters is only about 40% of the DGRDMSH inventory from Wilson (2016). However, the SAR projected inventory of total DHLW canisters is about 46% of the estimated total number of canisters for DHLW given in SNL (2014)— about 20,340 canisters. The difference between Wilson (2016) and SNL (2014) relates mainly to ~1200 more Hanford glass canisters and ~4000 more HIP calcine canisters (smaller-sized, alternate waste form) projected in Wilson (2016) versus SNL (2014).

The DHLW is grouped here into the several following categories:

- Savannah River tank waste, which is currently in the process of being vitrified into glass logs;
- Savannah River existing vitrified glass logs;
- Hanford tank waste, which is planned to be vitrified into glass logs;
- Calcine waste at Idaho, which is planned to be hot isostatic pressed into a glass ceramic waste form (note that direct disposal of untreated calcine was being considered potentially for Deep Borehole Disposal: SNL, 2014; DOE, 2014);
- German (FRG) glass logs stored at Hanford, which have no further planned treatment (added to DGRDMSH inventory in FY2017);
- Sodium bearing waste at Idaho, which is to be treated by fluidized bed steam reforming (to be added to DGRDMSH inventory the future); and
- Cs and Sr capsules at Hanford, which are planned to be vitrified (note that direct disposal of these untreated capsules was being considered potentially in the Deep Borehole Disposal concept: SNL, 2014; DOE 2014).

The number of waste canisters that will ultimately be available to be disposed for each these unprocessed wastes is uncertain. In some cases, the planned waste form pathway has changed, which leads to further variation in estimated numbers of canisters for a projected waste form. For example calcine waste at Idaho National Laboratory (INL) was planned to be vitrified for delivery to YM in ~2 ft diameter by ~10 ft height canisters (DOE, 2008, Section 1.5.1.2.1.2), but is now planned for hot isostatic pressing (DOE Record of Decision, 75 FR 137)[b]. Further, some projections include additional smaller volume wastes (e.g., sodium bearing wastes, German glass canisters), whereas others do not. All of these aspects have led to some variability in the projected canister totals in different reports (e.g., DOE, 2008; Carter et al., 2012; SNL 2014; Wilson, 2016), so it should be kept in mind that the values are approximate, and that projected canister counts should be explicit regarding which wastes are included to facilitate comparisons.

Table 2-1. Comparison of numbers of Projected HLW Canisters from the full received (though not to be disposed) inventory from the SAR (DOE, 2008) for Yucca Mountain and the current estimates from Wilson (2016).

Projected HLW Canisters		
Site	YM SAR Projection[a]	Current Projection[b]
Hanford	13,205 canisters	12,140 canisters
Savannah River Site	6,833 canisters	7,824 canisters
Idaho National Lab	1,190 canisters[c]	4,391 canisters[c]

a. These values represent best estimates of projected numbers of canisters that were to be delivered to the YM site at the time of the SAR (DOE, 2008), however only about 46% of them were to be disposed with the remainder slated for a second repository.
b. These estimates were developed by Wilson (2016) for the DGRDMSH inventory in support of preliminary design thermal and post-closure safety calculations for FY2016, and are based on current planning assumptions for waste treatment.
c. The estimate for Idaho National Lab HLW from the SAR included vitrification of calcine waste, whereas that from Wilson (2016) includes the assumption of an alternative calcine waste form which would be packaged for disposal in standard ~2 ft x ~10 ft cylindrical “glass” canisters.

[b] The baseline canister dimensions for the planned HIP calcine waste form are ~5.5 ft diameter by ~15 ft height (Kluk et al., 2011), whereas the HIP calcine from Wilson (2016) includes the assumption of an alternative waste form packaged for disposal in a standard ~2 ft x ~10 ft cylindrical “glass” canister.

The Yucca Mountain SAR (DOE, 2008, Section 1.5.1.2.1.2) included projections based on the best information available at the time. Wilson (2016) developed the DGRDMSH inventory for supporting design/engineering analyses, including thermal evolution, and safety assessments of a DGRDMSH. The two sets of projected canister values are presented in Table 2-1. It can be seen that there is some variability between the estimated values for Hanford and Savannah River Site DHLW glass canister projections, but the largest difference is in the values for the INL canisters. The difference in numbers of INL canisters is largely explained by the change to the planned waste form from vitrified calcine (SAR) to HIP calcine (current disposal pathway).

2.1.1.2 DGRDMSH Included DOE-managed SNF Inventory Compared to SAR Inventory

DOE production reactors, as well as foreign and domestic research reactors, have produced SNF with a very large range of physical characteristics. The spent fuel database (SFDB) for DOE managed SNF contains hundreds of entries with a wide range of fuel types that are managed by DOE currently, or are to be received by DOE at a later date from, for example, foreign research (DOE, 2007). Early SFDB work for the YMP SAR led to a grouping system that categorized the total DSNF inventory into 34 groups of DOE-managed SNF based in part on fuel matrix, cladding, cladding condition, and enrichment. These 34 DSNF fuel groups were the starting point for work leading up to the license application (LA) that DOE submitted to the NRC (DOE, 2008). The naval SNF, for example, is DOE SNF Group 32, separate from other DSNF. This DSNF grouping has proven to be very useful and is still in use today (DOE, 2007; SNL, 2014).

The canister counts and thermal output of the included inventory of DSNF for FY2016 DGRDMSH analyses are given in Wilson (2016). Appendix A presents a detailed tabulation of DSNF items that are included in this DGRDMSH inventory. The Appendix A table is organized using the 34 DSNF groups. The information was extracted from the supporting data for the inventory and thermal characteristics reported by Wilson (2016). The right hand column of the table identifies each DSNF item by name. The left hand column identifies the DOE fuel group for each item, the mass (MTHM) of items within the fuel group, and the projected/estimated number of waste containers within each DSNF group. There are no naval SNF containers included in the FY2016 preliminary inventory, but the coolest naval SNF containers ($\lesssim$1000 W) have been added in FY2017 (Section 2.3).

2.2 Identifying Potential Additional Waste Types and Waste Forms

Reviewing the materials on radioactive waste types within the DOE-managed realm has produced a number of potential candidates to add to those waste types and waste forms that were evaluated in the WFDOE (SNL, 2014). At this point in time, these candidates have only been identified but not added into the evaluations. Further consideration of these wastes in the future would determine which would be added to the list of DOE-managed (as) HLW and SNF to be populated in the OWL database. A brief summary is given here of the waste types that are presently identified.

Within the DOE-managed waste complex, many of the waste types have been included in SNL (2014), as well as their proposed disposition as waste forms. The WFDOE (SNL, 2014) inventory of wastes is a superset of the DGRDMSH inventory discussed in Section 2.1. Inclusion of additional wastes into the OWL database would be only a first step as new waste types would only be added to the DGRDMSH inventory based on input from the DOE.

Active research is being performed to evaluate a variety of high level waste glass compositional variations to address limitations of glass formulations due to chemical components such as Fe, Al, Cr, Bi, P, Zr, and S (e.g., Kruger et al., 2012; 2013). In many of these cases, each compositional variation of the glass does not yet appear to warrant specific tracking because these glass compositions are still within the R&D stage. One exception included below is a glass composition from the high sulfur waste streams.

Advanced fuels are being developed that will at some point need disposal dispositioning, for example at research reactors like the Transient Reactor Test Facility (e.g., Pope et al., 2014). Given the wide range of

fuel types existing within the DOE complex, such advanced fuels will only be considered after they are included into the DOE-managed SFDB as they would provide no immediate substantive difference for consideration.

Lastly, investigators are working to identify candidate waste forms for separated Tc waste streams, either directly from tank waste or from off-gassing as tank wastes are processed into glass (e.g., Westsik et al., 2014). Such waste forms include a wide variety of solids - borosilicate and iron phosphate glasses, cementitious grouts, geopolymers, phosphate-bonded ceramics, the fluidized bed steam reforming aluminosilicate waste form, the crystalline ceramic Synroc waste form, iron-technetium oxides, metal alloys, technetium oxides, silicate minerals, titanates, sulfides, phosphates, layered double hydroxides, and sulfur-based aerogels. One such waste form is included here because it has already been separated specifically, and is planned to be formed into the future. Additional tracking of potential waste types/forms for disposal disposition should only begin once the waste types/forms are actually generated.

Potential additions to the WFDOE (SNL, 2014) inventory include:

Hanford Tank Waste: Potential Additional Waste Types/Forms

- Existing separated waste
  - Demonstration of Cs-Tc removal from tank waste brines via ion exchange resins to be incorporated into high activity waste glass (existing separated waste; Hassan et al., 2000).
- Potential separated waste
  - Potential new glass formulations for projected high sulfur HLW streams from Hanford Tank Waste (likely separated waste; see Kruger et al., 2013).
- Potential separated waste type and waste form
  - Waste Treatment Plant (WTP) low activity waste vitrification facility off-gas condensate known as WTP Secondary Waste (WTP-SW) will be generated and enriched in volatile components such as ^{137}Cs, ^{129}I, ^{99}Tc, Cl, F, and SO_4 that volatilize at the vitrification temperature of 1150°C in the absence of a continuous cold cap (that could minimize volatilization). The current waste disposal path for the WTP-SW is to process it through the Effluent Treatment Facility (ETF). Fluidized Bed Steam Reforming (FBSR) is being considered for immobilization of the ETF concentrate that would be generated by processing the WTP-SW (Crawford et al., 2014).

2.3 Updated Inventory Information for FY2017 GDSA Evaluations

Based on the Sassani et al. (2016) recommendations, the primary FY2017 update to the preliminary DGRDMSH inventory is to include the additional possible DGRDMSH waste forms (DOE, 2014) that were not previously included in GDSA representations (e.g., Sevougian et al. 2016) and are most likely to expand the evaluation range of thermal and/or radionuclide inventory aspects compared to the previous analyses. Specifically, this entailed adding:

- The 340 Hanford Cs/Sr vitrified glass canisters (as detailed in Wilson, 2016, Table 2-6),
- The 34 glass canisters of Hanford Federal Republic of Germany (FRG) glass, which is material that has been managed as HLW (SNL, 2014), and may be disposed in a DGRDMSH,
- The planned waste form for calcine hot isostatically pressed (HIP) into HIP cans that are loaded/stacked into ~320 canisters (~5.5 ft diameter by ~15 ft height, naval canisters/waste packages containing ~10 HIP cans each; SNL 2014), and
- The naval SNF waste packages from the coolest thermal range (~13 naval SNF canisters using the ~1000W per canister thermal threshold for the upper bound—see Figure 3 of DOE, 2014; and SNL, 2014 naval waste package thermal binning listed in Appendix A, p. A-40).

Although most of these updates are relatively small from the standpoint of inventory mass, they may have some implications for analyses of thermal effects. This is because some of these added wastes tend to have higher average thermal loads per canister than the inventory previously evaluated in GDSA. Additionally, some of these waste forms represent larger waste packages, which may expand handling and emplacement considerations (i.e., naval SNF and planned calcine HIP waste form waste packages).

2.3.1 Hanford Vitrified Cs/Sr Glass Canisters

The definition of characteristics for vitrified Cs/Sr capsule glass canisters is given in Wilson (2016; Section 2.2; Table 2-6; Figure B-10; Tables B-16, -17) for thermal and compositional aspects of the projected 340 Cs/Sr glass canisters. These characteristics should be used to represent these Cs/Sr glass canisters. Note that these Cs/Sr glass canisters represent a somewhat higher thermal loading than all of the projected Hanford glass canisters, and virtually all of the Savannah River Site existing and projected glass canisters.

2.3.2 Hanford Federal Republic of Germany (FRG) Glass

As discussed in SNL (2014), the Pacific Northwest Laboratory prepared isotopic heat and radiation sources in 1986-87 to be part of the repository-testing program by the Federal Republic of Germany (FRG). Using a ceramic melter, thirty stainless steel canisters were filled with borosilicate glass spiked with ^{137}Cs and ^{90}Sr to achieve the desired heat and dose targets. In addition to the 30 canisters filled with radiation source glass, additional production included two demonstration canisters (these are low thermal, without included Cs/Sr) and two instrumented canisters for heat transfer studies (these two have lower activity loadings than the 30 radiation-source canisters). Each of the 34 canisters is ~1.2 m long by ~0.3 m in diameter, and therefore could be loaded in multiples into a smaller number of disposal containers. Note that it is thought that there is some level of transuranic radionuclide trace contamination in these glass canisters, and although these glass waste forms have been DOE-managed as HLW and may be disposed as such, they may also be reclassified as TRU waste. These FRG glass waste forms are reasonable to be included in GDSA for a DGRDMSH and they represent another higher thermal load waste form, similar to the Cs/Sr glass canisters, but of more limited mass.

The thermal and radionuclide compositions are summarized in SNL (2014, Table A-18) for the 34 FRG glass canisters with the inventory information corresponding to the year 1987 (as shown in supporting spreadsheet for SNL, 2014; and SNL, 2014 source references). Because these FRG glass canisters could be loaded (at least 3 each) into disposal containers, using overall averages for radionuclides and thermal loading is a reasonable approximation. Based on the values for the 30 radiation-source canisters, the average FRG glass canister has ~159,000 Ci of ^{137}Cs and ~119,300 Ci of ^{90}Sr (with their corresponding amounts of secular equilibrium daughter radionuclides, ^{137m}Ba and ^{90}Y), with an average thermal load of 1560 W. Applying these average values for all 34 FRG glass canisters would be a bounding estimate because the 2 heat transfer canisters have only about 2/3 of this thermal load and the 2 demo canisters are not loaded with Cs and Sr in a similar fashion (with correspondingly lower thermal loads). Alternatively, for the four FRG glass canisters that are not radiation-source canisters, applying an adjustment factor of 2/3 to the averages of the radiation-source canisters and using the result to represent the four non-production FRG glass canisters would be a reasonable, and still bounding, alternative approach.

2.3.3 Hot Isostatic Pressed (HIP) Calcine Waste Form

As summarized in SNL (2014, Appendix A, Section A-2.3.1), approximately 4,400 m^3 of calcine is currently stored in six Calcine Solids Storage Facility (CSSF) bin sets at the Idaho National Laboratory site. The currently selected waste form approach is hot isostatic pressing (HIP) technology to treat the calcine as summarized in SNL (2014). In the HIP process, the calcine is heated and then mixed with additives (e.g., amorphous silica, titanium metal and oxides, and calcium sulfate or elemental sulfur) and the mixture is placed in a stainless steel can (HIP can) which is then sealed with a lid with a vent tube. The HIP can is evacuated, the vent is sealed, and the can is placed in the HIP process vessel. The vessel

is pressurized with argon gas to 7,000 to 15,000 psi and is heated to 1,150 °C. At these processing conditions, the calcine is converted to a glass ceramic, and its degradation rate should be reasonably represented with glass degradation rates (Sassani et al., 2016).

Sandia (2014) evaluates the HIP calcine waste form canisters based on the larger-form HIP process that uses larger HIP cans (with initial ~60 ” diameter and ~30 ” height; SNL, 2014) that would be HIP processed and then loaded into naval long SNF canisters (~210 ” long by 66 ” diameter) after the HIP process. As a result, SNL (2014) provides ~3,200 HIP cans, with about 10 HIP cans loaded per naval long canister, for a total of ~320 HIP calcine canisters of the larger naval SNF canister dimension. In Wilson (2016), the calcine HIP canisters result from an alternative HIP process that uses smaller-form HIP cans (≲half-scale) loaded into standard DOE SNF canisters (external dimensions ~120 in long by ~24 in diameter). As a result, Wilson (2016; Table 2-5) has 4,391 canisters of HIP calcine, with each having calcine mass that is more comparable to that in the HIP can of SNL (2014) than for the one of the 320 HIP calcine canisters.

The estimated thermal loads for the larger HIP can used in SNL (2014) are 4 to 54 W/can, consistent with Wilson (2016; Table 2-5) showing all 4,391 HIP calcine canisters in the <50 W per canister thermal load bin—a single HIP *canister* in Wilson (2016) has slightly less than about ¾ of the calcine mass as one HIP *can* from SNL (2014). This would mean that the thermal load for a larger HIP calcine canister (SNL, 2014) would range from about 40 to 540 W. Using the calcine volume and thermal load data (referenced to 2016) given in Table 3 and 13 from Herbst (2005), the thermal load for the larger HIP cans ranges from about 13 to 52 W/can for averages of each of the six Calcine Solids Storage Facility bins. The overall average thermal load calculated for the 3,200 HIP cans is ~29 W, which corresponds to the average HIP calcine canister thermal load being ~290 W. The corresponding average radionuclide composition of this average HIP calcine canister can be directly derived from the total radionuclide inventory (year 2016) given in the last column of Table A-33 (SNL, 2014) divided by the 320 HIP calcine canisters.

2.3.4 Naval SNF Canisters

Naval SNF assemblies (see SNL, 2014, Appendix A, Section A-1.3) are composed of materials that keep temperatures low enough to maintain the integrity of the cladding. A decay heat limit of 11.8 kW was established for naval SNF canisters so that no canisters are shipped until the decay heat at the time of acceptance at the repository is less than or equal to 11.8 kW. This decay heat limit is sufficiently low that no aging is required prior to emplacement (DOE 2008). The average thermal load is 4,250 W per canister. As of March 2014, 93 naval SNF canisters have been loaded and are being temporarily stored at INL, pending shipment to a repository. Two naval SNF disposal canisters, one short and one long, were designed to accommodate different naval fuel assembly designs. Both canisters were sized to fit within the proposed design for the Yucca Mountain repository waste package. The outer diameter of the canister is 66 ” nominal (66.5 ” maximum). The maximum external dimensions ensure naval SNF canisters fit into the waste packages. The short canister is 185.5 ”. (nominal) in length (187 ” maximum), and the long canister is 210.5 ” (nominal) in length (212 ” maximum). With the exception of length, the other characteristics of the naval SNF canisters are identical. Approximately 400 canisters (310 long and 90 short) are planned to be packaged and temporarily stored pending shipment to a repository for disposal (DOE, 2008).

2.3.4.1 Thermal Averages for Coolest Naval SNF Waste Canisters

Approximately 400 naval SNF canisters (310 long and 90 short) are planned to be packaged and temporarily stored pending shipment to a repository for disposal (DOE 2008). As indicated in SNL (2014), the estimated average thermal load of a naval SNF canister is 4,250 W per canister (plans are that all 400 are loaded by 2035). Many of the naval SNF canisters may be at higher thermal load than would be desirable to include into a DGRDMSH with lower thermal constraints (DOE, 2014).

For the current GDSA analyses of a generic DGRDMSH, only the coolest naval SNF canisters (those in the 500 to ~1000 W thermal load range) are to be included as part of the DGRDMSH inventory for disposal analyses. This approximate thermal cutoff of ~1000 W is consistent with the recommendation (DOE 2014, p. ES-1) that DOE "pursue options" for "potentially including cooler naval SNF" in a DGRDMSH. To estimate the average thermal load of the ~13 naval canisters in this range, it is noted that these represent a small portion of the total (less than ~3 %) naval SNF canister thermal load. The total thermal output is ~1,700,000 W – calculated using the average thermal load per canister (4,250 W/canister) and the total number of projected naval SNF canisters (400) in 2035. Given that these ~13 naval SNF canisters represent such a small amount of the total thermal load, a simplifying assumption is made for GDSA analyses to use the value of 750 W/canister as representing the average thermal load of the naval SNF canister in the 500 – 1000 W/canister bin.

The average radionuclide composition for a naval SNF canister is given in SNL (2014; Table A-7), which provides the radionuclide inventory of a representative naval SNF canister five years after reactor shutdown. Using this to represent the average composition of the ~13 naval SNF canisters will be a bounding composition as these are the coolest canisters that are likely longer out of reactor and/or lower in content of the primary heat producing short-lived fission products. In terms of thermal decay history, using this radionuclide inventory directly would also conservatively represent the thermal aspects of these ~13 included naval SNF canisters. For a more explicit representation of the radionuclide inventory of these coolest naval SNF canisters, the average composition (SNL, 2014, Table A-7) adjusted their relative average thermal content compared to the average thermal content of the projected 400 naval SNF canisters. This ratio is about 750 W/4250 W, or approximately 17.6%. Only the concentrations of the short-lived major fission products ^{137}Cs and ^{90}Sr (and their corresponding daughter radionuclides ^{137m}Ba and ^{90}Y included as being in secular equilibrium) should be adjusted downward to reflect this different average thermal load for these ~13 naval SNF canisters. For any thermal decay evolution analyses, note that these short-lived radionuclides will all matter primarily for about only the first ~300 years of decay.

2.3.4.2 Performance for Naval SNF

For the naval SNF canister/waste form, degradation rates were represented previously (DOE, 2008; Section 2.4.2.3.2.2.4) as bounded by the degradation of commercial SNF (see also Figures 2.4-127, and -128 from DOE, 2008). Given those analyses, the GDSA representation should apply the degradation rates for SNF (i.e., UO_2 degradation rates) as bounding constraints for the naval SNF degradation rates.

2.4 Additional WF Characteristics Refinement Studies

During FY2017, a number of questions regarding the characteristics of various waste forms led to two studies on WF characteristics details. First, in our estimates of HLW glass compositions for postclosure safety analyses, we assume that all the ^{129}I in tank waste becomes part of the vitrified waste form. However, it is not clear if this quantitative assumption is correct, as the ^{129}I activity in the glass waste form does not appear to have been directly analyzed. Given that the Savannah River Site (SRS) has produced thousands of HLW glass logs, we initiated a study of the detailed documentation for the SRS vitrification process to see if it was possible to trace/quantify the potential sinks for ^{129}I in the various processes to form the HLW glass.

In addition to these uncertainties for SRS HLW glass logs, it was also noted that the inventory for the Cs/Sr capsules did not give the quantity of ^{135}Cs contained in the capsules. Nor did the reported inventory of Cs and Sr for the (Federal Republic of Germany – FRG) glass at Hanford provide the quantity of ^{135}Cs contained in those glass logs (SNL, 2014). Because quantities of these two long-lived fission products (half-life of ^{129}I is 1.57×10^7 years, half-life of ^{135}Cs is 2.3×10^6 years) were not readily available, we developed estimated quantities of both radionuclides in Savannah River glass (Section 2.4.1) and the FRG glass at Hanford (Section 2.4.2).

In FY2017 we began a third study to define characteristic isotopic ratios for various waste forms included in postclosure performance studies. This aspect arose due to questions regarding the relative contributions of radionuclides from disparate waste forms in DGRDMSH GDSA results, particularly, radionuclide contributions of DOE-managed SNF vs HLW glass. Depending on the design of the generic repository evaluated, it may be easy to assess such contributions proximal to the source terms if the waste forms are segregated. However, given the complexity of some geologic systems, isotopic ratios (two or more) that effectively tag their source waste form distinctly would facilitate such assessments at distal points. Using such ratios to define mixing lines may allow quantitative estimates of relative WF contributions to be "mined" from previously completed GDSA results, as long as the particular isotopes are tracked.

2.4.1 Evaluation of ^{129}I Sinks in Savannah River Site Glass Production

In our estimates of HLW glass compositions for postclosure safety analyses (SNL, 2014) we assume that all or most of the ^{129}I in Hanford tank waste becomes part of the Hanford vitrified HLW waste form (SNL, 2014; Sevougian et al., 2014). However, it may be that this quantitative assumption overestimates the ^{129}I activity in the Hanford HLW glass. Because ^{129}I tends to be mobile and a major radionuclide contributing to long term releases/dose in generic saturated system disposal analyses (e.g., Sevougian et al., 2016), source-term concentrations will have a direct effect on the calculated results. Given that the Savannah River Site (SRS) has produced over ~3,780 HLW glass logs through 2014 (Wilson, 2016), we initiated a study of the detailed documentation for the SRS vitrification processing to see identify and trace/quantify the potential sinks for ^{129}I in the SRS HLW glass production. There is some indication that not all (if any) of the ^{129}I in SRS tank HLW ends up in the SRS HLW glass waste form. For SRS HLW glass canisters already produced in macrobatches 1 through 8, there is no reported ^{129}I activity (see Table A-15 in SNL, 2014).

The SRS is required to report the inventory for radionuclides that have half-lives greater than 10 years and are present in the glass at greater than 0.05% of the total radionuclide inventory at any point up to 1,100 years after production, and SRS extends that specification to all isotopes exceeding 0.01% under those conditions (SNL, 2014). The ^{129}I activity was reported in sludge batch 1B (SB1B) of macrobatch 2, but was not of high enough activity in the other sludge batches of other macrobatches (Crawford and Diprete, 2014; Section 3.3 and Table 3-4). With enough quantitative information on the SRS processes, our goal would be to develop a better estimate of the fraction of ^{129}I activity/mass that would end up in the HLW glass waste forms. A summary status is provided here of our delineation of those process steps where conditions or materials may separate iodine from the fraction that ends up in the vitrified waste. Work is ongoing to identify the needed data to assess quantitatively iodine distribution within those processes and the general effects on the radionuclide inventory for ^{129}I in HLW glass waste forms. At this point, the major processes and their material outputs have been identified and summarized below.

The treatment process for HLW at the SRS is complex and requires multiple steps for worker health and safety purposes (Chew and Hamm, 2016), so it is a challenge to track quantitatively a specific radionuclide, like ^{129}I. The SRS tank wastes contain both salt waste supernate/salt cake (referred to as salt waste as it contains most of the soluble salts, including over 95% of the ^{129}I) and sludge, and these waste types are treated via separate processes prior to vitrification in the Defense Waste Processing Facility (DWPF; Kantelo et al., 1993; Chew and Hamm, 2016). Throughout the treatment processes, liquid waste (supernate) is evaporated (via boiling) to conserve tank space (Hang et al., 2003). Evaporating the liquid salt waste reduces it to 30% of its original volume, meaning recovery of over ~2.5 million gallons (combined) of tank space each year. The process begins with the liquid salt waste (supernate salt waste) being transferred from the feed tank to the evaporator vessel. The liquid salt waste is evaporated via boiling by a steam tube-bundle heat exchanger to attain a target specific gravity. The concentrated liquid (supernate) is recycled back into the tanks. The vapor generated passes through a de-entrainment unit to remove any liquids or solids, and is condensed and treated using a mercury removal system. The treated condensate is sampled and sent to the Effluent Treatment Facility (ETF; Hang et al., 2003; Chew and

Hamm, 2016). The evaporation process represents one place where iodine may separate into the vapor phase and be removed from materials that end up being incorporated into HLW glass at the DWPF.

Supernate is removed from the tanks using a pump system, and processed through the Actinide Removal Process (ARP) to remove strontium, specific actinide radionuclides (alpha-emitting actinides including plutonium, neptunium, and uranium), and any entrained sludge particles via adsorbants (Chew and Hamm, 2016). To start the ARP, the waste is transferred to an alkaline precipitate tank of sodium concentration of around 6 to 7 M. Here the solution is diluted with a 1.66 M caustic to adjust the sodium concentration to about 5.6 M. A 15 wt% monosodium titanate (MST) slurry is added to adsorb the strontium and actinides, allowing 24 hours for a steady state/equilibrium to be reached. The slurry is then filtered to remove the MST particles and any entrained sludge particulates. The filtered particles are concentrated to 5 wt%, washed (to reduce sodium concentration), and sent to the DWPF to be vitrified with additional sludge waste materials (Peters et al., 2011; Chew and Hamm, 2016; Hang, 2004). Any remaining supernate (salt waste) is sent to the Modular Caustic Side Solvent Extraction Unit (MCU).

Supernate salt waste sent to the MCU undergoes Caustic Side Solvent Extraction (CSSX) to remove cesium. The cesium is extracted using a specific solvent process that produces both the extracted Cs (which is concentrated with dilute nitric acid, and sent to the DWPF) and a treated liquid salt solution which goes to the Saltstone Production facility (SPF) to be treated/processed with the salt cake (Peters et al., 2011; Chew and Hamm, 2016).

Sludge waste in the tanks is diluted to suspend the sludge, and mixing pumps suspend the solids to transfer the slurry for processing. Depending on the origin/composition of the sludge, it is treated with the Low Temperature Aluminum Dissolution process, to remove aluminum solids at a temperature of 75°C (Chew and Hamm, 2016). All of the sludge waste goes through a sludge washing process in which the washed solids settle to the bottom. The sodium-rich supernate is removed and sent to the evaporator system to be concentrated and treated (see above). This sludge washing process is repeated until ~1.25 M Na is attained. The washed sludge waste then goes to the Sludge Receipt and Adjustment tank, where it is treated with formic acid to remove mercury, treated with nitric acid to be neutralized, concentrated, and made ready for the DWPF vitrification process.

This processed sludge and HLW components (e.g., Sr, Actindes, Cs) from the supernate processing are combined with borosilicate glass frit at the DWPF. The mixture is fed to a melter and heated to ~2,100°F (~1,150°C), creating a molten material. During the heating process gasses form (off-gas), which are collected and recycled into the overall processing system. The National Service Center for Environmental Publications (1979) indicates that “Iodine compounds dissociate at the temperature where glass melts,” so it is likely that most ^{129}I ends up in the off-gas at this point. The molten material is vacuum poured into stainless steel containers where it cools into solid glass. These canisters are then temporarily sealed and their outside is blast cleaned with and air-frit slurry for decontamination. After this they are permanently welded shut and stored in interim Glass Waste Storage Buildings (GWSBs), awaiting permanent geological disposal (Norton et al., 2002).

2.4.1.1 Estimated Quantity of ^{129}I in Savannah River Glass

The waste in tanks at Savannah River must be treated before disposal. The waste is a complex mixture of insoluble metal hydroxide solids, referred to as sludge, and soluble salt supernate (Chew and Hamm, 2016). ^{129}I partitions primarily into the supernate; as of January 2013, the supernate in all the tanks contained 11.6 curies of ^{129}I while the sludge in all the tanks contained 0.436 curies of ^{129}I (Li, 2013). The supernate and the sludge undergo different treatment processes. The supernate is stripped of actinides, strontium, and cesium; the waste that remains contains less than 1% of the radioactivity and is considered LLW. This LLW waste is turned into a solid by mixing it with cement, slag, and fly ash in the Saltstone Production Facility. The waste hardens in vaults at the Saltstone Disposal Facility at Savannah River (Chew and Hamm, 2016).

The actinides, strontium, and cesium that were removed from the supernate are mixed with the sludge and vitrified in the DWPF. The vitrification process involves mixing the waste with borosilicate glass frit and heating the mixture to 2,100°F (1,149 °C), creating a molten glass that is poured into stainless steel containers and allowed to cool, forming a solid glass HLW.

When reporting inventories of the glass HLW, only certain radionuclides are required to be reported. In general, certain uranium and plutonium isotopes must be reported, as must radionuclides with half-lives greater than 10 years that have concentrations greater than 0.01% of the total inventory from time of production through year 3115 (Crawford and DiPrete, 2013). While the half-life of ^{129}I is greater than 10 years, its concentration is usually less than 0.01%; therefore, the quantity of ^{129}I in the glass HLW produced at Savannah River was not always reported even though it is present in the glass (Crawford and DiPrete, 2013).

Our goal was to be able to estimate the quantity of ^{129}I in the glass waste produced at Savannah River. Most of the ^{129}I in the tanks is in the supernate, which is solidified into Saltstone. The processes that remove actinides, strontium, and cesium from the supernate do not remove iodine from the supernate, and iodine that is evaporated during Saltstone processing is captured and returned to the Saltstone Production Facility through the Effluent Treatment Project (Chew and Hamm, 2016). Therefore, we conclude that nearly all the ^{129}I in the supernate in the tanks is incorporated into the solid waste disposed of in the Saltstone Disposal Facility.

Some of the ^{129}I in the sludge is incorporated into the glass. Iodine is volatile under the conditions present during vitrification in the DWPF; during the vitrification process iodine is captured in the recycle stream, which is sent back to the tanks. Measurements indicate that concentrations of ^{129}I in the recycle condensate are about 38 times that in the tanks, indicating that a significant portion of the ^{129}I is not being incorporated into the glass (Bannochie, 2015). The fraction of ^{129}I that is incorporated into the glass during vitrification is unknown. Assuming that all the ^{129}I currently in the sludge in the tanks at Savannah River is incorporated into the glass overestimates the quantity of ^{129}I in the glass waste. However, in the absence of further information and at this stage of the development of the DGRDMSH, this assumption is sufficient.

2.4.2 Quantity of ^{135}Cs in Cs/Sr Capsules and in FRG Glass at Hanford

The waste known as “Cs/Sr Capsules” consists of 1,335 capsules that contain CsCl and 601 capsules that contain SrF_2. The cesium and strontium in the capsules was extracted between 1974 and 1985 from HLW stored in the underground tanks at Hanford to reduce the heat output of the tank waste. This operation took place at the B Plant, which had been repurposed for extracting cesium and strontium from the waste, and involved feed from the underground waste storage tanks as well as from the PUREX plant (Marceau et al., 2002).

Estimation of the quantity of ^{135}Cs in the capsules is based on ratios of the quantity of ^{135}Cs to the quantity of ^{137}Cs calculated to be in Hanford reprocessing waste (Wootan and Finfrock, 2002) and on measured values of Cs isotopes in cesium capsules that were destructively tested in 1984 (Sasmor et al., 1988).

The tanks at Hanford received waste from the T-plant, the B-plant, and the REDOX plant, and from the PUREX plant. Both the T-plant and the B-plant ceased operating as separations facilities in 1956, the REDOX plant ceased operation in 1967, and the PUREX plant ceased operating in 1988 (Marceau, et al., 2002). Production of cesium capsules ended in 1983, so not all the cesium separated in the PUREX plant ended up in the capsules.

Wootan and Finfrock (2002) estimated the radionuclide inventories produced by the reactors at Hanford by using ORIGEN2 and DKPRO. Inventories were calculated as a function of fuel type, fuel burnup, reactor type, decay, and the method of reprocessing, and were decayed to a date of January 1, 2001. The inventory estimates provided by Wootan and Finfrock (2002) represent the complete, best-estimate

Hanford Site processing waste stream inventory, and are summarized for the T-plant, the B-plant, the REDOX plant, and the PUREX plant. A total across all four reprocessing plants is also given.

The total quantity of ^{135}Cs and ^{137}Cs in the tanks as estimated by Wootan and Finfrock (2002) was used to calculate an activity (curie) ratio and a mass ratio (^{135}Cs/^{137}Cs) as of January 1, 2001.These ratios were then adjusted to the baseline inventory year for the capsules, 2016, to account for radioactive decay. The total mass of ^{135}Cs in the capsules in the baseline year was calculated by multiplying the baseline year mass ratio of ^{135}Cs to ^{137}Cs by the curies of ^{137}Cs in the capsules in the baseline year, as reported in the capsule-by-capsule inventory spreadsheet, and dividing by the specific activity of ^{137}Cs (curies/gram). The total activity of ^{135}Cs in the capsules was calculated by multiplying the baseline year curie ratio of ^{135}Cs to ^{137}Cs by the curies of ^{137}Cs in the capsules in the baseline year. This yielded a single estimate of the quantity of ^{135}Cs in the capsules in the baseline year.

Sasmor et al. (1988) destructively tested seven cesium capsules from Hanford in 1984 and measured the atom percent of multiple cesium isotopes (^{133}C, ^{134}Cs, ^{135}Cs, and ^{137}Cs) in the capsules, as reported in Table 11 of Sasmor et al. (1988). From this data, the mass ratio of ^{135}Cs to ^{137}Cs was calculated and the curie ratio of ^{135}Cs to ^{137}Cs was also calculated. Mass and curie ratios of ^{135}Cs to ^{137}Cs for the baseline year (2016) were then calculated. These ratios were then used as above to estimate the total activity of ^{135}Cs in the capsules in the baseline year and the total mass of ^{135}Cs in the capsules in the baseline year. This was done for each of the seven cesium capsules, yielding seven estimates of the ^{135}Cs content in the capsules in the baseline year.

The average ^{135}Cs inventory per capsule was calculated by averaging the eight estimates of total ^{135}Cs inventory and dividing this average by the number of cesium capsules (1,335). The average mass ratio of ^{135}Cs to ^{137}Cs in the baseline year is 0.87 and the average activity ratio of ^{135}Cs to ^{137}Cs in the baseline year is 1.16×10^{-5}.

The FRG glass consists of 34 glass logs produced by Pacific Northwest National Laboratory in 1987 to provide heat and radiation sources for repository testing by the Federal Republic of Germany in the Asse salt mine. Germany did not take delivery of the glass logs; they are currently stored at Hanford. The logs contain both strontium and cesium. The cesium in the glass logs was derived from the capsules described above. Therefore, the same ratios of ^{135}Cs to^{137}Cs activity and mass and the known activity and mass of ^{137}Cs can be used to estimate the quantity of ^{135}Cs in the FRG glass logs.

2.4.3 Delineating Characteristic Isotopic Ratios for Various WF

Late in July of FY2017, we began a second study to define characteristic isotopic ratios for various waste forms included in postclosure performance studies. Questions regarding the relative contributions of radionuclides from disparate waste forms in the previous generic DGRDMSH GDSA results for salt and granite systems (Sevougian et al., 2016) led to this evaluation initiation. Particularly, the question arose as to whether we could develop a direct method to deconvolve the radionuclide contributions of DOE-managed SNF vs HLW glass at various locations in a generic repository system represented in the GDSA.

Depending on the design of the generic repository evaluated, it may be easy to assess individual WF contributions proximal to the source terms if the various waste forms are segregated into sections of the repository. However, given the complexity of some geologic systems, isotopic ratios (two or more) that effectively tag their source waste form distinctly would facilitate such assessments at distal points. Using such ratios to define mixing lines may allow quantitative estimates of relative WF contributions to be “mined” from existing GDSA results, as long as the particular isotopes are tracked. This would be similar to geochemical techniques using natural isotopic ratios to assess mixing in natural systems. For future analyses, fictive particles singular to individual waste forms could be added to each WF inventory and used as tracking mechanisms, both unretarded and retarded fictive particles could be utilized.

In addition to the design of a generic repository, the inventory included therein also plays a role in how easily waste form contributions could be delineated. Given the GDSA for a generic salt and generic

granite DGRDMSH (Sevougian et al., 2016) contains both DOE-managed SNF and HLW glass, this case would provide an initial idealized test to check the efficacy of the approach. Useful isotopes for defining characteristic isotopic ratios of the waste forms of interest should have at least the following characteristics:

(a) they should be heavy molecular weight isotopes so fractionation processes are not an issue;

(b) they should have distinct signatures between the waste forms of interest (larger differences should provide more sensitivity); and

(c) they should have signatures that are not overwhelmed by the natural system background signatures, if those are included in the GDSA evaluations.

Isotopes of the actinides provide a direct means, for meeting the first and third above characteristics, with the level of distinction for each waste form inventory assessed preliminarily below.

A first-order assessment of the isotopes most likely to distinguishing HLW glass from DOE-managed SNF (DSNF) has been derived from Rechard and Stockman (2014; Figure 19) where it can be seen that

- Isotopes ^{230}Th, ^{233}U are about 30 to 100-fold the mass per package for DSNF than for HLW glass,
- Isotopes ^{234}U, ^{235}U, (^{238}U) are about 5 to 10-fold the mass per package for DSNF than for HLW glass, and
- Isotope ^{99}Tc which is about 1/10 the mass per package for DSNF than for HLW glass.

The results shown in Rechard and Stockman (2014) simply provide an initial starting point primarily because those waste form inventories represent averages of

a. various projected HLW glasses from a number of sites, and

b. all DSNF composition (not including naval SNF or Na-bonded fuel components).

The explicit waste form definition and inventory used in any particular GDSA evaluation should be the basis to define the appropriate discriminating isotopic ratios in order to quantitatively extract mixing data from that set of GDSA results. Isotopes other than those above (e.g., ^{229}Th, some Pu isotopes) may also be useful for some particular waste forms. Note that, as more waste forms are included into a GDSA analysis, more distinguishing characteristics (isotopic ratios) would likely be required to define quantitatively the mixing lines with enough independent constraints.

The next step in this analysis was to evaluate characteristic isotopic ratios based on the GDSA inventory specifics to see if there are ratios that discriminate among the waste forms, which may be useful for evaluating GDSA results away from the repository. Two uranium isotope ratios were assessed, ^{238}U/^{236}U and ^{234}U/^{233}U. For ^{238}U/^{236}U about 2.5 orders-of-magnitude (OoM) variation is observed for waste forms including Hanford and SRS HLW glasses, INL Calcine, and a number of the DOE SNF, whereas for ^{234}U/^{233}U there is about 3 OoM variation. Preliminary examination of ^{242}Pu/^{239}Pu indicates ~2 OoM variation, but with clustering around two values, indicating this may not be as discriminating as the uranium isotopic ratios, but possibly useful. Lastly the ^{135}Cs/^{129}I ratios were examined and it was found that this ratio is fairly similar for most of these waste forms (~0.5 Oom variation). This activity will continue evaluation of the various waste form inventories to define a set of ratios that allow quantitative evaluation of mixing of waste form contributions at various locations in the GDSA results.

2.5 Potential Variations of DGRDMSH Features/Concepts Related to DGRDMSH Inventory Characteristics

DOE-managed HLW and SNF in the DGRDMSH inventory differ from CSNF in a commercial repository inventory by smaller quantity (smaller mass and about 1/6 the volume), more relative heterogeneity, lower radionuclide inventory (HLW is weighted toward fission products), lower thermal

load per caniste, and waste form composition (including the presence of Research Conservation and Recovery Act (RCRA) regulated wastes, weapons-usable spent naval fuel, and water-soluble salts). DOE-managed HLW and SNF are similar to CSNF in proposed waste package composition (stainless steel) and range of waste package dimensions under consideration. Utilizing the detailed information on waste forms included in the Appendices of the WFDOE (SNL, 2014), considering these aspects helps define the data characteristics that facilitate evaluating the potential variations of DGRDMSH features and repository concepts from a repository concept that includes CSNF.

2.5.1 Implications for repository layout and design, FEPs screening, and PA implementation

A discussed in Section 2.1, the DGRDMSH inventory differs significantly from that considered in the SAR for YM (DOE, 2008). It also differs from the inventory of a generic repository described within the Generic Disposal Systems Analysis (GDSA) framework (Freeze et al., 2013; Sevougian et al., 2014; Mariner et al., 2015). Both the YM SAR and the GDSA framework assume a 70,000 MTHM inventory, most of which would be CSNF with small percentages of DOE-managed SNF and HLW. In these two cases, the relatively uniform CSNF inventory contains most of the radionuclides (SNL, 2014) and is the major driver behind repository layout and design choices, screening of features, events, and processes (FEPs), and performance assessment (PA) implementation. The concept of a low-temperature DGRDMSH, which has smaller radionuclide inventory and lower thermal load, results in the variations in waste forms being more pronounced because they are not overwhelmed by CSNF characteristics. For these reasons, reassessing design choices, FEPs screening, and PA implementation facilitates identifying aspects of a DGRDMSH that may be different than a repository containing mostly CSNF.

Within the GDSA framework three primary generic mined-repository reference disposal concepts (summarized in Section 1.3) are considered: in crystalline rock, in a salt deposit, and in an argillaceous formation (Clayton et al., 2011; Freeze et al., 2013; Sevougian et al,. 2014; Mariner et al., 2011; 2015; Wang et al., 2014; Jove Colon et al., 2014). Hardin et al. (2012) analyzed thermal load management for disposal of CSNF in variations of each of these disposal concepts, including “enclosed” concepts in crystalline, salt, and argillaceous host rocks with waste enclosed in backfill or buffer at the time of emplacement, and “open” concepts in which waste emplacement drifts are not backfilled/buffered for a period of time if at all. Open concepts are relevant to potential disposal of CSNF in large waste packages due to the high thermal load, but less relevant to a DGRDMSH, which with minimal exception, carries a much smaller thermal load.

In a generic repository, several key criteria affect repository layout and design, including temperature limits within the repository, the mechanical strength of the host rock, maintaining flexibility in design requirements and sequential construction ability, and maintaining retrievability, accounting, and control of the waste as required by law (Hardin et al., 2012; SNL, 2014). Limiting the maximum temperature in waste package walls, in the buffer/backfill, and in the host rock is required to maintain the hydrologic, chemical, and mechanical integrity of these materials. Temperature limits within the repository lead to constraints on surface storage (and/or drift ventilation) time, drift spacing, waste package spacing, and waste package size. Where temperature limits do not constrain drift spacing, drift spacing is constrained by the mechanical strength of the host rock. Maintaining flexibility in design and construction of the repository may be facilitated by segregating waste types (Hardin et al., 2012). Retrievability may be enhanced by certain emplacement options, such as the vertical emplacement boreholes in the KBS-3V design (SKB, 2009), which allow a single waste package to be retrieved without disturbing its neighbors.

In a DGRDMSH, additional criteria may become relevant to repository design, to FEPs screening, and to PA implementation. The WFDOE (SNL, 2014) identifies several issues specific to particular types of DOE-managed wastes, including: the corrosive effect of halide-containing salt waste; the formation of plutonium colloids from soluble plutonium wastes; criticality; pyrophoricity; various (some rapid) dissolution behaviors; and the presence of RCRA-regulated waste and highly-enriched weapons-usable

waste. It may be desirable to take into account these concerns in designing the DGRDMSH; segregating waste forms could, for instance, isolate corrosive waste from other waste packages; facilitate accounting and control of particular waste forms; and facilitate management of the thermal load. Performance assessment calculations can explicitly include waste stream heterogeneity by introducing waste forms with differing thermal loads, radionuclide inventory, and dissolution rates. FEPs screening may result in inclusion of additional FEPs due to concerns specific to a DGRDMSH and/or result in exclusion of FEPs due to exclusion of commercial SNF and the hotter DOE-managed wastes from a DGRDMSH.

2.5.1.1 Reference GDSA Disposal Concepts

Each of the generic mined repository concepts is being evaluated to define and refine reference cases for use in the GDSA PA analyses. There are summarized here to discuss the aspects that may be considered differently for a DGRDMSH.

Mined Repository in Salt

Extensive salt deposits exist as bedded (as at the WIPP; e.g., DOE, 2009) or domal (as at Gorleben; e.g., BMWi, 2008) formations. In the U.S., bedded salt formations hundreds of millions of years in age occur in deep sedimentary basins located in tectonically stable regions of the craton (Perry et al. 2014). Composed primarily of the mineral halite (NaCl), such formations have very low porosity (on the order of 0.01 to 0.05) and permeability (on the order of 10^{-23} m^2), which limit the amount of water present in the system and its ability to move; high thermal conductivity (3.1 to 4.7 $Wm^{-1}K^{-1}$), which promotes heat conduction away from waste packages; and the ability to self-heal through creep consolidation, which helps maintain the low permeability of the salt (Freeze et al., 2013; Hardin et al., 2012). Ambient porewater is saturated with respect to halite, which can help mitigate criticality concerns due to the high concentration of neutron-capturing Cl^- (SNL, 2014). At repository depth, reducing porewater conditions limit radionuclide solubility, and the lack of free oxygen makes pyrophoric behavior unlikely.

The GDSA (CSNF) salt reference case (Freeze et al. 2013; Sevougian et al. 2014; Mariner et al. 2015) builds upon experience at WIPP, and calls for a repository at 680 m depth, in a bedded salt formation 495 m thick. Carter et al. (2012) presented a reference case for disposal of DOE-managed wastes in a salt repository modeled after WIPP; in this reference case, access to the repository is through vertical shafts, and waste emplacement panels are backfilled to some height with crushed salt and closed with additional crushed salt. Carter et al. (2012) specify a distance of 100 ft between panels and a distance of 1 ft between waste packages, which are emplaced horizontally on the panel floor. The salt disposal concept relies on the very low permeability of the host rock to isolate radionuclides. It does not rely on waste package integrity, nor on the sorption capacity of halite, which is low. A reference case for a DGRDMSH in salt was developed by Sevougian et al. (2016).

Mined Repository in Crystalline Rock

A mined repository in crystalline rock would be placed several hundred meters below the land surface in sparsely fractured crystalline basement that either outcrops or subcrops near surface in a region where the topographic slope is < 1° (Wang et al., 2014). In such a location, the water table would be unconfined and topographically-controlled, and due to the limited topographic slope, little driving force for deep fluid flow would exist. This concept is consistent with international concepts of disposal in crystalline rock (e.g., SKB, 2011). Locations fitting this concept occur in the eastern half of the United States (Perry et al., 2014). Crystalline rock has very low matrix porosity (0.05) and permeability (on the order of 10^{-20} m^2) (Martino and Chandler, 2004; Cho et al., 2013). Fluid flow occurs in fractures, which have the potential to channel flow over long distances. Crystalline rock has moderate thermal conductivity (from ~2.3 to 3.8 $Wm^{-1}K^{-1}$; Hardin et al., 2012) and high mechanical strength. Ambient porewater may be fresh to brackish, and at repository depth is expected to be reducing, limiting radionuclide solubility (Mariner et al., 2011).

Within the GDSA framework, preliminary reference cases for this repository concept have been created both for disposal including CSNF and for a DGRDMSH (Wang et al., 2014; Stein et al., 2016a; 2016b). Access to a crystalline repository would likely be through a ramp (Hardin et al., 2012). References cases consider in-drift horizontal emplacement of waste and emplacement of waste in vertical emplacement boreholes drilled beneath the drift floor as in the KBS-3V concept (SKB, 2009).

The crystalline reference case relies on the engineered barrier to a greater extent than the does the salt reference case, in which more safety reliance is on the natural system barriers. This is primarily related to the fracture pathways in a crystalline system versus the relatively impermeable salt layers. Whether emplaced horizontally or vertically, waste packages in the crystalline reference case are surrounded by bentonite buffer, a material with low permeability and high sorption capacity, and drifts are buffered/backfilled with additional low permeability material (Wang et al., 2014; SKB, 2009). The Swedish crystalline safety case also relies upon the very slow corrosion rate of copper waste packages (SKB, 2011).

Although the strength of crystalline rock allows more flexibility in handling larger, heavier waste packages, the temperatures within a crystalline repository may require a greater degree of management because of the desire to not alter the bentonite buffer. Such management can be implemented via drift spacing and waste package spacing, as well as waste package total thermal load. This is also driven in part because the crystalline host rock has lower thermal conductivity than that in a salt repository, and by the low thermal conductivity of the bentonite buffer (0.4 $Wm^{-1}K^{-1}$ dry to 1.35 $Wm^{-1}K^{-1}$ saturated; Hardin et al., 2012).

Mined Repository in Argillaceous Formation

Clay-rich sedimentary strata (argillite) have been considered a potential medium for disposal of radioactive waste in the United States since the forerunner to the DOE introduced a program to develop radioactive waste disposal technology in 1976 (Shurr, 1977, Gonzales and Johnson, 1985). Clay-rich formations are an attractive disposal medium due to their low permeability (between 10^{-17} and 10^{-22} m^2; Jove Colon et al., 2014), high sorption capacity, typically reducing porewaters (which limit radionuclide solubility), and (if not indurated) ability to deform visco-plastically, which promotes self-healing of fractures. The U.S. hosts several marine sedimentary sequences containing thick beds of clay-rich sediments potentially suitable for deep geologic disposal of radioactive waste (Gonzales and Johnson, 1985; Perry et al., 2014).

The GDSA (with CSNF) clay reference case calls for in-drift, horizontal emplacement of waste with (as in the crystalline case) bentonite buffer and low permeability backfill (Jove Colon et al., 2014; Mariner et al., 2015). Access to the repository could be through either a ramp or shafts, depending on the strength of the particular argillite host rock—indurated host lithology tends to be stronger. Argillaceous sediments have lower thermal conductivity (1.3 to 2.7 $Wm^{-1}K^{-1}$; Hardin et al., 2012) than either crystalline rock or salt, making thermal management more challenging in the clay disposal concept than in either of the others. The clay reference case relies on the low permeability and high sorption capacity of both the host rock and the engineered buffer. In broad aspects of the geologic characteristics and engineered barriers, this repository concept is intermediate to the salt and crystalline reference cases.

DOE-Managed Wastes

The WFDOE (SNL, 2014) inventory included 43 waste types currently in existence and assigned them to 50 potential waste forms after taking into account alternate disposal pathways for several waste types (see Section 1.2). The 50 waste forms were further sorted into ten waste groups (WG), which were used to assess design aspects for each repository concept based primarily on expected post-closure degradation behavior assigned to each of those groups. Two of the groups comprised of CSNF are not relevant to the DGRDMSH inventory. The other eight WG contain waste types currently managed by DOE that are

potentially part of inventory for a DGRDMSH (Section 2 delineates the inventory for DGRDMSH analyses for FY2016). These eight are:

- WG3: vitrified HLW (including vitrified Cs and Sr capsules);
- WG4: other engineered HLW forms (including hot isostatic pressed calcine);
- WG5: metallic and non-oxide spent fuel (N-reactor is the largest waste in this group);
- WG6: untreated sodium-bonded spent fuels (these are not considered further as no direct disposal pathway was delineated – these would be processed via electrometallurgical treatment);
- WG7: DOE-managed oxide spent fuels;
- WG8: salt, granular solids, and powders (including untreated calcine waste and untreated Cs and Sr capsules);
- WG9: coated-particle spent fuel (e.g., TriSO particles); and
- WG10: Naval spent fuel.

Note that some waste types (e.g. calcine waste, and Cs and Sr capsules) appear in more than one waste group due to alternate disposal pathways.

Waste included in the current analysis (Sections 2.1 and 2.3) of a low-temperature, DGRDMSH is a subset of the waste managed by DOE, and primarily would include waste forms in WG3 (vitrified HLW, vitrified Cs/Sr capsules and FRG glass added in FY2017), WG4 (engineered HLW – HIP calcine, added in FY2017), WG5 (metallic SNF), WG7: (oxide spent fuels) and WG8 (salts, etc.), along with the coolest waste packages in group WG10 (Naval SNF –added in FY2017). For GDSA purposes, there are three waste-form degradation rate mechanisms included in the PA of a DGRDMSH into which each of these WF groups is mapped. The three rates are: (a) instantaneous degradation (e.g., metallic fuels like N-reactor); (b) glass waste degradation (DHLW glass and HIP calcine); (c) UO_2 degradation (e.g., naval SNF, and WG7). Section 3.2 discusses the post-closure degradation performance of these various waste forms and evaluates the mapping of waste forms to these groups.

2.5.1.2 Variations of a DGRDMSH from a CSNF-dominated repository

The WFDOE (SNL, 2014) defines waste groups on the basis of expected post-closure performance, radionuclide inventory, thermal characteristics, chemical characteristics, physical characteristics, packaging, and safeguards and security. All of these considerations come into play in assessing the characteristics of a DGRDMSH in terms of repository layout and design, FEPs screening, and PA implementation. Additionally, the total quantity of waste (e.g., numbers of canisters/packages, volumes) destined for disposal in such a repository is considered in these assessments as well.

Quantity of waste

The inventory of CSNF in 2048 estimated under the "no replacement scenario" (Carter et al., 2013) is projected to be 142,000 MTHM or ~183,900 m^3, enough to fill two repositories at about the 70,000 MTHM limit specified in the Nuclear Waste Policy Act (1983). The projected DOE-managed (as) HLW inventory is on the order of ~26,000 m^3 (SNL, 2014); the precise volume depends on whether certain waste forms are vitrified, or hot isostatic pressed. The projected DOE-managed SNF inventory volume is ~7,200 m^3 (SNL, 2014), over half of which would be excluded from a low-temperature, DGRDMSH. Assuming ~30,000 m^3 of waste in the DGRDMSH, such a repository would hold approximately one third the volume of waste held compared to a 70,000 MTHM commercial repository.

Radionuclide inventory

The radionuclide activity in a low-temperature DGRDMSH would be a small percentage of the activity in a 70,000 MTHM commercial repository. In addition to the difference in magnitude, the source of

radioactivity would differ. Radioactivity in CSNF comes in almost equal parts from long-lived transuranic isotopes (mainly ^{241}Pu, ^{238}Pu, and ^{241}Am) and short-lived fission products (^{137}Cs, ^{90}Sr, and their daughter products) (SNL 2014, Figures A-9 and A-10). Sources of radioactivity in DSNF (Wilson, 2016) are similar to those in CSNF, and depend on initial enrichment and burnup, with higher-burnup waste types having larger contributions from fission products (SNL, 2014). Radioactivity in some DOE-managed (as) HLW is almost entirely from ^{137}Cs, ^{90}Sr and their daughter products (i.e., fission products). Examples of these are Cs/Sr capsules, calcine waste, and Hanford and Savannah River glass and tank waste (SNL, 2014; Carter et al., 2013). Waste forms such as these whose radioactivity comes primarily from short-lived fission products will reach peak temperatures sooner than waste forms whose source of radioactivity is largely long-lived transuranics. The timing of temperature transients in a repository will affect the timing of multiple processes occurring in the system, including: re-saturation; buoyancy driven fluid flow; waste package degradation; waste form dissolution; buffer and host rock alteration; creep consolidation.

Thermal characteristics

DOE-managed (as) HLW and SNF generate less heat per canister than CSNF packed in any of the waste packages under consideration for a DGRDMSH. Assuming initial enrichment of 4.73 wt% ^{235}U and 60 GWd/MTHM burnup, pressurized water reactor (PWR) CSNF fifty years out of the reactor (OoR), generates approximately 1140 W/MTHM (Carter et al., 2013). The smallest CSNF waste package, holding four PWR assemblies (0.435 MTHM per assembly), will generate a thermal load of 1980 W, and a 12-PWR waste package, as assumed in GDSA calculations to date (Sevougian et al., 2014; Mariner et al., 2015; Stein et al., 2016a), will generate 5940 W at 50 years OoR. Approximately 13,440 12-PWR waste packages would fill a 70,000 MTHM commercial repository (Mariner et al., 2015), generating approximately 79.8 MW of heat.

By contrast, the DGRDMSH waste forms generating the greatest thermal load are:

- The Cs and Sr capsules, which generated up to 505 W per capsule in 2007 (SNL, 2014), and if packed 8 capsules to a waste package would generate as much as 800 W per waste package in year 2037 given the approximately 30 year half-lives of ^{137}Cs and ^{90}Sr;
- the ceramic waste form resulting from electrometallurgical treatment (EMT) of Na-bonded SNF, which the WFDOE (SNL, 2014) calculated to generate 2,240 W per waste package after treatment of 6-year-old waste, or 1,250 W per waste package after treatment of 20-year-old waste; and
- the naval SNF, which averages 4,250 W per waste package (currently only the coolest naval SNF packages have been added to the DGRDMSH inventory).

Most DGRDMSH HLW generates considerably less heat. At the time of projected production, 99.5% (~11,772 canisters) of Hanford glass HLW is expected to generate less than 200 W per canister, as is 89.9% (~7,037) of Savannah River glass HLW (Wilson, 2016). Similarly in the DGRDMSH inventory, 94.1% of DSNF (~2,337 canisters, excluding Naval SNF) generated less than 200 W per canister in 2010 (Wilson, 2016). Assuming a thermal load of 100 W per canister, the total ~21,146 canisters would generate only about 2 MW of heat, less than 3% of that expected in a commercial repository. In a DGRDMSH, waste package and drift spacing may not be defined based on temperature constraints, rather those design parameters may be delineated more by the mechanical strength of the host rock and other engineering concerns.

Because most DOE-managed waste generates less heat than commercial waste, the magnitude of peak temperatures in a DGRDMSH would be considerably less than peak temperatures in a repository with CSNF. For instance, Stein et al. (2016a) predicted peak temperatures just under 200°C in a crystalline CSNF repository, while Stein et al. (2016b) predicted peak temperatures of approximately 85°C in a similar repository layout containing only DOE-managed (as) HLW and SNF. The magnitude of peak

repository temperatures will affect multiple repository processes, including re-saturation timing; thermal buoyancy driven fluid flow; waste package degradation; waste form dissolution; buffer and host rock alteration; creep consolidation; and radionuclide solubility, sorption, and diffusion. For instance, Stein et al. (2016a) and (2016b) use the same temperature-dependent function for waste package degradation. In the commercial repository analyses results show that 50% of waste packages breach by ~22,000 years; whereas the DGRDMSH analyses results show 50% breach by ~44,000 years.

Chemical and physical characteristics

Commercial SNF consists of low enrichment UO_2 fuels plus a small amount of mixed oxide fuels (SNL, 2014). Though variations exist in assembly and cladding materials, initial enrichment, and discharge burnup, the CSNF waste stream is relatively homogeneous. The DGRDMSH inventory, by comparison, contains a large number of disparate waste forms, and the chemical and physical characteristics of many of them may require specific consideration in repository planning, FEPs screening, and PA implementation.

DGRDMSH waste forms can be sorted into three broad categories based on dissolution mechanism: 1) oxide fuels that will degrade in a similar fashion to CSNF (UO_2 degradation); 2) glass and ceramic waste forms that will experience rate-controlled dissolution over some portion of their lifespan (glass degradation); and 3) soluble salts and metals that will undergo essentially instantaneous dissolution. In all of the PA models, there is the degradation of the waste form representing a kinetic process followed by imposition of solubility limits for dissolved radionuclides controlled in part by bulk chemistry of the disposal environment. The variety of dissolution mechanisms for the DGRDMSH inventory will contribute to heterogeneity in the timing and nature of radionuclide release throughout the repository.

The dissolution of halide-containing salt wastes if disposed directly would be evaluated also for potential to generate a corrosive repository environment and adversely affect the performance of adjacent waste packages. Disposing of these wastes in a salt repository would mitigate this as an issue as the salt disposal concept does not rely upon waste package integrity. Segregation of these wastes may be desirable in other reference disposal concepts. Several DOE-managed (as) HLW waste types contain RCRA-regulated wastes. These include the tank waste at Hanford, calcine waste, Na-bearing waste at INL, and Cs and Sr capsules; the planned final waste forms will likely not be governed by RCRA, but some alternate disposal forms may need additional evaluation for RCRA. Some DSNF and the salt HLW resulting from EMT of Na-bonded SNF (though relatively small in volume) contain enough fissile material that criticality needs to be considered and/or managed. The high concentration of Cl^- in a salt repository would help prevent the occurrence of critical reactions when these wastes become wet. DGRDMSH waste forms containing soluble plutonium (i.e., direct-disposed salt waste from EMT of Na-bonded fuels) would be evaluated for the potential for formation of plutonium colloids. In a crystalline repository, such colloids if they moved through or around the bentonite buffer would have some potential to travel quickly and far in the fractured host rock.

Packaging

DOE plans to package most of its SNF (about 98% by MTHM) in multi-canister overpacks (MCOs) and standardized canisters suitable for storage, transport, and disposal (SNL, 2014). All such canisters share similar dimensions: MCOs are 24” in diameter and 166” long; standardized canisters are 18” or 24” inches in diameter and 10’ or 15’ long. MCOs and standardized canisters are smaller than the smallest proposed CSNF waste package, the 4-PWR waste package, which has a diameter of 0.82 m and length of 5 m (Hardin et al., 2012). Standardized canister internal basket assemblies will serve to control criticality by limiting the number of assemblies within a canister and by providing neutron absorbing material if necessary (SNL, 2014).

Glass HLW from Savannah River and Hanford sites is or will be packaged in canisters similar in size to the standardized canisters. Savannah River canisters are 24” diameter by 10’ long; Hanford canisters will

be 24” diameter by 15’ long. If calcine waste is vitrified, its canisters will be the same size as Savannah River canisters (SNL, 2014).

Naval SNF is, or will be, packaged in large canisters that were designed to fit inside the proposed waste package for the Yucca Mountain repository (SNL, 2014). These canisters are 66” in diameter and either 185.5” or 210.5” long, and are comparable in size to the largest canister proposed for CSNF disposal, the dual-purpose canister (DPC), which has a diameter of 2 m and a length of 5.13 m (Hardin et al., 2012). Naval SNF canisters and their internal components will provide shielding and control the risk of criticality (SNL, 2014). HIP calcine waste is planned to be packaged in the same size canister as Naval SNF (SNL, 2014; though an alternative HIP calcine waste form in 2’ by 10’ DOE SNF standard canisters in the DGRDMSH inventory of Wilson, 2016). The large openings (shafts and/or ramps) that facilitate emplacing the largest waste packages are more easily maintained in crystalline rock than in formations in which creep and visco-plastic deformation occur.

2.5.1.3 Summary of Variations on Disposal Concepts

A low-temperature DGRDMSH would differ in the following primary aspects compared to a repository including CSNF:

- A DGRDMSH would be smaller than a 70,000-MTHM CSNF repository due to the smaller waste volume.
- A DGRDMSH would contain a higher percentage of short-lived fission products than a CSNF repository. This alters the timing of peak repository temperatures and of transient temperature-dependent processes including re-saturation; buoyancy driven fluid flow; waste package degradation; waste form dissolution; buffer and host rock alteration; and creep consolidation.
- A DGRDMSH would experience a thermal load on the order of 3% of the thermal load in a 70,000 MTHM CSNF repository, allowing for smaller distances between drifts and waste packages. This would reduce issues regarding temperature-dependent processes including, for instance, waste package degradation and buffer and host rock alteration.
- A DGRDMSH may present unique challenges related to the chemical and physical characteristics of some waste forms. The effects of corrosive waste, highly soluble waste, and colloid-forming waste on repository performance should be considered. The presence of RCRA-regulated waste in some alternate waste form pathways may need to be considered.
- A DGRDMSH packaging plan results in a bimodal distribution of waste package sizes. Large waste packages may create engineering challenges in some disposal concepts.

These DGRDMSH considerations could be evaluated in more detailed design analyses and evaluations of those Features, Events, and Processes (FEPs) that would be handled substantively differently for a DGRDMSH Repository concept versus one that included CSNF. Because of the various reliance on engineered (most for crystalline/granite repository concepts and least for salt repository concepts), the list of altered FEPs could be different depending on the specific geologic disposal system being evaluated. The FEPs process allows for direct linkage to those aspects of the disposal option (combined waste forms and repository concept) that could be evaluated for a DGRDMSH.

3. STATUS FOR MANAGING INVENTORY DATA AND FOR POST-CLOSURE PERFORMANCE ASSESSMENTS OF REPOSITORY CONCEPTS

This section summarizes the progress made on designing and developing an on-line waste library (OWL) database to manage the information of the wastes and waste forms from the WFDOE (SNL, 2014), and describes the status of constraints on waste form degradation for post-closure performance assessments. In Section 3.1 the OWL database is described with updated status information from FY2017. Both the OWL database model (Appendix B) and an updated user's guide to the OWL database (Section 3.1.3) are provided. Unchanged from Sassani (2016), Section 3.2 provides various waste form performance constraints for postclosure performance assessment in the context of the GDSA effort.

3.1 Developing the Online Waste Library (OWL)

The OWL has two primary purposes: one purpose, already mentioned, is providing in one place information on the many different DOE-managed wastes that are likely to require deep geologic disposal, such that one can easily query the data. A second purpose is as the primary source for information on the waste types, inventory, and waste form characteristics necessary to develop a database of parameters for a performance assessment (PA) analysis for a repository safety assessment. The initial focus in this activity will be to develop the database with a user-friendly interface and to populate it with the information on waste types and waste forms. Linking OWL directly to performance modeling through a parameter database in order to facilitate PA analysis will occur in subsequent activities after the OWL is populated more comprehensively for waste types and forms.

The Siting Experience Archive (SEA) was developed at SNL for DOE in FY13 (Price et al., 2013). The SEA is a website and database of various experiences, primarily in the U.S., on siting large controversial projects. Although SEA cannot serve as an exact template, SEA has many of the attributes and features required for the implementation of OWL. To facilitate OWL development, the same team that designed the SEA database and interface has been engaged for developing OWL, such that desirable similarities are retained and development of OWL is efficient.

Although the OWL will likely be available through the world-wide web, initial FY2016 prototype development was restricted to the internal SNL network until FY2017 where the OWL has been put on an external interface (for testing by limited DOE and National Laboratory staff). The OWL is now functioning on the SNL External Collaboration Network (ECN). The ability to display various attributes of the information on waste forms was identified as an important function of OWL. The level of support for active databases will determine the type of arrangements that may be practical. As much as possible, the OWL will leverage existing databases to minimize duplication of effort.

3.1.1 Description

The OWL has been designed to contain information regarding DOE-managed high-level waste (HLW), spent nuclear fuel (SNF), and other wastes that are likely candidates for deep geologic disposal, with links to the current supporting documents for the data (when possible; note no classified or official-use-only data are planned to be included at this point). There may be up to several hundred different DOE-managed wastes that are likely to require deep geologic disposal. The DOE has a database (Spent Fuel Database-SFDB) that contains information regarding the SNF that DOE manages. We do not intend to replicate this database and the information in it, but would take advantage of that existing dataset to incorporate it into the on-line waste library for use in post-closure PA.

In addition to the data received from the SFDB, each waste (and its alternative waste forms) listed in the OWL could include (many already incorporated):

- Waste Characteristics
  - Narrative description of waste (some wastes that have variable processing characteristics, e.g., Savannah River tank waste, some of which has been processed and some of which has not; sodium-bonded fuel, some of which has been treated and some of which has not; Hanford tank waste once treatment starts such that some of it is treated and some is not)
  - Type of waste (HLW or SNF or other)
  - Origin of waste (commercial, DOE-managed (as), foreign, research, other?)
  - Total quantity of waste (volume and/or mass (as appropriate))
  - Physical form of waste (e.g., rods, plates, powder, liquid, glass)
  - Dimensional characteristic of waste (if a solid waste)
  - Radionuclide inventory and thermal information at specified times (e.g., at inception; at 2015; at 2048)
  - Bulk chemistry of the waste (noting hazardous constituents)
  - RCRA considerations (e.g., not an issue, characteristic, listed)
- Current storage information
  - Current storage location (e.g., INL, Hanford, perhaps more specific?)
  - Description of current storage method (e.g., tanks, canisters, high-integrity canisters, capsules)
  - Number of current containers
  - Dimensions of current storage method (per container, as appropriate)
  - Volume of current storage method (per container, as appropriate)
  - Mass of packaged waste as it currently exists (per container, as appropriate)
  - Radionuclide inventory and thermal information at specified times on a per-container basis (or as available)
  - Current status (e.g., awaiting treatment, awaiting packaging, ready for disposal)
- Planned processing and packaging for final disposition (identify which wastes have baseline processing and packaging plan with a yes/no field. Supply the information listed below for the baseline processing and packaging planned. If alternative processing and/or packaging options exist, provide information listed below for all alternative processing/packaging options)
  - Description of baseline/alternative processing and/or packaging for disposal, including options for processing and/or packaging
  - Number of baseline/alternative packages
  - Dimensions of baseline/alternative package
  - Volume of baseline/alternative package
  - Mass of baseline/alternative package
  - Will baseline/alternative package fit in a deep borehole? (yes/no)
  - Status of baseline/alternative planned processing (e.g., none, in progress, under development)
  - Status of baseline/alternative packaging (e.g., ready, being developed)
  - Radionuclide inventory and thermal information for treated/packaged waste at specified times on a per-package basis (or as available)

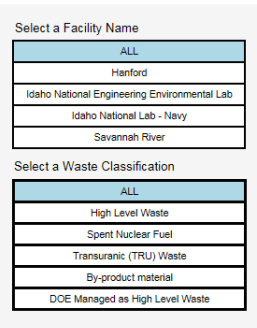
- Transportation considerations (e.g., certified transport canister exists (yes/no))
- Current base-line disposition pathway (e.g., deep geologic disposal in repository for HLW and/or SNF, WIPP, TBD)
- Copies of any Records of Decision (RODs) or agreements affecting the waste and its associated plans (linked to the specific data provided)
- Effects of RODs on waste (e.g., date of promised removal from state)
- Responsible contacts currently in charge of the waste types and forms (name, phone number, email address) for storage oversight, for processing, etc.

3.1.2 OWL Development Status

OWL is designed to contain information regarding all the radioactive waste that the DOE manages as high-level waste (DHLW) and the DOE managed spent nuclear fuel (DSNF), and to be able to disseminate that information. Currently, the OWL database contains information for eleven different wastes with fifteen potential (planned) waste form pathways and three existing waste forms defined.

Because of the way the database is structured, users can sort on waste by facility (Example: Hanford, Idaho, Savannah River, etc.), and waste classification (Example: HLW, SNF, etc.). This feature makes it easy to identify all the HLW types that are currently at Hanford, for example, which is similar to the DOE SFDB capabilities.

The following is a screenshot of the visual display where users can select wastes by Facility and/or Waste Classification as well as sort by Waste, Classification, or Storage Facility (using the up/down arrows).



Waste (BaseLine Inventory Date)		Waste Classification	Waste Description	Storage Facility
Calcine Waste (2016-01-01)	Details	High Level Waste	This waste is a solid granular material derived from liquid wastes produced by reprocessing SNF.	Idaho National Engineering Environmental Lab
Cesium and Strontium Capsules (2016-01-01)	Details	High Level Waste	This waste consists of 1335 CsCl capsules and 601 SrF2 capsules, each about 21 inches tall and 3 inches in diameter. They are currently managed as high-level waste and stored in pools at the Waste Encapsulation and Storage Facility at Hanford	Hanford
German Glass Waste (1987-01-01)	Details	Transuranic (TRU) Waste	This waste consists of 34 canisters of glass prepared by Pacific Northwest Laboratory to provide heat and radiation sources for repository testing by the Federal Republic of Germany in the Asse salt mine. This waste has been classfied as RH-TRU but does not meet the requirements of the WIPP Waste Acceptance Criteria and so cannot be disposed of at the WIPP. Two of the 34 canisters are thought to contain depleted uranium and natural thorium, but no cesium or strontium. Quantities of Cs-135 are unknown. The 34 canisters are currently stored in 6 CASTOR casks and 2 GNS casks.	Hanford
Hanford Tank Waste (CH-TRU) (2008-01-01)	Details	Transuranic (TRU) Waste	This waste is material that can be contact handled (CH) and is a subset of the 54.6 million gallons of liquid waste stored at Hanford. It may be transuranic (TRU) waste but has not officially been determined to be so by the DOE.	Hanford
Hanford Tank Waste (HLW) (2008-01-01)	Details	High Level Waste	This waste is a subset of the 54.6 million gallons of liquid waste stored at Hanford	Hanford
Hanford Tank Waste (RH-TRU) (2008-01-01)	Details	Transuranic (TRU) Waste	This waste is material that can be remotely handled (RH) and is a subset of the 54.6 million gallons of liquid waste stored at Hanford. It may be transuranic (TRU) waste, but has not officially been determined to be so by the DOE.	Hanford
N-Reactor Spent Fuel (1998-05-31)	Details	Spent Nuclear Fuel	This waste consists of 2,096 metric tons of N-Reactor spent fuel that is currently stored in about 388 multi-canister overpacks in the Canister Storage Building at Hanford.	Hanford
Naval spent fuel (2016-01-01)	Details	Spent Nuclear Fuel	This waste consists of about 20 metric tons (as of 2015) spent naval fuel that is currently stored at INL	Idaho National Lab - Navy
Savannah River Glass Waste (2016-04-04)	Details	High Level Waste	This waste consists of 4,000 vitrified glass logs that were formed by the Defense Waste Processing Facility at Savannah River from waste that was in tanks at Savannah River	Savannah River
Savannah River Tank Waste (2016-04-04)	Details	High Level Waste	This waste consists of approximately 36 million gallons of sludge, supernate, and salt currently stored in tanks at Savannah River.	Savannah River
Sodium Bearing Waste (2012-01-01)	Details	Transuranic (TRU) Waste	This waste is composed primarily of decontamination solutions, but includes small fractions of first (1%), second (2%) and third (4%) cycle extraction wastes from fuel reprocessing.	Idaho National Engineering Environmental Lab

Figure 3-1. Visual display of wastes, waste classification, description and storage facility

The OWL database is designed to capture the following information about a particular waste:

- Facility where it is currently located;
- Nominal waste classification (e.g., high level waste or spent nuclear fuel);
- Whether it was produced by the government;
- Whether it is mixed waste;
- The radionuclide inventory on a baseline date;
- The source of the waste;
- Contact information for a person knowledgeable about the waste;
- Contaminants present in the waste;
- Average, minimum, and maximum thermal output of a unit of the waste (the unit is waste specific);
- Dimensions of a container of the waste (the container is waste specific);
- Volume of the waste as currently stored;
- Radioactivity of the waste (as of the baseline date or calculated for another date);
- Radionuclide characteristics;
- The planned waste form to be used for disposal, and potential alternative waste forms;
- Average thermal output of the disposal waste form;
- Dimensions of the disposal waste form (including disposal packaging);
- Volume of the disposal waste form (including disposal packaging);
- Mass of the disposal waste form (including disposal packaging);

Because there is a large variety of waste information, the waste detail is organized into sections that can be selected for display.

Figure 3-2 provides a sample screenshot of the waste detail for Savannah River Glass Waste with Waste Characteristics and Disposal Waste Form information displayed.

OnLine Waste Library (OWL)

Waste Detail Report

Savannah River Glass Waste

Waste Classification	Waste Description	Storage Facility	Produced By	Is Mixed Waste?	Baseline Inventory Date	
High Level Waste	This waste consists of 4,000 vitrified glass logs that were formed by the Defense Waste Processing Facility at Savannah River from waste that was in tanks at Savannah River	Savannah River	Government	No	4/4/2016	Projected Inventory

Display Specific Waste Information by Expanding (+) the Type of Content Listed Below

⊟ 1. Waste Characteristics ⊟ 3. Disposal Waste Forms ⊞ 5. Radionuclide Inventory ⊞ 7. Waste Supporting Documents

⊞ 2. Waste Source ⊞ 4. Disposal Waste Form Characteristics ⊞ 6. Radionuclide Characteristics ⊞ 8. Waste Contacts

1. Waste Characteristics

Characteristic	Nuclear Waste Characteristic		Supporting Document
Average thermal output of a unit of the nuclear waste	Average thermal output of a canister of glass waste as of the baseline date	47 Watts	Open
Diameter of the nuclear waste container	Diameter of a container of glass waste	2 Feet	Open
Length of the nuclear waste container	Height of a container of glass waste at Savannah River	10 Feet	Open
Number of containers	Number of containers of glass waste at Savannah River	4,000	Open
Physical form of the waste	Physical form of the glass waste at Savannah River	Borosilicate glass	Open
Total radioactivity - the total curies of all the radionuclides in the waste as of the baseline date	Total radioactivity of glass waste at Savannah River	55,000,000 Curies	Open
Total volume of the waste as currently stored, including any packaging	Total volume of glass waste at Savannah River	124,000 Cubic Feet	Open

3. Disposal Waste Forms

Waste Form	Description	Planned or Existing	Preferred or Alternative	Quantity	Supporting Document
Glass waste	Glass logs in canisters	Existing	Preferred	4,000 2 ft. diameter, 10 ft. tall canisters	Open

Figure 3-2. Waste detail sections available for display with Waste Characteristics and Disposal Waste Form data selected.

To support Waste details, information on 86 radionuclides is also captured in the database. Figure 3-3 provides a screenshot from a database report showing a sample of the radionuclide information:

OnLine Waste Library (OWL)

Radionuclides (86)

Radionuclide	ID	Description	Half Life	Atomic Mass	Thermal Output	Parent Radionuclide	ID	Inventory Ratio	Supporting Document ID & Availability		
Ac-227	51	Actinium 227	21.77 Years	227.00					3172	Internal Full Document	Open
Al-26	46	Aluminum 26	717000 Years	26.00					3175	Internal Full Document	Open
Am-241	41	Americium 241	432.6 Years	241.00	32.450	Pu-241	39		1067	Internal Full Document	Open
Am-242	74	Americium 242	16.02 Hours	242.00		Am-242m	64	0.995	3176	Internal Full Document	Open
Am-242m	64	Americium 242 metastable	141 Years	242.00					3177	Internal Full Document	Open
Am-243	52	Americium 243	7370 Years	243.00					3178	Internal Full Document	Open
Ba137-m	1	Barium 137 metastable	2.552 Minutes	137.00	3.920	Cs-137	3	0.950	1068	Internal Full Document	Open
Bk-247	67	Berkelium 247	1380 Years	247.00					3179	Internal Full Document	Open
C-14	8	Carbon 14	5700 Years	14.00					3180	Internal Full Document	Open
Cd-113m	61	Cadmium 113 metastable	14.1 Years	113.00					3181	Internal Full Document	Open
Ce-144	24	Cerium 144	284.91 Days	144.00					3183	Internal Full Document	Open
Cf-249	68	Californium 249	351 Years	249.00					3184	Internal Full Document	Open
Cf-251	69	Californium 251	898 Years	251.00					3185	Internal Full Document	Open

Figure 3-3. OWL Database Report sample of radionuclides.

The OWL database can also calculate the inventory of a given waste/waste form in a given year (between 1950 and 6099). OWL database reports can be generated to provide the inventory in various units, such as

volumes, radioactivity, and/or thermal output of wastes as they currently exist, and in their planned (or proposed) disposal waste form(s).

Figure 3-4 provides an example screenshot of the projected inventory database report for Sodium Bearing Waste from the baseline inventory date to the selected target year 2200:

Radionuclide Inventory Calculation* - Target Year: 2200

Transuranic (TRU) Waste

Waste (Base Line Inventory Date)	Radionuclide	Half Life	BASELINE			PROJECTED		
			Inventory (curies)	Inventory (grams)	Thermal Ouput (watts)	Inventory (curies)	Inventory (grams)	Thermal Output (watts)
Sodium Bearing Waste (2012-01-01)	Americium 241	432.600 Years	3.16E+002	9.22E+001	1.03E+001	2.36E+003	6.87E+002	7.65E+001
	Barium 137 metastable	2.552 Minutes	1.57E+005	2.92E-004	6.15E+002	2.05E+003	3.81E-006	8.03E+000
	Cobalt 60	5.270 Years	2.16E+001	1.91E-002	0.00E+000	0.00E+000	0.00E+000	0.00E+000
	Cesium 134	2.065 Years	4.32E+000	3.34E-003	0.00E+000	0.00E+000	0.00E+000	0.00E+000
	Cesium 137	30.080 Years	1.66E+005	1.91E+003	1.84E+002	2.16E+003	2.49E+001	2.39E+000
	Europium 154	8.600 Years	1.78E+002	6.60E-001	0.00E+000	4.49E-005	1.66E-007	0.00E+000
	Niobium 94	20,300.000 Years	1.70E+001	9.08E+001	0.00E+000	1.69E+001	9.02E+001	0.00E+000
	Neptunium 237	2,144,000.000 Years	1.74E+000	2.47E+003	0.00E+000	1.74E+000	2.47E+003	0.00E+000
	Plutonium 238	87.700 Years	3.90E+003	2.28E+002	1.27E+002	8.79E+002	5.13E+001	2.86E+001
	Plutonium 239	24,110.000 Years	4.10E+002	6.61E+003	1.25E+001	4.08E+002	6.58E+003	1.24E+001
	Plutonium 240	6,561.000 Years	1.53E+002	6.74E+002	4.67E+000	1.50E+002	6.61E+002	4.58E+000
	Antimony 125	2.760 Years	1.21E+001	1.17E-002	0.00E+000	0.00E+000	0.00E+000	0.00E+000
	Strontium 90	28.900 Years	1.09E+005	7.93E+002	1.26E+002	1.19E+003	8.63E+000	1.38E+000
	Uranium 233	159,200.000 Years	3.60E-002	3.74E+000	0.00E+000	3.60E-002	3.73E+000	0.00E+000
	Uranium 234	245,500.000 Years	5.33E+000	8.57E+002	0.00E+000	5.33E+000	8.56E+002	0.00E+000
	Uranium 235	704,000,000.000 Years	1.27E-001	5.88E+004	0.00E+000	1.27E-001	5.88E+004	0.00E+000
	Uranium 236	23,420,000.000 Years	2.23E-005	3.45E-001	0.00E+000	2.23E-005	3.45E-001	0.00E+000
	Uranium 238	4,468,000,000.000 Years	1.25E-001	3.72E+005	0.00E+000	1.25E-001	3.72E+005	0.00E+000
	Yttrium 90	64.053 Hours	1.09E+005	2.01E-001	6.03E+002	1.19E+003	2.18E-003	6.56E+000
TOTAL			5.46E+005	4.44E+005	1.68E+003	1.04E+004	4.42E+005	1.40E+002

Figure 3-4. Visual display of calculated projected inventory from the baseline inventory to the target year 2200.

In addition to providing the ability to calculate projected inventory for a specific target year, the database now provides a calculation of the projected inventory by year for the next 200 years. The calculation can be plotted in a charts (Figure 3-5) that allow selection by waste, waste classification, and radionuclide.

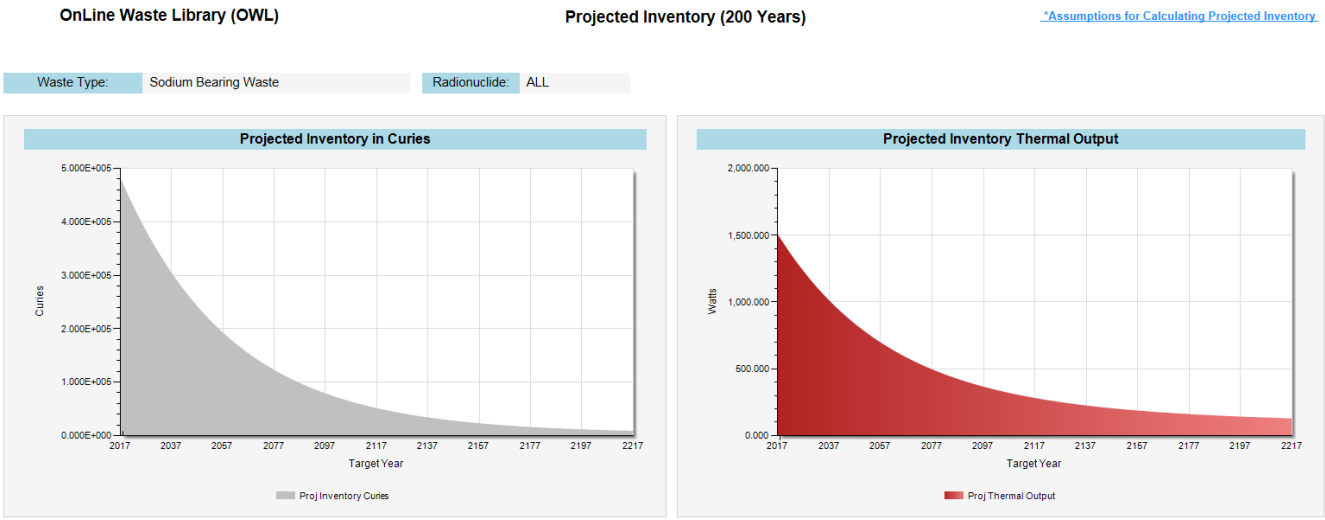


Figure 3-5. Visual display of calculated projected activity and thermal output for a waste by year for the next 200 years.

A further effort in FY2017 loaded 'supporting documents' into the OWL to provide the underpinning sources and to supplement the database content. There are currently 237 documents are integrated with the database content and can be viewed from within OWL. Figure 3-6 provides a screen shot sample of documents available.

Title
Canister Thermal Output
Capsule by Capsule Inventory
Capsule Info and Diagrams
Capsule Volume
Cs-135 Inventory
Cs Contaminants
Glass Canister Filled Mass
Glass Canister Total Volume
Hanford Glass Canister
Radioactivity Heat Dose Capsules
Sr Contaminants
Vitrification of Cs and Sr Capsules
Groundwork for Universal Canister System Development
Thermal Analysis of a Dry Storage Concept for Capsule Dry Storage Project
Evaluation of Options for Permanent Geologic Disposal of Spent Nuclear Fuel and High Level Radioactive Waste in Support of a Comprehensive National Nuclear Fuel Cycle Strategy: Volume II: Appendices
Tank Closure and Waste Management Environmental Impact Statement for the Hanford Site, Richland, Washington (TC & WM EIS), Appendix E

Figure 3-6. Database Report sample of supporting documents

Starting in FY2018, future work on the OWL database includes the following:

- Continue to add the full set of information regarding the other wastes from the WFDOE (SNL, 2014)(i.e., fully populate the OWL for previously identified waste types and waste form pathways)
    - The focus for FY2018 will be coordinating/synchronizing with the DOE SNF database at INL to leverage that dataset for purposes of GDSA assessment purposes
- Continue to refine the set of documentation for the OWL database architecture, including a comprehensive user's guide (see Section 3.1.3 for OWL user's guide)
- Continue the review and verification process to ensure information in the OWL is accurate and sourced correctly, including
    - Continuation/completion of the external BETA testing
    - It is intended in FY2018 to develop an external review process for the content of OWL
- Define an update processes (this will be done in conjunction with user review and feedback on the OWL) to
    - maintain current information linked to new or revised DOE documents
    - delineate additional features/capabilities to add to the OWL
    - add new waste types and waste forms as they are identified.

The activities in the first bullet above are a priority for FY2018 activities, as is completing the external BETA test for the OWL. The fourth bullet above represents the path for maintaining and expanding the utility of the OWL in the future. The OWL is intended to facilitate coherent analyses regarding the back end of the fuel cycle with respect to the full range of DGRDMSH wastes and waste forms.

3.1.3 User's Guide to OWL Database

The OWL is accessed through the external collaboration network (ECN) at Sandia National Laboratories, which requires an account with username and password to login to the network and for accessing the SharePoint and network facilities on which the OWL is implemented. The detailed model structure of the OWL is given in Appendix B, and an overview of the implementing architecture is shown in Figure 3-7.

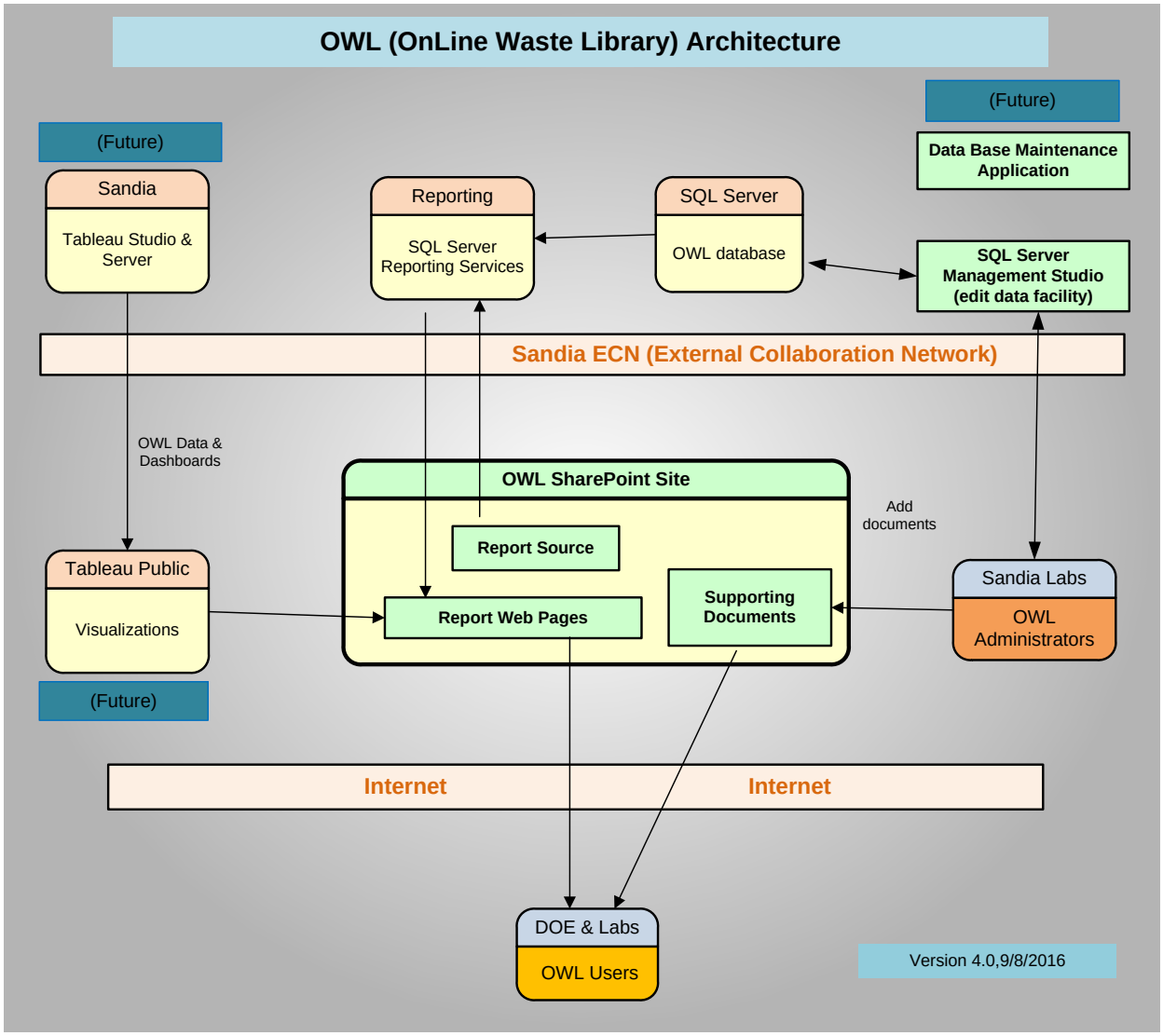


Figure 3-7. The high-level architecture of the OWL implementation as constructed on the Sandia National Laboratories External Collaboration Network (ECN).

The usage of the OWL is via straightforward access to a homepage within the SharePoint Site and a user's guide for the OWL is given here to demonstrate the various options for queries and reports from the database. The OWL home screen is shown in Figure 3-8. From this screen, the user can search for the information that is contained in the database.

Currently, users can:

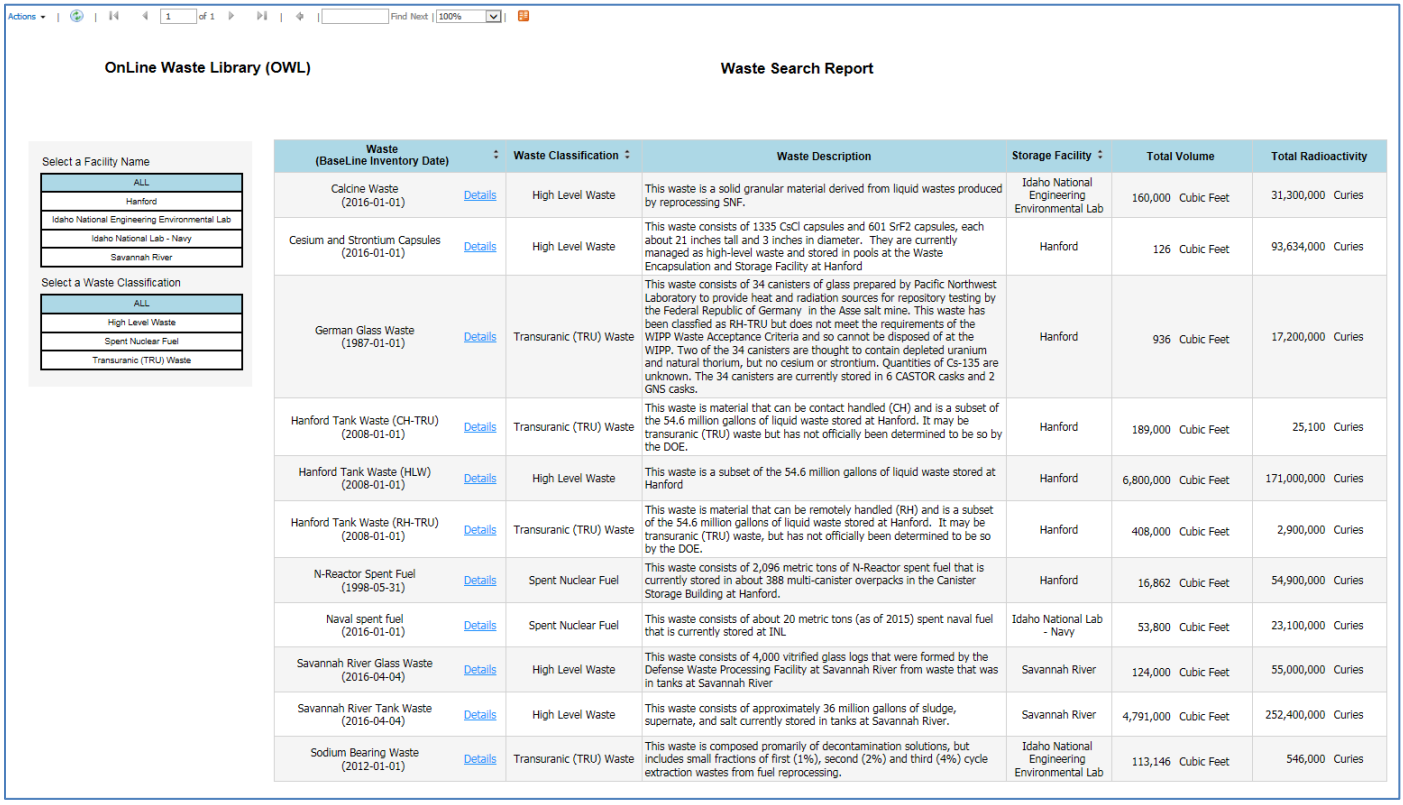
- Search on all the wastes, view waste details and supporting documents;
- Search waste disposal forms, their related wastes, and supporting documents;
- View radionuclides, their properties, and view supporting documents;
- Calculate the inventory of a selected waste in a chosen year;
- Display the projected inventory of wastes and radionuclides by year for the next 200 years;
- Display List of Supporting Documents with the ability to open the documents;

These capabilities as organized by the "Find Information About" report selectors shown in Figure 3-8 are described in more detail below (each report selector given in bold text below).

Figure 3-8. Home Screen for OWL.

DOE Managed Wastes – searching for wastes (see Figure 3-8)

From the home screen in Figure 3-8, selecting "DOE Managed Wastes" allows the user to see all the wastes that are currently in OWL. The user can search by waste location (Hanford, Idaho, Savannah River, etc.,) and waste classification (high level waste, spent nuclear fuel, Transuranic). The Waste Search Report is shown in Figure 3-9. This report gives a summary of each waste (classification, description, location, volume, radioactivity).



OnLine Waste Library (OWL)

Waste Search Report

Select a Facility Name

ALL
Hanford
Idaho National Engineering Environmental Lab
Idaho National Lab - Navy
Savannah River

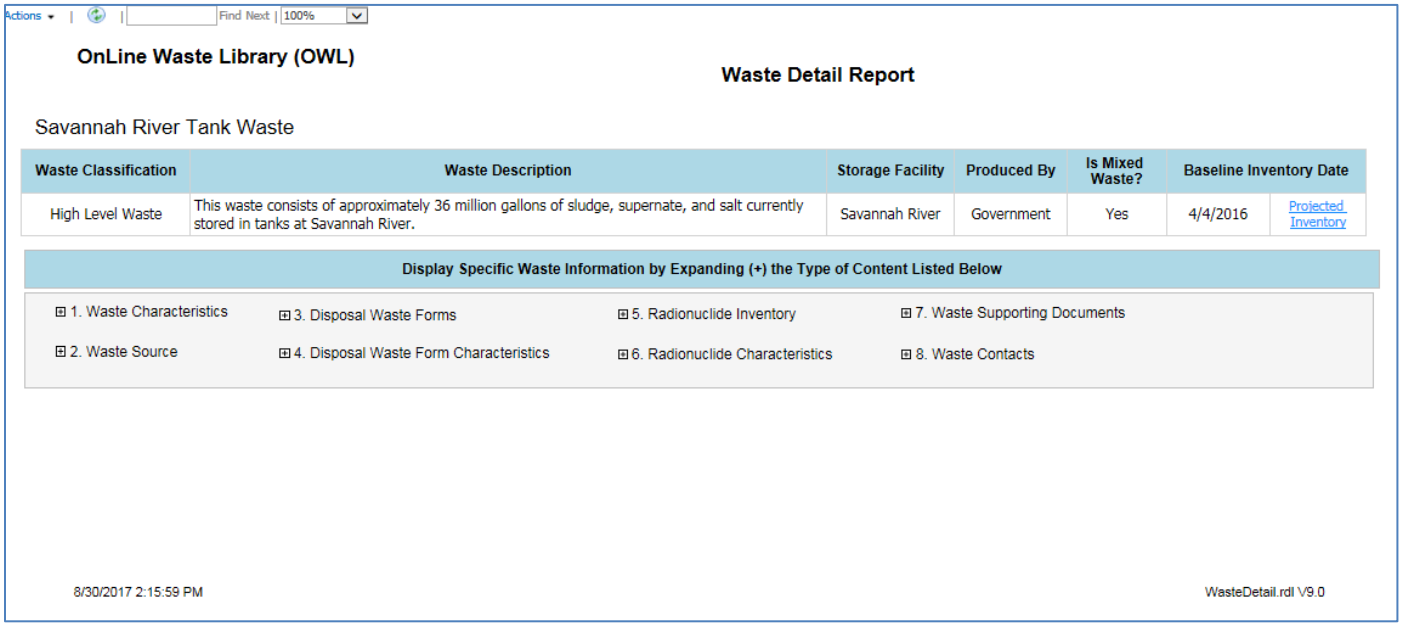
Select a Waste Classification

ALL
High Level Waste
Spent Nuclear Fuel
Transuranic (TRU) Waste

Waste (BaseLine Inventory Date)		Waste Classification	Waste Description	Storage Facility	Total Volume	Total Radioactivity
Calcine Waste (2016-01-01)	Details	High Level Waste	This waste is a solid granular material derived from liquid wastes produced by reprocessing SNF.	Idaho National Engineering Environmental Lab	160,000 Cubic Feet	31,300,000 Curies
Cesium and Strontium Capsules (2016-01-01)	Details	High Level Waste	This waste consists of 1335 CsCl capsules and 601 SrF2 capsules, each about 21 inches tall and 3 inches in diameter. They are currently managed as high-level waste and stored in pools at the Waste Encapsulation and Storage Facility at Hanford	Hanford	126 Cubic Feet	93,634,000 Curies
German Glass Waste (1987-01-01)	Details	Transuranic (TRU) Waste	This waste consists of 34 canisters of glass prepared by Pacific Northwest Laboratory to provide heat and radiation sources for repository testing by the Federal Republic of Germany in the Asse salt mine. This waste has been classified as RH-TRU but does not meet the requirements of the WIPP Waste Acceptance Criteria and so cannot be disposed of at the WIPP. Two of the 34 canisters are thought to contain depleted uranium and natural thorium, but no cesium or strontium. Quantities of Cs-135 are unknown. The 34 canisters are currently stored in 6 CASTOR casks and 2 GNS casks.	Hanford	936 Cubic Feet	17,200,000 Curies
Hanford Tank Waste (CH-TRU) (2008-01-01)	Details	Transuranic (TRU) Waste	This waste is material that can be contact handled (CH) and is a subset of the 54.6 million gallons of liquid waste stored at Hanford. It may be transuranic (TRU) waste but has not officially been determined to be so by the DOE.	Hanford	189,000 Cubic Feet	25,100 Curies
Hanford Tank Waste (HLW) (2008-01-01)	Details	High Level Waste	This waste is a subset of the 54.6 million gallons of liquid waste stored at Hanford	Hanford	6,800,000 Cubic Feet	171,000,000 Curies
Hanford Tank Waste (RH-TRU) (2008-01-01)	Details	Transuranic (TRU) Waste	This waste is material that can be remotely handled (RH) and is a subset of the 54.6 million gallons of liquid waste stored at Hanford. It may be transuranic (TRU) waste, but has not officially been determined to be so by the DOE.	Hanford	408,000 Cubic Feet	2,900,000 Curies
N-Reactor Spent Fuel (1998-05-31)	Details	Spent Nuclear Fuel	This waste consists of 2,096 metric tons of N-Reactor spent fuel that is currently stored in about 388 multi-canister overpacks in the Canister Storage Building at Hanford.	Hanford	16,862 Cubic Feet	54,900,000 Curies
Naval spent fuel (2016-01-01)	Details	Spent Nuclear Fuel	This waste consists of about 20 metric tons (as of 2015) spent naval fuel that is currently stored at INL	Idaho National Lab - Navy	53,800 Cubic Feet	23,100,000 Curies
Savannah River Glass Waste (2016-04-04)	Details	High Level Waste	This waste consists of 4,000 vitrified glass logs that were formed by the Defense Waste Processing Facility at Savannah River from waste that was in tanks at Savannah River	Savannah River	124,000 Cubic Feet	55,000,000 Curies
Savannah River Tank Waste (2016-04-04)	Details	High Level Waste	This waste consists of approximately 36 million gallons of sludge, supernate, and salt currently stored in tanks at Savannah River.	Savannah River	4,791,000 Cubic Feet	252,400,000 Curies
Sodium Bearing Waste (2012-01-01)	Details	Transuranic (TRU) Waste	This waste is composed promarily of decontamination solutions, but includes small fractions of first (1%), second (2%) and third (4%) cycle extraction wastes from fuel reprocessing.	Idaho National Engineering Environmental Lab	113,146 Cubic Feet	546,000 Curies

Figure 3-9. Waste Search Report

A user can obtain further details regarding a particular waste by clicking on "Details" next to the title of the particular waste. This opens another report, the Waste Detail Report (Figures 3-10), which provides more detailed information regarding the waste.



OnLine Waste Library (OWL)

Waste Detail Report

Savannah River Tank Waste

Waste Classification	Waste Description	Storage Facility	Produced By	Is Mixed Waste?	Baseline Inventory Date	
High Level Waste	This waste consists of approximately 36 million gallons of sludge, supernate, and salt currently stored in tanks at Savannah River.	Savannah River	Government	Yes	4/4/2016	Projected Inventory

Display Specific Waste Information by Expanding (+) the Type of Content Listed Below

⊞ 1. Waste Characteristics ⊞ 3. Disposal Waste Forms ⊞ 5. Radionuclide Inventory ⊞ 7. Waste Supporting Documents

⊞ 2. Waste Source ⊞ 4. Disposal Waste Form Characteristics ⊞ 6. Radionuclide Characteristics ⊞ 8. Waste Contacts

8/30/2017 2:15:59 PM

WasteDetail.rdl V9.0

Figure 3-10. Waste Detail Report

Figure 3-10 shows the top section of the Waste Detail Report, which gives the location of the waste, the entity that produced the waste, whether it is mixed, and the date of the baseline inventory. It also provides a link to display the "Projected Inventory" of the waste (the target year is selectable). More details will be provided below on this capability.

The next section provides a list of various waste information that can be displayed. To display the information, the user clicks on the [+] next to the description. The information will then be displayed in an area below the selection list and the [+] changes to [-] indicating it is being displayed. More than one category can be displayed at a time. Each new category of information selected is display in an area below the selection list. To remove the display of a category of information, click on the [-] next to the area that you want to remove from the display. The data is removed from the display and the [-] is then changed back to the [+].

Figure 3-11 shows an example of selecting "1. Waste Characteristics" and "3.Disposal Waste Forms"

Actions | Find Next | 100%

OnLine Waste Library (OWL)

Waste Detail Report

Savannah River Tank Waste

Waste Classification	Waste Description	Storage Facility	Produced By	Is Mixed Waste?	Baseline Inventory Date	
High Level Waste	This waste consists of approximately 36 million gallons of sludge, supernate, and salt currently stored in tanks at Savannah River.	Savannah River	Government	Yes	4/4/2016	Projected Inventory

Display Specific Waste Information by Expanding (+) the Type of Content Listed Below

⊟ 1. Waste Characteristics ⊟ 3. Disposal Waste Forms ⊞ 5. Radionuclide Inventory ⊞ 7. Waste Supporting Documents

⊞ 2. Waste Source ⊞ 4. Disposal Waste Form Characteristics ⊞ 6. Radionuclide Characteristics ⊞ 8. Waste Contacts

1. Waste Characteristics

Characteristic	Nuclear Waste Characteristic		Supporting Document
Diameter of the nuclear waste container	Nominal diameter of a Savannah River HLW tank	85 Feet	Open
Length of the nuclear waste container	Nominal height of a Savannah River HLW tank	32.8 Feet	Open
Number of containers	Number of tanks currently containing waste	43	Open
Physical form of the waste	Physical form of the tank waste	The tanks contain varying amounts of sludge, insoluble salts, and supernate.	Open
Total radioactivity - the total curies of all the radionuclides in the waste as of the baseline date	Total radioactivity of waste in SRS tanks	252,400,000 Curies	Open
Total volume of the waste as currently stored, including any packaging	Total volume of HLW currently in tanks at Savannah River	4,791,000 Cubic Feet	Open

3. Disposal Waste Forms

Waste Form	Description	Planned or Existing	Preferred or Alternative	Quantity	Supporting Document
Vitrified glass logs	Glass logs in canisters that contain the waste (sludge, supernate, and salt) that was in the SRS tanks as of 2013. The logs will be produced by the Defense Waste Processing Facility at Savannah River.	Planned	Preferred	4,200 2 ft. diameter, 10 ft. tall canisters	Open

Figure 3-11. Waste Detail Report – Waste Characteristics and Disposal Waste Forms

Waste characteristics can include contaminants, thermal output, dimensions, volume, and radioactivity of the waste. The Waste Forms include a description, whether the waste forms are planned, or existing, preferred or alternative, a description of the quantity and a supporting document (if available).

Many of the waste detail categories provide supporting documents that provide the basis for the information presented in the report, as well as documents that are general relevance to the waste. For

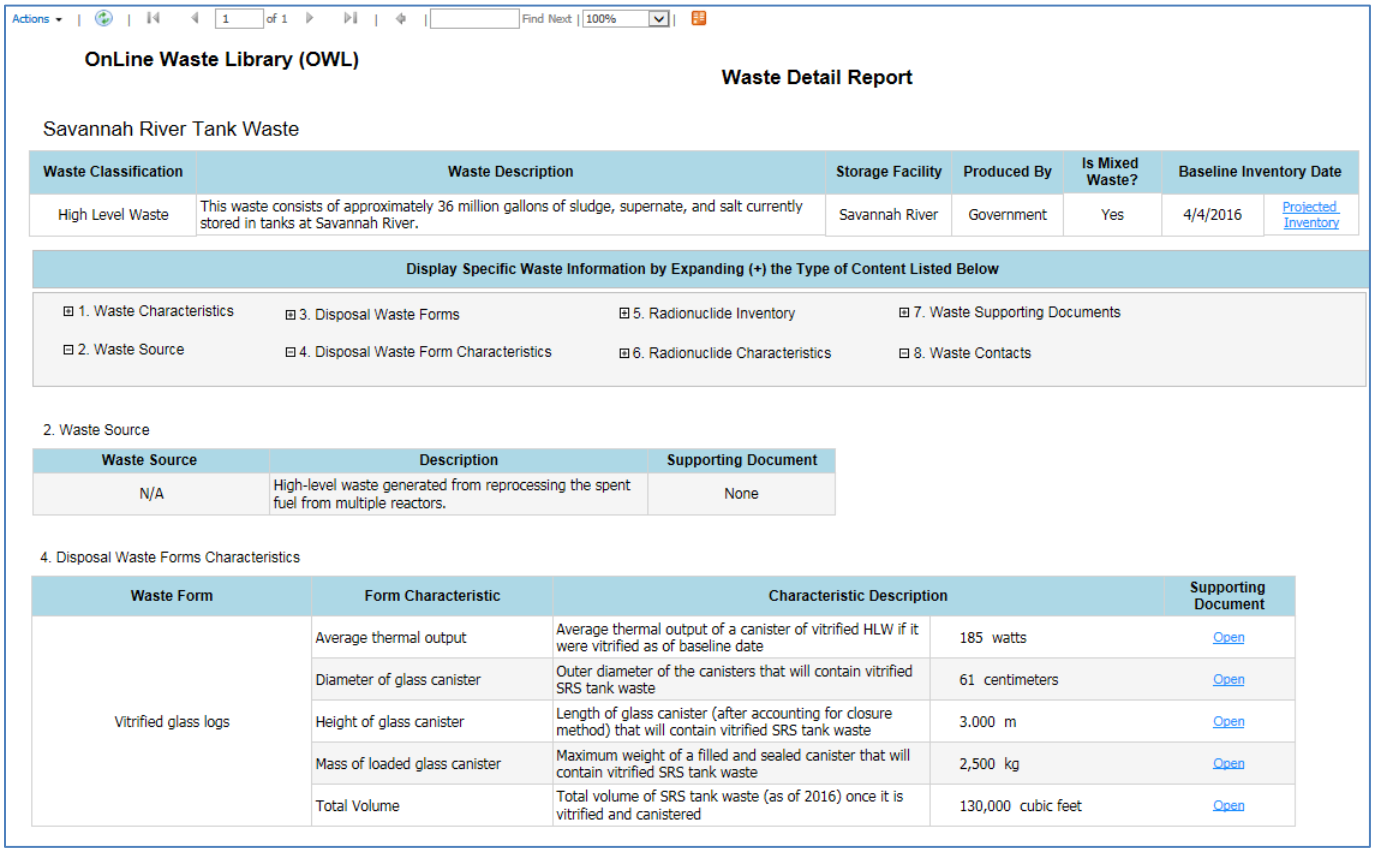
example, clicking on the "Open" under the supporting document column in the category "3. Disposal Waste Forms" for the waste form "Vitrified glass logs" will open a spreadsheet that calculates the number of expected canisters of vitrified waste.

Note that for each supporting document, clicking "Open" opens another browser window that allows the user to see the supporting document selected. Depending on the web browser being used, the document can be downloaded and/or saved, if desired.

In addition to Waste Characteristics and Waste Forms, the waste detail report provides:

- Waste Sources – a description of the source of the waste if available
- Disposal Waste Form Characteristics – waste form characteristics such as the average thermal output, dimensions of containers, etc. and supporting documents
- Radionuclide Inventory – list of the radionuclides in the waste and their inventory (in curies) and a link to the report where the radionuclide inventory can be projected to a selected target year.
- Radionuclide Characteristics – list of the radionuclides in the waste and their characteristics, which includes their half-life, etc.
- Waste Supporting Documents – data sheets for radionuclides in the waste, decay calculations, etc.
- Waste Contacts – names and contact information (email address, etc.) for persons who are designated to be the contact for the waste (if available).

Figure 3-12 is an example of the output for each of the information categories.



Actions | 1 of 1 | Find Next | 100%

OnLine Waste Library (OWL)

Waste Detail Report

Savannah River Tank Waste

Waste Classification	Waste Description	Storage Facility	Produced By	Is Mixed Waste?	Baseline Inventory Date	
High Level Waste	This waste consists of approximately 36 million gallons of sludge, supernate, and salt currently stored in tanks at Savannah River.	Savannah River	Government	Yes	4/4/2016	Projected Inventory

Display Specific Waste Information by Expanding (+) the Type of Content Listed Below

⊞ 1. Waste Characteristics ⊞ 3. Disposal Waste Forms ⊞ 5. Radionuclide Inventory ⊞ 7. Waste Supporting Documents

⊟ 2. Waste Source ⊟ 4. Disposal Waste Form Characteristics ⊞ 6. Radionuclide Characteristics ⊟ 8. Waste Contacts

2. Waste Source

Waste Source	Description	Supporting Document
N/A	High-level waste generated from reprocessing the spent fuel from multiple reactors.	None

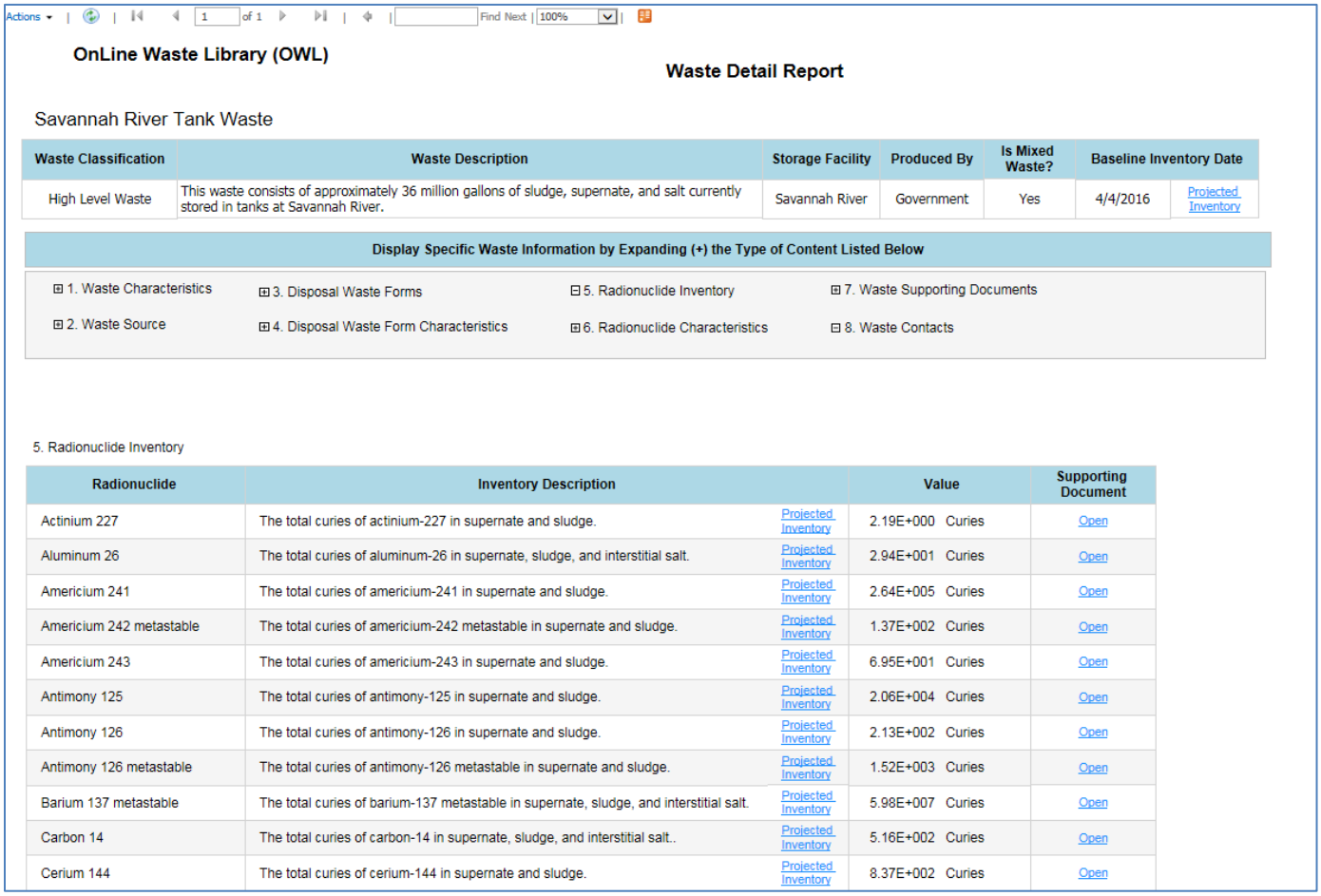
4. Disposal Waste Forms Characteristics

Waste Form	Form Characteristic	Characteristic Description		Supporting Document
Vitrified glass logs	Average thermal output	Average thermal output of a canister of vitrified HLW if it were vitrified as of baseline date	185 watts	Open
	Diameter of glass canister	Outer diameter of the canisters that will contain vitrified SRS tank waste	61 centimeters	Open
	Height of glass canister	Length of glass canister (after accounting for closure method) that will contain vitrified SRS tank waste	3.000 m	Open
	Mass of loaded glass canister	Maximum weight of a filled and sealed canister that will contain vitrified SRS tank waste	2,500 kg	Open
	Total Volume	Total volume of SRS tank waste (as of 2016) once it is vitrified and canistered	130,000 cubic feet	Open

Figure 3-12. Waste Detail Report – Waste Source and Disposal Forms Characteristics

Waste Sources – this provides a description of the source of the waste if available (e.g., Figure 3-12).

Disposal Waste Form Characteristics – this provides waste form characteristics such as the average thermal output, dimensions of containers, etc. and supporting documents (e.g., Figure 3-12).



Actions | 1 of 1 | Find Next | 100%

OnLine Waste Library (OWL)

Waste Detail Report

Savannah River Tank Waste

Waste Classification	Waste Description	Storage Facility	Produced By	Is Mixed Waste?	Baseline Inventory Date	
High Level Waste	This waste consists of approximately 36 million gallons of sludge, supernate, and salt currently stored in tanks at Savannah River.	Savannah River	Government	Yes	4/4/2016	Projected Inventory

Display Specific Waste Information by Expanding (+) the Type of Content Listed Below

⊞ 1. Waste Characteristics ⊞ 3. Disposal Waste Forms ⊟ 5. Radionuclide Inventory ⊞ 7. Waste Supporting Documents

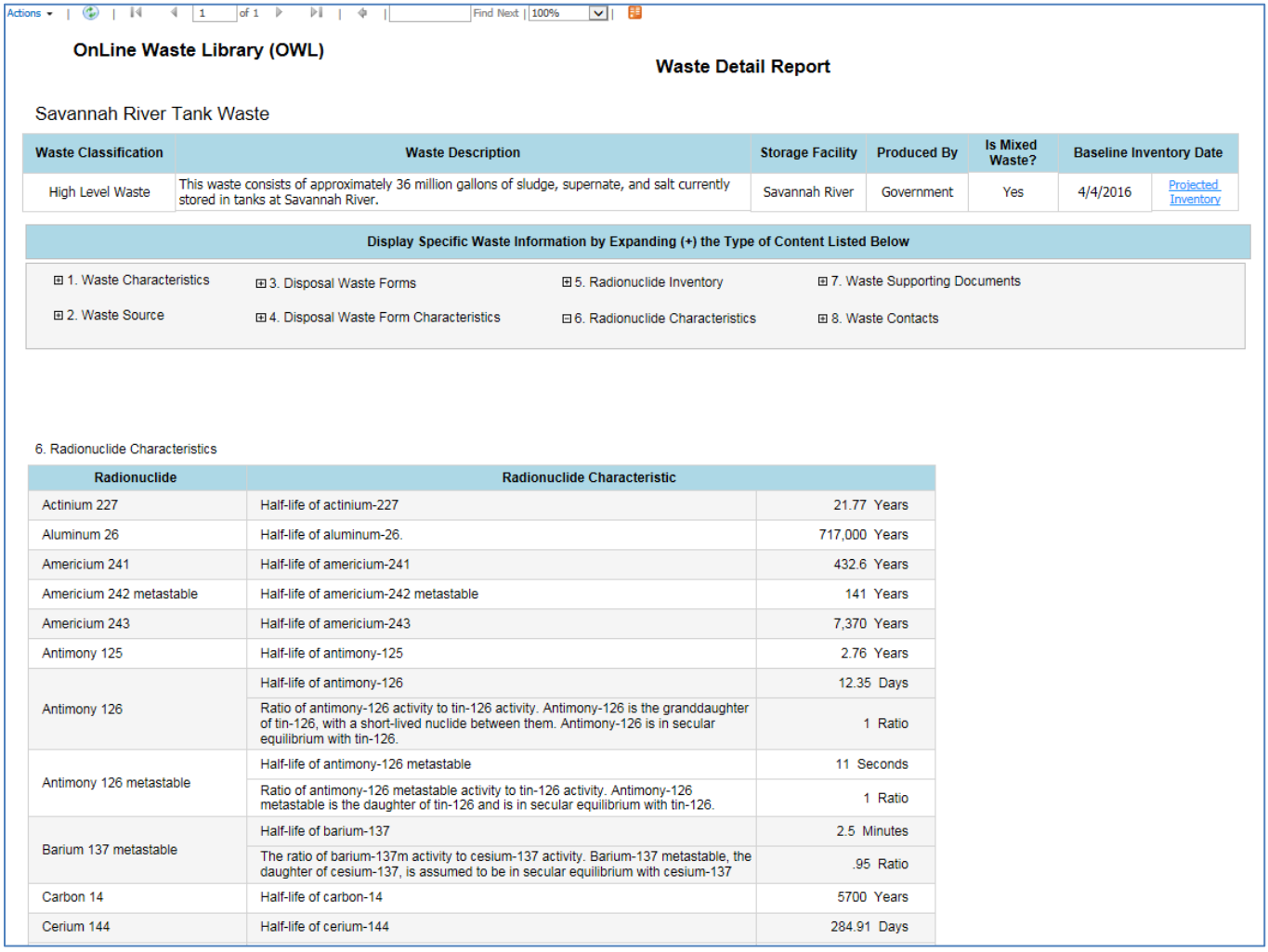
⊞ 2. Waste Source ⊞ 4. Disposal Waste Form Characteristics ⊞ 6. Radionuclide Characteristics ⊟ 8. Waste Contacts

5. Radionuclide Inventory

Radionuclide	Inventory Description		Value	Supporting Document
Actinium 227	The total curies of actinium-227 in supernate and sludge.	Projected Inventory	2.19E+000 Curies	Open
Aluminum 26	The total curies of aluminum-26 in supernate, sludge, and interstitial salt.	Projected Inventory	2.94E+001 Curies	Open
Americium 241	The total curies of americium-241 in supernate and sludge.	Projected Inventory	2.64E+005 Curies	Open
Americium 242 metastable	The total curies of americium-242 metastable in supernate and sludge.	Projected Inventory	1.37E+002 Curies	Open
Americium 243	The total curies of americium-243 in supernate and sludge.	Projected Inventory	6.95E+001 Curies	Open
Antimony 125	The total curies of antimony-125 in supernate and sludge.	Projected Inventory	2.06E+004 Curies	Open
Antimony 126	The total curies of antimony-126 in supernate and sludge.	Projected Inventory	2.13E+002 Curies	Open
Antimony 126 metastable	The total curies of antimony-126 metastable in supernate and sludge.	Projected Inventory	1.52E+003 Curies	Open
Barium 137 metastable	The total curies of barium-137 metastable in supernate, sludge, and interstitial salt.	Projected Inventory	5.98E+007 Curies	Open
Carbon 14	The total curies of carbon-14 in supernate, sludge, and interstitial salt..	Projected Inventory	5.16E+002 Curies	Open
Cerium 144	The total curies of cerium-144 in supernate and sludge.	Projected Inventory	8.37E+002 Curies	Open

Figure 3-13. Waste Detail Report – Radionuclide Inventory

Radionuclide Inventory – provides a list of the radionuclides in the waste and their inventory (in curies) and a link to the report where the radionuclide inventory can be projected to a selected target year. Note: The above sample screen shot (Figure 3-13) displays only some of the radionuclides.



Actions | 1 of 1 | Find Next | 100%

OnLine Waste Library (OWL)

Waste Detail Report

Savannah River Tank Waste

Waste Classification	Waste Description	Storage Facility	Produced By	Is Mixed Waste?	Baseline Inventory Date	
High Level Waste	This waste consists of approximately 36 million gallons of sludge, supernate, and salt currently stored in tanks at Savannah River.	Savannah River	Government	Yes	4/4/2016	Projected Inventory

Display Specific Waste Information by Expanding (+) the Type of Content Listed Below

⊞ 1. Waste Characteristics ⊞ 3. Disposal Waste Forms ⊞ 5. Radionuclide Inventory ⊞ 7. Waste Supporting Documents

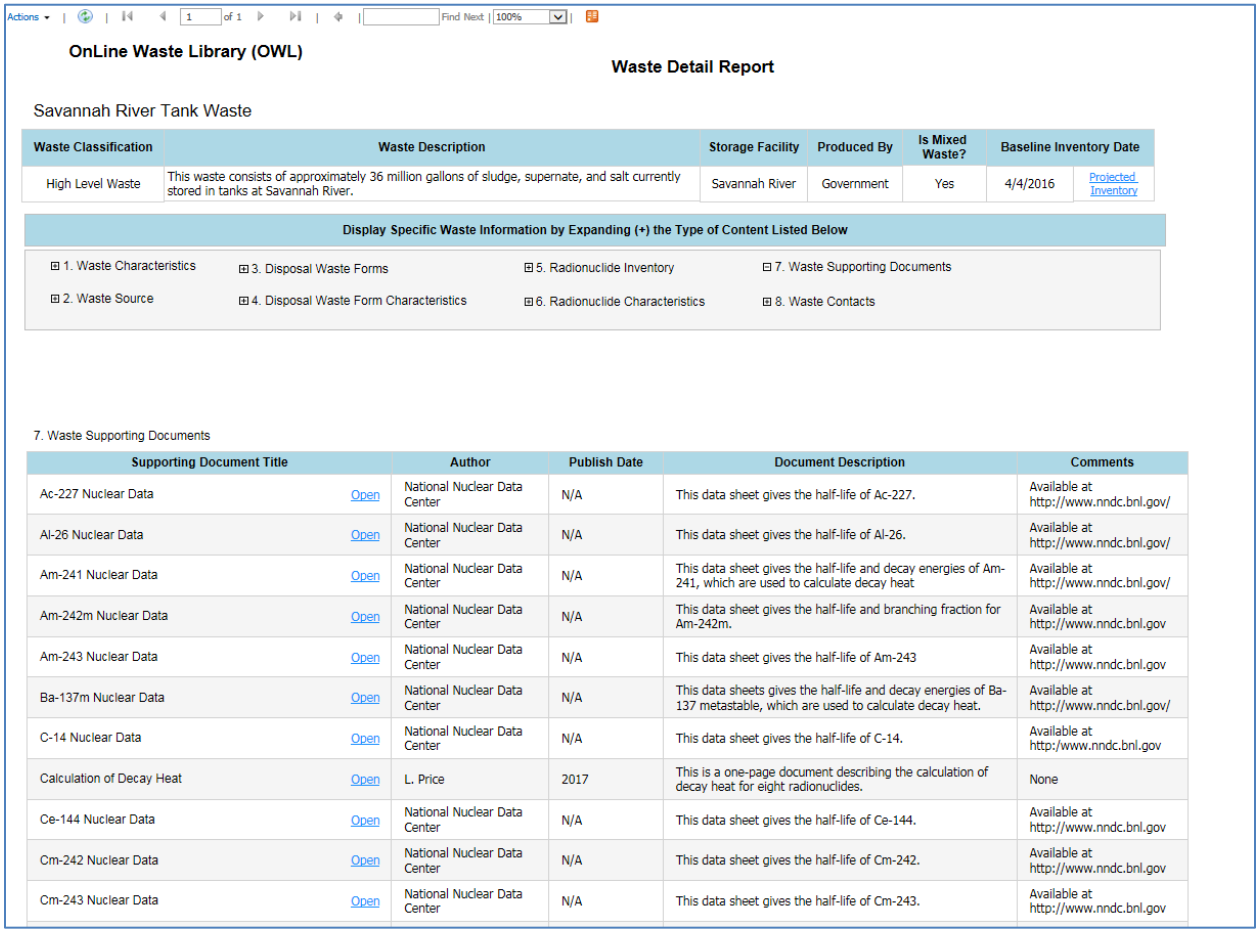
⊞ 2. Waste Source ⊞ 4. Disposal Waste Form Characteristics ⊟ 6. Radionuclide Characteristics ⊞ 8. Waste Contacts

6. Radionuclide Characteristics

Radionuclide	Radionuclide Characteristic	
Actinium 227	Half-life of actinium-227	21.77 Years
Aluminum 26	Half-life of aluminum-26.	717,000 Years
Americium 241	Half-life of americium-241	432.6 Years
Americium 242 metastable	Half-life of americium-242 metastable	141 Years
Americium 243	Half-life of americium-243	7,370 Years
Antimony 125	Half-life of antimony-125	2.76 Years
Antimony 126	Half-life of antimony-126	12.35 Days
	Ratio of antimony-126 activity to tin-126 activity. Antimony-126 is the granddaughter of tin-126, with a short-lived nuclide between them. Antimony-126 is in secular equilibrium with tin-126.	1 Ratio
Antimony 126 metastable	Half-life of antimony-126 metastable	11 Seconds
	Ratio of antimony-126 metastable activity to tin-126 activity. Antimony-126 metastable is the daughter of tin-126 and is in secular equilibrium with tin-126.	1 Ratio
Barium 137 metastable	Half-life of barium-137	2.5 Minutes
	The ratio of barium-137m activity to cesium-137 activity. Barium-137 metastable, the daughter of cesium-137, is assumed to be in secular equilibrium with cesium-137	.95 Ratio
Carbon 14	Half-life of carbon-14	5700 Years
Cerium 144	Half-life of cerium-144	284.91 Days

Figure 3-14. Waste Detail Report – Radionuclide Characteristics

Radionuclide Characteristics – provides a list of the radionuclides in the waste and their characteristics, which includes their half-life (e.g., Figure 3-14).



OnLine Waste Library (OWL)

Waste Detail Report

Savannah River Tank Waste

Waste Classification	Waste Description	Storage Facility	Produced By	Is Mixed Waste?	Baseline Inventory Date	
High Level Waste	This waste consists of approximately 36 million gallons of sludge, supernate, and salt currently stored in tanks at Savannah River.	Savannah River	Government	Yes	4/4/2016	Projected Inventory

Display Specific Waste Information by Expanding (+) the Type of Content Listed Below

⊞ 1. Waste Characteristics ⊞ 3. Disposal Waste Forms ⊞ 5. Radionuclide Inventory ⊟ 7. Waste Supporting Documents

⊞ 2. Waste Source ⊞ 4. Disposal Waste Form Characteristics ⊞ 6. Radionuclide Characteristics ⊞ 8. Waste Contacts

7. Waste Supporting Documents

Supporting Document Title		Author	Publish Date	Document Description	Comments
Ac-227 Nuclear Data	Open	National Nuclear Data Center	N/A	This data sheet gives the half-life of Ac-227.	Available at http://www.nndc.bnl.gov/
Al-26 Nuclear Data	Open	National Nuclear Data Center	N/A	This data sheet gives the half-life of Al-26.	Available at http://www.nndc.bnl.gov/
Am-241 Nuclear Data	Open	National Nuclear Data Center	N/A	This data sheet gives the half-life and decay energies of Am-241, which are used to calculate decay heat	Available at http://www.nndc.bnl.gov/
Am-242m Nuclear Data	Open	National Nuclear Data Center	N/A	This data sheet gives the half-life and branching fraction for Am-242m.	Available at http://www.nndc.bnl.gov
Am-243 Nuclear Data	Open	National Nuclear Data Center	N/A	This data sheet gives the half-life of Am-243	Available at http://www.nndc.bnl.gov
Ba-137m Nuclear Data	Open	National Nuclear Data Center	N/A	This data sheets gives the half-life and decay energies of Ba-137 metastable, which are used to calculate decay heat.	Available at http://www.nndc.bnl.gov/
C-14 Nuclear Data	Open	National Nuclear Data Center	N/A	This data sheet gives the half-life of C-14.	Available at http:/www.nndc.bnl.gov
Calculation of Decay Heat	Open	L. Price	2017	This is a one-page document describing the calculation of decay heat for eight radionuclides.	None
Ce-144 Nuclear Data	Open	National Nuclear Data Center	N/A	This data sheet gives the half-life of Ce-144.	Available at http://www.nndc.bnl.gov
Cm-242 Nuclear Data	Open	National Nuclear Data Center	N/A	This data sheet gives the half-life of Cm-242.	Available at http://www.nndc.bnl.gov
Cm-243 Nuclear Data	Open	National Nuclear Data Center	N/A	This data sheet gives the half-life of Cm-243.	Available at http://www.nndc.bnl.gov

Figure 3-15. Waste Detail Report – Waste Supporting Documents

Waste Supporting Documents – data sheets for radionuclides in the waste, decay calculations (e.g., Figure 3-15).

Details about a Particular Waste (see Figure 3-8)

This report provides the user the ability to select a particular waste from a drop down menu and obtain the Waste Detail report (Figure 3-16). This is an alternative to selecting the waste from the Waste Search Report.

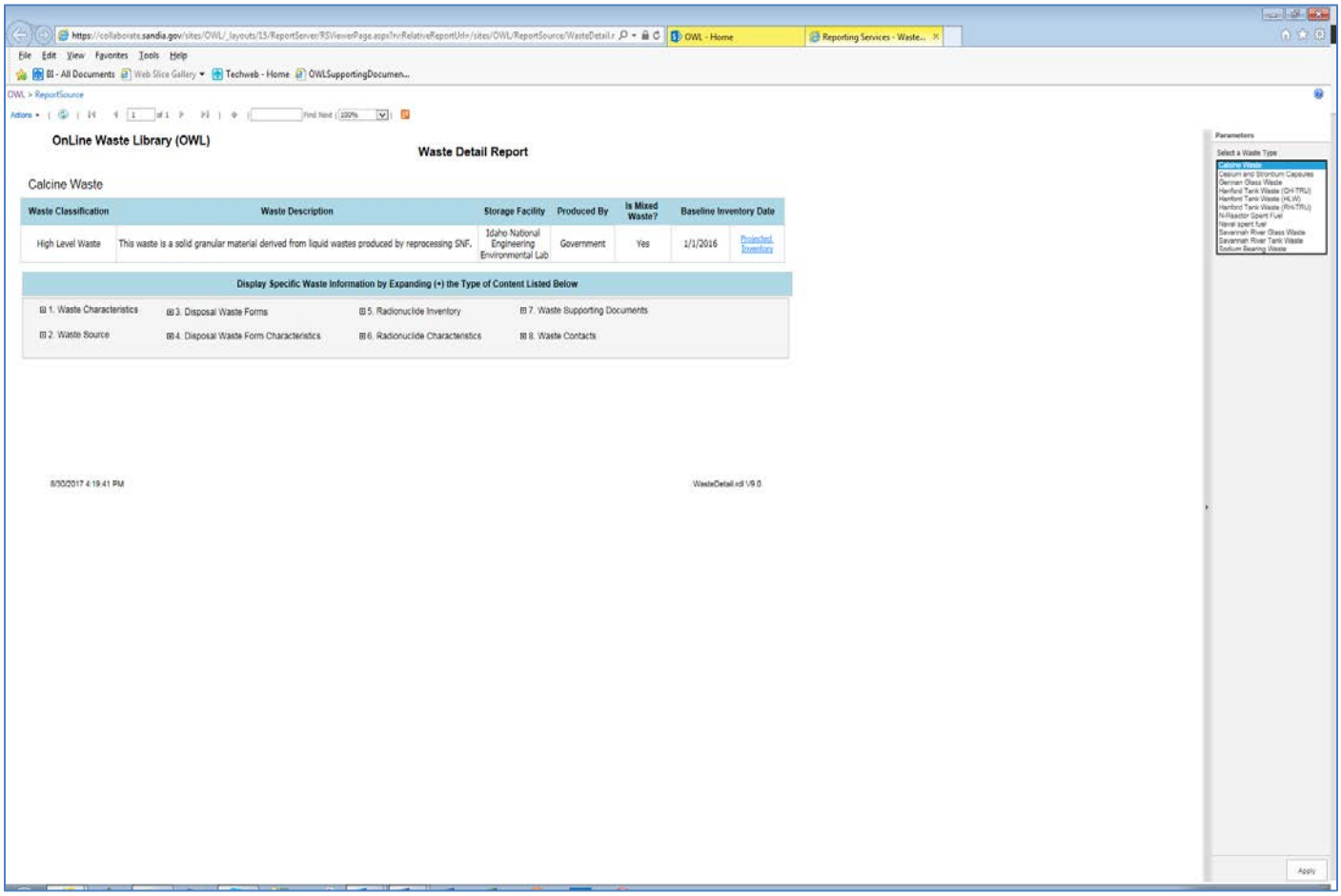


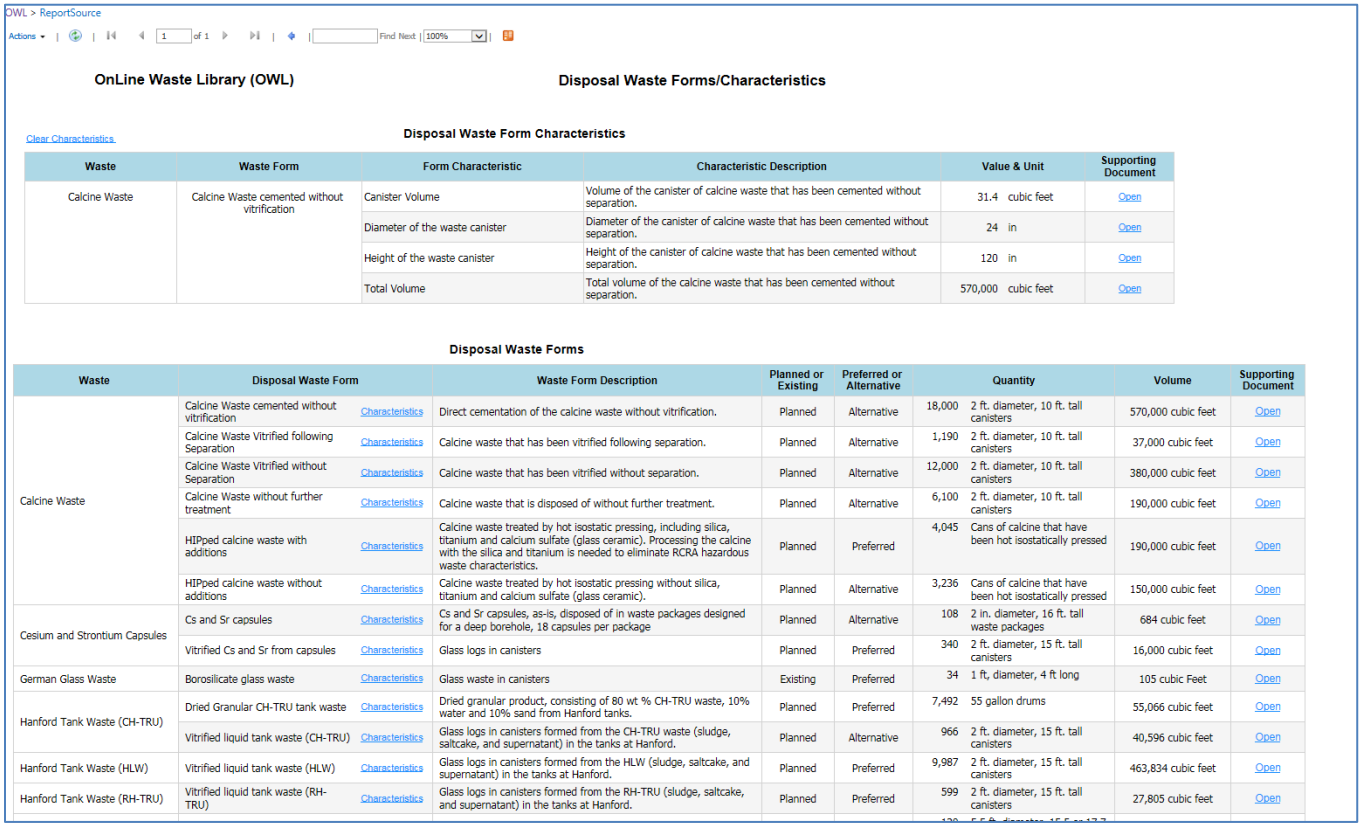
Figure 3-16. Waste Detail Report – Alternate selection.

Waste Forms (see Figure 3-8)

To list all the Wastes and their associated Disposal Waste Forms select the "Disposal Waste Form Characteristics". When selected, the "Characteristics" selector will display on the report of the list of Wastes/Waste Forms. Figure 3-17 is an example of the report with a selection of the Characteristics of the "Calcine Waste Cemented without vitrification" waste form in the "Calcine Waste".

Selecting another "Characteristic" will replace the previously selected characteristic. There is also the ability to clear the display of the characteristic by clicking on "Clear Characteristic" (top left above the Characteristic display – Figure 3-17).

The disposal waste form identifies the disposal waste forms associated with a waste. The waste form information includes a waste form description, indication of whether it is a planned or existing waste form, if it the preferred or alternative waste form, and its quantity and volume with a link to supporting documents.



OWL > ReportSource

Actions | 1 of 1 | Find Next | 100%

OnLine Waste Library (OWL) **Disposal Waste Forms/Characteristics**

Clear Characteristics

Disposal Waste Form Characteristics

Waste	Waste Form	Form Characteristic	Characteristic Description	Value & Unit	Supporting Document
Calcine Waste	Calcine Waste cemented without vitrification	Canister Volume	Volume of the canister of calcine waste that has been cemented without separation.	31.4 cubic feet	Open
		Diameter of the waste canister	Diameter of the canister of calcine waste that has been cemented without separation.	24 in	Open
		Height of the waste canister	Height of the canister of calcine waste that has been cemented without separation.	120 in	Open
		Total Volume	Total volume of the calcine waste that has been cemented without separation.	570,000 cubic feet	Open

Disposal Waste Forms

Waste	Disposal Waste Form		Waste Form Description	Planned or Existing	Preferred or Alternative	Quantity		Volume	Supporting Document
Calcine Waste	Calcine Waste cemented without vitrification	Characteristics	Direct cementation of the calcine waste without vitrification.	Planned	Alternative	18,000	2 ft. diameter, 10 ft. tall canisters	570,000 cubic feet	Open
	Calcine Waste Vitrified following Separation	Characteristics	Calcine waste that has been vitrified following separation.	Planned	Alternative	1,190	2 ft. diameter, 10 ft. tall canisters	37,000 cubic feet	Open
	Calcine Waste Vitrified without Separation	Characteristics	Calcine waste that has been vitrified without separation.	Planned	Alternative	12,000	2 ft. diameter, 10 ft. tall canisters	380,000 cubic feet	Open
	Calcine Waste without further treatment	Characteristics	Calcine waste that is disposed of without further treatment.	Planned	Alternative	6,100	2 ft. diameter, 10 ft. tall canisters	190,000 cubic feet	Open
	HIPped calcine waste with additions	Characteristics	Calcine waste treated by hot isostatic pressing, including silica, titanium and calcium sulfate (glass ceramic). Processing the calcine with the silica and titanium is needed to eliminate RCRA hazardous waste characteristics.	Planned	Preferred	4,045	Cans of calcine that have been hot isostatically pressed	190,000 cubic feet	Open
	HIPped calcine waste without additions	Characteristics	Calcine waste treated by hot isostatic pressing without silica, titanium and calcium sulfate (glass ceramic).	Planned	Alternative	3,236	Cans of calcine that have been hot isostatically pressed	150,000 cubic feet	Open
Cesium and Strontium Capsules	Cs and Sr capsules	Characteristics	Cs and Sr capsules, as-is, disposed of in waste packages designed for a deep borehole, 18 capsules per package	Planned	Alternative	108	2 in. diameter, 16 ft. tall waste packages	684 cubic feet	Open
	Vitrified Cs and Sr from capsules	Characteristics	Glass logs in canisters	Planned	Preferred	340	2 ft. diameter, 15 ft. tall canisters	16,000 cubic feet	Open
German Glass Waste	Borosilicate glass waste	Characteristics	Glass waste in canisters	Existing	Preferred	34	1 ft, diameter, 4 ft long	105 cubic Feet	Open
Hanford Tank Waste (CH-TRU)	Dried Granular CH-TRU tank waste	Characteristics	Dried granular product, consisting of 80 wt % CH-TRU waste, 10% water and 10% sand from Hanford tanks.	Planned	Preferred	7,492	55 gallon drums	55,066 cubic feet	Open
	Vitrified liquid tank waste (CH-TRU)	Characteristics	Glass logs in canisters formed from the CH-TRU waste (sludge, saltcake, and supernatant) in the tanks at Hanford.	Planned	Alternative	966	2 ft. diameter, 15 ft. tall canisters	40,596 cubic feet	Open
Hanford Tank Waste (HLW)	Vitrified liquid tank waste (HLW)	Characteristics	Glass logs in canisters formed from the HLW (sludge, saltcake, and supernatant) in the tanks at Hanford.	Planned	Preferred	9,987	2 ft. diameter, 15 ft. tall canisters	463,834 cubic feet	Open
Hanford Tank Waste (RH-TRU)	Vitrified liquid tank waste (RH-TRU)	Characteristics	Glass logs in canisters formed from the RH-TRU (sludge, saltcake, and supernatant) in the tanks at Hanford.	Planned	Preferred	599	2 ft. diameter, 15 ft. tall canisters	27,805 cubic feet	Open

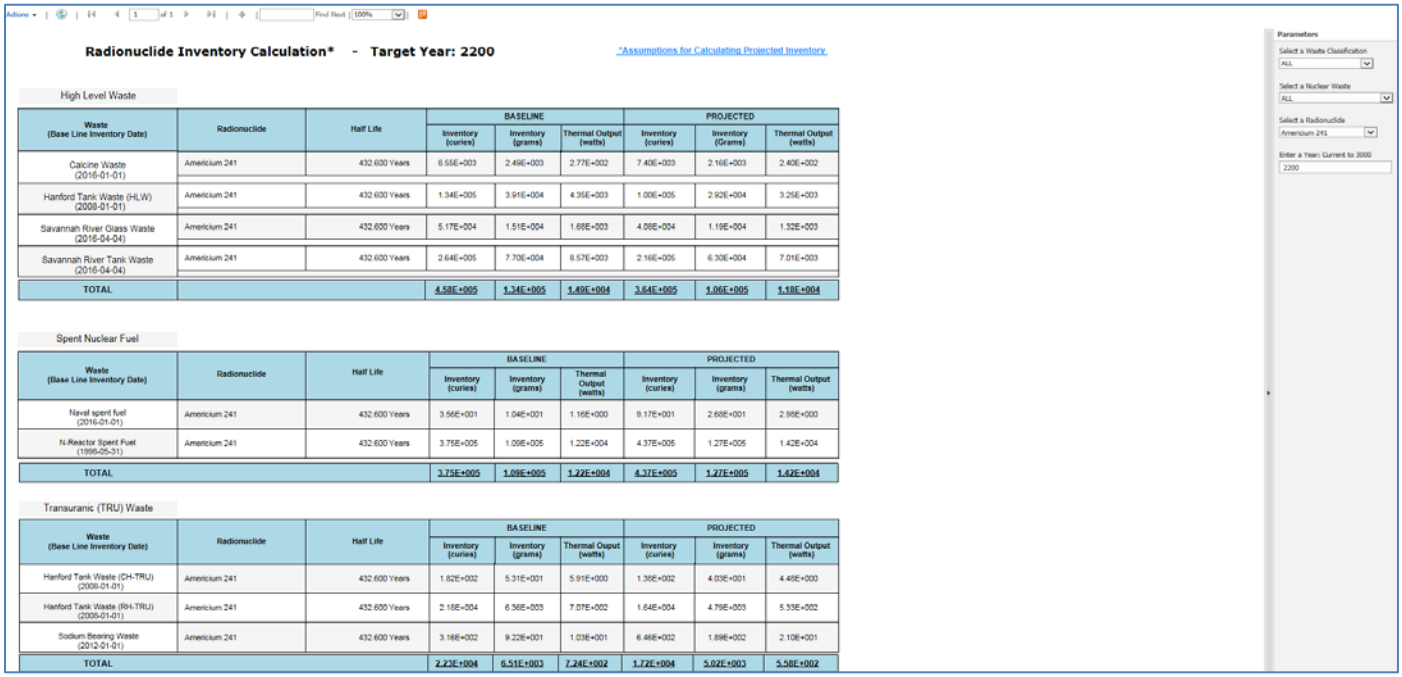
Figure 3-17. Waste Forms Report – after selecting a 'Characteristic'

The Disposal Waste Form "Characteristics" provides characteristics like dimensions (diameter, height, etc.) and volume as well as links to supporting documents.

Projected Radionuclide Inventory (see Figure 3-8)

This OWL report calculates the projected inventory in curies, grams, and watts (where applicable) of radionuclides to a selected target year (from the current year out to year 3000).

The report can provide summarized inventories by a selected waste classification, a selected nuclear waste, or a radionuclide. Selecting by a specific radionuclide will generate a report that provides the projected inventory for every waste the includes the selected radionuclide. Figure 3-18 is an example for the radionuclide Americium 241 projected inventory to the target year 2200 for "ALL" wastes.



Radionuclide Inventory Calculation* - Target Year: 2200

*Assumptions for Calculating Projected Inventory

High Level Waste

Waste (Base Line Inventory Date)	Radionuclide	Half Life	BASELINE			PROJECTED		
			Inventory (curies)	Inventory (grams)	Thermal Output (watts)	Inventory (curies)	Inventory (Grams)	Thermal Output (watts)
Calcine Waste (2016-01-01)	Americium 241	432.600 Years	8.55E+003	2.49E+003	2.77E+002	7.40E+003	2.16E+003	2.40E+002
Hanford Tank Waste (HLW) (2008-01-01)	Americium 241	432.600 Years	1.34E+005	3.91E+004	4.35E+003	1.00E+005	2.92E+004	3.25E+003
Savannah River Glass Waste (2016-04-04)	Americium 241	432.600 Years	5.17E+004	1.51E+004	1.68E+003	4.06E+004	1.19E+004	1.32E+003
Savannah River Tank Waste (2016-04-04)	Americium 241	432.600 Years	2.64E+005	7.70E+004	8.57E+003	2.16E+005	6.30E+004	7.01E+003
TOTAL			4.58E+005	1.34E+005	1.49E+004	3.64E+005	1.06E+005	1.18E+004

Spent Nuclear Fuel

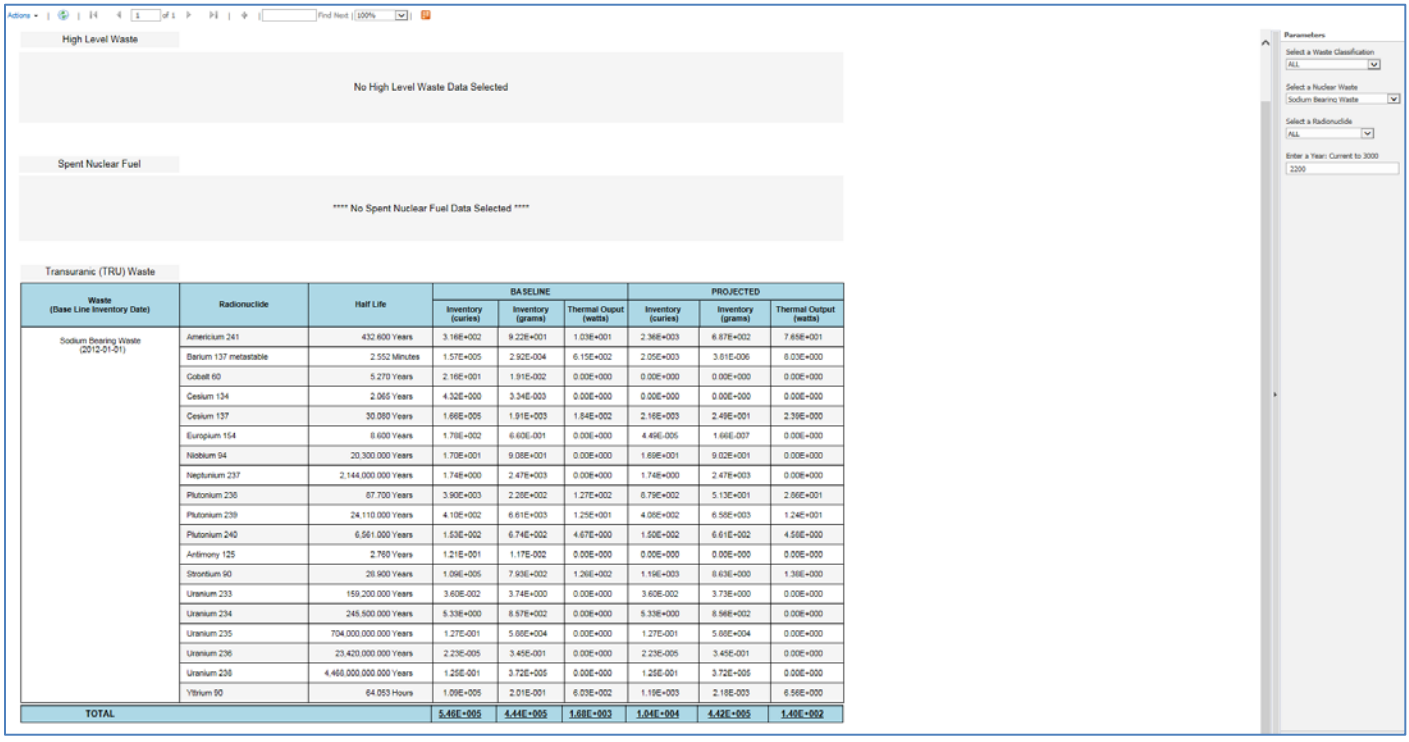
Waste (Base Line Inventory Date)	Radionuclide	Half Life	BASELINE			PROJECTED		
			Inventory (curies)	Inventory (grams)	Thermal Output (watts)	Inventory (curies)	Inventory (grams)	Thermal Output (watts)
Naval spent fuel (2016-01-01)	Americium 241	432.600 Years	3.56E+001	1.04E+001	1.16E+000	9.17E+001	2.68E+001	2.99E+000
N-Reactor Spent Fuel (1996-05-31)	Americium 241	432.600 Years	3.75E+005	1.09E+005	1.22E+004	4.37E+005	1.27E+005	1.42E+004
TOTAL			3.75E+005	1.09E+005	1.22E+004	4.37E+005	1.27E+005	1.42E+004

Transuranic (TRU) Waste

Waste (Base Line Inventory Date)	Radionuclide	Half Life	BASELINE			PROJECTED		
			Inventory (curies)	Inventory (grams)	Thermal Ouput (watts)	Inventory (curies)	Inventory (grams)	Thermal Output (watts)
Hanford Tank Waste (CH-TRU) (2008-01-01)	Americium 241	432.600 Years	1.82E+002	5.31E+001	5.91E+000	1.38E+002	4.03E+001	4.48E+000
Hanford Tank Waste (RH-TRU) (2008-01-01)	Americium 241	432.600 Years	2.18E+004	6.36E+003	7.07E+002	1.64E+004	4.79E+003	5.33E+002
Sodium Bearing Waste (2012-01-01)	Americium 241	432.600 Years	3.16E+002	9.22E+001	1.03E+001	6.46E+002	1.89E+002	2.10E+001
TOTAL			2.23E+004	6.51E+003	7.24E+002	1.72E+004	5.02E+003	5.58E+002

Figure 3-18. Radionuclide Projected Inventory calculation for a specific radionuclide. Target year entered was 2200

Figure 3-19 is a sample report where a specific waste is selected.



High Level Waste

No High Level Waste Data Selected

Spent Nuclear Fuel

**** No Spent Nuclear Fuel Data Selected ****

Transuranic (TRU) Waste

Waste (Base Line Inventory Date)	Radionuclide	Half Life	BASELINE			PROJECTED		
			Inventory (curies)	Inventory (grams)	Thermal Ouput (watts)	Inventory (curies)	Inventory (grams)	Thermal Output (watts)
Sodium Bearing Waste (2012-01-01)	Americium 241	432.600 Years	3.16E+002	9.22E+001	1.03E+001	2.36E+003	6.87E+002	7.65E+001
	Barium 137 metastable	2.552 Minutes	1.57E+005	2.92E-004	6.15E+002	2.05E+003	3.81E-006	8.03E+000
	Cobalt 60	5.270 Years	2.16E+001	1.91E-002	0.00E+000	0.00E+000	0.00E+000	0.00E+000
	Cesium 134	2.065 Years	4.32E+000	3.34E-003	0.00E+000	0.00E+000	0.00E+000	0.00E+000
	Cesium 137	30.080 Years	1.66E+005	1.91E+003	1.84E+002	2.16E+003	2.49E+001	2.39E+000
	Europium 154	8.600 Years	1.79E+002	6.60E-001	0.00E+000	4.49E-005	1.66E-007	0.00E+000
	Niobium 94	20,300.000 Years	1.70E+001	9.08E+001	0.00E+000	1.69E+001	9.02E+001	0.00E+000
	Neptunium 237	2,144,000.000 Years	1.74E+000	2.47E+003	0.00E+000	1.74E+000	2.47E+003	0.00E+000
	Plutonium 236	87.700 Years	3.90E+003	2.26E+002	1.27E+002	6.79E+002	5.13E+001	2.86E+001
	Plutonium 239	24,110.000 Years	4.10E+002	6.61E+003	1.25E+001	4.06E+002	6.58E+003	1.24E+001
	Plutonium 240	6,561.000 Years	1.53E+002	6.74E+002	4.67E+000	1.50E+002	6.61E+002	4.58E+000
	Antimony 125	2.760 Years	1.21E+001	1.17E-002	0.00E+000	0.00E+000	0.00E+000	0.00E+000
	Strontium 90	28.900 Years	1.09E+005	7.93E-002	1.26E+002	1.19E+003	8.63E+000	1.38E+000
	Uranium 233	159,200.000 Years	3.60E-002	3.74E+000	0.00E+000	3.60E-002	3.73E+000	0.00E+000
	Uranium 234	245,500.000 Years	5.33E+000	8.57E+002	0.00E+000	5.33E+000	8.56E+002	0.00E+000
	Uranium 235	704,000,000.000 Years	1.27E-001	5.66E-004	0.00E+000	1.27E-001	5.66E-004	0.00E+000
	Uranium 236	23,420,000.000 Years	2.23E-005	3.45E-001	0.00E+000	2.23E-005	3.45E-001	0.00E+000
	Uranium 238	4,468,000,000.000 Years	1.25E-001	3.72E-005	0.00E+000	1.25E-001	3.72E-005	0.00E+000
	Yttrium 90	64.053 Hours	1.09E+005	2.01E-001	6.03E+002	1.19E+003	2.18E-003	6.56E+000
TOTAL			5.46E+005	4.44E+005	1.68E+003	1.04E+004	4.42E+005	1.40E+002

Figure 3-19. Radionuclide Projected Inventory calculation for a specific waste. Target Year entered was 2200.

Waste Projected Inventory (200 Years) (see Figure 3-8)

This display (Figure 3-20) provides the projected inventory for a selected waste or radionuclide for each year (beginning with the current year) for the next 200 years. In Figure 3-20, there are 2 charts displayed:

on the left is the “Projected inventory in Curies” and on the right is the “Projected Inventory Thermal Output” for the particular waste type and “all” the radionuclides documented for it. The user can select any combination of waste and radionuclide on the right hand side selectors.

In addition, there is a link to a document that describes the assumptions and methodology for calculating the projected inventory accessed by clicking on the text: *Assumptions for Calculating Projected Inventory (top right of the display). Figure 3-20 is a sample report where the waste “Hanford Tank Waste (CH-TRU)” is selected.

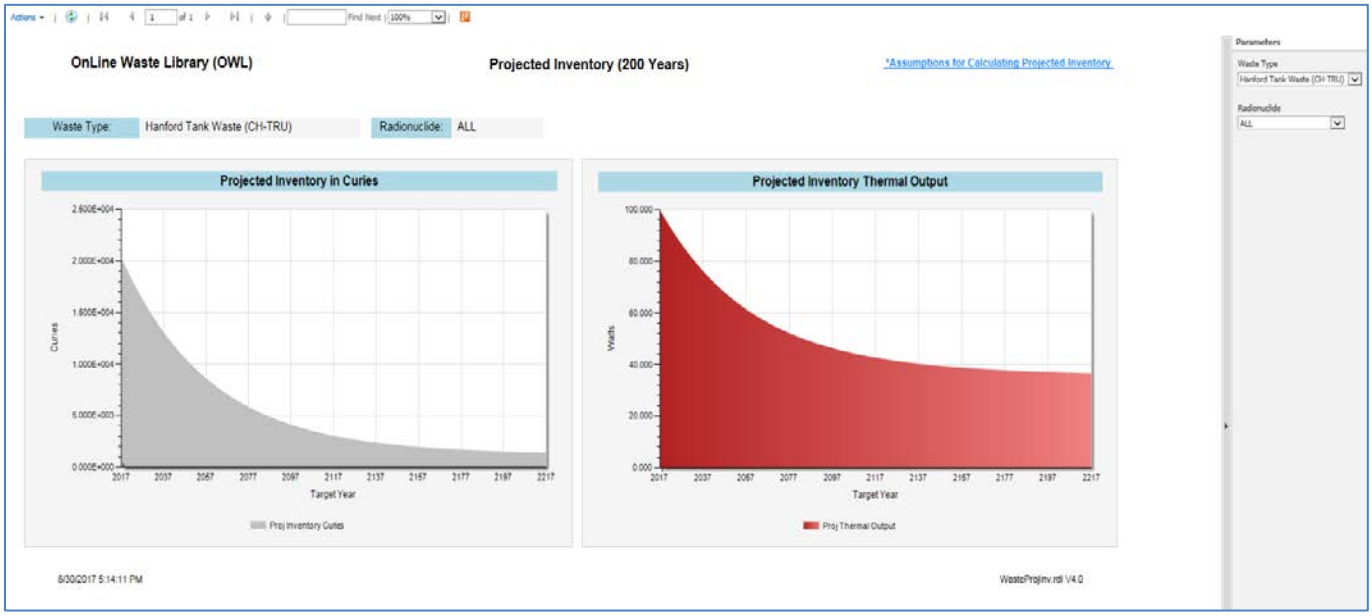


Figure 3-20. Projected inventory and thermal output for 200 years – specific waste selected

Shown in Figure 3-21 is an example where a single radionuclide is selected: Americium 241 that represents the total of that radionuclide in “All” the waste types in OWL.

Figure 3-21. Projected Inventory and thermal output for 200 years – specific radionuclide selected.

Baseline Radionuclide Inventory in Each Waste (see Figure 3-8)

This report is also referred to as the “Waste Search Radionuclides Report” because one can select a specific radionuclide and the report will list all the wastes that include the radionuclide. To select a

specific radionuclide click on the [+] next to the text “Expand to Select a Radionuclide”. A drop down list will display. Click on any radionuclide. When the report is returned the selection will be identified.

Besides providing the ability to select a specific radionuclide, the user can select a facility, which will then list all wastes and radionuclides in the wastes for that facility. Also provided is the ability to select a Waste Classification, which will filter the report to a specific waste classification. Multiple combinations of these selections can be made by the user. For example, a single facility can be selected along with a specific radionuclide. Another example is to limit the display to only High Level Waste in the Hanford facility.

The information display includes the Inventory in Curies for each radionuclide and there is also a link for each waste display to the “Waste Details”. Figure 3-22 shows an example for a selection of the Hanford facility, High Level Waste, and Radionuclide Cs-137:

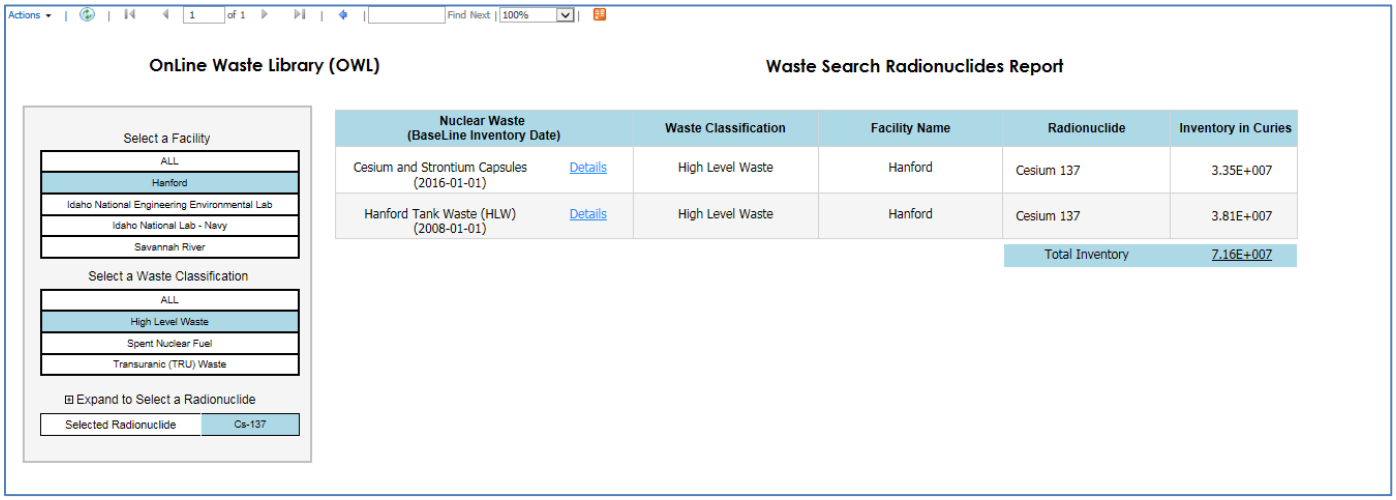


Figure 3-22. Baseline Radionuclide Inventory in each waste – in the Hanford facility, for High Level Waste, and the radionuclide Cs-137

Radionuclides (see Figure 3-8)

This report lists all the radionuclides and the basic information for each (Half-life, Atomic Mass, Thermal Output, etc.) contained within OWL. Although there are no filters to limit the display of radionuclides, a number of the columns can be used to sort the output: Atomic Mass, Thermal Output, Description, etc. The columns with small up/down arrows can be sorted. Click on the arrows to sort the report by that column. When the report is sorted the arrow will be changed to up (for ascending) or if clicked again, the report will be sorted by the column in descending sequence (down arrow). Also, for each radionuclide there is a supporting document about the radionuclide Figure 3-23 shows an example of the supporting document for Cs-137 and Figure 3-24 shows a portion of the report on the radionuclides listing in ascending order.

Nuclear Wallet Cards Results

Search parameters:
Nucleus:137CS

Results:

Nucleus	E(level) (MeV)	Jπ	Δ(MeV)	$T_{1/2}$	Abundance	Decay Modes
$^{137}_{55}Cs^{FF}$	0.0000	7/2+	-86.5459	30.08 y *0.09*		β- : 100.00 %

FF: ^{235}U thermal fission fragment with cumulative yield ≥ 10^{-6}

Figure 3-23. Supporting document for radionuclide Cs-137

Actions | 1 of 1 | Find Next | 100%

OnLine Waste Library (OWL) **Radionuclides (86)**

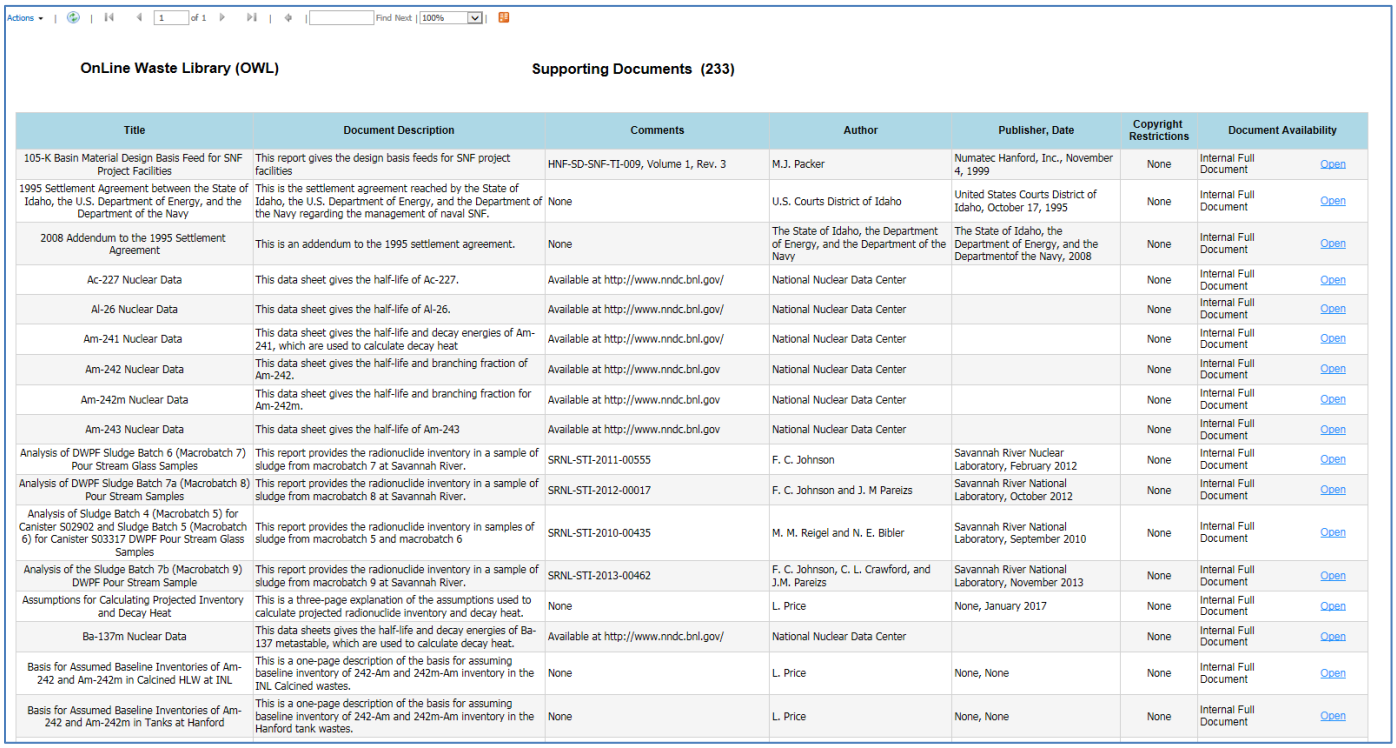
Radionuclide	ID	Description	Half Life	Atomic Mass	Thermal Output	Parent Radionuclide	ID	Inventory Ratio	Supporting Document ID & Availability		
Ac-227	51	Actinium 227	21.77 Years	227.00					3172	Internal Full Document	Open
Al-26	46	Aluminum 26	717000 Years	26.00					3175	Internal Full Document	Open
Am-241	41	Americium 241	432.6 Years	241.00	32.450	Pu-241	39		1067	Internal Full Document	Open
Am-242	74	Americium 242	16.02 Hours	242.00		Am-242m	64	0.995	3176	Internal Full Document	Open
Am-242m	64	Americium 242 metastable	141 Years	242.00					3177	Internal Full Document	Open
Am-243	52	Americium 243	7370 Years	243.00					3178	Internal Full Document	Open
Ba137-m	1	Barium 137 metastable	2.552 Minutes	137.00	3.920	Cs-137	3	0.950	1068	Internal Full Document	Open
Bk-247	67	Berkelium 247	1380 Years	247.00					3179	Internal Full Document	Open
C-14	8	Carbon 14	5700 Years	14.00					3180	Internal Full Document	Open
Cd-113m	61	Cadmium 113 metastable	14.1 Years	113.00					3181	Internal Full Document	Open
Ce-144	24	Cerium 144	284.91 Days	144.00					3183	Internal Full Document	Open
Cf-249	68	Californium 249	351 Years	249.00					3184	Internal Full Document	Open
Cf-251	69	Californium 251	898 Years	251.00					3185	Internal Full Document	Open
Cf-252	85	Californium 252	2.645 Years	252.00					3186	Internal Full Document	Open
Cl-36	70	Chlorine 36	301000 Years	36.00					3187	Internal Full Document	Open
Cm-242	42	Curium 242	162.8 Days	242.00		Am-242m	64	0.830	3188	Internal Full Document	Open
Cm-243	53	Curium 243	29.1 Years	243.00					3189	Internal Full Document	Open
Cm-244	43	Curium 244	18.1 Years	244.00					3190	Internal Full Document	Open
Cm-245	65	Curium 245	8423 Years	245.00					3191	Internal Full Document	Open
Cm-246	66	Curium 246	4706 Years	246.00					3192	Internal Full Document	Open
Cm-247	71	Curium 247	15600000 Years	247.00					3193	Internal Full Document	Open
Cm-248	72	Curium 248	348000 Years	248.00					3194	Internal Full Document	Open
Co-60	10	Cobalt 60	5.27 Years	60.00					3195	Internal Full Document	Open
Cs-134	23	Cesium 134	2.065 Years	134.00					3196	Internal Full Document	Open
Cs-135	2	Cesium 135	2300000 Years	135.00					3197	Internal Full Document	Open
Cs-137	3	Cesium 137	30.08 Years	137.00	1.110				1069	Internal Full Document	Open

Figure 3-24. Radionuclides report sorted by radionuclides in ascending sequence (only part of the report is displayed)

Supporting Documents (see Figure 3-8)

This report selection provides a complete listing of all the supporting documents used in the database. Currently there are 233 supporting documents. The documents include data sheets for radionuclides, descriptions of the assumptions used in providing the baseline inventory, design packaging criteria, and others. For every document, the report lists the Title, Document Description, Comments, Author, Publisher/Date, Copyright restrictions and the Document Availability. The document availability

indicates whether the entire document is available or only a summary (because of copyright restrictions). There is also a link to open the available documents presented. Figure 3-25 provides an example of the Supporting Documents report. Note it is only a partial sample since the report is extensive.



OnLine Waste Library (OWL) **Supporting Documents (233)**

Title	Document Description	Comments	Author	Publisher, Date	Copyright Restrictions	Document Availability
105-K Basin Material Design Basis Feed for SNF Project Facilities	This report gives the design basis feeds for SNF project facilities	HNF-SD-SNF-TI-009, Volume 1, Rev. 3	M.J. Packer	Numatec Hanford, Inc., November 4, 1999	None	Internal Full Document Open
1995 Settlement Agreement between the State of Idaho, the U.S. Department of Energy, and the Department of the Navy	This is the settlement agreement reached by the State of Idaho, the U.S. Department of Energy, and the Department of the Navy regarding the management of naval SNF.	None	U.S. Courts District of Idaho	United States Courts District of Idaho, October 17, 1995	None	Internal Full Document Open
2008 Addendum to the 1995 Settlement Agreement	This is an addendum to the 1995 settlement agreement.	None	The State of Idaho, the Department of Energy, and the Department of the Navy	The State of Idaho, the Department of Energy, and the Departmentof the Navy, 2008	None	Internal Full Document Open
Ac-227 Nuclear Data	This data sheet gives the half-life of Ac-227.	Available at http://www.nndc.bnl.gov/	National Nuclear Data Center		None	Internal Full Document Open
Al-26 Nuclear Data	This data sheet gives the half-life of Al-26.	Available at http://www.nndc.bnl.gov/	National Nuclear Data Center		None	Internal Full Document Open
Am-241 Nuclear Data	This data sheet gives the half-life and decay energies of Am-241, which are used to calculate decay heat	Available at http://www.nndc.bnl.gov/	National Nuclear Data Center		None	Internal Full Document Open
Am-242 Nuclear Data	This data sheet gives the half-life and branching fraction of Am-242.	Available at http://www.nndc.bnl.gov	National Nuclear Data Center		None	Internal Full Document Open
Am-242m Nuclear Data	This data sheet gives the half-life and branching fraction for Am-242m.	Available at http://www.nndc.bnl.gov	National Nuclear Data Center		None	Internal Full Document Open
Am-243 Nuclear Data	This data sheet gives the half-life of Am-243	Available at http://www.nndc.bnl.gov	National Nuclear Data Center		None	Internal Full Document Open
Analysis of DWPF Sludge Batch 6 (Macrobatch 7) Pour Stream Glass Samples	This report provides the radionuclide inventory in a sample of sludge from macrobatch 7 at Savannah River.	SRNL-STI-2011-00555	F. C. Johnson	Savannah River Nuclear Laboratory, February 2012	None	Internal Full Document Open
Analysis of DWPF Sludge Batch 7a (Macrobatch 8) Pour Stream Samples	This report provides the radionuclide inventory in a sample of sludge from macrobatch 8 at Savannah River.	SRNL-STI-2012-00017	F. C. Johnson and J. M Pareizs	Savannah River National Laboratory, October 2012	None	Internal Full Document Open
Analysis of Sludge Batch 4 (Macrobatch 5) for Canister S02902 and Sludge Batch 5 (Macrobatch 6) for Canister S03317 DWPF Pour Stream Glass Samples	This report provides the radionuclide inventory in samples of sludge from macrobatch 5 and macrobatch 6	SRNL-STI-2010-00435	M. M. Reigel and N. E. Bibler	Savannah River National Laboratory, September 2010	None	Internal Full Document Open
Analysis of the Sludge Batch 7b (Macrobatch 9) DWPF Pour Stream Sample	This report provides the radionuclide inventory in a sample of sludge from macrobatch 9 at Savannah River.	SRNL-STI-2013-00462	F. C. Johnson, C. L. Crawford, and J.M. Pareizs	Savannah River National Laboratory, November 2013	None	Internal Full Document Open
Assumptions for Calculating Projected Inventory and Decay Heat	This is a three-page explanation of the assumptions used to calculate projected radionuclide inventory and decay heat.	None	L. Price	None, January 2017	None	Internal Full Document Open
Ba-137m Nuclear Data	This data sheets gives the half-life and decay energies of Ba-137 metastable, which are used to calculate decay heat.	Available at http://www.nndc.bnl.gov/	National Nuclear Data Center		None	Internal Full Document Open
Basis for Assumed Baseline Inventories of Am-242 and Am-242m in Calcined HLW at INL	This is a one-page description of the basis for assuming baseline inventory of 242-Am and 242m-Am inventory in the INL Calcined wastes.	None	L. Price	None, None	None	Internal Full Document Open
Basis for Assumed Baseline Inventories of Am-242 and Am-242m in Tanks at Hanford	This is a one-page description of the basis for assuming baseline inventory of 242-Am and 242m-Am inventory in the Hanford tank wastes.	None	L. Price	None, None	None	Internal Full Document Open

Figure 3-25. Supporting Document Report (partial listing)

General User Directions

Navigation

Note that all reports are opened in a new browser window. This is true when linking to another report or opening a supporting document. Use the 'exit a window' browser function to return to a previous report or to the home page.

Actions

On every display of a report there is '**Actions**' text with a drop-down arrow. Selecting the drop down provides a menu of action choices. The only action that is active to be selected is "Export" (other actions will fail currently). The "Export" action when selected will also provide another menu of formats for the export. The recommendation is to select one of three formats, Excel, PDF or Word, for best results.

This will export the entire report (not just the current page displayed) in the format selected. Depending on the browser, you will be prompted to 'open' or 'save' (and to choose a file location). Figure 3-26 provides a sample display of the Action menu from the Supporting Documents report:

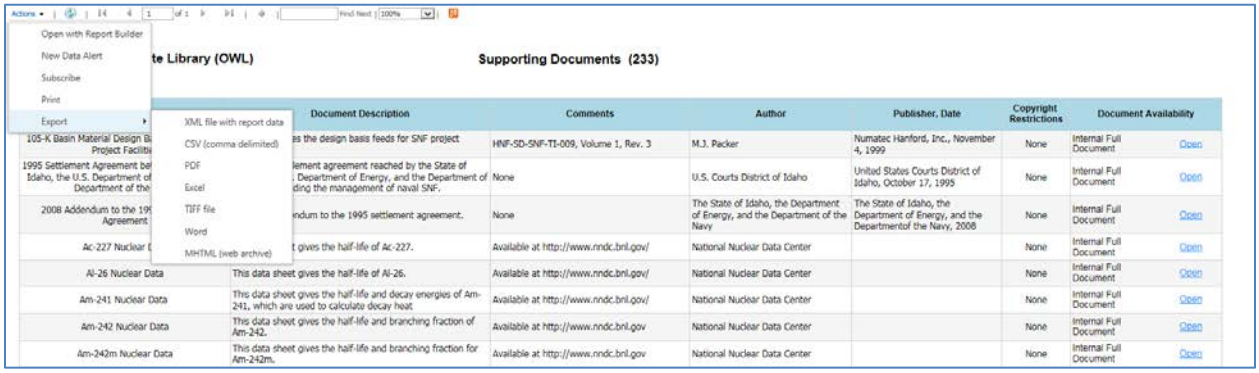


Figure 3-26. Action menu used to export report to a format – select PDF, Excel or Word

3.2 Summary of FY 2017 Updates to Online Waste Library (OWL) Status and Inventory Content

Throughout FY2017, the OWL database activities have focused in three areas. First, additional data for waste types (and their potential waste forms) and source documentation have been added to the OWL to flesh out its content covering DOE-managed HLW and SNF. In conjunction with further data entry, a process of checking the data entry into the OWL against the source documentation was launched to search for and rectify any errors in data entry. This checking was performed by technical individuals who were independent of the data entry process, who documented any issues noted, and resolved the issues with the data entry staff. As the OWL was modified throughout the year in terms of its interface and features, another process to assess the usability of the OWL was recently kicked-off. This process is referred to here as the External OWL BETA test and involves technical staff from within the DOE as well as at other National Laboratories. For each of these three areas, there is a summary update provided here, though all of these activities are ongoing and will be documented finally for FY2017 in milestone deliverable M2SF-17SN010501014 “Inventory and Waste Characterization Status Report”.

3.2.1 Update to OWL Inventory Content for FY 2017

The OWL database is itself the documentation and deliverable of the full array of information/data for the waste types and potential waste forms for DOE-managed (as) HLW and some DOE-managed SNF for use in GDSA evaluations for generic DGRDMSH performance, or any other generic, or otherwise, repository analyses. Because of its formative stature, the OWL database is updated continuously to add additional content, and capabilities to improve usability and facilitate research and development needs of the back end of the fuel cycle. As such, this report does not reproduce that set of information, but a content summary and update is covered.

As of the summer of FY2017, the OWL contains essentially completed primary (i.e., other than updating data values based on new information) datasets for nine (9) waste types (increased from 2 in the OWL prototype) and their planned and potential alternative waste forms.

These essentially completed primary datasets for waste types cover:

- The calcine waste at INL;
- Cs/Sr capsules at Hanford;
- The coolest ($\lesssim$1,000 W) packages of naval SNF;
- The DOE SNF from N-reactor (essentially DOE Group 1);
- Three (3) categories of Hanford tank wastes;

  - The HLW tank waste;
  - The contact handled (CH) tank waste; and
  - The remote handled (RH) tank waste;
- Two (2) categories of SRS HLW;
  - Existing HLW glass logs; and
  - HLW tank waste.

In addition, data entry has commenced on the waste types for sodium-bonded (Na-bonded) fuels from reactors EBR-II, FFTF, and Fermi, and their potential waste forms generated via electrometallurgical treatment (EMT; note that the DOE record of decision, DOE (2000a) only directs this processing for the first two of these, with Fermi Na-bonded blanket fuels awaiting disposition).

The OWL contains information/data on physical form, bulk composition, and content of over 85 radionuclides for these waste types and their waste forms, including concentrations as well as the related thermal output. In addition to the included information/data, the OWL contains over 200 source documents that supplement and support the database content, and which are available for review through links directly included within database content.

Additional improvements made to the OWL in FY2017 include:

- New reporting capabilities for:
  - Waste searching by classification or facility with links to waste detail;
  - Waste forms and characteristics with links to supporting documents;
  - Radionuclide inventory search by facility, classification, or name;
  - Radionuclide list with links to details;
  - Supporting document list with links to documents; and
  - Radionuclide inventory calculation by selected target date (primarily for hundreds of years timeframe);
- Improved reporting capabilities with:
  - New color and formatting themes (lighter, easier to read); and
  - Improved waste detail with selectable/customizable content types.

The other primary capability/quality improvements made for the OWL in FY2017 are the data entry checking process (Section 6.2) and kicking off the external OWL BETA Testing (Section 6.3).

3.2.2 Status of Data Entry Checking

A primary aspect of the OWL database is to provide as comprehensive a compilation of current information/data for DOE-managed (as) HLW and SNF that simultaneously provides direct links to the source documentation that underlies the content. This combination offers both readily checkable/verifiable information/data entries, as well as clear information paths, which can be updated expeditiously as new information/data is/are collected. Also, this provides as clear as possible derivation of the values being utilized with traceability to the source documentation. All of this facilitates maintaining clear understanding of the information/data content, as well as a direct method for finding/correcting errors in data entry. Lastly, because of the explicit link to the source documentation, consideration of, comparison to, and inclusion of, alternative data sources is simplified.

Given the additions to the content of the OWL, checking of the data entry process was initiated in FY2017 to verify the efficacy of the information/data already included in primary datasets for the essentially complete waste types. This primary goal of this process of data entry checking against the source documentation was to identify, and rectify, any errors in data entry using the database and source documents. This data entry checking is a continuous and iterative process of improvement for such a large set of information/data that is updated continuously. One key aspect of the checking process is that it is performed by technical individuals who are independent of the data entry process, and who document any issues noted, and resolve the issues with the data entry staff, with the aid of technical managers as needed.

The process for documenting any issues identified was for the checker to:

(a) print the OWL report to a MS Word file;

(b) highlight in the file all data entries as verified or potentially at issue (e.g., green highlight => verified; red or yellow highlight => potential issue); and

(c) summarize issues in an email to the data entry staff and the technical manager (at least).

Potential issues were then clarified/corrected via discussion and definition of summary solutions, with involvement from technical management as needed to define the path for correction.

For the data entry checking on the nine (9) primary datasets for the essentially complete waste types, th ere were 147 specific comments on potential issues.

These fell into the following types of comments/corrections:

- 82 – Typographical errors;
- 4 – Inconsistent units of measurement or presentation of information;
- 6 – Supporting document does not provide information it is supposed to provide;
- 31 – Supporting document is not clearly referenced or incorrectly referenced (e.g. wrong date, wrong document number);
- 21 – Information in OWL not matching the information in the supporting document (some cases where this was an issue for numerical values because of rounding of source values or because inventory content below a certain value was considered to be zero value);
- 2 – Information presented in an unclear fashion; and
- 1 – Link that does not work.

Virtually all comments were addressed directly and the issue was corrected. One issue is still being investigated for a solution. This issue involves assigning an inventory of zero for radionuclides with inventories less than about 10^{-7} curies. Only a few radionuclides fall into this issue category. The format of the data type for these values in OWL is floating point decimal and data entries can be input over the range from 10^{-7} to 10^{12} curies. Currently, work is ongoing to develop an approach that does not force an assumption of 0 curies for the inventory of those few radionuclides.

3.2.3 Kickoff of Online Waste Library (OWL) External Beta Testing

The first external BETA Test of the Online Waste Library (OWL) was launched in late July to solicit input from a small group of knowledgeable individuals throughout DOE Technical Staff (in DOE NE and EM) and at other national laboratories (INL and SRNL) who would be likely candidates to use the OWL and its content. These staff agreed to participate and take some time to access the OWL, use it as their time allowed, and provide input/feedback assessing its usability. We asked that they please compile their individual comments/input into a single file (MS-Word or text), and transmit it via email to SNL (to D. Sassani, L. Price and W. Walkow) prior to August 18, 2017. Although there is no hard end date to this

BETA Test of the OWL (it continues into FY2018), this permitted us to assess the preliminary input and develop a summary of feedback included below.

The OWL contains information and data on a variety of DOE-managed High-Level Wastes and their waste forms (both planned and alternative waste forms) for use in analyses of the back end of the fuel cycle. Additionally, we have included a small subset of the major DOE-managed SNF based on the information from the DOE SNF database. Future plans for the OWL include developing a methodology to leverage the information from the DOE SNF database for the back-end of the fuel cycle purposes in a proficient manner.

The database also contains links to the source reports for the content in the OWL, providing an efficient route to the primary information. The focus of this BETA Testing of the OWL, is on its usability and utility as a tool that facilitates access to this vast set of information for use throughout the DOE complex. A request was sent to the BETA testers to focus their comments/inputs/feedback primarily on how useful and straightforward it is to access the information and the source materials.

Examples questions to the BETA testers are:

a) What you found most likeable/easy about using the OWL?

b) What you do not like about the way it works/presents the information?

c) What improvements you would like to see in its usability?

d) What additions to data/information representation you would most like to see?

e) What aspects made using it more difficult than expected?

Although the focus of the OWL BETA Testing is on its utility/ease of use and information access/presentation, we welcome comments/corrections of any nature that BETA Testers want to provide. With any large set of information, there may be errors incurred during data entry/processing, we encouraged the participants to please note any errors as they saw them. As discussed above we have had staff conduct data entry checking of the information in the OWL against the source information. Following the OWL BETA Testing (to be planned for FY18), we are hoping to perform a larger data review to assess whether there are updates needed to the contents of the OWL, in addition to planning the updates for any major identified needs to the OWL functionality.

3.2.3.1 Preliminary OWL External Beta Test Feedback

The early feedback on the OWL consisted of 30 comments from the reviewers and are summarized as follows:

- 10 - Comments not requiring modifications to the OWL (e.g., compliments, connection problems that have already been resolved, general questions requiring a response but no change to the OWL);
- 8 - Problems with navigation (e.g., looking for an easy way to “Return to Home,” finding the parameters with which to filter database reports, presence of SharePoint headers and links, size of the collapse/expand box);
- 5 - Suggestions and questions regarding content (e.g., additional background information regarding OWL, adding MTU as a waste characteristic, adding the remaining DSNF, whether information is current);
- 4 - Comments regarding information presentation and selection (e.g., selecting items with which to filter or sort a database report, the units used to describe parameter values);
- 1- Link not working;

- 1 - Problem opening Excel spreadsheet; and
- 1 - Typographical error.

Two of the comments have already been addressed (links that did not work and typographical errors). Most of the other comments will be addressed via modifications to the OWL, either to the database or to the website, within the first half of FY2018. A few comments are the subject of ongoing efforts for expanding OWL content. One example is the inclusion of DSNF items in OWL, which will be handled by efforts to synchronize with the DOE SNF database at INL. These last aspects will likely take most of FY2018 to address. The external BETA testing will continue into FY2018, with inputs expected about the midpoint of the first quarter.

3.3 Waste Form Performance Constraints for Post-Closure Safety Assessments

Each waste form included into a performance assessment (PA) has characteristic degradation behavior assigned within the PA analyses to evaluate the release of radionuclides from the waste form (after waste packages are breached) over geologic time. In the current PA for the Generic Disposal System Analyses (GDSA), there are three types of degradation behavior: Instantaneous Degradation (ID); Used Nuclear Fuel (UNF or UO_2 or Spent NF-SNF) Degradation; and Glass Waste Degradation (GWD). The constraints on these are given here, followed by assessments of the basis.

3.3.1 Degradation of Potential Waste Forms

The purpose of this section is to document degradation rates of potential waste forms for current and future GDSA performance assessments. The waste forms considered include: (1) used nuclear fuel (UNF – also termed UO_2); (2) high level waste (HLW) glass and (3) DOE-managed spent nuclear fuel (DSNF). The degradation of UNF is included because naval UNF is expected to degrade similar to UNF and its degradation has been previously modeled as UNF in the Yucca Mountain TSPA. Hot isostatic pressed calcine waste is treated as degrading similar to HLW glass as evaluated below. Other waste forms considered but not modeled in the current GDSA include untreated granular calcine waste in case it is determined that it should be included in future PA.

3.3.1.1 UNF Source Term

In geologic repository modeling, UNF is generally assumed not to degrade in volume or structure until after containment breach. After containment failure and exposure to water (or humid air), release of radionuclides is typically modeled by two processes: instant release, discussed in the next subsection, and kinetically-controlled dissolution, discussed in the two sections below that.

UNF Instant Release Fractions

Within an intact fuel rod, volatile fission products collect at grain boundaries and in the gap between the fuel pellet and fuel rod. After containment breach, these products can quickly move to the surrounding environment. In repository modeling, an instant release fraction model is generally used at the time of breach to transfer radionuclides in the gap and grain boundaries to the transport domain.

Measurements of instant release fractions for UNF at different burnups and for a variety of environmental conditions are collected and evaluated in Sassani et al. (2012). For PWR fuel and 60 MWd/kgHM burnup, the Sassani et al. (2012) study recommends the instant release fractions summarized in Table 3-1.

Data compiled in Kienzler et al. (2012) indicate that there may also be an initial dissolution of around 10^{-5} of the UNF waste form at the time of containment breach. This initial dissolution can be included in a simulation by including an additional instant release fraction of 0.001% for all radionuclides in the waste form.

Table 3-1. UNF instant release fractions for PWR (60 MWd/kgHM burnup)

Source	Instant Release Fraction (%)	Comments
Johnson et al. (2005)	C: 10 Cl: 5 Sr, Tc: 7(11) I, Cs: 10(16)	Best estimates (pessimistic estimates in parentheses); Sassani et al. (2012) recommends using the best estimates for 60 MWd/kgHM burnup, pessimistic estimates for 75 MWd/kgHM burn-up, and a linear relationship for fuel with burnups that fall between

UNF Dissolution in Groundwater in Crystalline Rock

Groundwater at the depth of a potential mined repository (about 500 m) in crystalline rock is generally brackish. In Sweden, Finland, and Canada, groundwater at 500 m is dominated by sodium, calcium, and chloride with total dissolved solids in the range of 1 to 10 g L^{-1} or higher (Mariner et al. 2011, Table 2-1). Below 3,000 m, as in the case of deep borehole disposal, Na-Ca-Cl solutions continue to dominate but are likely to be brines with total dissolved solids in the range of 100 g L^{-1} or higher (Brady et al. 2009). UNF dissolution in brines is addressed in the next section.

Groundwater at depth in crystalline rock is reducing. Reducing conditions are maintained by limited mixing of infiltrating waters and an abundance of oxygen-consuming reactants along the flow path. At Olkiluoto, iron oxyhydroxides are observed in fractures only in the top few meters of rock (Posiva 2010, Section 6.2.5). At approximately 300 m at Olkiluoto, reducing conditions are strong enough to reduce sulfate to sulfide. Below 300 m, concentrations of methane rise and conditions are strongly reducing, e.g. -300 mV below 3,000 m, pH of 8 to 9 (Anderson 2004). Adding to the naturally reducing conditions, corrosion of steel is expected to further reduce the redox potential in the vicinity of a breached waste package. Radiolysis induces oxidizing conditions at an exposed UNF surface but is not expected to significantly affect the overall local redox potential.

Studies that measure UNF dissolution rates under strongly reducing conditions (imposed using H_2(g) or metallic iron) show that they result in very low UO_2 dissolution rates despite the oxidizing effects of radiolysis (Röllin et al. 2001; Werme et al. 2004). Table 3-2 presents UNF dissolution rates measured and used for reducing conditions. The first two references, SKB (2006, 3.3.7) and Pastina and Hellä (2010, 1.4.6), establish fractional rates (10^{-8} to 10^{-6} yr^{-1}) used in the performance assessments of the repository programs in Sweden and Finland. These rates are supported by the third reference, Ollila (2008), and others (Grambow et al. 2000; Werme et al. 2004; Carbol et al. 2006). Ollila (2008) studied UO_2 doped with ^{233}U at concentrations representative of alpha dose levels expected at 3,000 to 10,000 years for a BWR fuel rod. The fourth reference, Röllin et al. (2001), provides a forward dissolution rate for a transition state theory (TST) model. This rate should not be used as an overall long-term rate because flow-through conditions keep aqueous U(IV) concentrations far below saturation.

Actual UNF dissolution rates are expected to vary over time as a function of competing processes and changes in environmental conditions. Important processes and parameters include:

- generation of radiolytic oxidants,
- generation of H_2(g) and Fe^{2+} from degradation of steel,
- catalyzed oxidation of H_2(g),
- precipitation of secondary phases,
- complexation of uranyl bicarbonate,
- oxidation of Fe^{2+},
- temperature variations, and
- diffusion of chemical species at the interface.

Except for the generation of $H_2(g)$ and Fe^{2+}, all of these processes are included in version 2.3 of the Fuel Matrix Degradation Model (FMDM) (Jerden et al. 2015). The FMDM dissolution rate is calculated in units of mg m^{-2} yr^{-1}. A specific surface area of approximately 0.001 m^2 g^{-1} may be used to convert the FMDM rate to a fractional dissolution rate (Cachoir and Mennecart 2011; Jerden, J., pers. comm.).

Coupling the FMDM to a repository model, as done using PFLOTRAN (Mariner et al. 2015), allows the UNF dissolution rate to be calculated mechanistically over time as a function of changing conditions. For example, as the dose rate decreases by orders of magnitude over thousands years, the generation of radiolytic oxidants decrease accordingly and reduce the rate of UO_2 oxidation. Other processes and conditions that reduce dissolution rates over time include decreasing temperatures and the buildup of secondary mineral phases at the fuel surface.

Until the FMDM or other mechanistic model is fully developed, measurements and analyses are used to establish UNF dissolution rate distributions for repository modeling. Such analyses need to consider the expected time frame of containment breach and the environmental and radiolytic conditions after breach. Assuming strongly reducing conditions and lower dose rate after breach, the distribution of UNF dissolution rates used in the Swedish performance assessment (SKB 2006, 3.3) (see Table 3-2) is reasonable for a mined repository in crystalline rock.

UNF Dissolution in Brines

Measurements and data on the dissolution of UNF in brines are available from several studies (e.g., Grambow et al. 2000; Loida et al. 2005; Metz et al. 2008; Ollila 2008; Kienzler et al. 2012). Rates from these studies are summarized in Table 3-3. Many of these studies report rates in terms of “fraction of inventory in the aqueous phase” (FIAP) per day.

It is important to note that reported rates in these studies are often average rates over the durations of the experiments. The trend in Fig. 18 of Kienzler et al. (2012) is consistent with a low dissolution rate after the first few days. This is illustrated in Figure 3-27 where a line is superimposed on a copy of the Kienzler et al. (2012) figure to show how the data would track if the initial concentration in the aqueous phase remained constant for the remainder of the experiment, i.e., a zero dissolution rate from that point on.

Table 3-2. UNF dissolution rates relevant to contact with groundwater in crystalline rock under reducing conditions

Source	Rates	Units	Comments
SKB (2006, 3.3.7)	10^{-8} (min) 10^{-7} (mode) 10^{-6} (max)	yr^{-1}	Log-triangular distribution based on Werme et al. (2004)
Pastina and Hellä (2010, 1.4.6)	10^{-7} (reference)	yr^{-1}	Based on model by Werme et al. (2004) and data by King and Shoesmith (2004), Ollila and Oversby (2005), Carbol et al. (2006), and Ollila (2008) that show absence of radiolysis effects in presence of metallic iron (strongly reducing conditions); considered pessimistic (p. 138)
Ollila (2008)	Anoxic: 8.1×10^{-7} (min) 2.2×10^{-6} (max) Reducing: 4.3×10^{-8} (min) 2.2×10^{-7} (max)	yr^{-1}	Static batch dissolution tests, isotope dilution, 0.01 M NaCl; UO_2 doped with 0, 5 and 10% ^{233}U; anoxic conditions from N_2 and 1 ppm S^{-2} (E_h ~ -200 mV); reducing conditions from N_2 and Fe (E_h ~ -400 mV); 2 $cm^2\ g^{-1}$ geometric surface area
Röllin et al. (2001)	$6 \times 10^{10} \times U_{max}$	$mg\ m^{-2}\ d^{-1}$	U_{max} is the aqueous solubility of $UO_{2(c)}$ in $mol\ L^{-1}$; 300 $cm^2\ g^{-1}$; reducing conditions (~8×10^{-4} $mol\ L^{-1}$ $H_{2(g)}$); forward reaction rate because measured under flow-through conditions; very low flow rates provided insufficient flux of $H_{2(g)}$ to maintain reducing conditions
Jerden et al. (2015)	FMDM	$mg\ m^{-2}\ yr^{-1}$	The FMDM code is coupled with PFLOTRAN to calculate the UNF dissolution rate as a function of environmental conditions and surface precipitation (see text); 0.001 $m^2\ g^{-1}$ specific surface area recommended (Cachoir and Mennecart 2011; Jerden, J., pers. comm.)

Grambow et al. (2000, WP III.1) observed very low rates (< 10^{-9} day^{-1}) at the end of a 4.4-year experiment on 50 MWd/kgHM burnup spent fuel pellets in 5 molal NaCl solution in the presence of metallic iron powder. That study showed slowly changing ^{90}Sr FIAP measurements toward the end of the experiment where "the progress of matrix dissolution seems to stop."

The dissolution study by Ollila (2008) of ^{233}U-doped UO_2 indicates that increasing ionic strength may noticeably reduce dissolution rates. Under reducing conditions, the range of dissolution rates was lower in 0.5 and 1 M NaCl solutions (2.2×10^{-8} to 1.6×10^{-7} yr^{-1}) than in 0.01 M NaCl (4.3×10^{-8} to 2.2×10^{-7} yr^{-1}). The degree of doping in these experiments was designed to produce alpha dose rates of BWR fuel of ages 3,000 and 10,000 years.

Maximum dissolution rates for spent fuel decrease nearly in proportion with fuel age (Nielsen et al. 2008). This relationship is shown in Figure 3-28 for fuels of different burnup (Ollila 2011, Table 2-3). Ollila (2011) concludes that an activity of at least 1.8×10^7 to 3.3×10^7 Bq g^{-1} is needed to observe alpha radiolysis effects on UNF dissolution in a 0.001 M carbonate solution under anoxic conditions. Ollila (2011) also concludes that the presence of carbonate reduces UNF dissolution rates as bicarbonate scavenges hydroxyl radicals.

The K8 fuel pellet data of Loida et al. (2005) for a 5.6 molal NaCl solution with a H_2(g) overpressure of 3.2 bar, as best depicted in Fig. 7 of Metz et al. (2008), indicate a dissolution rate of approximately 2×10^{-7} FIAP d^{-1} over 1,095 days (3.0 years). This rate is approximately half the rate (4×10^{-7} FIAP d^{-1})

calculated over the first 213 days (Loida et al. 2005, Fig. 2). These rates are much higher than those of Ollila (2008) as the alpha dose rate is much higher.

Data from Metz et al. (2008) indicate that the presence of 10^{-4} to 10^{-3} molal bromide significantly increases the dissolution of spent nuclear fuel pellets. The measured rates (10^{-6} to 10^{-5} FIAP d^{-1}) are shown to decrease with time over the length the study (Metz et al. 2008, Fig. 7 and 8). The effect of bromide appears to be that it reduces the protective H_2 effect as it reacts with beta/gamma radiolysis products (Loida et al. 2007). Because beta/gamma activity diminishes more quickly than alpha activity and alpha activity dominates the radiation field in the long term, this effect may only be significant for spent fuel in canisters that fail at early times.

Because (1) the dose rate is a major factor in the rate of UNF dissolution, (2) the dose rate decreases by orders of magnitude over thousands of years, and (3) UNF in repository concepts is generally not expected to be exposed to water (or humid air) for thousands of years, UNF dissolution rates for repository concepts after containment breach are expected to be much lower than rates measured for current spent fuel. As noted in the previous section, until the FMDM or other model is fully developed to account for the major processes, measurements and analyses are used to establish UNF dissolution rate distributions. For UNF dissolution in brines after containment breach, the rates reported in Grambow et al. (2000, WP III.1) and Ollila (2008) for brine solutions are expected to be particularly relevant. The measurements from these studies are in the same general range as the rates used in SKB (2006, 3.3.7) and Pastina and Hellä (2010, 1.4.6) for deep groundwater in crystalline rock (Table 3-2). While there appears to be a decrease in UNF dissolution rate as salinity increases (Ollila 2008), the decrease is not great. Thus, until salinity and/or bromide concentration is shown to be a major factor for aged fuel (e.g., >1,000 years), or until a model such as the FMDM is fully developed and coupled to the repository model, it is reasonable to use the distribution of SKB (2006, 3.3.7) (Table 3-2) for UNF dissolution rates in brine.

Table 3-3. UNF dissolution rates in brine

Source	Fractional Rates	Units	Comments
Grambow et al. (2000, WP III.1)	$< 10^{-9}$	day^{-1}	5 molal NaCl solution, 50 MWd/kgHM, in presence of metallic iron powder
Ollila (2008)	0.5 M NaCl: 5.4×10^{-8} to 1.6×10^{-7} 1.0 M NaCl: 2.2×10^{-8} to 5.4×10^{-8}	yr^{-1}	0.5 and 1 M NaCl, static batch dissolution tests, isotope dilution, 79 days; UO_2 doped with 0, 5 and 10% ^{233}U; reducing conditions from N_2 and Fe (E_h ~ -400 mV); 2 cm^2 g^{-1} geometric surface area
Kienzler et al. (2012, Fig. 18)	2×10^{-9} to 10^{-5}	FIAP d^{-1}	Range of values for brines compiled and plotted in Kienzler et al. (2012, Fig. 18), also shown in Figure 3-27
Loida et al. (2005)	4×10^{-7} (213 d) 2×10^{-7} (1,095 d)	FIAP d^{-1}	5.6 molal NaCl solution at strongly reducing conditions (3.2 bar H_2(g)); overall average rates (see text)
Metz et al. (2008)	10^{-6} to 10^{-5}	FIAP d^{-1}	5.3 molal NaCl solution at strongly reducing conditions (0.32 MPa H_2(g)) in presence of 10^{-4} to 10^{-3} molal Br^-

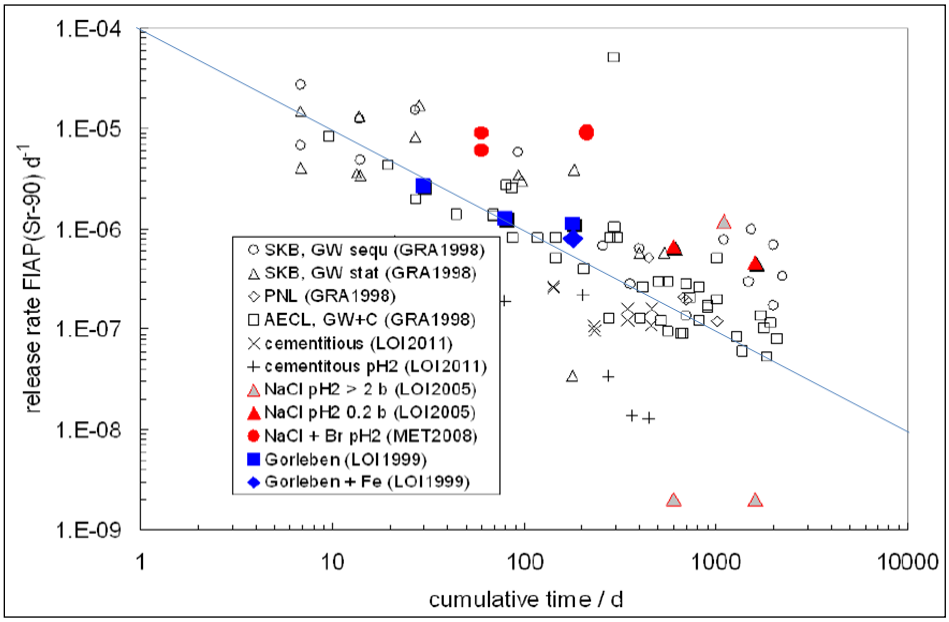


Figure 3-27. Compilation of UO_2 dissolution rate measurements displayed in Kienzler et al. (2012, Fig. 18). Blue line, which is superimposed on the Kienzler et al. (2012) figure, indicates the trend if zero dissolution occurs after starting at any point on the line.

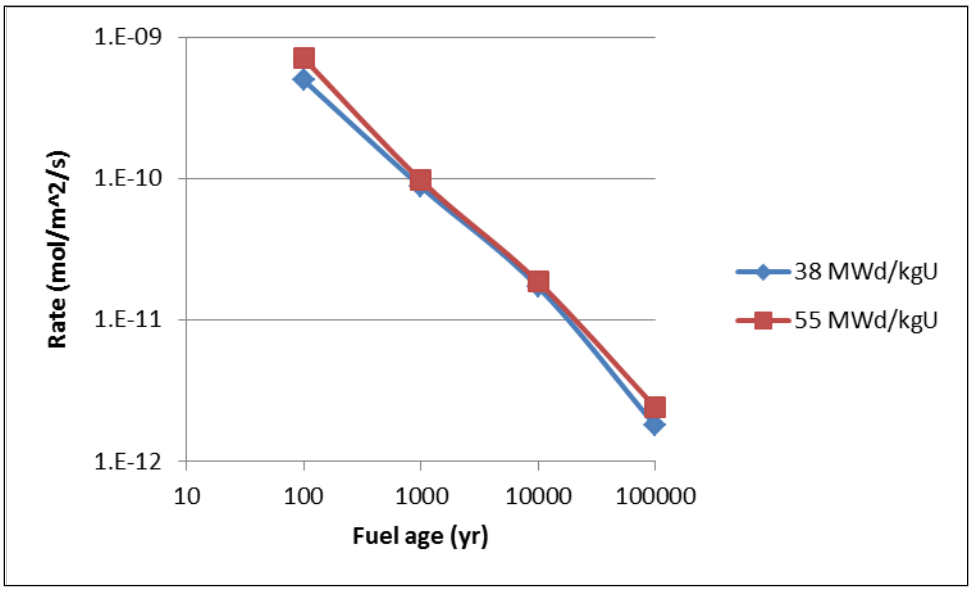


Figure 3-28. Maximum fuel dissolution rate calculated as a function of fuel age (Ollila 2011, Table 2-3).

3.3.2 HLW Glass Source Term

As in the case of UNF, HLW glass is typically assumed not to degrade until exposed to water. Instant release fractions for HLW glass are expected to be small and are typically not simulated. After containment breach, dissolution rates are often calculated as a function of temperature, specific surface area, and water composition.

The dissolution rate per unit surface area for HLW glass is a function of water composition, ion exchange, precipitation of alteration products, and transport processes across an alteration layer. Section 3.2.2.1

addresses the major competing processes and summarizes two dissolution models used in performance assessment.

To calculate an overall dissolution rate the glass surface area is needed. Surface area is a function of cracking, but dissolution in cracks is limited by diffusion. Because cracks and their properties are highly important to calculating bulk dissolution, Section 3.2.2.2 discusses HLW glass surface area and the effects of cracking on overall dissolution rates.

3.3.2.1 HLW Glass Dissolution

The evolution of glass dissolution rates over time can be described as having three stages (Vienna et al. 2013). In stage I, aqueous silica concentrations are below saturation and glass dissolution is rapid. As water near the glass surface approaches saturation with respect to silica, rates decrease markedly until aqueous silica concentrations reach saturation and alteration products of silica begin to precipitate. At this point, stage II begins and glass dissolution rates are low. After a period of time at stage II, a stage III dissolution rate can potentially occur where rates increase significantly. Stage III dissolution is poorly understood and is generally excluded in repository modeling (Vienna et al. 2013).

Table 3-4 summarizes two rate models used in repository performance assessment. These models are stage II models. In stage II, though the solution at the interface is essentially saturated with respect to silica alteration products, the glass continues to dissolve and alteration products continue to accumulate. Dissolution at this stage is driven by the thermodynamic instability of HLW glass.

Each of the models in Table 3-4 calculates a long-term dissolution rate that can be used for both dilute and saline solutions in repository simulations. The first of the two models is an empirical exponential equation fitted to temperature (Kienzler et al. 2012, Eq. 6, p. 17). The second is a more analytical model that includes the additional effects of water composition and thermodynamics (Strachan 2004, 8.0). Each model is fitted to observed behavior in long-term laboratory studies.

Table 3-4. HLW glass dissolution rate models used in repository performance assessment

Source	Rates	Units	Comments
Kienzler et al. (2012, Eq. 6, p. 17)	$560 \cdot exp\left(\frac{-7397}{T}\right)$	kg m^{-2} d^{-1}	T is temperature in Kelvin. Rate based on measurements in water and in salt solutions. Illustrated in Fig. 5 of Kienzler et al. (2012)
Strachan (2004, 8.0)	$k \cdot 10^{\varphi \cdot pH} exp\left(\frac{-E_a}{RT}\right)$	g m^{-2} d^{-1}	The larger of two calculations ("acidic" and "alkaline") is used for a given pH. For the "acidic" calculation, k is 1.15 × 10^7 g m^{-2} d^{-1}, φ is -0.49, and E_a is 31 kJ mol^{-1}. For the "alkaline" calculation, k is 3.47 × 10^4 g m^{-2} d^{-1}, φ is +0.49, and E_a is 69 kJ mol^{-1}. T is temperature, and R is the universal gas constant.

3.3.2.2 HLW Glass Surface Area

The surface area of a HLW glass cylinder is a function of container geometry, void space, and the number and size of exposed cracks. Cracking is expected to largely be the result of cooling as the glass hardens after it is poured into its canister. Rough handling may also cause cracking. Chemical processes typically do not cause cracking, but they can cause cracks to grow or, alternatively, cement existing cracks.

The exposed surface area of HLW glass is generally calculated from the following relationship:

$$S = f_{exposure} A$$

where $f_{exposure}$ is the exposure factor and A is the nominal geometric surface area. The $f_{exposure}$ parameter is non-dimensional and accounts for increased surface area due to cracking and surface roughness. This parameter is the key parameter used in repository performance assessment to establish the effective surface area of the HLW glass. The value of $f_{exposure}$ is greater than one but is restrained as

needed to account for reduced dissolution rates in cracks. Used and recommended values for this factor are shown in Table 3-5.

The value of the surface area (A) changes over time and can be calculated as the product of the geometric specific surface area (s_a) and the mass of glass remaining (M):

$$A = s_a M.$$

For COGEMA glass R7T7, Kienzler et al. (2012) estimates an initial geometric surface area of 1.7 m^2 (1.08 m in length and 0.42 m in diameter), an initial mass of 412 kg, and an exposure factor of 10. These values imply a geometric specific surface area of 4.1×10^{-3} m^2 kg^{-1} and a total exposed specific surface area of 0.041 m^2 kg^{-1}.

For U.S. HLW glass, Strachan (2004, 6.5.4) estimates initial masses and volumes for three proposed canisters:

- 1,682 kg and 0.626 m^3 for Defense Waste Processing Facility (DWPF) glass
- 1,900 kg and 0.704 m^3 (2.49 m in length and 0.61 m in diameter) for West Valley Demonstration Project (WVDP) glass, and
- 1,650 kg and 0.626 m^3 at 825°C for Hanford "long" glass canisters.

The geometric specific surface area for the DWPF and WVDP glass is 2.8×10^{-3} m^2 kg^{-1}, and that of the Hanford "long" canister glass is 2.6×10^{-3} m^2 kg^{-1} (Strachan 2004, 6.5.4). Strachan (2004, 8.2.1) recommends a triangular distribution for $f_{exposure}$, with a value of 4 for both the minimum and most probable value and 17 for the maximum value. The maximum value is a weighted average wherein all glass undergoes thermal cracking and 1 out of 100 glass cylinders experiences impact cracking. For the maximum value of 17, all cracks are assumed to be fully accessible and reactive. The minimum and mode value of 4 is calculated as the maximum value of 17 reduced by a factor of approximately 4 to account for reduced accessibility and reactivity of cracks.

A more straightforward representation of cracking and effective surface area is provided by the relationship:

$$f_{exposure} = f_{crack} f_{reactivity}$$

where f_{crack} is the ratio of the total surface (with cracking) to the geometric area alone (e.g., cylinder), and $f_{reactivity}$ is the effective fraction of total surface area that dissolves as fast as the outer surface of the glass. Based on the analysis by Strachan (2004, 6.5.4), 99% of the glass cylinders would have a f_{crack} value of 12 (for thermal cracking only) and 1% would have a value of 480 (for both thermal and impact cracking).

The value of $f_{reactivity}$ accounts for reduced dissolution resulting from reduced crack accessibility and reduced diffusion of glass components to the bulk solution. Like the value of f_{crack}, the value of $f_{reactivity}$ is uncertain. Perez and Westsik (1981) performed static leach tests with small polished borosilicate glass cylinders at different spacing to simulate different sizes of cracks. They demonstrated that glass surfaces with no space between them do not contribute to glass dissolution while a spacing of 0.038 cm contributes at a rate that is two to five times slower than the outer surface of the glass. Based on the Perez and Westsik (1981) study, $f_{reactivity}$ is clearly less than one. How much less depends on the apertures and depths of the cracks in HLW glass and the transfer rate of glass components away from the glass. Strachan (2004, 6.5.4) effectively used a value of 1 for $f_{reactivity}$ when calculating the maximum for $f_{exposure}$ and a value of 0.25 (0.5×0.5) when calculating the minimum and mode.

Much work remains to improve confidence in the distribution of $f_{exposure}$ for HLW glass. In the meantime, the distribution of Strachan (2004) is adequate for repository modeling.

Table 3-5. HLW glass $f_{exposure}$ values

Source	Value	Comments
Kienzler et al. (2012, Table A-1)	10	COGEMA glass R7T7
Strachan (2004, 8.0)	4 (min) 4 (mode) 17 (max)	Triangular distribution; conservatively calculated (see text)

3.3.3 Evaluation of Bases for Assigning Post-Closure Performance Constraints

The models for degradation of both UO_2 and HLW glass given above are currently being used within the GDSA for PA modeling of post-closure system evolution. The waste forms in the current DGRDMSH analyses have been mapped into those models as either performing similarly or being bounded by a particular model degradation behavior. For example, the HIP calcine waste form is assigned to degrade as the HLW glass degradation. For waste forms that do not have substantial waste form lifetimes (i.e., generally only 10,000 years, or less), the instantaneous degradation rate is used. In all cases the waste form degradation is the initial, kinetic step, and the dissolved radionuclides are evaluated against solubility limits based in part on the geologic environment.

The current assignments for degradation rates of the DSNF in the DGRDMSH inventory are based on the work in the YM SAR (DOE, 2008), which assigned virtually all the DSNF to the instantaneous degradation rate model except for the naval SNF. This was based primarily on the small amounts the other than naval DSNF represented in the YM SAR relative to the mass of CSNF. Because the DSNF represents a fractionally larger portion of the radionuclides in the DGRDMSH, we have reviewed the bases for the PA groupings from the YM SAR and some prior analyses to see if there may be some of the DSNF waste forms that have a basis for better performance in post-closure (Section 3.2.3.1). In addition, the assumption of glass degradation being assigned to the HIP calcine waste form was evaluated as well.

3.3.3.1 DOE-managed Spent Nuclear Fuel (DSNF) Grouping and Associated Degradation Models

Background of DSNF Grouping in Support of Performance Assessment and Disposal Concepts

A number of published reported and meeting documents have focused on the management of the more than 200 DOE-managed spent nuclear fuel (DSNF) types into groups for specific purposes, such as disposition in geological repositories. A representative example of such attempts to selectively group DSNF was documented in 1997 in the report *Grouping Method to Minimize Testing for Repository Emplacement of DOE UNF* (DOE-EM, 1997). This report suggested the partition of DSNF into 11 groups for testing purposes, based on the examination of available data and information and associated degradation models of DSNF. The behaviour of DSNF in terms of time-to-failure and release rate was found to be primarily influenced by fuel matrix and cladding, while seven other parameters (i.e., burnup, initial enrichment, cladding integrity, fuel geometry, radionuclide inventory, fission gas release, and moisture content) had only limited impact on fuel behaviour (DOE-EM, 1997; DOE-EM, 1998a). However, subsequent discussions suggested that this 11-group partition is not suitable for other analyses, such as criticality evaluations in support of DSNF repository disposal, and a new partition into 34 intermediate condensed DSNF groups was proposed based on fuel matrix, cladding, cladding condition, and enrichment (DOE-NSNFP, 2002).

For the purpose of total system performance assessment (TSPA), those 34 DSNF groups could be reduced to 16 groups for the TSPA, with the seminal rationale for such partition documented in the report *DOE UNF Information in Support of TSPA-VA* (cf. Figure 5-1 in DOE-EM, 1998b). Further details for

grouping covered in the report *DOE UNF Grouping in Support of Criticality, DBE, and TSPA-LA* (DOE-EM, 2000). According to the DOE grouping team assessment, the 34 intermediate condensed DSNF groups in support of the postclosure safety case could be further reduced to 13 groups for the purpose of post-closure performance assessment (PA) analyses (DOE-NSNFP, 2002), with a subsequent refinement to 11 DSNF groups for TSPA [by placing the plutonium/uranium nitride fuels in the "miscellaneous fuel" group (Group 10 below) due to their small quantity and the uranium beryllium oxide fuels into the "uranium oxide" group (Group 8 below) owing to their similarities]. The final DSNF TSPA grouping in support of the YM SAR for the purpose of postclosure safety is given below:

Group 1 - Naval spent nuclear fuel (Classified UNF from surface ship/submarine assemblies)

Group 2 - Plutonium/uranium alloy (Fermi Core 1 and 2 UNF)

Group 3 - Plutonium/uranium carbide (Fast Flux Test Facility-Test Fuel Assembly UNF)

Group 4 - Mixed oxide and plutonium oxide (Fast Flux Test Facility-Demonstration Fuel Assembly/Fast Flux Test Facility-Test Demonstration Fuel Assembly UNF)

Group 5 - Thorium/uranium carbide (Fort St. Vrain UNF)

Group 6 - Thorium/uranium oxide (Shippingport light water breeder reactor UNF)

Group 7 - Uranium metal (N Reactor UNF)

Group 8 - Uranium oxide (Three Mile Island-2 core debris)

Group 9 - Aluminum-based UNF (Foreign Research Reactor UNF)

Group 10 - Miscellaneous Fuel

Group 11 - Uranium-zirconium hydride (Training Research Isotopes–General Atomics (TRIGA) UNF).

The aforementioned 11 DSNF groups were used in the TSPA-SR/LA in FY 1999 (cf. details in DOE-NSNFP, 2002).

Recently, a new grouping of waste forms was introduced in the context of the various disposal concepts being considered in the WFDOE (SNL, 2014). As discussed in Section 2.3.1.1, the waste groups (WG) are based on expected postclosure performance, radionuclide inventory, thermal characteristics, chemical characteristics, physical characteristics, packaging, and considerations of safeguards and security. Within those groups the DGRDMSH DSNF inventory is captured in WG5 (metallic SNF), WG7 (oxide spent fuels), WG9 (coated-particle spent fuel, e.g., TriSO particles) and WG10 (naval SNF).

Preliminary postclosure PA analyses within the GDSA for a DGRDMSH in the various representative disposal concepts under consideration (i.e. mined repositories in three geologic media—salt, clay/shale rocks, and crystalline (e.g., granitic) rocks—and deep borehole disposal in crystalline rocks) are currently underway.

Degradation Models for the DSNF Groups

Actual postclosure analyses carried out as part of the FY 1999 TSPA demonstrated that, for the aforementioned 11 DSNF groups considered for TSPA, a U-metal spent fuel surrogate can accurately represent DSNF properties for the base case in TSPA (DOE, 2000), except for Naval spent nuclear fuel

(Group 1) owing to it significantly different and robust design which allows this UNF to remain essentially intact beyond several hundred-thousand years, therefore significantly delaying release from naval SNF (DOE-NSNFP, 2002). In order to provide a conservative simplification for the TSPA, the commercial light water reactor UNF (i.e., UO_2–type UNF) was used as a surrogate for naval UNF under the range of expected repository environmental conditions (DOE-OCRWM, 2004). Therefore, only two release/degradation models – i.e., instantaneous (Groups 2-11) and UO_2–type (Group 1) release/degradation models - were used to simulate radionuclide release from those 11 DSNF groups in the TSPA-LA model (DOE-OCRWM, 2004).

A similar mapping of the DSNF inventory for a DGRDMSH into two release/degradation models, namely UO_2–type UNF and instantaneous models, has been adopted for initial GDSA DGRDMSH post-closure analyses comprising Naval UNF is assumed to degrade as UO2–type UNF (following the conservative assumption made previously for DSNF TSPA Group 1), while it can be inferred that all other DSNF will release/degrade instantaneously (as was assumed for DSNF TSPA Groups 2 to 11).

This conservative selection of only two upper-limit release/degradation models to represent the DSNF properties was specifically tied to the base case in TSPA (DOE 2000b), where inventory was dominated by CSNF. Because the DGRDMSH inventory is quite different from that (Section 2), it is desirable to evaluate the degradation models to see if DSNF degradation properties are appropriately captured, or if additional degradation behavior would be appropriate to add into GDSA. In order to achieve this, a close reexamination of the various initial release/degradation models for the 11 TSPA DSNF groups (DOE-NSNFP, 2002) was undertaken. Summaries of DSNF wet dissolution models from DOE-NSNFP (2002) of upper-limit degradation models, and best-estimate degradation models developed for each of the 11 TSPA DSNF groups from DOE-OCRWM (2004) are presented in Table 3-6 and Table 3-7, respectively.

Table 3-6. DOE UNF wet dissolution models (adapted from DOE-NSNFP 2002)

Fuel Group	Fuel Matrix	Typical Fuel in the Group	Wet Dissolution Model
1	Naval fuel	Surface Ship/Submarine Assemblies	Commercial model
2	Pu/U alloy	FERMI Core 1 and 2 standard fuel assembly fuel	U-<8 wt% Mo/water model
3	U/Pu carbide	Fast Flux Test Facility (FFTF-TFA-AC-3) carbide fuel	100x U-metal model
4	MOX	Fast Flux Test Facility (FFTF-DFA/TFA) oxide fuel	Commercial model
5	U/Th carbide	Fort St. Vrain fuel	10x U-metal model
6	U/Th oxide	Shippingport LWBR fuel	Ceramic model (Ringwood)
7	U-metal	N-Reactor fuel	U-metal/water model
8	U-oxide	Three Mile Island fuel Shippingport PWR fuel	Commercial model
9	Al-based	Foreign Research Reactor fuel	Aluminum alloy model
10	Miscellaneous UNF	Miscellaneous fuel	U-metal
11	U-Zr-Hx	Training Research Isotopes—General Atomic fuel	0.1x Commercial model

As shown in Table 3-6, eight variants of dissolution/degradation models (including multiples of those models) were considered:

- the commercial UO_2–type model (Groups 1, 4, 8),
- the 0.1x commercial UO_2–type model (Group 11),
- the U–metal model (Groups 7 and 10),
- the 10x U–metal model (Group 7),
- the 100x U–metal model (Group 3),
- the U-<8 wt% Mo/water model (Group 2),
- the ceramic model (Ringwood) (Group 6), and
- the aluminum alloy model (Group 9).

Based on composition alone, those variants can be further regrouped into only five main dissolution/degradation models, namely, the commercial UO_2–type model (Groups 1, 4, 8 and 11), the U–metal model (Groups 3, 5, 7 and 10), the U-<8 wt% Mo/water model (Group 2), the ceramic model (Ringwood) (Group 6), and the aluminum alloy model (Group 9).

Table 3-7. DSNF, Naval UNF, Plutonium Disposition Release/Degradation Models (adapted from DOE-OCRWM 2004).

	Upper-Limit Model		Best-Estimate Model
DSNF Group	**Model**	**Surrogate**	**Model**
1. Naval	Commercial UNF	UO_2–type	Commercial UNF
2. Plutonium / Uranium Alloy	Instantaneous release upon exposure to groundwater	uranium - molybdenum	(semi-empirical) rate (mg metal/cm^2/h) = 1.15×10^8 exp{(-66,500 ± 12,200 J/mol)/RT} [100–178°C] rate (mg metal/cm^2/h) = 1.58×10^6 exp{(-80,500 ± 10,600 J/mol)/RT} [304-440°C] (Linear interpolation between 178°C and 304°C)
3. Plutonium / Uranium Carbide	Instantaneous release upon exposure to groundwater	uranium metal	100 × Unirradiated uranium metal best-estimate: k (mg/m^2-day) = $100 \times \{1.21 \times 10^{15}$ exp(-66.4 ± 2.0 kJ/mol /RT)}
4. Mixed Oxide and Plutonium Oxide	Instantaneous release upon exposure to groundwater	light water reactor UNF	(semi-empirical) uranium oxide best-estimate model
5. Thorium / Uranium Carbide	Instantaneous release upon exposure to groundwater	SiC	(semi-empirical) R (kg/m^2-s) = 0.6×10^{-12}
6. Thorium / Uranium Oxide	Instantaneous release upon exposure to groundwater	Synroc	(semi-empirical) k (mg/m^2·day) = $82.0 \times 10^{(-1,000/TK)}$
7. Uranium Metal-Based	Instantaneous release upon exposure to groundwater	N Reactor	(semi-empirical) 2.52×10^{10}exp (-66,400/RT) mg/cm^2-hr R = 8.314 J/mol-K
8a. Intact Uranium Oxide	Instantaneous release upon exposure to groundwater	light water reactor UNF	(semi-empirical) uranium oxide best-estimate model
8b. Damaged Uranium Oxide	Instantaneous release upon exposure to groundwater	Three Mile Island-2 debris	(surface area enhancement factor of 100 is based on professional judgment) 100 × uranium oxide best-estimate
9. Aluminum-based	Instantaneous release upon exposure to groundwater	Savannah River Site uranium/ aluminum UNF in J-13 well water	(empirical) 1.38 mg metal/m^2·day at 25°C 13.80 mg metal/m^2·day at 90°C
10. Miscellaneous	Instantaneous release upon exposure to groundwater	N/A	(empirical) rate (mg metal/cm^2/h) = 1.15×10^8 exp{(-66,500 ± 12,200 J/mol)/RT} [100–178°C] rate (mg metal/cm^2/h) = 1.58×10^6 exp{(-80,500 ± 10,600 J/mol)/RT} [304°C to 440°C]
11. Uranium-Zirconium Hydride	Instantaneous release upon exposure to groundwater	Training Research Isotopes–General Atomic	(empirical) 0.1 × uranium oxide best estimate

For the DSNF in WG5, WG7, WG9 and WG10 a potential remapping to the behaviors for the 11 groups above is given in Table 3-8. The WG10 (naval SNF) corresponds to Group 1 and will continue to be represented with the UO_2–type degradation model. DSNF in WG5 (metallic and non-oxide spent fuels)

comprise aspects of Group 2 (Pu/U alloy, with U-<8 wt% Mo/water degradation model), Group 7 (U-metal, with instantaneous degradation model), Group 9 (Al-based, with aluminum-alloy degradation model), Group 10 (miscellaneous UNF, with instantaneous degradation model). So there may be some waste forms within that group that could have various models assigned in future GDSA PA analyses if desired. The DSNF in WG7 (DOE oxide spent fuels) will include fuel belonging to Group 4 (MOX, with UO_2–type degradation model), Group 6 [U/Th oxide, with ceramic degradation model (Ringwood)], and Group 8 (U-oxides, with UO_2–type degradation model) and Group 11 (U-Zr-Hx, with UO2–type degradation model). Finally, DSNF from WG9 would correspond to Group 3 (U/Pu carbide, with instantaneous degradation model) and Group 5 (U/Th carbide, with instantaneous degradation model). This tentative remapping, with respect to degradation/dissolutions, of DSNF in WG5, WG7, WG9 and WG10 into Group 1 through Group 11 allows consideration of more specific assignments for PA analyses. This would only be undertaken if there was a need for such detail based on post-closure performance assessment results.

Table 3-8. Possible remapping of DSNF in WG5, WG7, WG9 and WG10 into Groups 1-11

Waste Group	Fuel Group	Fuel Matrix	Typical Fuel in the Group	Degradation Model
WG5	2	Pu/U alloy	FERMI Core 1 and 2 standard fuel assembly fuel	U-<8 wt% Mo/water model
	7	U-metal	N-Reactor fuel	Instantaneous degradation model
	9	Al-based	Foreign Research Reactor fuel	Aluminum alloy model
	10	Miscellaneous UNF	Miscellaneous fuel	Instantaneous degradation model
	11	U-Zr-Hx	Training Research Isotopes—General Atomic fuel	UO_2–type degradation model
WG7	4	MOX	Fast Flux Test Facility (FFTF-DFA/TFA) oxide fuel	Commercial model
	6	U/Th oxide	Shippingport LWBR fuel	Ceramic model (Ringwood)
	8	U-oxide	Three Mile Island fuel Shippingport PWR fuel	UO_2–type degradation model
WG9	3	U/Pu carbide	Fast Flux Test Facility (FFTF-TFA-AC-3) carbide fuel	Instantaneous degradation model
	5	U/Th carbide	Fort St. Vrain fuel	Instantaneous degradation model
WG10	1	Naval fuel	Surface Ship/Submarine Assemblies	UO_2–type degradation model

As discussed above, those 11 TSPA DSNF groups resulted from successive down-selections of the initial 34 intermediate condensed DSNF groups in support of OCRWM's postclosure safety case into 16 groups for the TSPA (DOE-EM, 1998b), followed by a reduction to 13 groups for PA analyses (DOE-NSNFP, 2002). In addition to the aforementioned degradation models discussed for 11 TSPA DSNF groups, a dissolution model was used for each of the 16 groups for the TSPA to represent the fuel's radionuclide release rate to the repository's unsaturated zone and eventual transport to the receptor. Details of the rationale for the use of such dissolution models can be found in DOE-EM (1998b). The level of details regarding the dissolution models used for the DSNF of WG5, WG7, WG9 and WG10 tentative remapping into Groups 1-11 (Table 3-8). A second analysis of the degradation/dissolution of the DSNF in WG5, WG7, WG9 and WG10 can be achieved by mapping the waste forms in these groups to those 16 groups initially considered for TSPA.

A one-to-one correspondences exist between Groups 1, 3, 4, 6, 7, 10, 11 of Table 3-6 and their counterparts in the 16 initial TSPA partitioning. Two of the 16 groups considered have been eliminated

(i.e. “Canyon Stab.” and “Na-Bonded Fuel” because these would be processed into other waste forms). Four of the 16 groups have been consolidated (i.e. “U-Zr fuels” and “U-Mo fuels” have been merged into Group 2, and “U/Th carbide high-integrity” and “U/Th carbide low-integrity” have been included in Group 5). Some of the DSNF have been rearranged in the remaining groups. Those rearrangements resulted in Group 8 containing both “U oxide intact fuel” and “U oxide failed/decladed fuel” (also referred to as Group 8a and 8b, as shown in Table 3-7). Many of the changes were driven by the state or composition of the fuel cladding. As a result, in the context of PA (i.e. with zero credit given to the fuel cladding in terms of degradation), the mapping proposed above between the DSNF of WG5, WG7, WG9 and WG10 and Groups 1-11 in the TSPA-SR/LA of FY 1999 appears to contain a sufficient level of detail.

The various DSNF groupings proposed in support of performance assessment and disposal concepts have been reviewed and analyzed. While as a crude first approximation DSNF can utilize either UO_2–type UNF or instantaneous degradation models, it was shown that some of the recently introduced groupings from the WFDOE (SNL, 2014) can be mapped to a wider variety of degradation/dissolution models previously established for the 11 DSNF groups considered in the early work of the YM SAR. A finer remapping of into the original 16 groups considered is not expected to provide additional useful information in terms of degradation at the PA level, although future work may elucidate fuel degradation/dissolution models at the level of the 34 condensed DSNF groups.

3.3.3.2 Calcine Waste and Associated Degradation Behavior

Background on Calcine Waste

Spent nuclear fuel was reprocessed to recover enriched uranium and other radionuclides at the Idaho Nuclear Technology and Engineering Center (INTEC), located at INL in southeastern Idaho. Reprocessing operations ran from 1953 to 1994 and produced highly radioactive aqueous wastes that were temporarily stored in underground tanks. Fluidized-bed calcination was then used at INTEC to solidify the aqueous acidic metal nitrate radioactive wastes. In the calcination process, the liquid wastes are sprayed using air-atomizing nozzles into a fluidized bed of heated spherical calcine particles, evaporating water and nitric acid in the wastes, and leaving behind solid-phase metal oxides and fluorides known as calcine.

Calcination operations ran from 1963 to 2000 and produced approximately 4,400 m^3 of calcine that is stored in a total of 6 Calcine Solids Storage Facilities (CSSF). A CSSF consists of several stainless-steel storage bins that are housed within concrete vaults and are commonly referred to as “bin sets.” Each CSSF has between three and twelve bins containing the calcine (Staiger and Swenson 2011). Different fuel configurations and the use of different fuel-cladding materials led to the generation of several chemically distinct liquid wastes during reprocessing and consequently led to several different calcine compositions. For example, “aluminum” and “zirconium” wastes are so named because each was generated from the reprocessing of aluminum- and zirconium-clad fuels respectively. Sodium-bearing waste (SBW) is a term used to describe wastes that contain relatively high concentrations of sodium salts. The compositions of four primary types of calcine waste stored at INTEC are provided in Table 3-9.

Initially DOE intended to immobilize the calcine waste in a vitrified (glass) waste form before shipping it to a geologic repository. INTEC proposed to implement its vitrification program in 2020 and complete it in 2035 (DIRS 103497- INEEL 1998, pp. A-39 to A-42). For this reason, it was assigned the properties of HLW glass in terms of its dissolution rate in the Yucca Mountain TSPA. More recently, in the 2010 Record of Decision (ROD) 75 FR 137, DOE selected hot isostatic pressing as the technology to treat the calcine and create a new waste form that is suitable for disposal. The hot isostatic pressing process uses calcine retrieved from the CSSF and heat-treated at temperatures up to 600°C to remove moisture and NOx. After heating, the calcine is mixed with silica, titanium and calcium sulfate (or elemental sulfur), and the mixture is placed in a stainless steel can which is then sealed with a lid with a vent tube. The can is evacuated, the vent is sealed, and the can is placed in the hot isostatic pressing process vessel. The

vessel is pressurized with argon gas to between 7,200 and 15,000 psi and is heated to between 1,050°C and 1,200°C.

At these processing conditions, the calcine is converted to a glass ceramic consisting of a mixture of titanates, sulfides, glass/quartz, and nepheline (CDP, 2012). It is expected that this glass ceramic has properties consistent with HLW borosilicate glass. ROD 75 FR 137 also retains an option to hot isostatic press the calcine without the addition of the silica, titanium and calcium sulfate. It is expected that this would provide additional volume reduction of up to approximately 50%. However, this alternative calcine waste form would release RCRA waste constituents and therefore would require disposal at a facility that accepts RCRA wastes. Yet a third option under consideration is the direct disposal of calcine waste without additional treatment. Similar to the additive-free HIP calcine waste, it is expected that this waste form would release RCRA waste constituents and would require disposal at a facility that accepts RCRA wastes.

Table 3-9. Typical Compositions of the Four Types of Calcine

Element/ Chemical Species	Units	Type of Calcine			
		Aluminum[a]	Zirconium[a]	Fluorinel/SBW Blend[a]	Aluminum Nitrate/SBW Blend[a]
Al	wt%	47	8.1	7.5	38
B	wt%	0.1	1.0	1.0	0.1
Cd	wt%	—[b]	—	5.0	0.2
Ca	wt%	—	28	27	3.2
Cl	wt%	—	—	0.1	0.4
Cr	wt%	0.1	0.3	0.1	0.1
F	wt%	--	25	17	1.7
Fe	wt%	0.8	0.1	0.3	0.6
Hg	wt%	1.9	—	—	—
NO_3	wt%	2.5	0.8	6.0	5.9[c]
O	wt%	42	16	17	38
K	wt%	0.2	0.1	0.7	1.8[c]
Na	wt%	1.3	0.4	2.9	8.4[c]
SO_4	wt%	1.8	2.0	3.5	0.3
Sn	wt%	—	0.3	0.2	—
Zr	wt%	0.1	17	11	1.3

NOTES: [a] Column totals are not 100% because of rounding values and the exclusion of trace components. [b]..A dash within a cell indicates an insignificant quantity. [c] The aluminum nitrate/SBW blend nitrate value is a high-temperature (600°C) calcination value. Nitrate values were higher and alkali (sodium and potassium) values were lower when SBW was calcined at 500°C. SBW = sodium-bearing waste. Source: Staiger and Swenson (2011).

3.3.3.3 Degradation Model for Hot Isostatic Pressed (HIP) Calcine Waste with Additives

A literature survey revealed very little research has been done to establish the long-term dissolution rates of HIP calcine waste under repository conditions. However, Begg et al., 2005 studied HIP simulated zirconia calcine samples at various loadings of glass additives to create a set of simulated glass-ceramic waste materials that are intended to represent HIP zirconia calcine waste forms. The glass-ceramic samples were prepared with the simulated zirconia calcine at various loadings from 60 wt% to 90 wt% with proportionate amounts of glass additives. In addition, a densified zirconia calcine was prepared at 100% loading (no additives). These simulated waste forms were then subjected to the Product Consistency Test (PTC-B) (ASTM C 1285-95); a leach test designed to determine the chemical durability of nuclear waste glasses. The PTC-B test results show high chemical durability with waste loadings of up to 80% as indicated by the retention of numerous elements within the simulated waste forms including B, Na, Cs, Mo, Sr, Gd, Al, Ca, Cr, F, Fe, Mg, Si and Zr. Figure 3-29 shows that Na release rates are well below the environmental assessment (EA) glass release limit in samples where the simulated zirconia calcine loadings are below 80 wt %. It is important to note that the HIP and fully densified 100% zirconia calcine sample exceeds the EA glass release rate limit for Na.

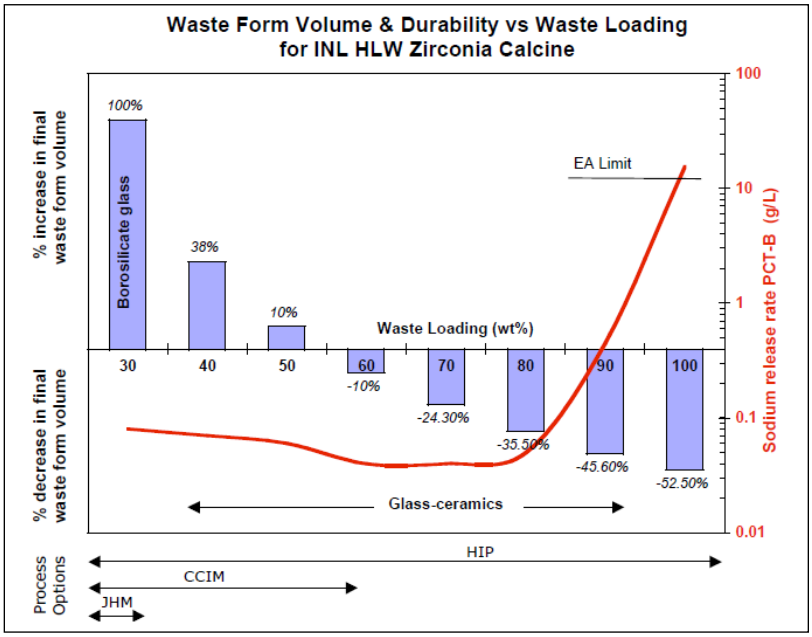


Figure 3-29. above from Begg et al., 2005 shows the relationship between various waste form alternatives including process options as a function of waste loading and chemical durability (PCT-B: Sodium release rate, g/L).

With the very limited amount of data available on calcine degradation it is difficult to assign a dissolution rate to HIP calcine waste. However, Knecht and Berreth, 1989 assert that the overall durability of the resulting glass ceramic is expected to be similar to a HLW glass. Further, the work by Begg et al. 2005 suggests HIP calcine waste with loading below 80 wt% may perform as well as the HLW glass waste. In the best case, the recommended glass dissolution rates above can be used to model the performance of HIP calcine waste. Conservatively, instantaneous dissolution may be assumed. The behavior of HIP calcine is very likely bounded by these two rates.

3.3.3.4 Degradation Model for HIP Calcine Waste without Additives

Once again, a literature survey revealed very little research has been done to establish the degradation rates and leachability of HIP calcine waste in the absence of glass additives. As noted above, Begg et al., (2005) showed that leach testing on fully densified 100% zirconia calcine sample exceeds the EA glass

release rate limit for Na. In the absence of long-term degradation rates under geologically relevant conditions, an instantaneous dissolution rate is recommended.

3.3.3.5 Degradation Model for Direct Disposal of Granular Calcine Waste

Available data on untreated granular calcine dissolution behavior, leachability and degradation rates is also limited and little has been done to examine long-term degradation rates under geologically relevant conditions. However, a comparison of the leach rates of glass waste forms to calcine waste indicate calcine leach rates range from 10^{-1} to 10^{-2} g/cm^2-day and are 4 to 6 orders of magnitude higher than glass leach rates (Stewart, 1985). In addition, several papers summarized below provide data on short-term (days to several weeks) leaching data in distilled water and dilute nitric acid.

Granular alumina calcine produced in the Waste Calcining Facility (WCF) at the INTEC was leached continuously in laboratory experiments with distilled water at 25°C and 80 to 90°C and in dilute (0.25 to 0.5M) nitric acid at 25°C (Paige, 1966). In this study, more than 95% of the Cs and 33% of the Sr was leached by distilled water at 25 °C from the alumina calcine in seven weeks; most of the leaching occurred during the first two or three days. Only 0.01% of the Al leached in a similar period, and the Ce and Ru were leached effectively at the same rate as the Al. During six weeks of leaching with dilute HNO_3 (0.25 to 0.5M) at 25 °C, the alumina calcine disintegrated, and more than 99% of the alumina dissolved.

More recently, Staples et al.1979, examined the leaching characteristics of both alumina and zirconia calcine wastes. They concluded that leaching characteristics of both alumina and zirconia calcines by distilled water are similar. Cesium and strontium were selectively leached at significant rates, although cesium leached much more completely from the alumina calcine than from the zirconia calcine. After 2,000 hours, about 95 percent of the cesium and 33 percent of the strontium leached from the alumina calcine. In this same time period nearly 60 percent of the cesium and 33 percent of the strontium leached from the zirconia calcine. Cesium and strontium are probably contained in both calcines as nitrate salts and also as fluoride salts in zirconia calcine, all of which are at least slightly soluble in water. Radionuclides of cerium, ruthenium, and plutonium in both calcines were much more resistant to leaching and leached at rates similar to or less than those of the matrix elements. For example, after 1,300 hours of continuous leaching, 0.1 percent of the total plutonium in the zirconia calcine had been removed and the rate of removal became extremely slow.

Chipman (1990) reported the leaching characteristics of Fluorinel/SBW calcines produced at INTEC. The samples tested included two non-radioactive pilot plant calcines as well as a radioactive Fluorinel-SBW calcine sample. The leaching methods employed were the Environmental Protection Agency's Extraction Procedure (EP) Toxicity Test and the Materials Characterization Center's (MCC) MCC-1 Static Leach Test at 25°C.

The MCC-1 leach test results on the non-radioactive pilot plant calcines show that total mass loss and component mass loss are affected by solution temperature, initial concentration of calcine in water, and time. Total mass loss increases rapidly and reaches a maximum after about 3 to 7-days and then decreases as some species that are initially leached into solution precipitate as time continues. In the test using the lowest initial concentration of calcine in water (0.001 g calcine/ml water) the total mass loss reached a maximum of about 45% after a 7-day period. Further, MCC-1 testing revealed that NO_3 and Cl were totally leached from the calcine within 1-day of water contact and about 90% of the Na and K leached from the calcine within 1-day of water contact at 25°C. Only a few tenths of a percent of Zr, Cd, and F species are leached from the calcine after 28-days. Partial re-precipitation of a phase containing Al, B, Ca, Cr, and SO_4 was also observed. Additionally, the EP Toxicity Tests on both pilot calcines showed that the limit of toxicity was exceeded by a factor of about 10 to 70 for Cr, and about 170 to 850 for Cd.

The MCC-1 static leach testing on the radioactive Fluorinel/SBW (4.7:1 blend of high-level waste and sodium-bearing liquid waste) revealed a similar behavior in total mass loss with respect to time as the

pilot-plant calcines. The total mass loss increases rapidly and reaches a maximum of about 50 wt% after one day. At intermediate times out to seven days, the total mass loss decreases and then slowly increases to about 45 wt% at the conclusion of the 28-day tests. Similar to the simulated calcines, leaching followed by re-precipitation of some components was also observed. Analysis of the leachate shows that about 93 wt% of the Cs, which accounts for about one-half of the total β activity for this age of calcine used, leaches from the calcine after one day. The quantity leached varies slightly for the remainder of the 28-day test. About 65 wt% of the Sr-90 leaches from the calcine after one day, and this quantity increases up to about 86 wt% after 28-days. Only a small amount of the α activity leaches (0.060% gross α) in the 28-day test.

In summary, the leach studies on alumina calcine, zirconia calcine and Fluorinel/SBW summarized above all indicate the rapid and substantial leaching of soluble species such as Cs, Tc and Sr in distilled water at 25°C while actinides including Pu, Am, and Cm are leached at slower rates. Meanwhile, leach studies on alumina calcine in nitric acid (0.25 to 0.5M) revealed nearly all of the alumina dissolved into solution. EP toxicity tests on Fluorinel/SBW, exceed the limit of toxicity for the RCRA metals Cr and Cd. Based upon the studies summarized above and the absence of long-term degradation rates under geologically relevant conditions, an instantaneous dissolution rate is recommended.

4. Summary

This report provides an update to the Sassani et al. (2016) and includes

(1) an updated set of inputs (Sections 2.3) on various additional waste forms (WF) covering both DOE-managed spent nuclear fuel (SNF) and DOE-managed (as) high-level waste (HLW) for use in the inventory represented in the geologic disposal safety analyses (GDSA);

(2) summaries of evaluations initiated to refine specific characteristics of particular WF for future use (Section 2.4);

(3) updated development status of the Online Waste Library (OWL) database (Section 3.1.2) and an updated user guide to OWL (Section 3.1.3); and

(4) status updates (Section 3.2) for the OWL inventory content, data entry checking process, and external OWL BETA testing initiated in fiscal year 2017.

As such, this report represents completion of milestone deliverable M2SF-17SN010501014 "Inventory and Waste Characterization Status Report" (SFWD-SFWST-2017-000014), as the final report on FY2017 activities for the work packages SF-17SN01050101 and SF-17SN01050102. Note that content included from Sassani et al. (2016) in unchanged form is summarized below these updates summaries for inclusiveness.

Based on the Sassani et al. (2016) recommendations, the primary update to the preliminary DGRDMSH inventory is to add the additional possible DGRDMSH waste forms (DOE, 2014) that were not previously included in GDSA representations, for which GDSA evaluation of thermal or radionuclide inventory aspects may be somewhat expanded compared to the previous analyses. Specifically, this entails the following:

- Adding the 340 Hanford Cs/Sr vitrified glass canisters (as detailed in Wilson, 2016, Table 2-6)
- Adding the 34 glass canisters of Hanford Federal Republic of Germany (FRG) glass, which is material that has been managed as HLW (SNL, 2014), and may be disposed in a DGRDMSH,
- Adding the planned waste form for calcine hot isostatically pressed (HIP) into HIP cans that are loaded/stacked into ~320 canisters (~5.5 ft diameter by ~15 ft height, naval canisters/waste packages containing ~10 HIP cans each; SNL 2014).
- Adding the naval SNF waste packages from the coolest thermal range (~13 naval SNF canisters using the ~1000W per canister thermal threshold for the upper bound—see Figure 3 of DOE, 2014; and SNL, 2014 naval waste package thermal binning listed in Appendix A, p,. A-40),

Although most of these updates are relatively small from the standpoint of inventory mass, they may have some implications for analyses of thermal effects. This may be the case because some of these added wastes tend to have higher average thermal loads per canister than the inventory previously evaluated in GDSA. Additionally, some of these additions represent larger waste packages that may expand handling and emplacement considerations (i.e., naval SNF and planned calcine HIP waste form waste packages).

During FY2017, a number of questions regarding the characteristics of various waste forms led to three ongoing studies on WF characteristics details (Section 2.4). First, in our estimates of HLW glass compositions for postclosure safety analyses, we assume that all the ^{129}I in tank waste becomes part of the vitrified waste form. However, it is not clear if this quantitative assumption is correct, as the ^{129}I activity in the glass waste form is not high enough to warrant direct analysis. Given that the Savannah River Site (SRS) has produced thousands of HLW glass logs, we initiated a study of the detailed documentation for the SRS vitrification process to see if it was possible to trace/quantify the potential sinks for ^{129}I in the various processes to form the HLW glass.

In addition to these uncertainties for SRS HLW glass logs, it was also noted that the inventory for the Cs/Sr capsules did not give the quantity of ^{135}Cs contained in the capsules. Nor did the reported inventory of Cs and Sr for the (Federal Republic of Germany – FRG) glass at Hanford provide the quantity of ^{135}Cs contained in those glass logs (SNL, 2014). Because quantities of these two long-lived fission products (half-life of ^{129}I is 1.57×10^7 years, half-life of ^{135}Cs is 2.3×10^6 years) were not readily available, we developed estimated quantities of both radionuclides in Savannah River glass (Section 2.4.1) and the (FRG) glass at Hanford (Section 2.4.2).

We began a third study to define characteristic isotopic ratios for various waste forms included in postclosure performance studies (Section 2.4.3). This aspect arose due to questions regarding the relative contributions of radionuclides from disparate waste forms in DGRDMSH GDSA results, particularly, radionuclide contributions of DOE-managed SNF vs HLW glass. Depending on the design of the generic repository evaluated, it may be easy to assess such contributions proximal to the source terms if the waste forms are segregated. However, given the complexity of some geologic systems, isotopic ratios (two or more) that effectively tag their source waste form distinctly would facilitate such assessments at distal points. Using such ratios to define mixing lines may allow quantitative estimates of relative WF contributions to be "mined" from GDSA results, as long as the particular isotopes are tracked.

Throughout FY2017, the OWL database activities have focused in three areas (Section 3.2). First, additional data for waste types (and their potential waste forms) and source documentation have been added to the OWL to flesh out its content covering DOE-managed (as) HLW and SNF (Section 3.2.1). In conjunction with further data entry, a process of checking the data entry into the OWL against the source documentation was launched to search for and rectify any errors in data entry (Section 3.2.2). This checking was performed by technical individuals independent of the data entry process, who documented any issues noted, and resolved the issues with the data entry staff. As the OWL was modified throughout the year in terms of its interface and features, another process to assess the usability of the OWL was recently kicked-off. This process is referred to here as the External OWL BETA test (Section 3.2.3) and involves technical staff from within the DOE (both NE and EM), as well as at other National Laboratories, using the OWL and providing feedback on its utility and content. Preliminary feedback is summarized herein, with feedback to be continued into the first quarter of FY2018. Each of these three OWL update activities is ongoing into FY2018.

The online waste library (OWL) has been designed to contain information regarding DOE-managed (as) high-level waste (HLW), spent nuclear fuel (SNF), and other wastes that are likely candidates for deep geologic disposal, with links to the current supporting documents for the data (when possible; note no classified or OUO data are planned to be included at this point). There may be up to several hundred different DOE-managed wastes that are likely to require deep geologic disposal. The DOE has a database (Spent Fuel Database-SFDB) that contains information regarding the SNF that DOE manages. We do not intend to replicate this database and the information in it, but intend to take advantage of that existing dataset to incorporate it efficiently into the on-line waste library for use in post-closure PA. A status of the OWL database is provided with updates on the OWL content (greatly expanded with additional data for wastes beyond the Cs/Sr capsule waste and its two alternate waste forms contained in the prototype). Both the OWL database model (Appendix B) and a user's guide to the OWL (Section 3.1.3) are provided.

Starting in FY2018, future work on the OWL database includes the following:

- Continue to add the full set of information regarding the other wastes from the WFDOE (SNL, 2014)(i.e., fully populate the OWL for previously identified waste types and waste form pathways)
    - The focus for FY2018 will be coordinating/synchronizing with the DOE SNF database at INL to leverage that dataset for purposes of GDSA assessment purposes

- Continue to refine the set of documentation for the OWL database architecture, including a comprehensive user's guide (see Section 3.1.3 for OWL user's guide)
- Continue the review and verification process to ensure information in the OWL is accurate and sourced correctly, including
    - Continuation/completion of the external BETA testing
    - It is intended in FY2018 to develop an external review process for the content of OWL
- Define an update processes (this will be done in conjunction with user review and feedback on the OWL) to
    - maintain current information linked to new or revised DOE documents
    - delineate additional features/capabilities to add to the OWL
    - add new waste types and waste forms as they are identified

The activities in the first bullet above are a priority for FY2018 activities, as is completing the external BETA test for the OWL. The fourth bullet above represents the path for maintaining and expanding the utility of the OWL in the future. The OWL is intended to facilitate coherent analyses regarding the back end of the fuel cycle with respect to the full range of DGRDMSH wastes and waste forms.

Summary of Content Included but Unchanged from FY2016

Sassani et al. (2016) provided the other content of this report including (1) developing a preliminary DGRDMSH included inventory for engineering/design/safety analyses (updated with additions herein as described above); (2) assessing the major differences of this included inventory relative to that in other analyzed repository systems and the potential impacts to disposal concepts (unchanged); (3) designing and developing the prototype on-line waste library (OWL) to manage the information of all those wastes and their waste forms (updated as discussed above); and (4) constraining post-closure waste form degradation performance for safety assessments of a DGRDMSH (unchanged). In addition, Sassani et al. (2016) reported on identifying potential candidate waste types/forms to be added to the full list from the WFDOE (SNL, 2014 – see Table C-1), which also may be added to the OWL in the future (unchanged). The summaries of these included original contents are given here for convenience.

DGRDMSH Included Preliminary Inventory for GDSA

Wilson (2016) provides the preliminary inventory for the analyses of a DGRDMSH for FY2016 and includes both DHLW and DSNF waste canister counts and thermal information (Tables 2-1, and 2-3 thru 2-6 from Wilson, 2016). The Wilson (2016) report describes each waste form in terms of both average radionuclide content and overage thermal output evolution. The tabulation includes canister counts and ranges of thermal characteristics for each DHLW and DSNF waste form considered (Wilson, 2016). For the preliminary DGRDMSH inventory used in this report, the various types of DSNF are listed in Appendix A, which are contained in the ~2485 DSNF canisters (see Table 2-1 from Wilson, 2016). The included DHLW canister counts are given in Wilson (2016) in Tables 2-3 thru 2-6, respectively, for Savannah River glass (7824 canisters), Hanford glass (11,800 canisters), INL hot isostatic pressed (HIP) calcine (4391 canisters), and Hanford vitrified Cs/Sr capsules (340 canisters- see SNL, 2014 also).

Disposal Concepts Information Evaluation

A low-temperature DGRDMSH would differ in the following primary aspects compared to a repository including CSNF:

- A DGRDMSH would be smaller than a 70,000-MTHM CSNF repository due to the smaller waste volume.

- A DGRDMSH would contain a higher percentage of short-lived fission products than a CSNF repository. This alters the timing of peak repository temperatures and of transient temperature-dependent processes including resaturation; buoyancy driven fluid flow; waste package degradation; waste form dissolution; buffer and host rock alteration; and creep consolidation.
- A DGRDMSH would experience a thermal load on the order of 3% of the thermal load in a 70,000 MTHM CSNF repository, allowing for smaller distances between drifts and waste packages. This would reduce issues regarding temperature-dependent processes including, for instance, waste package degradation and buffer and host rock alteration.
- A DGRDMSH may present unique challenges related to the chemical and physical characteristics of some waste forms. The effects of corrosive waste, highly soluble waste, and colloid-forming waste on repository performance should be considered. The presence of RCRA-regulated waste in some alternate waste form pathways may need to be considered.
- A DGRDMSH packaging plan results in a bimodal distribution of waste package sizes. Large waste packages may create engineering challenges in some disposal concepts.

Waste Form Performance Constraints

The models for degradation of both UO_2 and HLW glass given above (Section 3.3) are currently being used within the GDSA for PA modeling of post-closure system evolution. The waste forms in the current DGRDMSH analyses have been mapped into those models as either performing similarly or being bounded by a particular model degradation behavior. For example, the HIP calcine waste form is assigned to degrade as the HLW glass degradation. For waste forms that do not have substantial waste form lifetimes (generally ~10,000 years or less), the instantaneous degradation rate is used. Note that in all cases the waste form degradation is the initial, kinetic step, and the dissolved radionuclides are evaluated against solubility limits based in part on the geologic environment.

The current assignments for degradation rates of the DSNF in the DGRDMSH inventory are based on the work in the YM SAR (DOE, 2008), which assigned virtually all the DSNF to the instantaneous degradation rate model except for the naval SNF. This was based primarily on the small amounts the other than naval DSNF represented in the YM SAR relative to the mass of CSNF. Because the DSNF represents a fractionally larger portion of the radionuclides in the DGRDMSH, we have reviewed the bases for the PA groupings from the YM SAR and some prior analyses to see if there may be some of the DSNF waste forms that have a basis for better performance in post-closure (Section 3.3.3.1). In addition, the assumption of glass degradation being assigned to the HIP calcine waste form was evaluated as well. These provide input to potential adjustments to the GDSA models, if appropriate.

The various DSNF groupings proposed in support of performance assessment and disposal concepts have been reviewed and analyzed. While as a crude first approximation DSNF can utilize either UO_2–type UNF or instantaneous degradation models, it was shown that some of the recently introduced groupings from the WFDOE (SNL, 2014) can be mapped to a wider variety of degradation/dissolution models previously established for the 11 DSNF groups considered in the early work of the YM SAR. A finer remapping of into the original 16 groups considered is not expected to provide additional useful information in terms of degradation at the PA level, although future work may elucidate fuel degradation/dissolution models at the level of the 34 condensed DSNF groups.

Studies of the degradation performance of HIP calcine (with additives) provide information that allows assigning glass degradation rates to the glass ceramic calcine waste form as a reasonable approach. The use of instantaneous degradation rates for the HIP calcine waste form would represent a conservative bounding approach. For untreated calcine, or calcine HIP without additives, instantaneous degradation rates should be used in PA analyses.

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APPENDICES

Appendix A. Included Inventory of DOE-managed SNF for Defense Repository Analyses

Included Defense Repository DSNF Inventory[a]	
DSNF Group	Included Inventory Item
DOE Fuel Group 01	HWCTR RMT & SMT
MTHM = 2096	HWCTR TWNT
Containers = 388 (388.41)	HWCTR ETWO
	N REACTOR
DOE Fuel Group 02	HWCTR IMT
MTHM = 7.65	SINGLE PASS REACTOR FUEL
Containers = 4 (4.15)	MISCELLANEOUS RSWF FUEL
DOE Fuel Group 03	CP-5 CONVERTER CYLINDERS
MTHM = 6.71	EBWR ENRICHED HEAVY
Containers = 18 (18.05)	HWCTR DRIVER
	HWCTR SPR
	HWCTR TFEN
	EBWR ENRICHED THIN
	EBWR ET-11
	EBWR NORMAL HEAVY
	EBWR NORMAL THIN
	HWCTR IS
DOE Fuel Group 04	HWCTR 3EMT-2
MTHM = 0.0105	SPEC (ORME)
Containers = 1 (1.16)	
DOE Fuel Group 05	TREAT DRIVER
MTHM = 0.0533	VBWR
Containers = 18 (18.34)	EBWR (SPIKES)
	BR-3
DOE Fuel Group 06	EBWR PURE 6% UO2
MTHM = 1.90	PULSTAR - SUNY BUFFALO (CANNED)
Containers = 7 (6.93)	BR-3 FUEL
	SAXTON
DOE Fuel Group 07	EBWR PURE NORMAL
MTHM = 31.30	HWCTR SPRO

Included Defense Repository DSNF Inventory[a]

DSNF Group	Included Inventory Item
Containers = 69 (68.77)	HWCTR SOT
	LOFT CENTER FUEL MODULE (A1,A2,A3,F1)
	LOFT CORNER FUEL MODULE
	LOFT SQUARE FUEL MODULE
	PULSTAR-N.C. STATE UNIV.
	PULSTAR - SUNY BUFFALO (ASSEMBLIES)
	HWCTR OT
	SURRY
	DRCT (TN-24P)
	DRCT (VSC-17)
	HWCTR SPRO
	N.S. SAVANNAH
	DRESDEN I (E00161)
	HWCTR IRO
	CANDU
	SURRY (T11 RODS)
DOE Fuel Group 08	APPR (AGE-2)
MTHM = 0.14	BORAX V (SUPERHEATER)
Containers = 9 (8.76)	ML-1 (GCRE)
	GCRE (1B SERIES)
	GCRE (1Z SERIES)
DOE Fuel Group 09	PBF DRIVER CORE
MTHM = 0.69	ACRR (PULSED CORE)
Containers = 12 (11.93)	SAXTON
DOE Fuel Group 10	FFTF-TFA-ABA-1 THRU 6
MTHM = 0.44	FFTF-TFA-WBO18 & WBO42
Containers = 2 (1.66)	HWCTR SPRO
DOE Fuel Group 11	BMI (CPI-38)
MTHM = 0.701	GCRE CAN (1B-8T 1&2)
Containers = 195 (194.94)	GCRE PELLETS (1B-7T-1)
	GETR FILTERS
	HTRE (ANP)
	SM-1A
	SPSS (SPERT)
	TORY-IIA
	TORY-IIC
	VBWR (GENEVA)

Included Defense Repository DSNF Inventory[a]

DSNF Group	Included Inventory Item
	FRR TARGET (ARGENTINA)
	ANP
	FRR TARGET (CANADA)
	FRR TARGET (INDONESIA)
	EBWR (FUEL FOLLOWER)
	BMI (CPI-24)
DOE Fuel Group 12	SPERT-III
MTHM = 0.156	PNL MIXED MATERIAL EXP.DCC-1
Containers = 5 (5.36)	PNL MIXED MATERIAL EXP.DCC-2
	PNL MIXED MATERIAL EXP.DCC-3
	SP-100 FUEL
	LOFT CENTER FUEL MODULE FP-2 REMAINS
DOE Fuel Group 13	LOOSE FUEL ROD STORAGE BASKET (LFRSB)
MTHM = 82.21	HANFORD COMMERCIAL TEST SCRAP
Containers = 361 (361.42)	HANFORD LWR SCRAP
	H. B. ROBINSON RODS
	TMI-2 CORE DEBRIS
	LOFT FUEL RODS
	LWR SNF SCRAP
	SURRY (T11 SCRAP RODS)
DOE Fuel Group 14	BSR
MTHM = 1.84	HFBR
Containers = 208 (208.19)	HFIR (INNER)
	NIST
	OMEGA WEST (204)
	OMEGA WEST (236)
	OMEGA WEST (250)
	ORR
	HFBR
	HFIR (OUTER)
	NIST
	ORR
	ORR
	HFBR
DOE Fuel Group 15	ORR SPECIAL
MTHM = 0.3315	RSG-GAS (INDONESIA)
Containers = 9 (8.43)	FRR MTR-C (PERU)

Included Defense Repository DSNF Inventory[a]	
DSNF Group	Included Inventory Item
	FRR MTR-S (PERU)
	SAR-GRAZ (AUSTRIA)
	FRG-1 (GERMANY)
	FRR FRJ (GERMANY)
	FRJ (GERMANY)
DOE fuel Group 16	ANLJ
MTHM = 7.35	ARMF (PLATES)
Containers = 626 (625.62)	ARMF/CFRMF MARK I
	ARMF/CFRMF MARK I LL
	ARMF/CFRMF MARK II
	ARMF/CFRMF MARK III
	ADVANCED TEST REACTOR (ATR)
	ADVANCED TEST REACTOR (ATR)
	ATSR
	BNL MEDICAL RX (BMRR)
	GTRR
	GENTR
	JMTR 93% ENRICHED (JAPAN)
	MIT
	MIT
	MURR (COLUMBIA)
	MURR (COLUMBIA)
	MURR (COLUMBIA)
	OHIO STATE
	PURDUE UNIVERSITY
	RHF (FRANCE)
	RINSC
	UNIV OF FLORIDA (ARGONAUT)
	UNIV OF MASS-LOWELL
	UNIV OF VIRGINIA
	UNIV OF WASHINGTON
	FRR MTR (CANADA)
	SLOWPOKE (CANADA)
	GRR (GREECE)
	SAPHIR (SWITZERLAND)
	JRR-4 (JAPAN)
	FRR MTR (JAPAN)
	ASTRA (AUSTRIA)
	ENEA SALUGGIA (ITALY)
	FMRB (GERMANY)
	FRR MTR-C (GERMANY)

Included Defense Repository DSNF Inventory[a]

DSNF Group	Included Inventory Item
	FRR MTR-S (GERMANY)
	FRR MTR-S (GERMANY)
	FRR MTR-S (GERMANY)
	IAN-R1 (COLUMBIA)
	KUR (JAPAN)
	FRR MTR (JAPAN)
	FRR MTR (JAPAN)
	JRR-2 (JAPAN)
	FRR MTR (NETHERLANDS)
	HFR PETTEN HEU (NETHERLANDS)
	FRR MCMASTER MNR HEU MTR-C (CANADA)
	MCMASTER MNR/PTR UALX HEU (CANADA)
	FRR MTR (ZPRL, TAIWAN)
	THOR (TAIWAN)
	FRR MTR-C (PORTUGAL)
	FRR MTR-S (PORTUGAL)
	TRR-1 (THAILAND)
	RA-6 (ARGENTINA)
	RA-3 (ARGENTINA)
	PRR-1 (PHILIPPIINES)
	FRR MTR-C (ISRAEL)
	FRR MTR-O (TURKEY)
	FRR MTR-C (TURKEY)
	FRR MTR-S (TR-2, TURKEY)
	ASTRA (AUSTRIA)
	FRR MTR-S (ISRAEL)
	MOATA ARGONAUT (AUSTRALIA)
	FRR PIN CLUSTER (CANADA)
	FRR PIN CLUSTER (CANADA)
	FRR PIN CLUSTER (CANADA)
	FRR SLOWPOKE (CANADA)
	FRR SLOWPOKE (CANADA)
	FRR SLOWPOKE (MONTREAL)
	FRR SLOWPOKE (CANADA)
	FRR SLOWPOKE (CANADA)
	HIFAR (AUSTRALIA)
	FRR FRJ (GERMANY)
	FRR TUBES (AUSTRALIA)
	FRR FRJ (GERMANY)
	RECH-1 80% (CHILE)
	HOR (NETHERLANDS)
	DR-3 (DENMARK)

Included Defense Repository DSNF Inventory[a]

DSNF Group	Included Inventory Item
	FRR MCMASTER MNR HEU MTR-S (CANADA)
	FRG-1 (GERMANY)
	BER-II [HMI] (GERMANY)
	ESSOR (ITALY)
	IOWA ST. UNIV.
	JEN-1 (SPAIN)
	R-2 SVTR (SWEDEN)
	FRM (GERMANY)
	FRM (GERMANY)
	ADVANCED TEST REACTOR (ATR)
	UMRR (ROLLA)
	JRR-2 (JAPAN)
	JMTR 45% ENRICHED (JAPAN)
	FRJ (GERMANY)
	MURR (COLUMBIA)
	FRJ (GERMANY)
DOE Fuel Group 17	UNIV OF MICHIGAN
MTHM = 1.997	WORCESTER POLY INSTITUTE
Containers = 69 (69.38)	FRR TUBES (AUSTRALIA)
	R-2 SVTR (SWEDEN)
	FRR MTR-C (PORTUGAL)
	FRR MTR-O (PORTUGAL)
	FRR MTR-S (PORTUGAL)
	IEA-R1 (BRAZIL)
	FRR MTR (ARGENTINA)
	FRR MTR (TTR-1, JAPAN)
	FRR MTR-C JRR-3M (JAPAN)
	FRR MTR-S JRR-3M (JAPAN)
	ZPRL (TAIWAN)
	FRR MTR (THAR, TAIWAN)
	RU-1 (URAGUAY)
	PRR-1 (PHILLIPPINES)
	JEN-1 (SPAIN)
	ENEA SALUGGIA (ITALY)
	RV-1 (VENEZUELA)
	RPI (PORTUGAL)
	JRR-3M (JAPAN)
DOE Fuel Group 18	UMRR (ROLLA)
MTHM = 6.15	OHIO STATE
Containers = 215 (215.27)	ORR

Included Defense Repository DSNF Inventory[a]

DSNF Group	Included Inventory Item
	PURDUE UNIVERSITY
	RINSC
	UNIV OF MASS-LOWELL
	FRR MTR-C KUR (JAPAN)
	FRR PIN CLUSTER (SO. KOREA)
	SAPHIR (SWITZERLAND)
	FRR MTR-S KUR (JAPAN)
	JMTR (JAPAN)
	FRR MTR-S (JAPAN)
	FRR MCMASTER MNR LEU MTR-C (CANADA)
	FRR MCMASTER MNR LEU MTR-S (CANADA)
	FRR BER II [HMI] MTR-C (GERMANY)
	FRR BER II [HMI] MTR-S (GERMANY)
	FRR MTR-C2 (TURKEY)
	FRR MTR-S (TURKEY)
	FRR PIN CLUSTER (SO. KOREA)
	FRR PIN CLUSTER (CANADA)
	ASTRA (AUSTRIA)
	FRG-1 (GERMANY)
	NEREIDE (FRANCE)
	DR-3 (DENMARK)
	ORR
	R-2 SVTR (SWEDEN)
	ORR
	SAPHIR (SWITZERLAND)
	UNIV OF VIRGINIA
	IOWA ST. UNIV.
DOE Fuel Group 19	GA HTGR FUEL
MTHM = 0.0184	HTGR (PEACH BOTTOM SCRAP)
Containers = 3 (2.62)	
DOE Fuel Group 21	EBR-II, FFTF & MTR EXPERIMENTS
MTHM = 0.0765	FFTF-TFA-FC-1
Containers = 5 (5.14)	FFTF CARBIDE FUEL EXPER. (AC-3)
	FFTF-TFA-ACN-1 RODS
	FAST REACTOR FUEL
	FFTF-TFA PINS (AC-3)
DOE Fuel Group 22	EBWR PURE MOX
MTHM = 1.218	GE TEST

Included Defense Repository DSNF Inventory[a]

DSNF Group	Included Inventory Item
Containers = 5 (5.43)	SAXTON
DOE Fuel Group 23	BABCOCK & WILCOX SCRAP
MTHM = 10.65	EPRI
Containers = 139 (138.84)	FFTF-DFA/TDFA
	LWR SAMPLES (MOX)
	ORR-BW-1
	FFTF-TFA-AB-1
	FFTF-TFA PINS
	FFTF-TFA-ACN-1 PINS
	FFTF-TFA-CRBR-3 & CRBR-5
	FFTF-DFA/TDFA PINS
	FFTF-TFA-DEA-2
	FFTF-TFA-ACO-2, 4 THRU 16
	FFTF-TFA-MFF-1 & 1A (CDE)
	FFTF-TFA-P0-2,4 & 5
	FFTF-TFA-SRF-3&4
	FFTF-TFA-UO-1
	EBR-II OXIDE FUEL EXPER
	FFTF OXIDE EXPERIMENTS (FO-2 & ACO-3)
	SODIUM LOOP SAFETY FAC.
	US/UK FUEL PINS
	EBR-II OXIDE FUEL EXPER
	SODIUM LOOP SAFETY FAC.
	PNL MOX FUEL
	PNL MOX FUEL (7010)
	PNL MOX FUEL (7055)
	PNL-3
	PNL MOX STAR 7
	PNL MOX STAR 3
	PNL MOX STAR 4
	PNL MOX STAR 5
	PNL MOX STAR 6
	EBR-II & TREAT EXPERIMENTS
	SAXTON
DOE Fuel Group 24	MOX SCRAP SNF
MTHM = 0.1096	MISCELLANEOUS TREAT FUEL
Containers = 1 (1.45)	PNL MOX FUEL (7057)
	PNL MOX PELLETS (7057)
	PNL MOX PINS (7057)

Included Defense Repository DSNF Inventory[a]

DSNF Group	Included Inventory Item
DOE Fuel Group 26	ERR
MTHM = 5.04	FAST REACTOR FUEL
Containers = 11 (10.63)	
DOE Fuel Group 27	BER-II TRIGA (GERMANY)
MTHM = 0.153	TRIGA FLIP (TAMU)
Containers = 17 (17.38)	TRIGA HEU (OSU)
	TRIGA FLIP (TAMU)
	TRIGA FLIP (UNIV OF WISCONSIN)
	TRIGA FLIP HEU (WSU)
	TRIGA HIGH POWER HEU (ROMANIA)
	TRIGA FFCR SST (NRF AT HANFORD)
	TRIGA FLIP
	TRIGA FLIP (AUSTRIA)
	TRIGA FLIP (MEXICO)
	TRIGA FLIP (SO. KOREA)
	TRIGA FLIP (LJUBLJANA, SLOVENIA)
	TRIGA HEU FFCR (OSU)
	TRIGA FLIP (GA)
	TRIGA FLIP FFCR (SO. KOREA)
	TRIGA FLIP (DAMAGED) (SO. KOREA)
	TRIGA FLIP (NRAD)
	TRIGA HIGH POWER HEU (ROMANIA)
	TRIGA FLIP-HEU FFCR (GA)
	TRIGA HEU TEST STD OR IFE (GA)
	TRIGA HEU (IFE) (OSU)
DOE Fuel Group 28	TRIGA STD (U OF AZ)
MTHM = 1.053	GA RERTR
Containers = 60 (59.50)	TRIGA SST (OSU AT HANFORD)
	TRIGA STD
	TRIGA STD SST (GA)
	TRIGA SST (CORNELL UNIV.)
	TRIGA STD
	TRIGA STD
	TRIGA SST STD/IFE (GA)
	TRIGA STD
	TRIGA STD
	TRIGA STD SST CLUSTER RODS (TAMU)
	TRIGA STD

Included Defense Repository DSNF Inventory[a]

DSNF Group	Included Inventory Item
	TRIGA STD
	TRIGA STD
	TRIGA STD SST (UNIV OF TEXAS)
	TRIGA STD (WSU)
	TRIGA STD (GERMANY)
	TRIGA SS (NRF AT HANFORD)
	TRIGA STD
	TRIGA STD
	TRIGA FFCR (UNIV OF ILLINOIS)
	TRIGA STD SST (UNIV OF ILLINOIS)
	TRIGA STD (AUSTRIA)
	TRIGA FLIP (BANGLADESH)
	TRIGA STD (FINLAND)
	TRIGA STD (HANNOVER)
	TRIGA STD (GERMANY)
	TRIGA SST 8.5% (BANDUNG INDONESIA)
	TRIGA SST RC-1 (ROME, ITALY)
	TRIGA STD SST (MUSASHI, JAPAN)
	TRIGA ACPR (JAPAN)
	TRIGA STD (MEXICO)
	TRIGA STD (SO. KOREA)
	TRIGA STD (ENGLAND)
	TRIGA STD (ZAIRE)
	TRIGA SST (LJUBLJANA, SLOVENIA)
	TRIGA STD (THAILAND)
	TRIGA STD (TURKEY)
	TRIGA FLIP (THAILAND)
	TRIGA FLIP (MALAYSIA)
	TRIGA FLIP (TAIWAN)
	TRIGA FFCR (MNRC)
	TRIGA STD (MNRC)
	TRIGA FFCR RC-1 (ROME, ITALY)
	TRIGA FFCR (SO. KOREA)
	TRIGA FFCR (ZAIRE)
	TRIGA FFCR (MNRC)
	TRIGA STD (REED COLLEGE)
	TRIGA STD (ARRR)
	TRIGA FFCR (PENN. STATE UNIV.)
	TRIGA STD (MSU)
	TRIGA SST (UC BERKLEY)
	TRIGA STD (ACPR)
	TRIGA SST IFE RC-1 (ROME, ITALY)

Included Defense Repository DSNF Inventory[a]

DSNF Group	Included Inventory Item
	TRIGA ACPR (LJUBLJANA, SLOVENIA)
	TRIGA FFCR (LJUBLJANA, SLOVENIA)
	TRIGA STD (USGS)
	TRIGA FFCR (AFRRI)
	TRIGA (DEMOUNTABLE) (U OF AZ)
	TRIGA STD (IFE) (U OF AZ)
	TRIGA STD (IFE) (U OF AZ)
	TRIGA FFCR (U OF AZ)
	TRIGA FFCR (ENGLAND)
	TRIGA SST 20/30 (GA)
	TRIGA ACPR PENN. STATE UNIV.
	TRIGA LEU FFCR (GA)
	TRIGA STD FFCR (OSU)
	TRIGA STD (IFE) (OSU)
	TRIGA STD (IFE) (ENGLAND)
	TRIGA STD (HEIDELBERG)
	TRIGA FFCR (HEIDELBERG)
	TRIGA FFCR (UC-IRVINE)
	TRIGA STD (IFE) (UC-IRVINE)
	TRIGA STD (MNRC)
	TRIGA FFCR (MNRC)
DOE Fuel Group 29	TRIGA STD
MTHM = 0.325	TRIGA STD
Containers = 16 (16.22)	TRIGA STD
	TRIGA STD (HANNOVER)
	TRIGA AL (NRF AT HANFORD)
	TRIGA STD AL (UNIV OF ILLINOIS)
	TRIGA STD (AUSTRIA)
	TRIGA STD (FINLAND)
	TRIGA STD (HEIDELBERG)
	TRIGA STD (GERMANY)
	TRIGA AL RC-1 (ROME ITALY)
	TRIGA AL (LJUBLJANA, SLOVENIA)
	TRIGA STD (BRAZIL)
	TRIGA AL (RIKKYO UNIV. JAPAN)
	TRIGA STD (SO. KOREA)
	TRIGA STD (ZAIRE)
	TRIGA STD (U OF UTAH)
	TRIGA AL STD OR IFE (GA)
	TRIGA STD (KSU)
	TRIGA STD AL (GA)

Included Defense Repository DSNF Inventory[a]	
DSNF Group	Included Inventory Item
	TRIGA STD (KSU)
	TRIGA STD (HANFORD)
	TRIGA STD AL (UNIV OF TEXAS)
	TRIGA STD (MSU)
	TRIGA STD (DOW)
DOE Fuel Group 30	SNAP
MTHM = 0.0298	
Containers = 6 (6.15)	
DOE Fuel Group 32	NAVAL (S1W1)
MTHM = 0.00018	
Containers = 0	
DOE Fuel Group 34	DOE TEST (EBR-II, FFTF, LWR)
MTHM = 0.416	HWCTR TMT-1-2 & 1-3
Containers = 5 (5.06)	TRIGA AL (CORNELL UNIV.)
	EBR-II NITRIDE FUEL EXPER
	MIXED PLUTONIUM & URANIUM TEST
	TRU SCRAP SNF
	MISCELLANEOUS TREAT FUEL

a. From Wilson (2016) with red text indicating DOE-managed SNF that may be reconsidered for inclusion/exclusion in future work on a Defense Waste Repository (DGRDMSH). Note that there are materials for which final disposition for a DGRDMSH has not been made. This included inventory is being used for preliminary technical analyses of both thermal design aspects, and postclosure safety assessments and any final inventory for a DGRDMSH would need to be directed by the US DOE. Note also that this includes no naval SNF package, but using a thermal cutoff of ~ 1000 W/canister, a number of naval SNF packages (<~15) would be included in this inventory also.

Appendix B. OWL Prototype Database Model Details

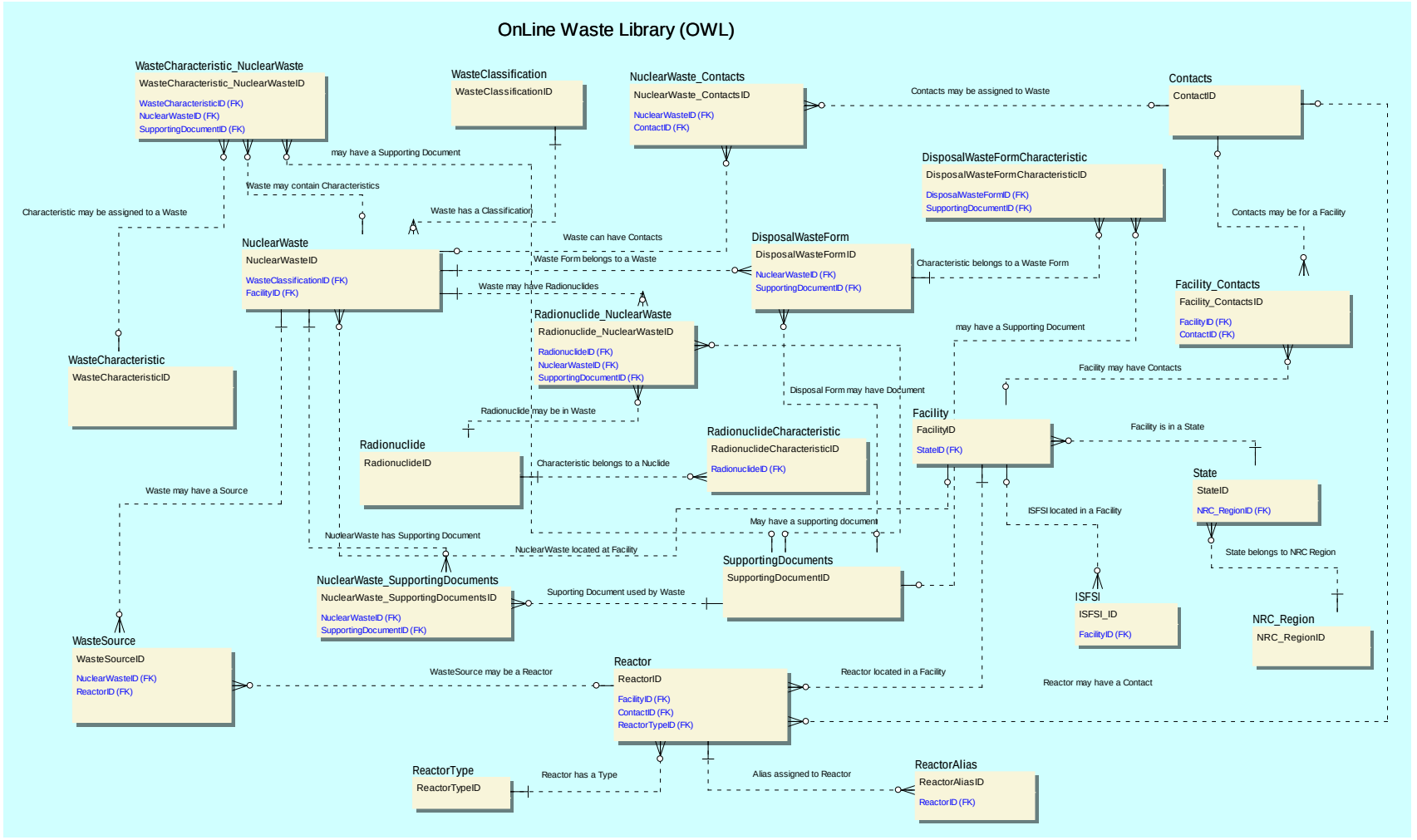
Information Model

OWL (Online Waste Library)
Version 8.2

Sandia National Labs
Walter Walkow
Last Modified 7/5/2016

Table of Contents

Main Model Image



Entity Detail Reports

Contacts

Primary Keys ContactID
Definition Provides information about Contacts that may be assigned to Nuclear Waste, Facilities, etc..

Attributes

Attribute/Logical Rolename	Domain	Datatype	Null	Definition
ContactID		INTEGER	N	Uniquely identifies a Contact. It is an integer that begins with 1 and is incremented on each new addition.
ContactName		VARCHAR(100)	N	The name of the Waste Type
PhoneNumber		VARCHAR(20)	Y	Optionally provides a phone number in the format (area code) - xxx-xxxx
Comments		VARCHAR(4000)	Y	
Status		CHAR(10)	N	

4

DisposalWasteForm

Primary Keys DisposalWasteFormID
Definition Provides information about the form that waste will be used for Disposal

Attributes

Attribute/Logical Rolename	Domain	Datatype	Null	Definition
DisposalWasteFormID		INTEGER	ID	Unique ID of the Disposal Waste Form
DisposalWasteForm		VARCHAR(100)	N	The name of the Disposal Waste Form
FormDescription		VARCHAR(4000)	N	Provides a textual description of the Disposal Waste Form
NuclearWasteID		INTEGER	N	This is the ID of the Nuclear Waste is the basis of the Disposal Waste Form.
PlannedOrExisting		VARCHAR(50)	Y	State of the Disposal Waste Form: Planned or Existing. Valid values are 'Planned' or 'Existing'
PreferredOrAlt		VARCHAR(50)	Y	Preference of Disposal Waste Form: Preferred or Alternative
UnitOfMeasure		VARCHAR(100)	Y	Unit of Measure in describing the nature of the Disposal Waste Form
UnitOfMeasureValue		VARCHAR(100)	Y	Value for the Unit of Measure that describes the nature of the waste in the form used in disposal
Status		VARCHAR(10)	Y	Staus of the data: 'Active' or 'Inacative'.

SupportingDocumentD	INTEGER	Y	Default is Active Uniquely identifies a document, Assigned by the System, beginning with 1 and incremented by 1

DisposalWasteFormCharacteristic

Primary Keys DisposalWasteFormCharacteristicID
Definition Provides information about specific Waste Characteristics

Attributes

Attribute/Logical Rolename	Domain	Datatype	Null	Definition
DisposalWasteFormCharacteristicID		INTEGER	ID	Unique ID of the DIsposal Waste Form Characteristic
DisposalWasteFormID		INTEGER	N	ID of the Disposal Waste Form that the characteristic is associated.
FormCharacteristic		VARCHAR(200)	N	The name of the Waste Characteristic
CharacteristicDescription		VARCHAR(4000)	Y	Provides a textual description of the waste characteristic
UnitOfMeasure		VARCHAR(100)	Y	
UnitOfMeasureValue		VARCHAR(100)	Y	
SupportingDocumentID		INTEGER	Y	ID of a Supporting Document if it exists.
Status		VARCHAR(10)	N	

Facility

Primary Keys FacilityID
Definition Provides information about the facilities where Waste is stored or may be the source of the Waste

Attributes

Attribute/Logical Rolename	Domain	Datatype	Null	Definition
FacilityID		INTEGER	N	
FacilityName		VARCHAR(200)	N	The name of the Location
lattitude_d		DECIMAL(20, 12)	Y	
longitude_d		DECIMAL(20, 12)	Y	
StateID		INTEGER	N	ID of the state code.
Facility_Abbr		VARCHAR(200)	Y	
Comments		VARCHAR(4000)	Y	Provides a textual description of the waste item
City		VARCHAR(50)	Y	
Status		VARCHAR(10)	Y	Status of the Waste. Default is 'Active'. Possible values are 'Active' and 'Inactive'
IsFederalGovt		VARCHAR(10)	Y	

Facility_Contacts

Primary Keys Facility_ContactsID
Definition Identifies the Contacts for a specfied Facility

Attributes

Attribute/Logical Rolename	Domain	Datatype	Null	Definition
Facility_ContactsID		INTEGER	ID	Uniquely identifies a Contact for a Facility. This is an integer assigned by the System, beginning with a 1 and incremented by 1.
FacilityID		INTEGER	N	The ID of the Facility to which the Contact is assigned
ContactID		INTEGER	Y	ID of the Contact assigned to the Facility
Description		VARCHAR(4000)	Y	Provides a textual description of the Contact for the Facility

ISFSI

Primary Keys ISFSI_ID
Definition Independent Spent Fuel Storage Installation (ISFSI) - The ISFSI Must be licensed by the NRC in accordance with 10CFR2. This table lists the facilities that provide the storage facility for spent nuclear fuel.

Attributes

Attribute/Logical Rolename	Domain	Datatype	Null	Definition
ISFSI_ID		INTEGER	ID	Uniquely identifies an ISFSI. This is an integer assigned by the System, beginning with a 1 and incremented by 1.
ISFSI		VARCHAR(100)	N	Name of the ISFSI. EX: Diablo Canyon
FacilityID		INTEGER	Y	The ID of the Facility in which the ISFSI is located.
EIA_Nbr		VARCHAR(50)	Y	EIA (U.S. Energy Information Administration) Number of the ISFSI. EX: 3501D for Diablo Canyon

NRC_Region

Primary Keys NRC_RegionID
Definition Stores information about the NRC Regions. NRC Regions are assigned to States

Attributes

Attribute/Logical Rolename	Domain	Datatype	Null	Definition
NRC_RegionID		INTEGER	ID	Uniquely identifies each NRC Region with an integer that begins with 1 and is incremented by 1
NRC_Region		VARCHAR(200)	N	The name of the NRC Region..
Comments		VARCHAR(2000)	N	Comments

NuclearWaste

Primary Keys NuclearWasteID
Definition Provides generral information about specific Nuclear Wastes.

Attributes

Attribute/Logical Rolename	Domain	Datatype	Null	Definition
NuclearWasteID		INTEGER	ID	This is the nuclear waste item. It contains the basic information about the Nuclear waste. More specific details are found in the related entities.
WasteType		VARCHAR(100)	N	The Nuclear Waste type
WasteClassificationID		INTEGER	N	Uniquely identifies a waste classification. It is an integer that begins with 1 and is incremented on each new addition.
WasteDescription		VARCHAR(2000)	Y	Provides a textual description of the waste item
ProducedBy		VARCHAR(50)	Y	Is it Government produced or Commerically produced. There is no default
IsMixedWaste		VARCHAR(10)	Y	Is it mixed waste? Possilble values are 'Yes', 'No'., or 'N/A'. Default is 'N/A'
Status		VARCHAR(10)	Y	Status of the Waste. Default is 'Active'. Possible values are 'Active' and 'Inactive'

WasteBaselineInventoryDate	DATE	Y	The date of the baseline activity inventory for the Waste. The default is January 1, 2016
FacilityID	INTEGER	Y	ID of the Facility where the Waste is located

NuclearWaste_Contacts

Primary Keys NuclearWaste_ContactsID
Definition Identifies the Contacts for a specified Nuclear Waste

Attributes

Attribute/Logical Rolename	Domain	Datatype	Null	Definition
NuclearWaste_ContactsID		INTEGER	ID	Uniquely identifies a Contact for a specified Nuclear Waste
NuclearWasteID		INTEGER	Y	Provides the ID of a Responsible Contact for the specified Nuclear Waste
ContactID		INTEGER	Y	Identifies the Nuclear Waste for which the Responsible contact is assigned.
Description		VARCHAR(2000)	Y	Provides a textual description of the waste item

NuclearWaste_SupportingDocuments

Primary Keys NuclearWaste_SupportingDocumentsID
Definition Identifies where a Supporting Document is Used and describes the usage. Each entry identifies the SupportingDocumentID, an entity where it is used, and the ID of the entry in the entity. Example: SupportingDocument ID = 1, entity = NuclearWaste, entityID = 1.

Attributes

Attribute/Logical Rolename	Domain	Datatype	Null	Definition
NuclearWaste_SupportingDocumentsID		INTEGER	ID	Uniquely identifies the linkage of a supporting document to a Nuclear Waste. This is an integer created by the system, begining with a 1 and incremented by 1
NuclearWasteID		INTEGER	N	ID of the Nuclear Waste that is linked to a supporting document
SupportingDocumentID		INTEGER	N	ID of the Supporting Document
DocumentUsageDescription		VARCHAR(2000)	Y	Breif description of the document's content.

Radionuclide

Primary Keys RadionuclideID
Definition Provides information about Radionuclides that may be the nature of Nuclear Waste.

Attributes

Attribute/Logical Rolename	Domain	Datatype	Null	Definition
RadionuclideID		INTEGER	ID	Unique ID of the Radio Nuclide
Radionuclide		VARCHAR(100)	N	The name of the Radionuclide. Example Cs 135
RadionuclideDescription		VARCHAR(4000)	Y	Provides a textual description of the Radionuclide
Status		VARCHAR(10)	N	Status of the data. Default is 'Active'. Other value is 'InActive'.
HalfLife		FLOATN	Y	Half Life in years
AtomicMass		DECIMAL(10, 2)	Y	Atomic Mass in grams
ThermalOutput		DECIMAL(10, 2)	Y	Thermal output in watts/kilocurie

Radionuclide_NuclearWaste

Primary Keys Radionuclide_NuclearWasteID
Definition Associates the Radionuclides to specific Nuclear Wastes and provides the inventory of the radionuclides in the Waste.

Attributes

Attribute/Logical Rolename	Domain	Datatype	Null	Definition
Radionuclide_NuclearWasteID		INTEGER	ID	Unique ID of the Radionuclide_Nuclear Waste record. Assigned by the system beginning with 1 incremented by 1
RadionuclideID		INTEGER	N	Unique ID of the Radio Nuclide
NuclearWasteID		INTEGER	N	ID of the Nuclear Wastefrom which the Nuclide originates
InventoryUnitofMeasure		VARCHAR(50)	Y	
InventoryValue		INTEGER	Y	
InventoryDescription		VARCHAR(500)	Y	
SupportingDocumentID		INTEGER	Y	ID of the Supporting Document

RadionuclideCharacteristic

Primary Keys RadionuclideCharacteristicID
Definition Provides information about specific Radiio Nuclides that may be the nature of Nuclear Waste.

Attributes

Attribute/Logical Rolename	Domain	Datatype	Null	Definition
RadionuclideCharacteristicID		INTEGER	ID	Unique id that identifies this data. It is an integer assigned by the system, beginning with 0 and incremented by 1.
RadionuclideID		INTEGER	N	Unique ID of the Radio Nuclide which has the Radio Nuclide characteristic
UnitOfMeasure		VARCHAR(100)	Y	Unit of Measure for the Radio Nuclide Characteristic. EXAMPLE year for a Characteristic of Half-life
UnitOfMeasureValue		VARCHAR(100)	Y	Value for the unit of measure of the Radio Nuclide Characteristic. EX: 2,300,000 may be value for unit of measure: Years
Radionuclide_CharacteristicDescription		VARCHAR(200)	Y	Description of the characteristic that applies to the identified Radionuclide
Status		VARCHAR(10)	N	

Reactor

Primary Keys ReactorID
Definition Provides generral information about Reactors, including Location

Attributes

Attribute/Logical Rolename	Domain	Datatype	Null	Definition
ReactorID		INTEGER	ID	Provides information about the Reactor source of the waste
FacilityID		INTEGER	N	
ReactorName		VARCHAR(100)	N	The name of the Nuclear Waste
NRC_ReactorName		VARCHAR(200)	Y	NRC name for the Reactor
EIA_Number		VARCHAR(200)	Y	The ID of the Reactor as assigned by the U.S. Energy Information Agency (EIA).
CoreSize		VARCHAR(10)	Y	Description of the core size
ThermalCapacityMWTh		INTEGER	Y	Thermal Capacity Mega Watt Therrmal
ElelectricCapacityMWe		INTEGER	Y	Electric capacity in Mega Watts of electricity
ContactID		INTEGER	Y	ID of the primary Contact
Comments		VARCHAR(4000)	Y	Provides a textual description of the waste item
Status		VARCHAR(10)	Y	Status of the Waste. Default is 'Active'. Possible values are 'Active' and 'Inactive'
ReactorTypeID		INTEGER	N	Uniquely identifies each reactor type with

	an integer that begins with 1 and is incremented by 1

ReactorAlias

Primary Keys ReactorAliasID
Definition Identifies all the reactor aliases for a reactor

Attributes

Attribute/Logical Rolename	Domain	Datatype	Null	Definition
ReactorAliasID		INTEGER	ID	Uniquely identifies each reactor alias with an integer that begins with 1 and is incremented by 1
ReactorID		INTEGER	N	The Reactor ID of the reactor
AliasName		VARCHAR(200)	N	Alias name for the Reactor
Comments		VARCHAR(4000)	Y	Provides a textual description of the waste item

ReactorType

Primary Keys ReactorTypeID
Definition Identifies the type of reactor. EX: (PWR) Pressurized Water Reactor, (BWR) Boiling Water Reactor

Attributes

Attribute/Logical Rolename	Domain	Datatype	Null	Definition
ReactorTypeID		INTEGER	ID	Uniquely identifies each reactor type with an integer that begins with 1 and is incremented by 1
ReactorType		VARCHAR(100)	N	The name of the Reactor Type: Boiling Water Reactor, etc...
Comments		VARCHAR(4000)	Y	Provides a textual description of the waste item
Status		VARCHAR(10)	Y	Status of the Waste. Default is 'Active'. Possible values are 'Active' and 'Inactive'

State

Primary Keys StateID
Definition Stores information about US states, used by Facility table. Also provides the NRC region for the state

Attributes

Attribute/Logical Rolename	Domain	Datatype	Null	Definition
StateID		INTEGER	ID	Uniquely identifies each State with an integer that begins with 1 and is incremented by 1
StateName		VARCHAR(200)	N	The name of the Reactor Type: Boiling Water Reactor, etc...
StateAbbreviation		VARCHAR(3)	N	Provides the state abbreviation
NRC_RegionID		INTEGER	N	ID of the NRC Region that is assigned to the State

SupportingDocuments

Primary Keys SupportingDocumentID
Definition Provides information about the Supporting Documents that may be used by the various information tables. Includes information about the title, author, publisher, etc.

Attributes

Attribute/Logical Rolename	Domain	Datatype	Null	Definition
SupportingDocumentID		INTEGER	ID	Uniquely identifies a document, Assigned by the System, beginning with 1 and incremented by 1
Title		VARCHAR(200)	N	Title of the Document
Author		VARCHAR(100)	Y	Author or the document. May be a one or more individuals or an organization
Publisher		VARCHAR(100)	Y	Publishing company or organization
PublishDate		VARCHAR(100)	Y	
DocumentDescription		VARCHAR(4000)	Y	Breif description of the document's content.
URL_Address		VARCHAR(300)	Y	URL of the location of the document. Document may be internal of external.
DocumentAvailability		VARCHAR(50)	Y	Identifies whether the document is located within OWL (Internall Full Document or Internal Summary) or outside of OWL, on internet, etc... (External), ot Not Available.

CopyrightRestrictions	VARCHAR(200)	Y	Description of any copyright restrictions
Comments	VARCHAR(2000)	Y	Provides a textual description of the waste item
Status	VARCHAR(10)	Y	Status of the Waste. Default is 'Active'. Possible values are 'Active' and 'Inactive'

WasteCharacteristic

Primary Keys WasteCharacteristicID
Definition Provides information about Waste Characteristics that can be associated with one or more Nuclear Wastes

Attributes

Attribute/Logical Rolename	Domain	Datatype	Null	Definition
WasteCharacteristicID		INTEGER	ID	Uniquely identifies a Waste Characteristic by a number, beginning with 1 and incremented by 1.
WasteCharacteristic		VARCHAR(100)	N	The name of the Waste Characteristic
UnitOfMeasure		VARCHAR(100)	Y	Unit of Measure for the Waste Characeristic. Example 'Inches' for a diameter characteristic
Status		VARCHAR(10)	N	Status of the data: Active or Inactive. Default is Active

WasteCharacteristic_NuclearWaste

Primary Keys WasteCharacteristic_NuclearWasteID
Definition Links Waste Characteristics to Nuclear Wastes and provides a Value for the Nuclear Waste to the waste characteristic unit of measure

Attributes

Attribute/Logical Rolename	Domain	Datatype	Null	Definition
WasteCharacteristic_NuclearWasteID		INTEGER	ID	Uniquely identifies a Waste Characteristic by a number, beginning with 1 and incremented by 1.
WasteCharacteristicID		INTEGER	Y	The name of the Waste Characteristic
NuclearWasteID		INTEGER	Y	ID of the Nuclear Waste that the characteristic describes
NuclearWasteCharacteristicDescription		VARCHAR(2000)	N	Provides a textual description of the waste characteristic
UnitOfMeasureValue		VARCHAR(100)	Y	Value for the Waste Characteristic (based on Unit of Measure). EX: 2.15 for the Unit of Measure 'inches'
SupportingDocumentID		INTEGER	Y	ID of the Supporting Document, if it exists

WasteClassification

Primary Keys WasteClassificationID
Definition Provides information about Waste Types

Attributes

Attribute/Logical Rolename	Domain	Datatype	Null	Definition
WasteClassificationID		INTEGER	ID	Uniquely identifies a waste classification. It is an integer that begins with 1 and is incremented on each new addition.
WasteClassification		VARCHAR(100)	N	The name of the Waste Classification. EX: High Level Waste, Spent Nuclear Fuel
Description		VARCHAR(2000)	Y	Provides a textual description of the waste type
Status		VARCHAR(10)	Y	

WasteSource

Primary Keys WasteSourceID
Definition Provides information about the source of the Waste (which reactores, etc..)

Attributes

Attribute/Logical Rolename	Domain	Datatype	Null	Definition
WasteSourceID		INTEGER	ID	Uniquely identifies the source for the speacific Nuclear Waste (NuclearWasteID). If the source is a reactor, the ID of the Reactor is provided
NuclearWasteID		INTEGER	N	This is the nuclear waste item. It contains the basic information about the Nuclear waste. More specific details are found in the related entities.
ReactorID		INTEGER	Y	ID of the Reactor that is the source of the Nuclear Waste
Description		VARCHAR(4000)	N	Provides comments about the source of the waste
Status		VARCHAR(10)	Y	