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The On-line Waste Library (OWL): Usage and Inventory Status Report

Fuel Cycle Research & Development

Prepared for
U.S. Department of Energy
Used Fuel Disposition Campaign
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APPENDIX E

FCT DOCUMENT COVER SHEET [1]

Name/Title of Deliverable/Milestone/Revision No. -- The On-line Waste Library (OWL): Usage and Inventory Status Report/ M2FT-16SN080501042/ Rev = initial (00)

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This deliverable was prepared in accordance with Sandia National Laboratories
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☒ DOE Order 414.1 ☐ NQA-1-2000 ☐ Other

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Technical Review (TR)

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Robert Moore (SNL)

☐ Peer Review

Peer Review (PR)

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CONTENTS

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ACRONYMS and ABBREVIATIONS

BWR	boiling water reactor
DOE	U.S. 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
HEU	highly enriched uranium
HIP	hot isostatic pressing
HLW	high-level radioactive waste
INL	Idaho National Laboratory
INTEC	Idaho Nuclear Technology and Engineering Center
LEU	low-enriched uranium
LLW	low-level waste
MCO	multicanister overpack
MEU	medium enriched uranium
MOX	mixed oxide (fuel)
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
SRS	Savannah River Site
U.S.	United States
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

USED FUEL DISPOSITION CAMPAIGN/DISPOSAL RESEARCH –DOE-MANAGED HLW AND SNF RESEARCH

1. INTRODUCTION

The U.S. DOE evaluated the need for disposing of only DOE-Managed high-level-waste (HLW, which includes DOE-managed spent nuclear fuel [SNF] in this report) in a geologic repository by using the Nuclear Waste Policy Act 1982 (NWPA, as amended) and stated (DOE, 2015) "A geologic repository for permanent disposal of Defense HLW could be sited, licensed, constructed, and operated more quickly than a Common NWPA Repository and would provide valuable experience to reduce the cost of a future repository and the time needed to develop it." Based in part on that report, the President of the United States issued a memorandum stating that such a repository was required (see website with content at https://www.whitehouse.gov/the-press-office/2015/03/24/presidential-memorandum-disposal-defense-high-level-radioactive-waste-se). This furthered the recommendation from DOE (2014) to "…begin implementation of a phased, adaptive, and consent-based strategy with development of a separate mined repository for some DOE-managed HLW and cooler DOE-managed SNF, potentially including some portion of the inventory of naval SNF." The work in this report summarizes the status of activities within the Used Fuel Disposition Campaign (UFDC) to begin assessing the information needs for evaluating the safety of such a repository, considering both the waste forms being disposed and the repository concepts being considered. This report is milestone M2FT-16SN080501042 performed under the UFDC work package FT-16SN08050104 to provide a status of the progress made towards this objective.

1.1 Purpose and Scope

The Waste Form Disposal Options Evaluation Report (SNL 2014) evaluated 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 Used Fuel Disposition Campaign. That work covered a comprehensive inventory and a wide range of disposal concepts. The primary goal of this work is to evaluate the information needs for analyzing disposal solely of a subset of those wastes in a Defense Repository (DRep; i.e., those wastes that are either defense related, or managed by DOE but are not commercial in origin). A potential DRep also appears to be safe in the range of geologic mined repository concepts but may have different concepts and features because of the very different inventory of waste that would be included.

The focus of this status report is to cover the progress made in FY16 toward: (1) developing a preliminary DRep included inventory for engineering/design analyses; (2) assessing the major differences of this included inventory relative to that in other analyzed repository systems and the potential impacts to disposal concepts; (3) designing and developing an on-line waste library (OWL) to manage the information of all those wastes and their waste forms (including CSNF if needed); and (4) constraining post-closure waste form degradation performance for safety assessments of a DRep. In addition, some continuing work is reported on identifying potential candidate waste types/forms to be added to the full list from SNL (2014 – see Table C-1) which also may be added to the OWL in the future. The status for each of these aspects is reported herein.

1.2 Background

The Waste Forms Disposal Options Evaluation report (SNL, 2014) provided part of the technical basis for the DOE (2014) assessment of disposal options. The SNL (2014) work provides the starting point for information consideration of a repository concept for only DOE-managed HLW and SNF disposal. Both the wastes and waste forms considered in that previous study, as well as summaries of the disposal concepts evaluated, are given below.

1.2.1 Waste Types and Waste Forms Considered

The scope of the waste in the Waste Forms Disposal Options Evaluation (SNL, 2014) includes all existing SNF from commercial, defense, and research reactors, and SNF from reasonably foreseeable operations of existing reactors (projected to 2048). That study also includes existing HLW (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, that study includes consideration of 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. That study acknowledges existing plans, commitments, and requirements where applicable, but evaluates options for disposal based primarily on technical, rather than programmatic or regulatory constraints.

The SNL (2014) waste inventory was classified 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 disposed of as at least one, and possibly more than one, waste form in a deep geologic disposal concept (e.g., Hanford tank wastes; commercial spent fuels, HLW glass). A "waste form" is the end-state material as packaged that is to be disposed of in a deep geologic disposal concept. 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."

Considering the alternative treatment options for some of the 43 waste types, 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 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 in this study. The aggregation into waste groups allowed a high-level identification of any waste forms that may need to be considered as a separate group due to outstanding qualities in any one of these characteristics. Those same 10 groupings, lacking those solely containing CSNF (WG1 and WG2), are utilized below in this study to consider information needs regarding features of a D-Wastes Repository Concept.

Major assumptions and considerations used in SNL (2014) 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 all 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.

These assumptions apply also to the present work, which builds off the work done in SNL (2014) but focusses solely on the disposal of DOE-managed High-Level Waste and Spent Nuclear Fuel. As such, the CSNF aspects assessed previously are not included in this consideration. 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 initial work is assessing any needed additions to the D-Wastes to be added to the list compiled previously (see Tables C-1 and ES-1, SNL, 2014), layout the preliminary structure of the on-line waste library (OWL) to manage the waste types/forms information, and develop the information needs for delineating any needed changes to repository concept features relative to a concept that includes CSNF.

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

1.3 Disposal Concepts Considered

The Waste Forms Disposal Options Evaluation report (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 had 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, Sweden and Finland both have license applications 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 WIPP, and Germany has extensive experience with the design of a mined repository for SNF and HLW in salt. France, Switzerland, and Belgium have completed detailed safety assessments for proposed SNF and HLW repositories in clay and shale media. No nations are currently planning deep borehole repositories, but the concept has been evaluated in multiple programs since the 1970s, and remains viable for waste forms small enough for emplacement.

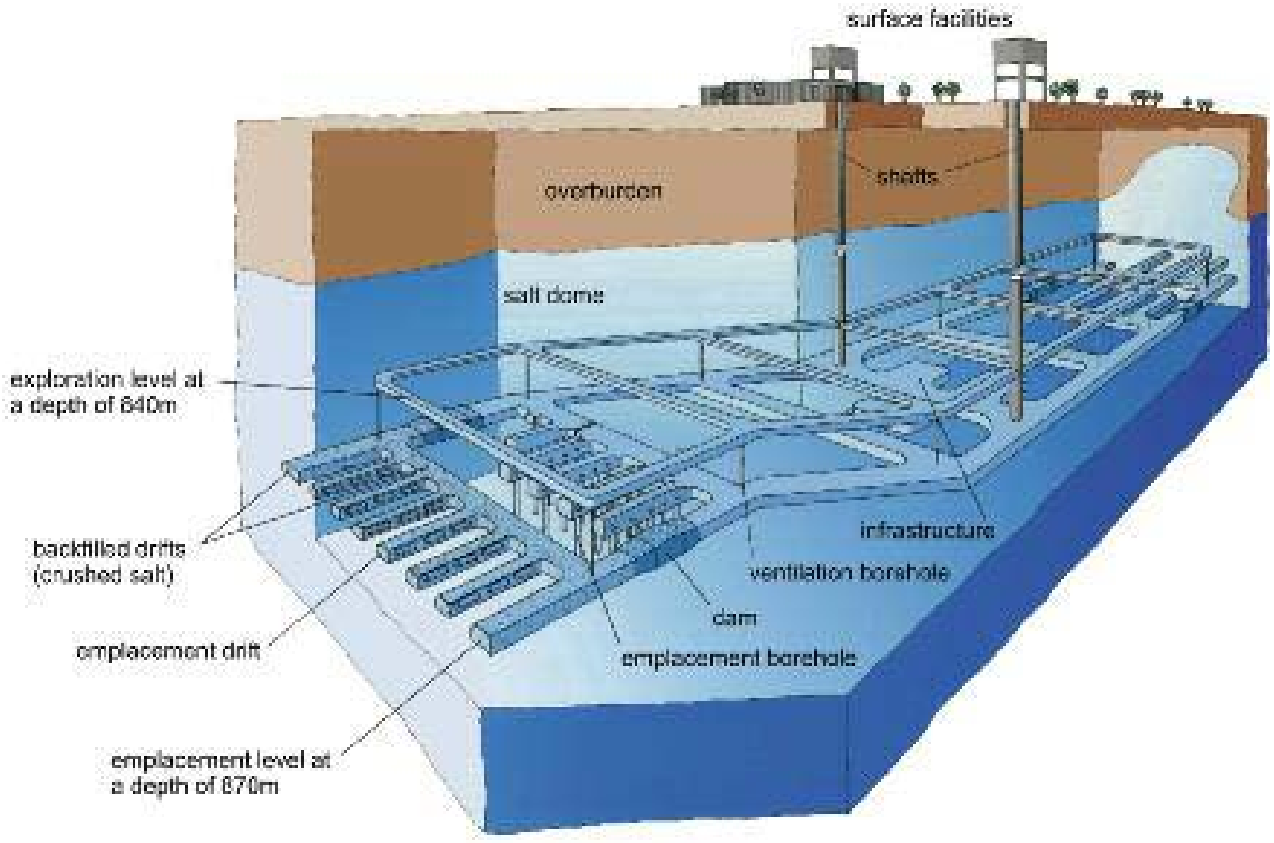
Variants of the four primary concepts are also considered where appropriate. For example, as described by Hardin et al. (2012a), some mined repository concepts can, in principle, be implemented in both open modes (i.e., with active ventilation during the operational period) and closed modes (i.e., with early emplacement of backfill), depending 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). Although these concepts have unique features that distinguish them from the four selected for consideration within UFDC, attributes of the four concepts discussed here are representative of a broad range of other disposal concepts.

Mined Repositories in Salt

The primary references for mined repositories in salt come from the U.S. WIPP program (DOE 1996b; DOE 2009) which is an operating repository disposing of defense-related transuranic waste, and the

proposed German repository at Gorleben (e.g., BMWi 2008). Figure 2-1 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 boreholes drilled from 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 a relatively high thermal conductivity that allows conductive transfer of heat away from the waste, and the plastic creep behavior of salt under pressure that causes it to flow, closing fractures and allowing for seal systems in access shafts that will compact under lithostatic loads to achieve extremely low permeabilities. Bedded salt, which occurs in horizontal layers of nearly pure sodium chloride originally deposited from shallow salt-saturated sea water, can contain both small quantities of trapped brine and interbedded layers of clays and other evaporite minerals such as anhydrite (calcium sulfate). Domal salt, which has moved from its original bedded form into dome-shaped structures due to plastic flow over geologic time, tends to have less water and fewer impurities and interbeds, but also occurs in more restricted geographic settings.



Source: BMWi 2008, Figure 15.

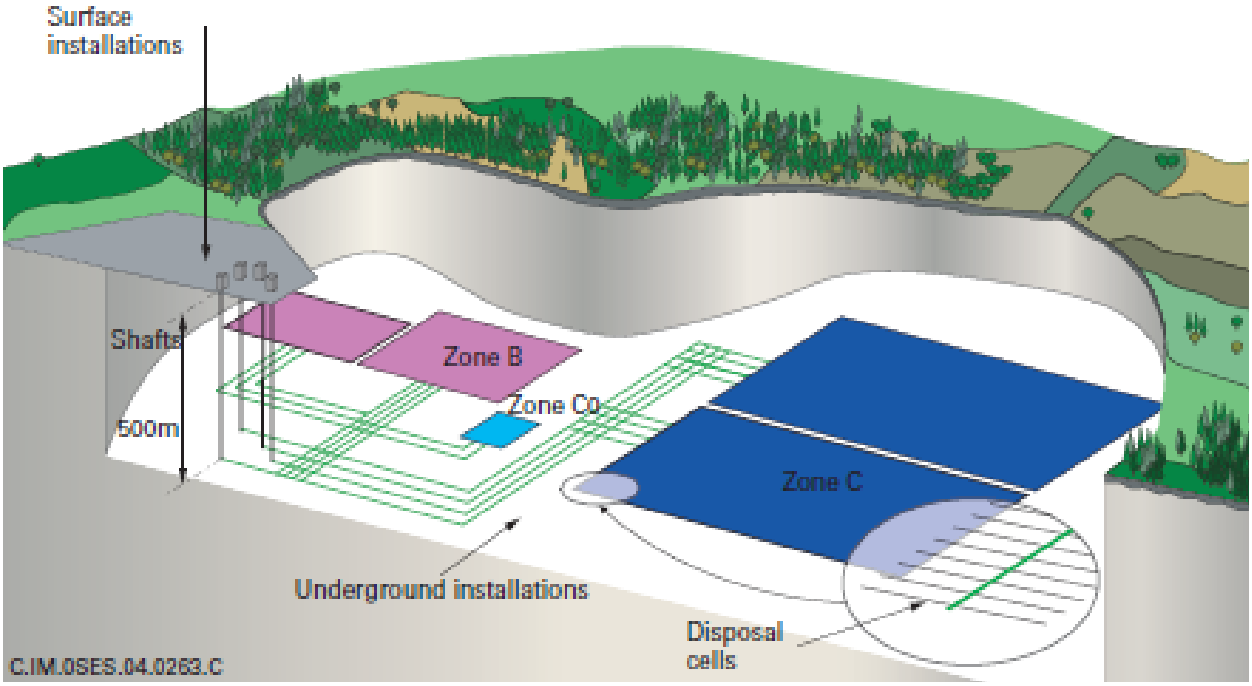
Figure 1-1. Schematic representation of a mined repository in salt

To the extent that sufficient water may be present to saturate the waste emplacement region in salt repositories, it will be a salt-saturated brine and chemical conditions will be reducing, with any free oxygen being consumed by corrosion of metal in the waste packages or other engineered systems. Salt creep will tend to close emplacement regions relatively rapidly (perhaps within decades) after waste emplacement, complicating the implementation of design concepts that call for extended periods of ventilation. However, the relatively high thermal conductivity of salt significantly reduces the need for ventilation to remove heat, compared to other potential media.

Because of the essentially impermeable nature of the host rock and very low potential for advective transport of radionuclides away from the disposal region, salt repository concepts place little or no reliance on the long-term performance of either the waste form or the waste packaging.

Mined Repositories in Clay and Shale Rocks

The primary references 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). Figure 2-2 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 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.

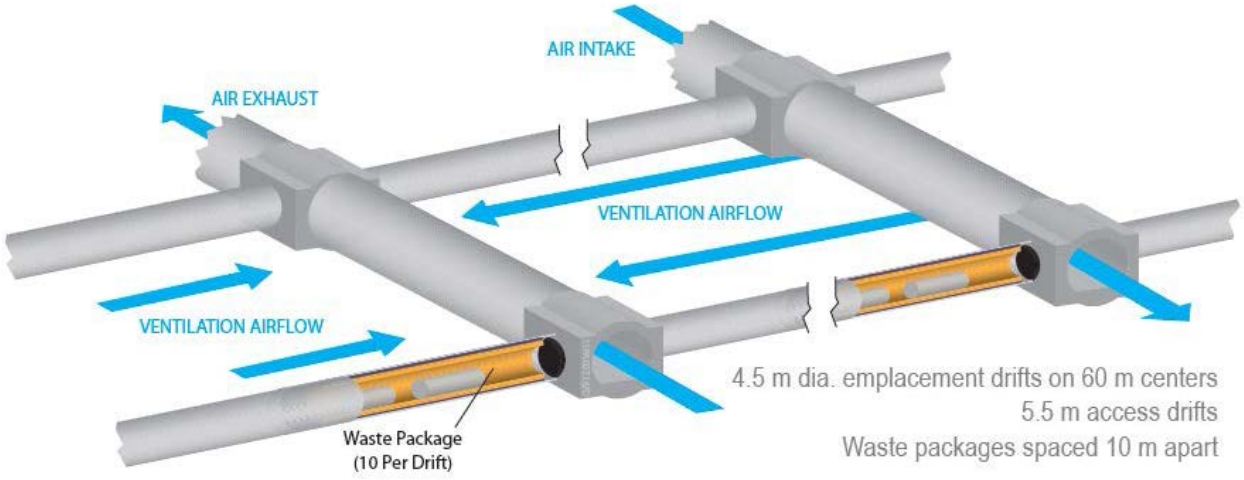


Source: ANDRA 2005b.

Figure 1-2. Schematic representation of a mined repository in argillaceous rock

Argillaceous rocks display a broad range of physical properties from weakly indurated clays capable of plastic flow (e.g., the formation being evaluated for disposal in Belgium), to laminated shales common in many sedimentary basins including in the U.S., to strongly indurated and massive argillites such as that being evaluated for disposal in France. All are characterized by extremely low permeability that will lead to diffusion-dominated release pathways and by an abundance of clay minerals that contribute to radionuclide sorption. All also have lower thermal conductivity than salt, and 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

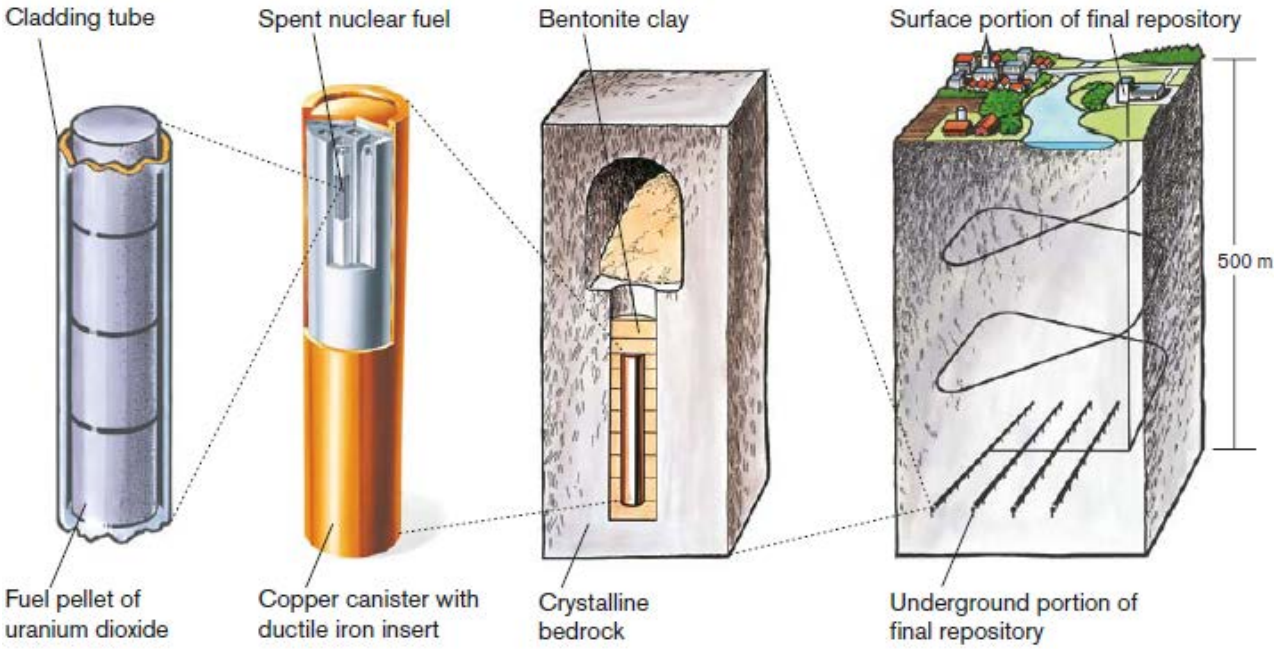
shale by considering an "open-emplacement" design concept in which emplacement drifts remain unbackfilled and open to allow extended ventilation to remove decay heat, as illustrated in Figure 2-3. Backfilling and sealing of access drifts occurs at the time of repository closure, with the option of leaving the emplacement drifts unbackfilled permanently if the operational constraints so dictate.



Source: Hardin et al. 2012a, Figure 1.5-3.
Figure 1-3. Schematic of shale unbackfilled open emplacement concept

Mined Repositories in Crystalline Rock

The primary references 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. Multiple other nations are also conducting research on mined repositories in crystalline rock, including Canada, Japan, Korea, China, and the Czech Republic. Figure 2-4 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. Isolation is provided by long-lived corrosion-resistant waste packages (copper in this case, which is thermodynamically stable in the chemically reducing environment), by the durability of the uranium oxide waste form (also stable in reducing conditions) disposed of in the Swedish repository concept, and by the high sorption capability of the bentonite clay buffer that surrounds the waste packages. 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 are breached, and design concepts therefore call for avoiding emplacement in regions intersected by fractures and for surrounding waste packages with a low-permeability bentonite clay buffer.



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 designs for waste packaging and will allow for more rapid degradation of UO_2 waste forms. The same would be true for other reduced waste forms, especially metallic waste forms, which would also have higher potential for pyrophoric 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.

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); subsequent analyses have considered disposal of solid wastes of various types, including glass HLW and surplus weapons-grade plutonium. 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, in part because of the availability of proven mining technologies at the time that national policy decisions were made, and in part because of

concerns about the feasibility of retrieving waste from deep boreholes. Advances in drilling technologies over the last several decades (e.g., Beswick 2008) suggest that the construction of deep boreholes should no longer be viewed as a greater technical challenge than deep mines, and 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, and if permanent disposal is not intended, deep boreholes should not be a preferred option.

As described by Arnold et al. (2011; 2012) and illustrated in Figure 2-5, 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, 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.

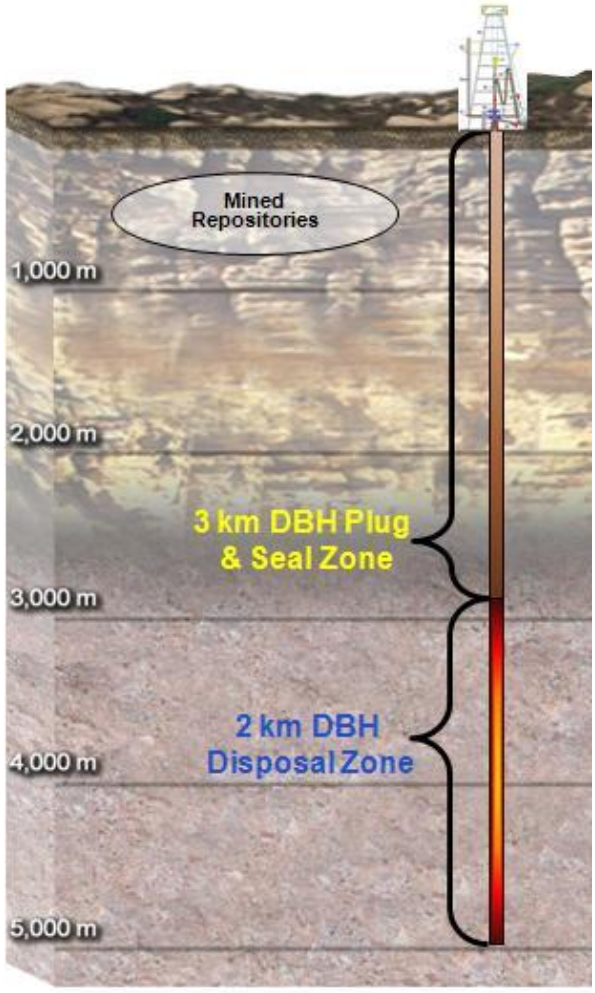


Figure 1-5. Schematic representation of a deep borehole repository

The deep borehole disposal 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; there are seven companies in the U.S. operating such rigs. 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.

Isolation of the waste would be provided by the extremely low permeability of crystalline rocks at these depths (significantly deeper than the depths proposed for mined repositories), and by the long pathway for diffusive transport upward through the borehole seal system. 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 the 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 the bentonite backfill capabilities.

For the purposes of evaluating a DOE-managed HLW and SNF disposal concept, only the three mined geologic repository concepts in crystalline rocks (e.g., granite), argillite, and salt are going to be considered in detail because the deep borehole concept is currently being considered primarily for a subset of small waste forms, many of which could be disposed in deep boreholes with diameters much less than 17-in, with consideration of potential alternate disposal pathways that allow flexibility for the disposal mission. The work below focuses, in part on assessing those aspects of the mined repositories that may need to be modified in a DOE-managed HLW and SNF only repository relative to the reference cases for those repositories including commercial spent nuclear fuel.

2. Inventory Included in a Defense Repository and Considerations of Resultant Disposal Conceptual Variations

The conclusion of the Waste Forms Disposal Options Evaluation (SNL 2014) were 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 this 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 concepts). (NOTE that 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 SNL (2014) study indicate that some disposal options form 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 Waste Isolation Pilot Plant 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 greater.

c) Mined repositories in crystalline rocks may offer operational advantages because of the rock strength, which allows robust openings to be easily maintained providing 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 work in SNL (2104), 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 DRep (Carter et al., 2012, 2013). These studies also inform the analyses being done this fiscal year for a DRep within the UFD Campaign, and the previous DRep inventory estimates were synthesized and integrated through Wilson (2016) for use in FY2016 UFD Campaign scoping analyses. It should be noted that the DRep inventory define in Wilson (2016) is only a preliminary one for use in the prototype analyses for FY2016 as it (a) does not include the cooler naval SNF waste packages (<~15 using 1000W per canister as the threshold) that are intended for a DRep; (b) has only a preliminary basis for thermal binning of DHLW glass canisters from Savannah River and Hanford; and (c) may change regarding DSNF to be included in a potential DRep based on DOE decisions for those waste forms. The updates to the DRep inventory will be one focus of activities in FY2017 work. Given the preliminary DRep inventory, the broad generalities for disposal concepts defined above are assessed for changes to the reliance on system features and variations on design concepts for the range of mined geologic repository concepts for a DRep containing no CSNF and only a small number of lower-thermal-load naval SNF canisters.

2.1 Included DOE-Managed HLW and SNF Inventory for Defense Repository Analyses

This section provides the overview of the inventory included for DRep analyses in the activities for a DRep for this year FY2016). Some of the DOE-managed HLW and SNF would not necessarily be included in the inventory to be placed within a DRep because it has been designated as commercial in origin. As stated in DOE (2014), “The inventory of DOE-managed SNF has also been augmented by

materials not contemplated in the early 1980s, including foreign research reactor fuel and several hundred tons of commercial SNF accepted by DOE under its Atomic Energy Act responsibilities (e.g., SNF from the Fort St. Vrain reactor in Colorado and the damaged core from the Three Mile Island Unit 2 reactor). Under the terms of the NWPA, this commercial-origin SNF is not a candidate for disposal in a separate repository for DOE-managed wastes." Similarly, West Valley Glass (275 canisters) is designated as of commercial origin (DOE, 2014; see the Table 1 note). Because the designation of some of the DOE-managed SNF is not completely finalized, there is some uncertainty regarding what to include or exclude for analyses of a DRep. The included materials are identified for the inventory used herein for preliminary scoping analyses in FY2016. The included set of materials for the inventory may change in the future based on the designation decisions made by DOE. The included inventory in this report is only for use in analyses of a potential generic DRep, and there is no intent to indicate whether any waste form is, or is not, commercial waste.

Wilson (2016) provides the preliminary inventory for the analyses of a DRep for FY2016 and includes both DOE-managed SNF and HLW. There are both average radionuclide content and ranges of thermal output provided for the included waste forms (Wilson, 2016). For our preliminary DRep inventory, the various types of DSNF are listed in Appendix A as included in the ~2485 DSNF canisters (based on Table 2-1 from Wilson, 2016). The primary included DHLW canister counts are given in Wilson (2016; Tables 2-3 thru 2-6) for Savannah River glass (7824 canisters), Hanford glass (11,800 canisters), INL Hip'd calcine (4391 canisters), and Hanford vitrified Cs/Sr capsules (340 canisters- see SNL, 2014 also).

Planned updates in FY2017 to this preliminary DRep inventory (Wilson, 2016) include (a) adding the cooler naval SNF waste packages (~12 naval SNF canisters based on ~1000W per canister as thermal threshold—see Figure 3 of DOE, 2014), (b) adding the 34 glass canisters of "German" (generated for FRG testing) glasses (SNL, 2014), (c) swapping in the planned HIP'd waste form for calcine in ~320 canisters (~5.5 ft diameter by ~15 ft height, naval canisters; SNL 2014); and revising the list of DSNF materials included in the inventory based on any applicable DOE decisions. Though most of these updates are relatively small from the standpoint of inventory mass, they may have some implications for thermal aspects (naval SNF and FRG glasses) and handling considerations (naval SNF and planned calcine waste forms) as discussed in Section 2.3.

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

As discussed in detail below in Section 2.3, major variations in the inventory included for a particular repository concept may influence more than simply the total radionuclide content. The inventory also affects the total thermal input, the potential thermal distribution, the numbers of packages to be handled, and the range of size/mass of packages to be handled in a geologic disposal 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 (YMP; DOE, 2008). The SAR inventory slated for a repository at YM included a large portion of commercial SNF (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 included waste inventory for DRep analyses listed above, differs from the SAR inventory in three primary ways.

1. There is no Commercial SNF (CSNF)
2. There are larger quantities of the various DHLW Glass included
3. There are smaller quantities of DSNF included (this will be updated in FY2017, but will still not include all DSNF and will include only the ~12 coolest naval spent fuel waste packages)

Any Defense Waste Repository will not include any CSNF (DOE, 2014; 2015). This results in a very large reduction in the total radionuclide content and the thermal mass relative to the YMP SAR inventory.

The DRep inventory from Wilson (2016) includes about 2.5 times as many HLW canisters as was planned for the YMP repository[a]. (The Nuclear Waste Policy Act placed a legal limit on the amount of radioactive waste (MTHM) that could be disposed of in a Yucca Mountain repository. A portion of this limit (4,667 MTHM) was allocated to HLW glass (DOE, 2008, Table 1.5.1-1). The SAR projected a total of 21,228 HLW 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 HLW canisters would be included in the Yucca Mountain 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 allocation represented less than half of the projected HLW inventory. The current DRep 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, and The current estimates of the DOE-managed SNF inventory total 2,336 MTHM (DOE 2014, Section 2.2.1). The current DRep inventory includes most, but not all of the DSNF, and will be 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 DRep inventory and the differences from the SAR inventory.

2.1.1.1 Defense Repository Included DHLW Inventory Compared to SAR Inventory

High Level Waste (HLW) has been generated by the reprocessing of 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/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).

The DHLW can be grouped into several 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 to be vitrified into glass logs
- Calcine waste at Idaho, which is planned to be hot isostatically pressed (HIP'd) into a glass ceramic waste form (note that direct disposal of calcine is being considered potentially for Deep Borehole Disposal: SNL, 2014; DOE, 2014)
- German (FRG) glass logs stored at Hanford, which have no further planned treatment (to be added to DRep inventory in FY2017)

[a] Note that this means the SAR projected inventory of total DHLW canisters is only about 40% of the DRep 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'd calcine canisters (smaller-sized, alternate waste form) projected in Wilson (2016) versus SNL (2014).

- Sodium bearing waste at Idaho, which is to be treated by fluidized bed steam reforming (to be added to DRep inventory in FY2017)
- Cs and Sr capsules at Hanford, which are planned to be vitrified (note these are being considered potentially for direct disposal 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.

The Yucca Mountain SAR (DOE, 2008, Section 1.5.1.2.1.2) included projections based on the best information that was available at the time. Wilson (2016) developed the DRep inventory for supporting design/engineering analyses, including thermal evolution, and safety assessments of a DRep. 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. This difference is largely explained by the change to planned waste form for calcine given in above paragraph, but there may also be additional wastes beyond calcine included in the SAR estimate.

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. Note that these values represent best estimates of projected numbers of canisters at the time the SAR (DOE, 2008) that were to be delivered to the YM site, 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 DRep inventory in support of preliminary design thermal and post-closure safety calculations for FY2016. These estimates are based on current planning assumptions.
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'd calcine waste form are ~5.5 ft diameter by ~15 ft height (Kluk et al., 2011), whereas the HIP'd 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.

2.1.1.2 Defense Repository 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 of 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 done on the YM Viability Assessment (VA), the Site recommendation (SR) and 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 DRep analyses are given in Wilson (2016) and Appendix A presents a detailed tabulation of DSNF items that are included in this DRep inventory, The Appendix A table is organized using each of 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 left hand column also includes the mass (MTHM) of all items within that fuel group and the projected number of waste containers within each DSNF group. Note that there are not any naval SNF containers included in this preliminary inventory and the coolest naval SNF containers (<~1000 W) are planned to be added in FY2017. As stated at the beginning of Section 2, the portion of existing DSNF that would be included in a DRep is uncertain, and this preliminary DRep inventory is being used for scoping calculations in FY2016 and is planned to be updated based on DOE review and comment on those items.

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 Waste Forms Disposal Options Evaluation (SNL, 2014). At this point in time, these candidates have only been identified but not added into the evaluations. Consideration of these wastes further in the FY2017 would determine which would be added to the full list of DOE-managed HLW and SNF to be populate in the OWL as discussed below. A brief summary is given here of the waste types that have been 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. That SNL (2014) inventory of wastes is a larger one than the DRep inventory discussed in Section 2.1, which is a subset. Inclusion of additional wastes into the OWL would be only a first step as each added waste would only be added to the DRep inventory later 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 on glass formulations due to components such as Fe, Al, Cr, Bi, P, Zr, and S (e.g., Kruger et al., 2012; 2013). In many of these cases, the compositional variation of the glass for some of its elemental components does not appear to warrant a separate tracking as yet because these are still within the R&D stage. One exception from high sulfur waste streams is included below.

Additionally, some advanced fuels are being developed that will at some point need disposal dispositioning for research reactors like at the Transient Reactor Test Facility (e.g., Pope et al., 2014). Given the wide range of fuel types considered within the DOE, such advanced fuels will only be considered as they are included into the DOE-managed SFDB as they would provide no immediate substantive difference for consideration. Lastly, many investigations 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 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 type/form is included here as it has already been separated and is planned to be formed. Such considerations may suggest additional tracking of potential waste types/forms, however this should only be engaged once the waste types/forms are actually generated.

Potential additions to the 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
  - WTP LAW 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 Potential Variations of Conceptual Defense Repository Features Related to Included Inventory Characteristics

DOE-managed HLW and SNF in the DRep inventory differ from CSNF in quantity (there's a lot less) heterogeneity (more so), radionuclide inventory (HLW is weighted toward fission products), thermal load (generally less per waste package), and waste form composition (including the presence of 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 within the Appendices of SNL (2014), considerations of these parametric aspects helps define the data characteristics that would be most central to evaluating the potential changes to features of repository concepts.

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

AS discussed in Section 2.1, the DRep inventory differs significantly from that considered in the SAR for Yucca Mountain (DOE, 2008). It also differs from that inventory in a generic repository assumed 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 scenarios, 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 DRep, which has smaller radionuclide inventory and lower thermal load, results in the variations in waste forms more pronounced because they are not overwhelmed by the CSNF. For these reasons, reassessing design

choices, FEPs screening, and PA implementation facilitates identifying aspects of a DRep that may be different than a repository containing 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 disposal of defense-related waste, 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 the ability to construct the repository sequentially, 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 in order 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 and other controls on waste inventory 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 DRep, additional criteria may become relevant to repository design, to FEPs screening, and to PA implementation. The Waste Forms Disposal Options Evaluation (SNL 2014) identifies several issues specific to particular types of defense-related waste, 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 repository; 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 defense-repository and/or result in exclusion of FEPs due to exclusion of commercial waste and hotter defense-related wastes from the repository.

2.3.1.1 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 DRep.

Mined Repository in Salt

Extensive salt deposits can occur as bedded (as at the Waste Isolation Pilot Plant (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) 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 defense-related waste 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 DRep in salt is being developed currently.

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 (2.3–3.8 $Wm\text{-}1K^{-1}$; 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 DRep (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 reference). 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 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.

Defense-Related Waste

The inventory covered in SNL (2014) 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 DRep inventory. The other eight WG contain waste types currently managed by DOE that are potentially part of inventory for a DRep (Section 2 delineates the inventory for DRep analyses for FY2016). These eight are WG3: vitrified HLW (including vitrified Cs and Sr capsules); WG4: other engineered HLW forms (including hot isostatically-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 (Section 2) of a low-temperature, DRep is a subset of the waste managed by DOE, and primarily would includes waste forms in WG3 (vitrified HLW), WG4 (engineered HLW – HIP'd calcine), WG5 (metallic SNF), WG7: (oxide spent fuels) and WG8 (salts, etc.), along with the coolest waste packages in group WG10 (Naval SNF – to be added in FY2017). For GDSA purposes, there are three waste form degradation rate mechanisms included in the PA of a DRep into which each of these 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'd 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.

2.3.1.2 Variations from a CSNF-dominated repository

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 criteria come into play in assessing the characteristics of a DRep 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 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 the 70,000 MTHM limit specified in the Nuclear Waste Policy (1983). The projected defense-related 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 HIP'd. The projected DOE-managed SNF inventory is ~7200 m^3 (SNL 2014), over half of which would be excluded from a low-temperature, defense-only repository. Assuming ~30,000 m^3 of waste in the defense-only repository, 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, defense-only repository 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 defense-related HLW is almost entirely (e.g. Cs and Sr capsules, calcine waste), or in large part (Hanford and Savannah River glass and tank waste) from ^{137}Cs, ^{90}Sr and their daughter products (SNL 2014; Carter et al. 2013).

Waste forms 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 the repository will affect the timing of multiple processes occurring in the repository, including: resaturation; buoyancy driven fluid flow; waste package degradation; waste form dissolution; buffer and host rock alteration; creep consolidation.

Thermal characteristics

Defense-related HLW and SNF generate less heat per canister than CSNF packed in any of the waste packages under consideration. 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 DREp waste forms generating the greatest thermal load are 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 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 SNL (2014) calculated to generate 2240 W per waste package after treatment of 6-year-old waste or 1250 W per waste package after treatment of 20-year-old waste; and naval SNF, which averages 4250 W per waste package (currently only the naval coolest naval SNF packages would be added to the DRep inventory in FY2017).

Most DRep 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% (~7037) of Savannah River glass HLW (Wilson 2016). Similarly in the DRep inventory, 94.1% of DSNF (~2337 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 DRep, 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 defense-related waste generates less heat than commercial waste, the magnitude of peak temperatures in a DRep would be considerably less than peak temperatures in a commercial repository. 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 defense-related HLW and SNF. The magnitude of peak repository temperatures will affect multiple repository processes, including resaturation 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 defense-repository 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 DRep 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.

DRep waste forms can be sorted into three broad categories on the basis of 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 DRep 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 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 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. DRep 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 multicanister 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). Hot isostatically-pressed calcine waste is planned to be packaged in the same size canister as Naval SNF (SNL, 2014; though an alternative HIP'd calcine waste form in 2' by 10' glass canisters in the DRep inventory of Wilson, 2016). The large openings (shafts and/or ramps) that facilitate emplacing these largest waste packages are more easily maintained in crystalline rock than in formations in which creep and plastic deformation occur.

2.3.1.3 Summary of Variations on Disposal Concepts

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

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

b. A DRep 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.

c. A DRep 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.

d. A DRep 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.

e. A DRep inventory packaging plans result in a bimodal distribution of waste package sizes. Large waste packages may create engineering challenges in some disposal concepts.

These DRep considerations are being evaluated in more detailed design analyses and evaluations of those Features, Events, and Processes (FEPs) that would be handled substantively differently for a DRep 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 would be explicitly evaluated for a DRep. These detailed evaluation activities will be more mature in FY2017 and should begin to define the set of FEPs that appear affected, followed by the revised evaluations of those for a DRep.

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) to manage the information of all the wastes and waste forms from SNL (2014; including CSNF if needed). A prototype OWL database is described and has been populated with data for the Cs/Sr capsule waste and two alternate waste forms for disposal. Both the OWL database model (Appendix B) and a user's guide to the OWL prototype (Section 3.1.3) are provided.

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 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 fully operational.

The Siting Experience Archive (SEA), which is a database of various experiences primarily in the U.S. on siting large controversial projects, was developed at SNL for DOE in FY13. 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 development has been restricted internal to SNL until an external interface is needed. The prototype OWL is 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 online waste library (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 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 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 On-line Waste Library could include:

- 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, defense, 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 Protoype Development Status

OWL is designed to contain information regarding all the high level radioactive waste that the DOE manages, and to be able to disseminate that information. Currently, the OWL prototype database contains information for one waste, the cesium and strontium capsules at Hanford, with two potential waste form pathways defined – (1) vitrification of the capsules to glass waste form (current DOE planned disposal pathway) and (2) direct disposal of the capsules (alternative disposal pathway). The OWL prototype database is designed to capture the following information about a particular waste:

- Facility where it is currently located
- Waste classification (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
- Volume of the disposal waste form
- Mass of the disposal waste form

Because of the way the database is structured, users can sort on waste by facility (Hanford, Idaho, or Savannah River), and waste classification (HLW or SNF). This makes it easy to identify all the HLW types that are currently at Hanford, for example (similar to the DOE SNF database capabilities). 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 that give the inventory in various units, such as volumes, radioactivity, and/or thermal output of wastes as they currently exist, and in their planned (or planned or proposed) disposal waste form(s). From these data, visual displays are easily generated. An example of a visual display that shows the curies of the five radionuclides in the cesium and strontium capsules is shown in Figure 10. Other visual displays comparing waste volumes, radioactivity, or thermal output can be generated once the data for additional wastes have been entered into the OWL database.

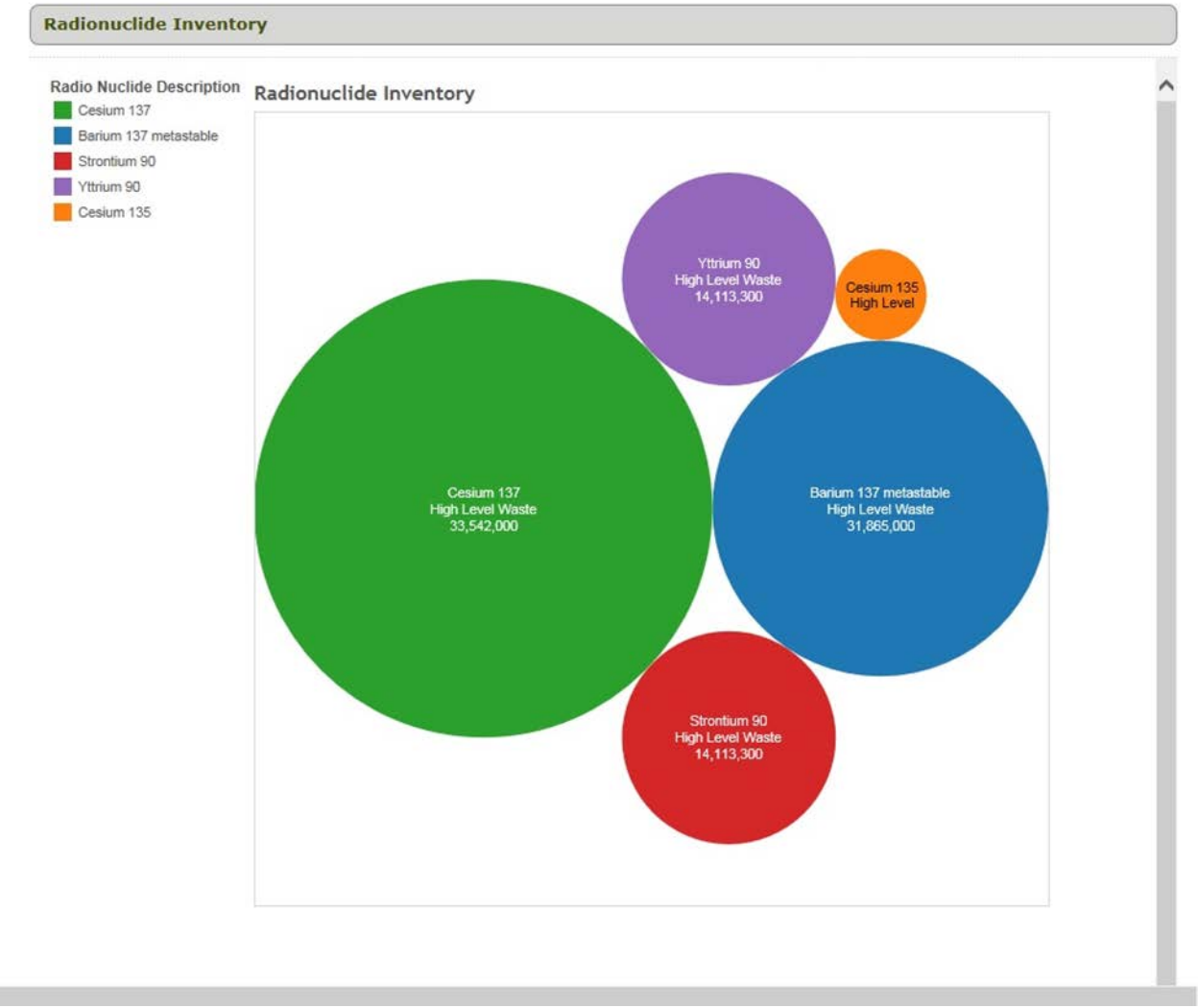


Figure 3-1. Example of visual display of OWL data (total curies of primary radionuclides in the cesium and strontium capsules as of January 1, 2016).

Starting in FY2017, the future work on the OWL database includes the following:

- Add the full set of information regarding the other wastes from SNL (2014) – i.e., fully populate the OWL for previously identified waste types and waste form pathways
- Develop the complete set of documentation for the OWL database architecture, including a comprehensive user's guide (see Section 3.1.3 for prototype user's guide)
- Develop a review and verification process to ensure information in the OWL is accurate and sourced correctly
- Define an update processes (this will be done in conjunction with user review and feedback on the prototype) 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 is a priority for FY2017 activities, as is making the OWL available to a set of users that can provide direct feedback to the activities listed in the second and third bullets above. 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 wastes and waste forms.

3.1.3 User's Guide to OWL Prototype

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 3-2.

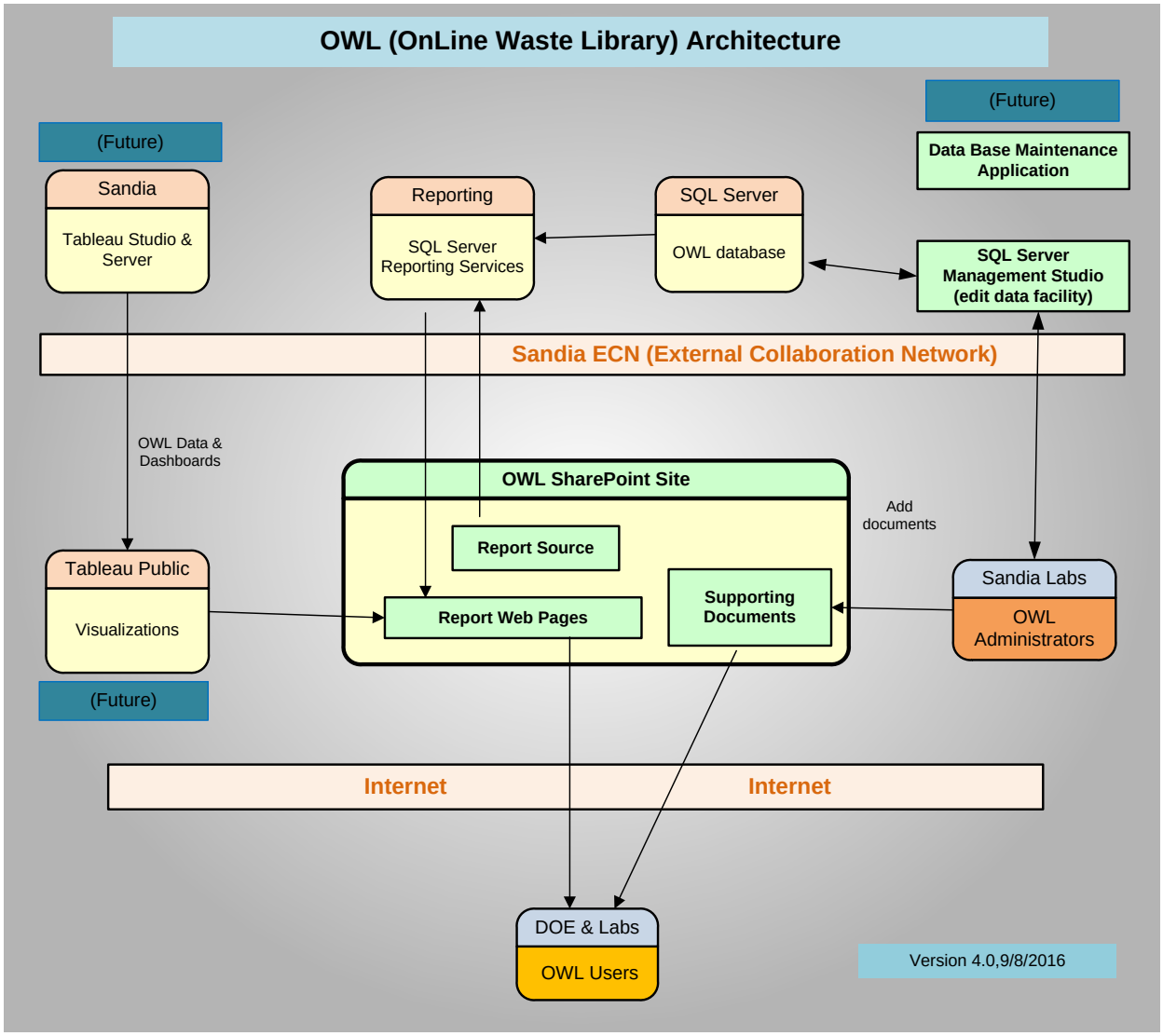


Figure 3-2. The high-level architecture of the OWL implementation as constructed on the Sandia National Laboratories External Collaboration Network (ECN).

The usage of the OWL is via straightforward access to a homepage within the SharePoint Site and a user's guide for the prototype OWL is given here to demonstrate the various options for queries and reports from the database. The OWL home screen is shown in Figure 3-3. From this screen, the user can search for the information that is contained in the database. Currently, users can search on all the wastes, search on the radionuclides in each waste, look at the supporting documents for each waste, and calculate the inventory of a particular waste in a chosen year. These search functions are described in more detail below.

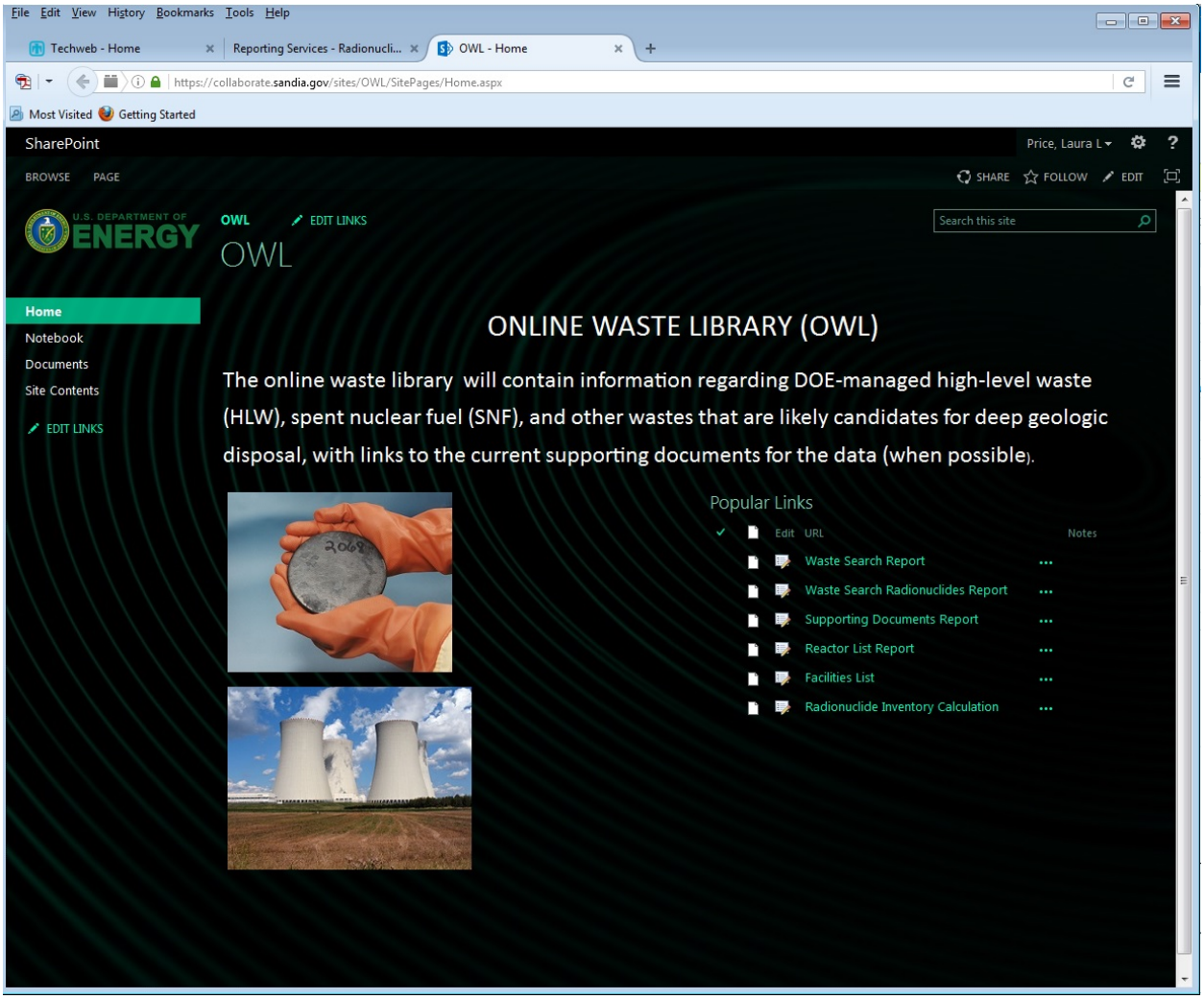


Figure 3-3. Home Screen for OWL.

Search for a Waste

From the home screen in Figure 3-3, selecting "Waste Search Report" allows the user to see all the wastes that are currently in OWL. The user can search by waste location (Hanford, Idaho, Savannah River) and waste classification (high level waste or spent nuclear fuel). The Waste Search Report is shown in Figure 3-4. Currently, the only waste for which information has been entered into the database is cesium and strontium capsules; hence this is the only waste that appears in the report. This report gives a short description of each waste (classification, location, volume, radioactivity). 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 3-5 through 3-8), which provides more detailed information regarding the waste.

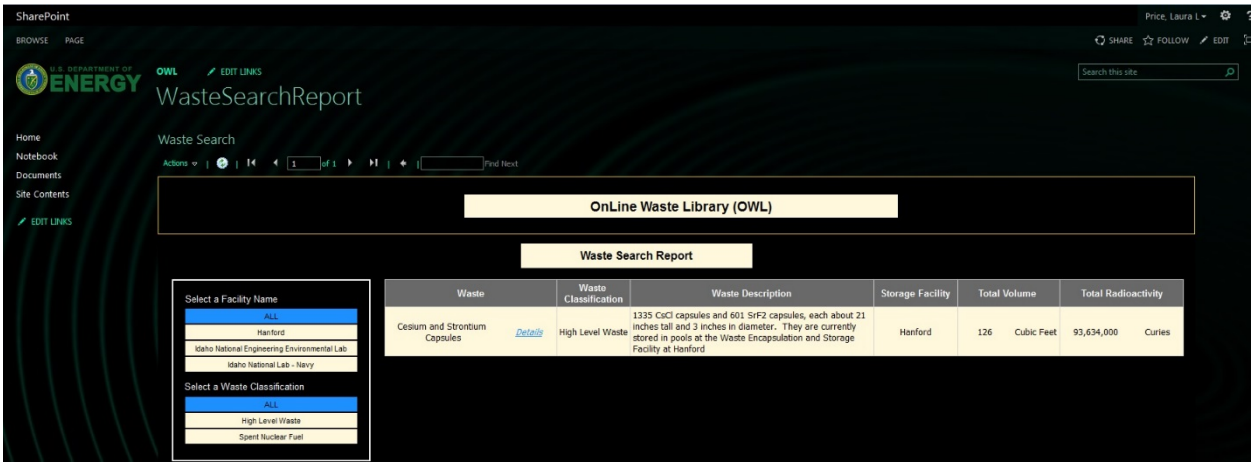


Figure 3-4. Waste Search Report.

Figure 3-5 shows the top portion 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 gives a list of documents that provide the basis for the information presented in the report, as well as documents that are general relevance to the waste. For each document, clicking on "Display" in the right hand column 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.

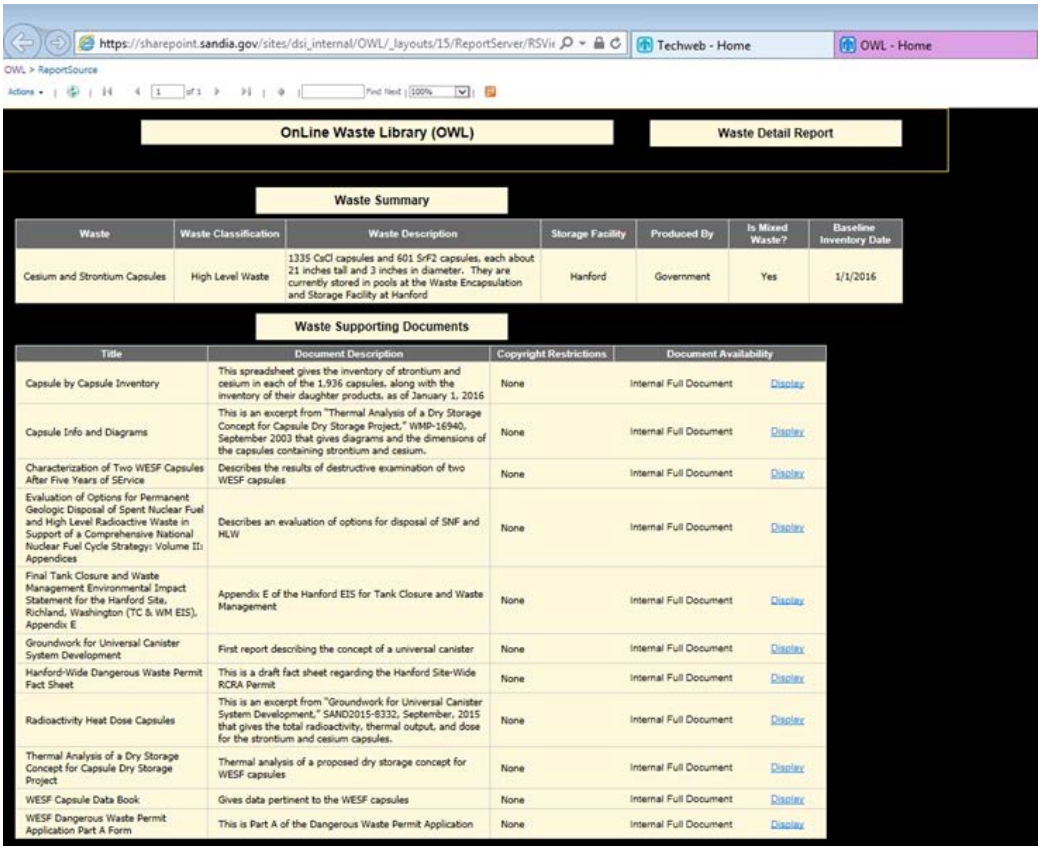
Figure 3-6 shows the second portion of the Waste Detail Report, which gives the sources of the waste, a person who can be contacted regarding the waste, and waste characteristics. Waste characteristics include contaminants, thermal output, dimensions, volume, and radioactivity of the waste. In the Waste Characteristics table, clicking on "Display" in the right hand column in the same line as a particular characteristic opens another browser window that will allow the user to see the document that provides the basis for the value of that characteristic. For example, clicking on "Display" in the line that gives the average thermal output of the Cs capsules opens a spreadsheet that gives the inventory of each capsule, as well as the average thermal output for all the capsules. Frequently what is displayed is a table or a few pages excerpted from a report. In such cases, the excerpted pages are shown along with the title page of the report; the entire report is available by clicking on "Display" in the Waste Supporting Documents, as shown in Figure 3-5.

Figure 3-7 shows the third portion of the Waste Detail Report, which gives the radionuclide characteristics and radionuclide inventory of the radionuclides in the waste. Characteristics include half-life and activity ratios, for those radionuclides that are in secular equilibrium with a parent radionuclide. The inventory is as of the baseline date; calculations of inventory for another date are described below.

Figure 3-8 is the bottom portion of the Waste Detail Report, which gives the planned and alternative forms for disposing of the waste, as well as characteristics of those forms. Characteristics of the waste forms for disposal include average thermal output, dimensions, mass, and volume.

Search for Particular Radionuclides in Waste

From the home screen in Figure 3-3, selecting "Waste Search Radionuclides Report" allows the user to see all the wastes that are currently in OWL and to search by radionuclide, in addition to searching by location and waste classification (Figure 3-9), as described above. For example, a user will be able to see which wastes contain ^{239}Pu (once the OWL database is populated with more waste than just the cesium and strontium capsules). As described above, clicking on "Details" next to the name of a nuclear waste opens the Waste Detail Report (Figure 3-5).



OnLine Waste Library (OWL)

Waste Detail Report

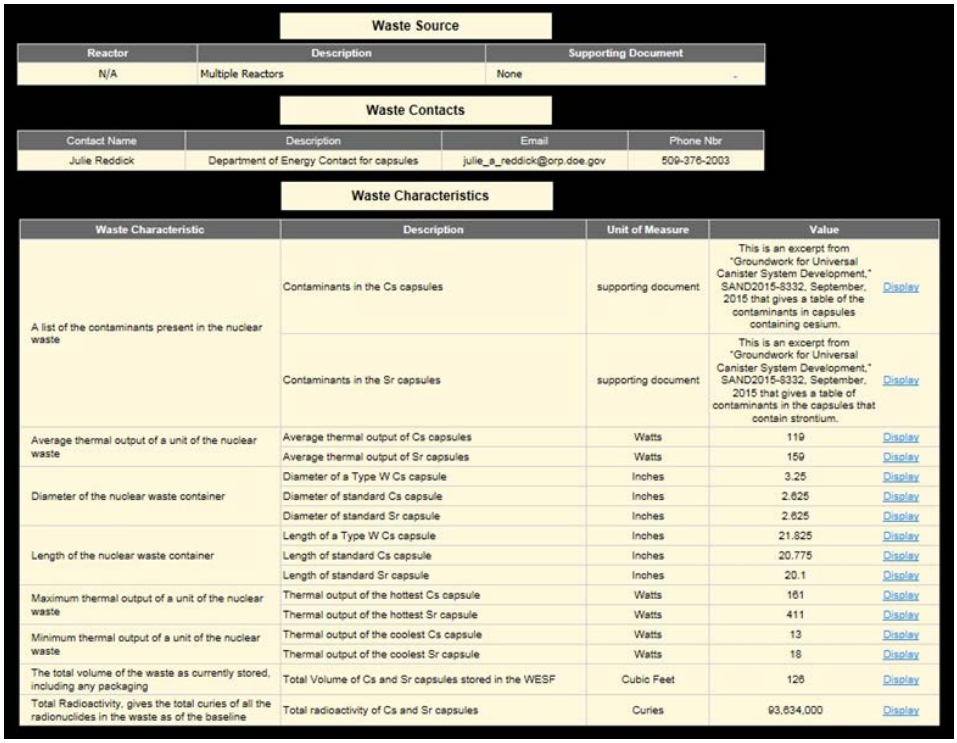
Waste Summary

Waste	Waste Classification	Waste Description	Storage Facility	Produced By	Is Mixed Waste?	Baseline Inventory Date
Cesium and Strontium Capsules	High Level Waste	1335 CsCl capsules and 601 SrF2 capsules, each about 21 inches tall and 3 inches in diameter. They are currently stored in pools at the Waste Encapsulation and Storage Facility at Hanford	Hanford	Government	Yes	1/1/2016

Waste Supporting Documents

Title	Document Description	Copyright Restrictions	Document Availability	
Capsule by Capsule Inventory	This spreadsheet gives the inventory of strontium and cesium in each of the 1,936 capsules, along with the inventory of their daughter products, as of January 1, 2016	None	Internal Full Document	Display
Capsule Info and Diagrams	This is an excerpt from "Thermal Analysis of a Dry Storage Concept for Capsule Dry Storage Project," WMP-16940, September 2003 that gives diagrams and the dimensions of the capsules containing strontium and cesium.	None	Internal Full Document	Display
Characterization of Two WESF Capsules After Five Years of SErvice	Describes the results of destructive examination of two WESF capsules	None	Internal Full Document	Display
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	Describes an evaluation of options for disposal of SNF and HLW	None	Internal Full Document	Display
Final Tank Closure and Waste Management Environmental Impact Statement for the Hanford Site, Richland, Washington (TC & WM EIS), Appendix E	Appendix E of the Hanford EIS for Tank Closure and Waste Management	None	Internal Full Document	Display
Groundwork for Universal Canister System Development	First report describing the concept of a universal canister	None	Internal Full Document	Display
Hanford-Wide Dangerous Waste Permit Fact Sheet	This is a draft fact sheet regarding the Hanford Site-Wide RCRA Permit	None	Internal Full Document	Display
Radioactivity Heat Dose Capsules	This is an excerpt from "Groundwork for Universal Canister System Development," SAND2015-8332, September, 2015 that gives the total radioactivity, thermal output, and dose for the strontium and cesium capsules.	None	Internal Full Document	Display
Thermal Analysis of a Dry Storage Concept for Capsule Dry Storage Project	Thermal analysis of a proposed dry storage concept for WESF capsules	None	Internal Full Document	Display
WESF Capsule Data Book	Gives data pertinent to the WESF capsules	None	Internal Full Document	Display
WESF Dangerous Waste Permit Application Part A Form	This is Part A of the Dangerous Waste Permit Application	None	Internal Full Document	Display

Figure 3-5. First (of four) portion of Waste Detail Report.



Waste Source

Reactor	Description	Supporting Document
N/A	Multiple Reactors	None

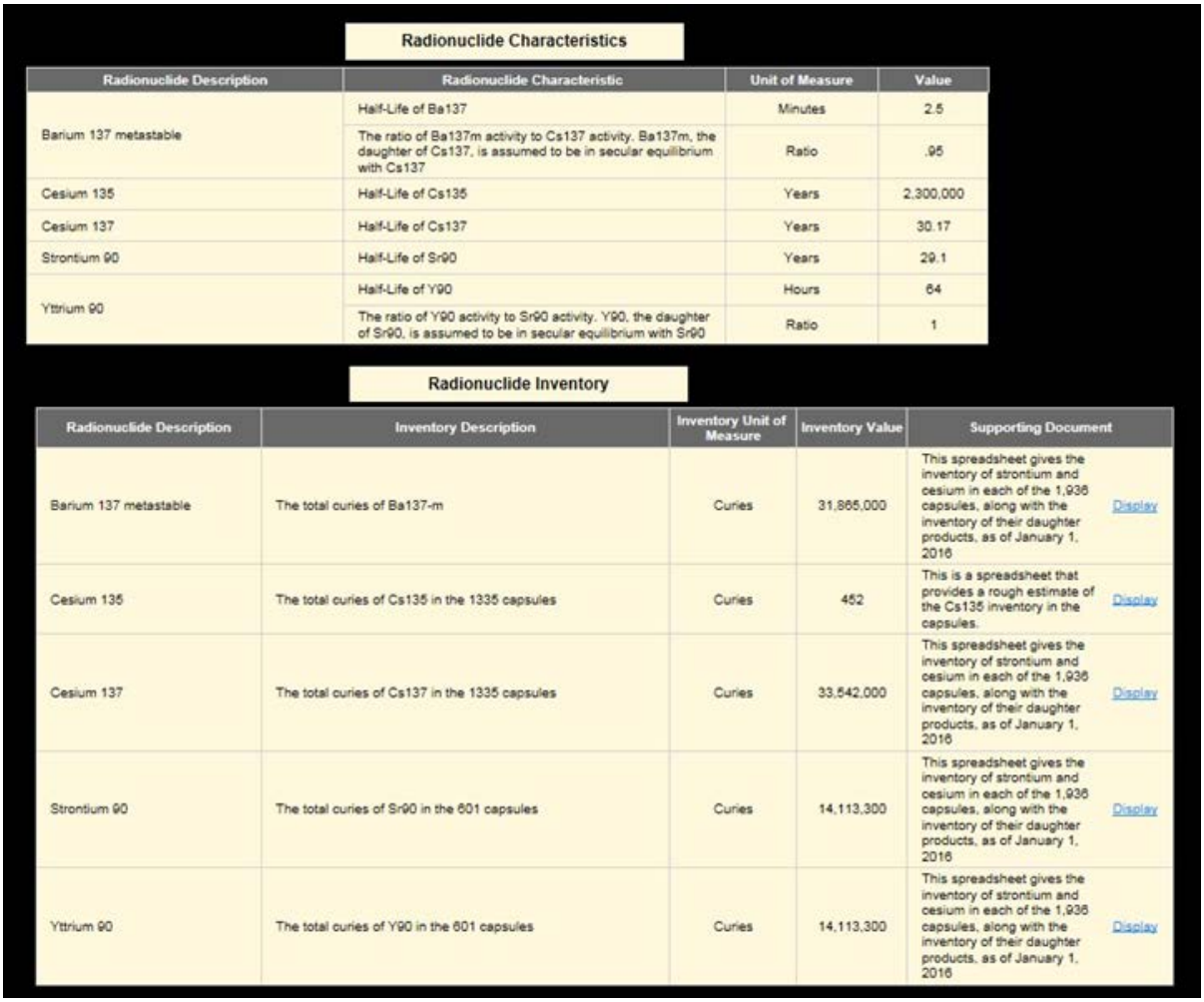
Waste Contacts

Contact Name	Description	Email	Phone Nbr
Julie Reddick	Department of Energy Contact for capsules	julie_a_reddick@orp.doe.gov	509-376-2003

Waste Characteristics

Waste Characteristic	Description	Unit of Measure	Value	
A list of the contaminants present in the nuclear waste	Contaminants in the Cs capsules	supporting document	This is an excerpt from "Groundwork for Universal Canister System Development," SAND2015-8332, September, 2015 that gives a table of the contaminants in capsules containing cesium.	Display
	Contaminants in the Sr capsules	supporting document	This is an excerpt from "Groundwork for Universal Canister System Development," SAND2015-8332, September, 2015 that gives a table of contaminants in the capsules that contain strontium.	Display
Average thermal output of a unit of the nuclear waste	Average thermal output of Cs capsules	Watts	119	Display
	Average thermal output of Sr capsules	Watts	159	Display
Diameter of the nuclear waste container	Diameter of a Type W Cs capsule	Inches	3.25	Display
	Diameter of standard Cs capsule	Inches	2.625	Display
	Diameter of standard Sr capsule	Inches	2.625	Display
Length of the nuclear waste container	Length of a Type W Cs capsule	Inches	21.825	Display
	Length of standard Cs capsule	Inches	20.775	Display
	Length of standard Sr capsule	Inches	20.1	Display
Maximum thermal output of a unit of the nuclear waste	Thermal output of the hottest Cs capsule	Watts	161	Display
	Thermal output of the hottest Sr capsule	Watts	411	Display
Minimum thermal output of a unit of the nuclear waste	Thermal output of the coolest Cs capsule	Watts	13	Display
	Thermal output of the coolest Sr capsule	Watts	18	Display
The total volume of the waste as currently stored, including any packaging	Total Volume of Cs and Sr capsules stored in the WESF	Cubic Feet	126	Display
Total Radioactivity, gives the total curies of all the radionuclides in the waste as of the baseline	Total radioactivity of Cs and Sr capsules	Curies	93,634,000	Display

Figure 3-6. Second (of four) portion of Waste Detail Report.



Radionuclide Characteristics

Radionuclide Description	Radionuclide Characteristic	Unit of Measure	Value
Barium 137 metastable	Half-Life of Ba137	Minutes	2.5
	The ratio of Ba137m activity to Cs137 activity. Ba137m, the daughter of Cs137, is assumed to be in secular equilibrium with Cs137	Ratio	.95
Cesium 135	Half-Life of Cs135	Years	2,300,000
Cesium 137	Half-Life of Cs137	Years	30.17
Strontium 90	Half-Life of Sr90	Years	29.1
Yttrium 90	Half-Life of Y90	Hours	64
	The ratio of Y90 activity to Sr90 activity. Y90, the daughter of Sr90, is assumed to be in secular equilibrium with Sr90	Ratio	1

Radionuclide Inventory

Radionuclide Description	Inventory Description	Inventory Unit of Measure	Inventory Value	Supporting Document	
Barium 137 metastable	The total curies of Ba137-m	Curies	31,865,000	This spreadsheet gives the inventory of strontium and cesium in each of the 1,936 capsules, along with the inventory of their daughter products, as of January 1, 2016	Display
Cesium 135	The total curies of Cs135 in the 1335 capsules	Curies	452	This is a spreadsheet that provides a rough estimate of the Cs135 inventory in the capsules.	Display
Cesium 137	The total curies of Cs137 in the 1335 capsules	Curies	33,542,000	This spreadsheet gives the inventory of strontium and cesium in each of the 1,936 capsules, along with the inventory of their daughter products, as of January 1, 2016	Display
Strontium 90	The total curies of Sr90 in the 601 capsules	Curies	14,113,300	This spreadsheet gives the inventory of strontium and cesium in each of the 1,936 capsules, along with the inventory of their daughter products, as of January 1, 2016	Display
Yttrium 90	The total curies of Y90 in the 601 capsules	Curies	14,113,300	This spreadsheet gives the inventory of strontium and cesium in each of the 1,936 capsules, along with the inventory of their daughter products, as of January 1, 2016	Display

Figure 3-7. Third (of four) portion of the Waste Detail Report.

Disposal Waste Forms

Waste Form	Description	Planned or Existing	Preferred or Alternative	Unit Of Measure	Value	Supporting Document
Vitrified Cs and Sr from capsultes	Glass logs in canisters	Planned	Preferred	canisters	340	This is an excerpt from "Final Tank Closure and Waste Management Environmental Impact Statement for the Hanford Site, Richland, Washington (TC & WM EIS)" that describes the process for vitrifying the contents of the Cs and Sr capsules. Display

Disposal Waste Form Characteristics

Waste Form	Form Characteristic	Characteristic Description	Unit of Measure	Value	Supporting Document
Vitrified Cs and Sr from capsultes	Average thermal output	Average thermal output of canister of vitrified Cs and Sr	watts	748	This one-page document describes the calculation of the thermal output of a canister containing vitrified waste from the strontium and cesium capsules Display
	Diameter of glass canister	Diameter of the canister into which the molten glass containing Cs and Sr is poured	feet	2	This is an excerpt from "Final Tank Closure and Waste Management Environmental Impact Statement for the Hanford Site, Richland, Washington (TC & WM EIS)" DOE/EIS-0391 that briefly describes the canisters into which the molten glass is to be poured. Display
	Height of glass canister	Height of the canister into which molten Cs and Sr is poured	feet	15	This is an excerpt from "Final Tank Closure and Waste Management Environmental Impact Statement for the Hanford Site, Richland, Washington (TC & WM EIS)" DOE/EIS-0391 that briefly describes the canisters into which the molten glass is to be poured. Display
	Mass of loaded canister	Mass of a canister, filled with vitrified Cs and Sr	pounds	8,217	This is an excerpt from "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" SAND2014-0189P that gives the projected mass of a canister filled with vitrified glass from the Hanford site. Display
	Total Volume	Total Volume of Vitrified Cs and Sr from capsules	cubic feet	15,980	This is a one-page document describing the calculation of the total volume of vitrified Cs and Sr from capsules, including the waste package. Display

Figure 3-8. Fourth (of four) portion of the Waste Detail Report.

OnLine Waste Library (OWL)

Waste Search Radionuclides Report

Select a Radionulcide
ALL
Ba137-m
Cs-135
Cs-137
Sr-90
Y-90

Select a Facility
ALL
Hanford
Idaho National Engineering Environmental Lab
Idaho National Lab - Navy
Savannah River

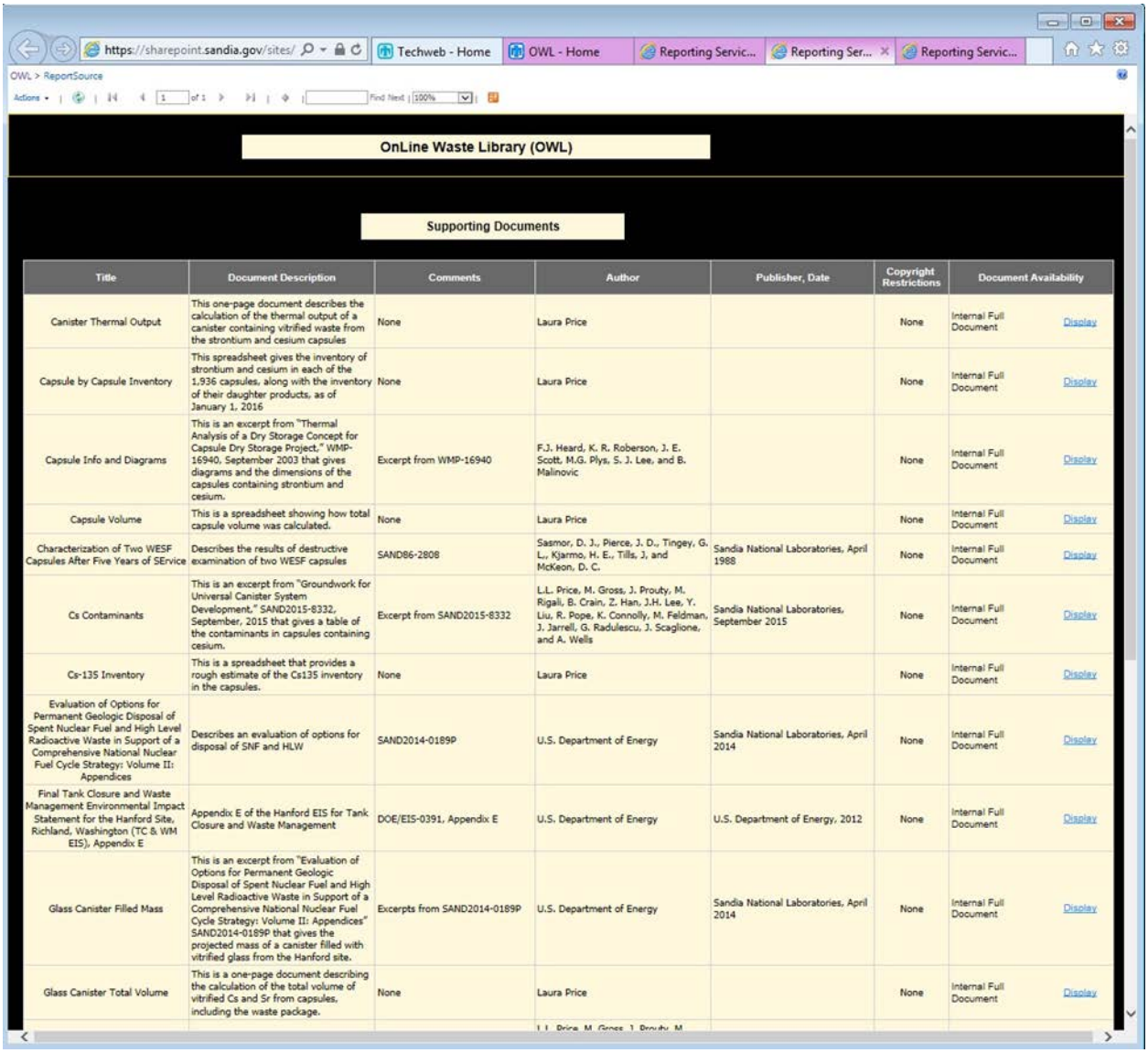
Select a Waste Classification
ALL
High Level Waste
Spent Nuclear Fuel

Nuclear Waste	Waste Classification	Facility Name	Radionuclide	Inventory in Curries
Cesium and Strontium Capsules Details	High Level Waste	Hanford	Barium 137 metastable	31,865,000
			Cesium 135	400
			Cesium 137	33,542,000
			Strontium 90	14,113,300
			Yttrium 90	14,113,300
			Total Inventory	93,634,000

Figure 3-9. Waste Search Radionuclides Report.

Supporting Documents Reports

From the home screen (Figure 3-3), selecting "Supporting Documents Report" allows the user to see a list of supporting documents (Figure 3-10). This list of supporting documents differs from the one shown in Figure 3-5 in that it includes full documents, excerpts from documents, and spreadsheets. That is, it contains both those documents that directly support data (e.g., a waste dimension) as well as those that provide more general information regarding the waste. As before, clicking on "Display" in the same line as a desired document opens another browser window that displays the document. From that browser window, the document can be saved or printed, depending on the capabilities of the web browser being used.



Title	Document Description	Comments	Author	Publisher, Date	Copyright Restrictions	Document Availability
Canister Thermal Output	This one-page document describes the calculation of the thermal output of a canister containing vitrified waste from the strontium and cesium capsules	None	Laura Price		None	Internal Full Document Display
Capsule by Capsule Inventory	This spreadsheet gives the inventory of strontium and cesium in each of the 1,936 capsules, along with the inventory of their daughter products, as of January 1, 2016	None	Laura Price		None	Internal Full Document Display
Capsule Info and Diagrams	This is an excerpt from "Thermal Analysis of a Dry Storage Concept for Capsule Dry Storage Project," WMP-16940, September 2003 that gives diagrams and the dimensions of the capsules containing strontium and cesium.	Excerpt from WMP-16940	F.J. Heard, K. R. Roberson, J. E. Scott, M.G. Plys, S. J. Lee, and B. Malinovic		None	Internal Full Document Display
Capsule Volume	This is a spreadsheet showing how total capsule volume was calculated.	None	Laura Price		None	Internal Full Document Display
Characterization of Two WESF Capsules After Five Years of SErvice	Describes the results of destructive examination of two WESF capsules	SAND86-2808	Sasmor, D. J., Pierce, J. D., Tingey, G. L., Kjarmo, H. E., Tills, J, and McKeon, D. C.	Sandia National Laboratories, April 1988	None	Internal Full Document Display
Cs Contaminants	This is an excerpt from "Groundwork for Universal Canister System Development," SAND2015-8332, September, 2015 that gives a table of the contaminants in capsules containing cesium.	Excerpt from SAND2015-8332	L.L. Price, M. Gross, J. Prouty, M. Rigali, B. Crain, Z. Han, J.H. Lee, Y. Liu, R. Pope, K. Connolly, M. Feldman, J. Jarrell, G. Radulescu, J. Scaglione, and A. Wells	Sandia National Laboratories, September 2015	None	Internal Full Document Display
Cs-135 Inventory	This is a spreadsheet that provides a rough estimate of the Cs135 inventory in the capsules.	None	Laura Price		None	Internal Full Document Display
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	Describes an evaluation of options for disposal of SNF and HLW	SAND2014-0189P	U.S. Department of Energy	Sandia National Laboratories, April 2014	None	Internal Full Document Display
Final Tank Closure and Waste Management Environmental Impact Statement for the Hanford Site, Richland, Washington (TC & WM EIS), Appendix E	Appendix E of the Hanford EIS for Tank Closure and Waste Management	DOE/EIS-0391, Appendix E	U.S. Department of Energy	U.S. Department of Energy, 2012	None	Internal Full Document Display
Glass Canister Filled Mass	This is an excerpt from "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" SAND2014-0189P that gives the projected mass of a canister filled with vitrified glass from the Hanford site.	Excerpts from SAND2014-0189P	U.S. Department of Energy	Sandia National Laboratories, April 2014	None	Internal Full Document Display
Glass Canister Total Volume	This is a one-page document describing the calculation of the total volume of vitrified Cs and Sr from capsules, including the waste package.	None	Laura Price		None	Internal Full Document Display

Figure 3-10. Supporting Documents Report.

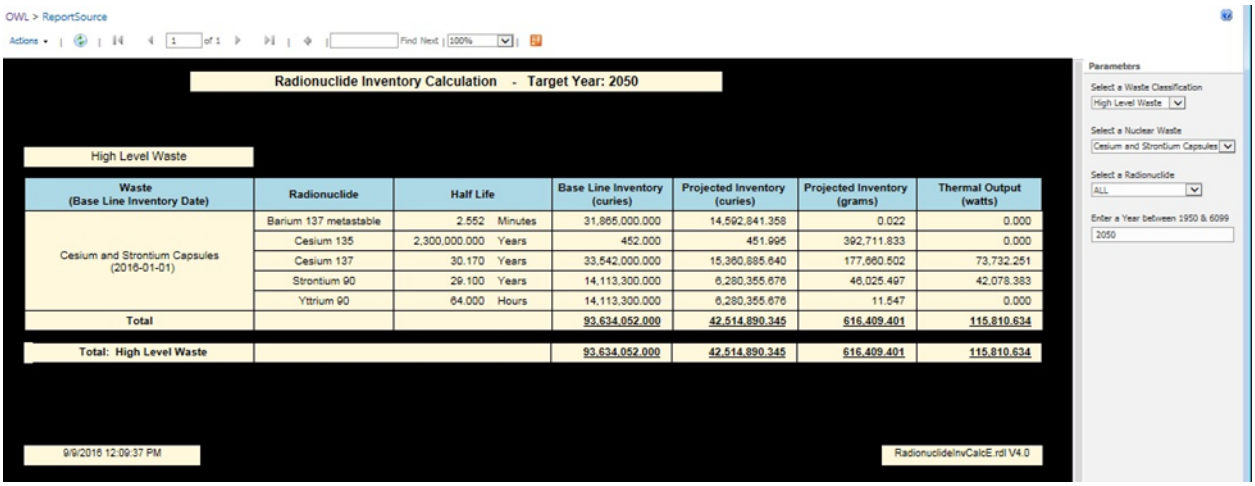


Figure 3-11. Radionuclide Inventory and Thermal (Heat) Content Calculation.

Radionuclide Inventory Calculations

From the home screen, selecting "Radionuclide Inventory Calculation" opens another browser window that allows the user to calculate the inventory of a particular waste (curies and grams) and its heat output for a given year (between 1950 and 6099) (Figure 3-11). The user can select the waste(s) and radionuclides for which calculations are to be performed as well as the year, as shown on the right hand side of Figure 3-11. It should be noted that the heat output of ^{90}Y is included in the calculation of the heat output of ^{90}Sr; likewise, the heat output of ^{137m}Ba is included in the calculation of the heat output of ^{137}Cs.

3.2 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.

3.2.1 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 (HIP'd) 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.

3.2.1.1 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 3-12 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 3-13 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 3-12
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^-

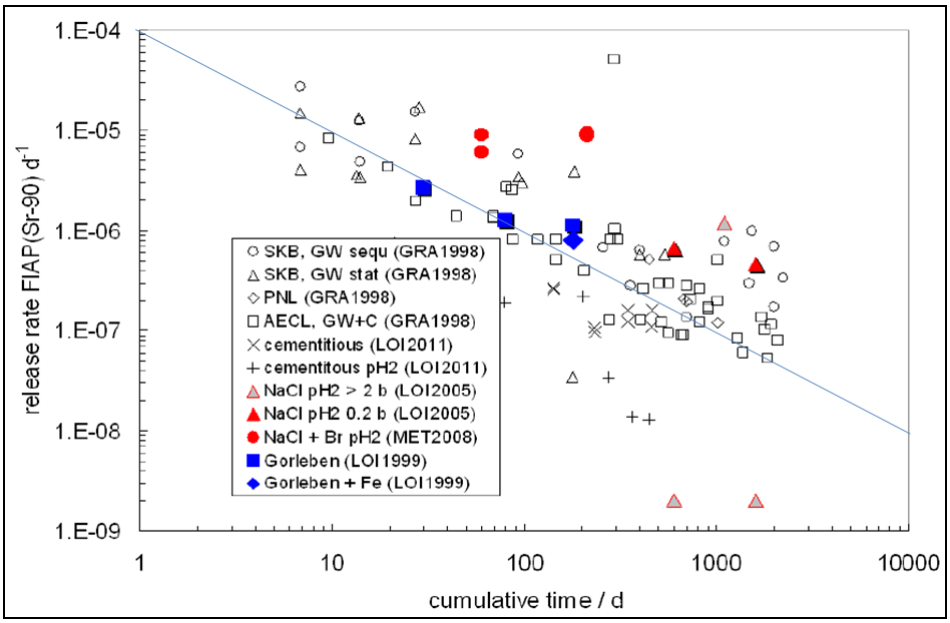


Figure 3-12. Compilation of UO_2 dissolution rate measurements displayed in Kienzler et al. (2012, Fig. 18). Blue line, which is superimposed on the Kienzler et al. (2012) figure, indicates the trend if zero dissolution occurs after starting at any point on the line.

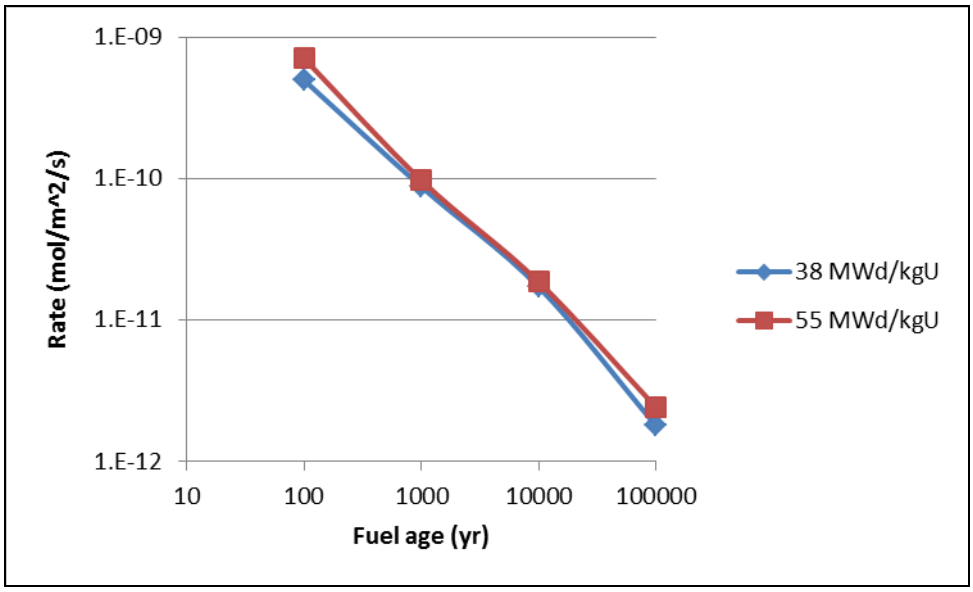


Figure 3-13. Maximum fuel dissolution rate calculated as a function of fuel age (Ollila 2011, Table 2-3).

3.2.2 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.

3.2.2.1 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.

3.2.2.2 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)

3.2.3 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 DRep analyses have been mapped into those models as either performing similarly or being bounded by a particular model degradation behavior. For example, the HIP'd calcine waste form is assigned to degrade as the HLW glass degradation. For waste forms that do not have substantial waste form lifetimes (> 10,000 years), 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 DRep 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 DRep, 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'd calcine waste form was evaluated as well. These provide input to potential adjustments to the GDSA models in FY2017, if appropriate.

3.2.3.1 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 behavior 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 behavior (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 postclosure 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 for the disposal options (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 DRep 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 DRep 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 DRep into two release/degradation models, namely UO_2–type UNF and instantaneous models, has been adopted for initial GDSA DRep 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 2000), where inventory was dominated by CSNF. Because the DRep 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

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 $\pm$ 12,200 J/mol)/RT} [100–178°C] rate (mg metal/cm^2/h) = 1.58×10^6 exp{(-80,500 $\pm$ 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

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).

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 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.

3.2.3.2 Calcine Waste and Associated Degradation Behavior

Background on Calcine Waste

UNF 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 (HIP) as the technology to treat the calcine and create a new waste form that is suitable for disposal. The HIP 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 HIP 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 HIP 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

NOTE: [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.

3.2.3.3 Degradation Model for Hot Isostatic Pressed (HIP'd) Calcine Waste with Additives

A literature survey revealed very little research has been done to establish the long-term dissolution rates of HIP'd calcine waste under repository conditions. However, Begg et al., 2005 studied HIP'd 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'd 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 3-14 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'd and fully densified 100% zirconia calcine sample exceeds the EA glass release rate limit for Na.

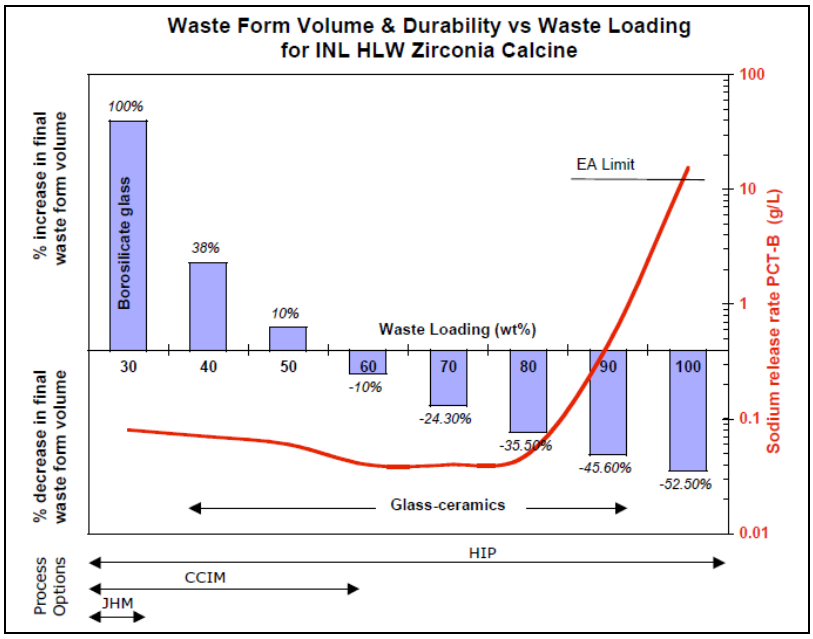


Figure 3-14. above from Begg et al., 2005 shows the relationship between various waste form alternatives including process options as a function of waste loading and chemical durability (PCT-B: Sodium release rate, g/L).

With the very limited amount of data available on calcine degradation it is difficult to assign a dissolution rate to HIP'd 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'd 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'd calcine waste. Conservatively, instantaneous dissolution may be assumed. The behavior of HIP'd calcine is very likely bounded by these two rates.

3.2.3.4 Degradation Model for HIP of 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'd 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.

3.2.3.5 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 and Objectives for Future Work

This status report covers the progress made in FY16 toward: (1) developing a preliminary DRep included inventory for engineering/design/safety analyses; (2) assessing the major differences of this included inventory relative to that in other analyzed repository systems and the potential impacts to disposal concepts; (3) designing and developing an on-line waste library (OWL) to manage the information of all those wastes and their waste forms (including CSNF if needed); and (4) constraining post-closure waste form degradation performance for safety assessments of a DRep. In addition, some continuing work is reported on identifying potential candidate waste types/forms to be added to the full list from SNL (2014 – see Table C-1), which also may be added to the OWL in the future. The status for each of these aspects is reported above.

DRep Included Inventory

Wilson (2016) provides the preliminary inventory for the analyses of a DRep for FY2016 and includes both DOE-managed SNF and HLW. There are both average radionuclide content and ranges of thermal output provided for the included waste forms (Wilson, 2016). For our preliminary DRep inventory, the various types of DSNF are listed in Appendix A as included in the ~2485 DSNF canisters (based on Table 2-1 from Wilson, 2016). The primary included DHLW canister counts are given in Wilson (2016; Tables 2-3 thru 2-6) for Savannah River glass (7824 canisters), Hanford glass (11,800 canisters), INL Hip'd calcine (4391 canisters), and Hanford vitrified Cs/Sr capsules (340 canisters- see SNL, 2014 also).

Planned updates in FY2017 to this preliminary DRep inventory (Wilson, 2016) include (a) adding the cooler naval SNF waste packages (~12 naval SNF canisters based on ~1000W per canister as thermal threshold—see Figure 3 of DOE, 2014), (b) adding the 34 glass canisters of "German" (generated for FRG testing) glasses (SNL, 2014), (c) swapping in the planned HIP'd waste form for calcine in ~320 canisters (~5.5 ft diameter by ~15 ft height, naval canisters; SNL 2014); and revising the list of DSNF materials included in the inventory based on any applicable DOE decisions. Though most of these updates are relatively small from the standpoint of inventory mass, they may have some implications for thermal aspects (naval SNF and FRG glasses) and handling considerations (naval SNF and planned calcine waste forms) as discussed in Section 2.3.

Disposal Concepts Information Evaluation

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

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

b. A DRep 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.

c. A DRep 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.

d. A DRep 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.

e. A DRep inventory packaging plans result in a bimodal distribution of waste package sizes. Large waste packages may create engineering challenges in some disposal concepts.

These DRep considerations are being evaluated in more detailed design analyses as input to identifying those FEPs that would be handled substantively differently for a DRep 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 would be explicitly evaluated for a DRep. These detailed evaluation activities will be more mature in FY2017 and should begin to define the set of FEPs that appear affected, followed by the revised evaluations of those for a DRep.

On-line Waste Library (OWL) Prototype

The online waste library (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 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 would take advantage of that existing dataset to incorporate it into the on-line waste library for use in post-closure PA. A prototype OWL database is described and has been populated with data for the Cs/Sr capsule waste and two alternate waste forms for disposal. Both the OWL database model (Appendix B) and a user's guide to the OWL prototype (Section 3.1.3) are provided.

Starting in FY2017, the future work on the OWL database includes the following:

- Add the full set of information regarding the other wastes from SNL (2014) – i.e., fully populate the OWL for previously identified waste types and waste form pathways
- Develop the complete set of documentation for the OWL database architecture, including a comprehensive user's guide (see Section 3.1.3 for prototype user's guide)
- Develop a review and verification process to ensure information in the OWL is accurate and sourced correctly
- Define an update processes (this will be done in conjunction with user review and feedback on the prototype) 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 is a priority for FY2017 activities, as is making the OWL available to a set of users that can provide direct feedback to the activities listed in the second and third bullets above. 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 wastes and waste forms.

Waste Form Performance Constraints

The models for degradation of both UO_2 and HLW glass given above (Section 3.2) are currently being used within the GDSA for PA modeling of post-closure system evolution. The waste forms in the current DRep analyses have been mapped into those models as either performing similarly or being bounded by a particular model degradation behavior. For example, the HIP'd calcine waste form is assigned to degrade as the HLW glass degradation. For waste forms that do not have substantial waste form lifetimes (>10,000 years), 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 DRep 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 DRep, 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'd calcine waste form was evaluated as well. These provide input to potential adjustments to the GDSA models in FY2017, 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 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'd 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'd calcine waste form would represent a conservative bounding approach. For untreated calcine, or calcine HIP'd 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 (DRep). Note that there are materials for which final disposition for a DRep 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 DRep 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

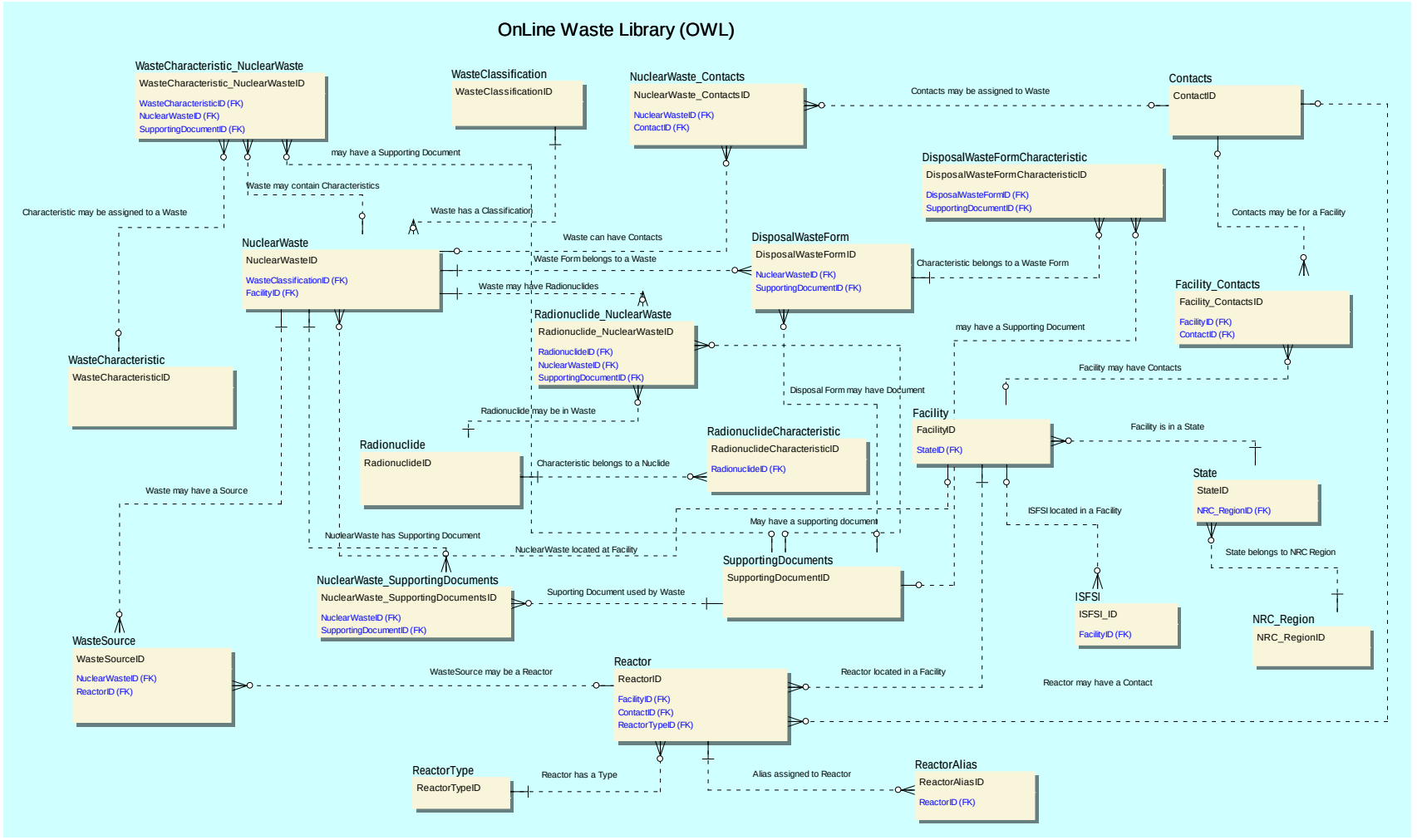
Information Model

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	

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	