SANDIA REPORT
SAND2015-8332
Unlimited Release
September 2015

Groundwork for Universal Canister System Development

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Laura Price, Mike Gross, Jeralyn Prouty, Mark Rigali

Argonne National Laboratory
Brian Craig, Zenghu Han, John Hok Lee, Yung Liu, Ron Pope

Oak Ridge National Laboratory
Kevin Connolly, Matt Feldman, Josh Jarrell, Georgeta Radulescu,
John Scaglione, Alan Wells

Prepared by
Sandia National Laboratories
Albuquerque, New Mexico 87185 and Livermore, California 94550

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Sandia National Laboratories
P.O. Box 5800
Albuquerque, New Mexico 87185-0747

Abstract

The mission of the United States Department of Energy's Office of Environmental Management is to complete the safe cleanup of the environmental legacy brought about from five decades of nuclear weapons development and government-sponsored nuclear energy research. Some of the wastes that that must be managed have been identified as good candidates for disposal in a deep borehole in crystalline rock (SNL 2014a). In particular, wastes that can be disposed of in a small package are good candidates for this disposal concept. A canister-based system that can be used for handling these wastes during the disposition process (i.e., storage, transfers, transportation, and disposal) could facilitate the eventual disposal of these wastes. This report provides information for a program plan for developing specifications regarding a canister-based system that facilitates small waste form packaging and disposal and that is integrated with the overall efforts of the DOE's Office of Nuclear Energy Used Fuel Disposition Campaign's Deep Borehole Field Test.

Wastes to be considered as candidates for the universal canister system include capsules containing cesium and strontium currently stored in pools at the Hanford Site, cesium to be processed using elutable or nonelutable resins at the Hanford Site, and calcine waste from Idaho National Laboratory. The initial emphasis will be on disposal of the cesium and strontium capsules in a deep borehole that has been drilled into crystalline rock.

Specifications for a universal canister system are derived from operational, performance, and regulatory requirements for storage, transfers, transportation, and disposal of radioactive waste. Agreements between the Department of Energy and the States of Washington and Idaho, as well as the Deep Borehole Field Test plan provide schedule requirements for development of the universal canister system.

Future work includes collaboration with the Hanford Site to move the cesium and strontium capsules into dry storage, collaboration with the Deep Borehole Field Test to develop surface handling and emplacement techniques and to develop the waste package design requirements, developing universal canister system design options and concepts of operations, and developing system analysis tools. Areas in which further research and development are needed include material properties and structural integrity, in-package sorbents and fillers, waste form tolerance to heat and postweld stress relief, waste package impact limiters, sensors, cesium mobility under downhole conditions, and the impact of high pressure and high temperature environment on seals design.

Acknowledgments

This work was supported by the US Department of Energy Office of Environmental Management.

The authors gratefully acknowledge Laura Connolly (Sandia National Laboratories) for helping with document production and Robby Joseph (Oak Ridge National Laboratory) for support with the CURIE website. The authors would also like to thank those who reviewed this work: Steve Bellamy (Savannah River National Laboratory), Joel Case (US Department of Energy, Idaho Operations Office), Tim Gunter (US Department of Energy Office of Nuclear Energy), Julie Reddick (US Department of Energy, Richland Operations Office), Jim Shuler (US Department of Energy, Office of Packaging and Transportation), and Michael Swenson (Idaho National Laboratory). This report benefitted greatly from conversations with Bob MacKinnon and Ernie Hardin of Sandia National Laboratories. Steve Gomberg of the US Department of Energy Office of Environmental Management provided leadership, review, and oversight throughout.

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Nomenclature

ALARA	as low as reasonably achievable
ANL	Argonne National Laboratory
BUSS	Beneficial Uses Shipping System
CAP	Capsule Advisory Panel
CDSP	Capsule Dry Storage Project
CFR	Code of Federal Regulations
CNWRA	Center for Nuclear Waste Regulatory Analyses
CoC	Certificate of Compliance
CSSF	Calcine Solids Storage Facility
CST	crystalline silicotitanate
DBFT	Deep Borehole Field Test
DNFSB	Defense Nuclear Facilities Safety Board
DOE	Department of Energy
DOE M	DOE Manual
DOE O	DOE Order
DOT	Department of Transportation
EM	Office of Environmental Management
EPA	Environmental Protection Agency
HAC	hypothetical accident conditions
HIP	hot isostatic pressing
HLW	high-level radioactive waste
IAC	Idaho Administrative Code
IBC	in-bed combustion
ID	inside diameter
INL	Idaho National Laboratory
INTEC	Idaho Nuclear Technology and Engineering Center
LAW	low activity waste
LAWPS	Low Activity Waste Pretreatment System
NCT	normal conditions of transport
NE	Office of Nuclear Energy
NEPA	National Environmental Policy Act
NO_x	nitrogen oxides
NRC	Nuclear Regulatory Commission
NWCF	New Waste Calcining Facility
NWPA	Nuclear Waste Policy Act
OD	outside diameter

ORNL	Oak Ridge National Laboratory
RCRA	Resource Conservation and Recovery Act
ROD	Record of Decision
SBW	sodium-bearing waste
SCC	stress corrosion cracking
SCW	special case waste
SNF	spent nuclear fuel
SNL	Sandia National Laboratories
sRF	spherical resorcinol formaldehyde
SS	stainless steel
US	United States
WAC	Washington Administrative Code
WBS	work breakdown structure
WCF	Waste Calcining Facility
WESF	Waste Encapsulation and Storage Facility
WTP	Waste Treatment and Immobilization Plant

1 Introduction and Background

The mission of the United States (US) Department of Energy's (DOE) Office of Environmental Management (EM) is to complete the safe cleanup of the environmental legacy brought about from five decades of nuclear weapons development and government-sponsored nuclear energy research. The wastes produced over these decades vary greatly in their characteristics, such as radionuclide inventory, thermal output, physical dimensions, physical properties, chemical reactivity, etc. However, some of the wastes have been identified as good candidates for disposal in a deep borehole in crystalline rock (SNL 2014a). In particular, wastes that can be disposed of in a small package are good candidates for this disposal concept. DOE–EM has determined that a canister-based system that can be used for handling these wastes during the disposition process (i.e., storage, transfers, transportation, and disposal) could facilitate the eventual disposal of these wastes. This report provides information for a program plan for developing specifications regarding a canister-based system that facilitates small waste form packaging and disposal and that is integrated with the overall efforts of the DOE's Office of Nuclear Energy (NE) Used Fuel Disposition Campaign's Deep Borehole Field Test (DBFT).

The remainder of Section 1 presents the objective, purpose, and scope of this report. The subsequent sections discuss the wastes to be considered as candidates for the canister being described herein (Section 2); the applicable waste management functions (Section 3); the technologies that are already available, potentially will be available, or have been proposed in the past (Section 4); possible system concepts (Section 5); key requirements (Section 6); risks and technical challenges (Section 7), and a summary, including future tasks for the near term and research and development needs (Section 8).

1.1 Objectives

The long-term objective of the Universal Canister Project is to develop a universal canister system concept for disposal of small waste forms. It is universal in the sense that it can be used for multiple waste forms; it can be used for storage, transfers, transportation, and disposal; and it can be used for multiple disposal systems (e.g., deep borehole and mined repository). Given the flexibility that might be required of the universal canister system, it may be that, ultimately, variations of universal canisters will be designed and built. For example, some canisters could have different dimensions, and some may have different internal arrangements, such as baskets or inserts, for the purpose of accommodating a particular waste.

A systems engineering approach will be used to identify the specific functions that the universal canister must be able to perform and the requirements that it must meet in performing those functions. It is anticipated that the process of preliminary/final specification development will be used to define and procure the family of universal canisters.

1.2 Purpose

The purpose of this report is to lay the groundwork for developing specifications for a canister system that would enable disposal of small waste forms in a deep borehole—specifications that are also suitable for procuring canisters and associated overpacks and casks from qualified suppliers. To that end, this report (1) describes available technologies for storage, transfers, transportation, and disposal; (2) identifies key requirements, parameters, and components for storage, transfers, transportation, and disposal; (3) identifies some system concepts; (4) identifies important interfaces, risks and technical challenges going forward; and (5) identifies near-term and longer-term activities that should be initiated to enable decision making to facilitate deep borehole disposal.

1.3 Scope

The wastes to be considered as candidates for the universal canister system that is the subject of this report include capsules containing strontium (Sr) and cesium (Cs) at the Waste Encapsulation and Storage Facility (WESF) at the Hanford Site, Cs to be processed using elutable or nonelutable resins from the Waste Treatment and Immobilization Plant (WTP) at the Hanford Site, and calcine waste from Idaho National Laboratory (INL). These wastes are described in more detail in Section 2. However, consistent with a phased approach to develop the universal canister system, this report focuses on disposal of the Cs and Sr capsules. This focus is appropriate because disposal of the Cs and Sr capsules is part of two milestones, M-092-00 and M-092-05, under the Tri-Party Agreement Action Plan (HFFACO 2015) for the Hanford Site; these milestones are described in more detail in Table 4-1. Of equal importance for this report is that the universal canisters, as part of the universal canister system, be capable of storing waste, transferring waste, transporting waste, and disposing of waste without the canisters having to be opened again and having their contents re-packaged. Therefore, the requirements associated with these four waste management functions (i.e., storage, transfers, transportation, and disposal) are identified in this report.

With respect to disposal concepts considered, the initial emphasis is on disposal of these wastes in a deep borehole that has been drilled into crystalline rock. However, the design of the universal canister system should not preclude disposal in a mined geologic repository. Available and proposed disposal technologies are discussed in greater detail in Section 4.

In preparing the groundwork for developing specifications for a universal canister system, the following assumptions are made:

- None of the waste considered is suitable for near-surface disposal and will be disposed of using one or more deep geologic disposal techniques.
- The universal canister may have an insert or fill materials for a specific waste type and still be considered “universal.”

- The universal canister may be designed to be compatible with multiple storage overpacks, each of which may be specific to a particular storage site.
- The universal canister may be designed to be compatible with multiple transfer overpacks.
- The universal canister may be designed to be compatible with multiple transportation overpacks, each of which may be specific to a particular mode of transportation.
- The universal canister may be designed to be directly disposed of or compatible with multiple disposal overpacks, each of which may be specific to a particular disposal geology.

1.4 Stakeholders and Their Responsibilities

Multiple organizations have a stake in the universal canister system. They are shown below, along with their responsibilities with respect to developing a universal canister system.

1.4.1 DOE Office of Environmental Management

DOE–EM has the responsibility to ensure the safe cleanup and ultimate disposition of nuclear waste at federal facilities. In this role, DOE–EM will design and procure universal canisters for its waste, manage the waste on-site, and place into transportation vehicles for off-site disposal in a deep borehole.

1.4.2 DOE Office of Nuclear Energy

As a part of its Used Fuel Disposition Campaign, the DOE–NE is investigating the possibility of using deep boreholes to dispose of spent nuclear fuel (SNF) and high-level radioactive waste (HLW), and is currently funding the DBFT project to test this idea. In this role, DOE–NE will be responsible for the following:

- Collaborate with the Universal Canister Project to develop and test the system for waste handling and emplacement operations
- Collaborate with the Universal Canister Project to design waste packages to be used for deep borehole disposal

1.4.3 Hanford

The Hanford Site has possession of all the Cs and Sr capsules under consideration in this report as well as the Cs to be processed using either elutable or nonelutable resins. In this role, Hanford will manage the Cs and Sr capsules in a manner that is compatible with developing a universal canister system in which the capsules may ultimately be stored, transported, and disposed of. Hanford is currently seeking to move the capsules into a new dry storage facility (DOE 2015b);

the Universal Canister Project will collaborate with the Hanford Site to develop a dry storage system that is compatible with the universal canister system.

1.4.4 Idaho National Laboratory

The INL has possession of all of the calcine waste that is under consideration in this report. In this role, INL will manage the calcine waste in a manner that is compatible with developing a universal canister system in which the calcine may ultimately be stored, transported, and disposed of.

1.4.5 Nuclear Regulatory Commission

The Nuclear Regulatory Commission (NRC) will provide the regulatory oversight for disposal of the waste contained in a universal canister; this is discussed further in Sections 6.2.3 and 7.5. The NRC has no regulatory authority over defense waste storage at DOE sites. Transportation packages will be certified to NRC requirements.

1.4.6 States of Washington and Idaho

The States of Washington and Idaho have agreements with the DOE regarding the DOE-managed wastes in their respective states, and have the authority to implement the requirements of the Resource Conservation and Recovery Act (RCRA) in their respective states. As such, the States will be responsible for the following:

- Working with the DOE to ensure that changes in the status of the DOE-managed wastes in their respective states are consistent with their agreements
- Reviewing and granting RCRA permits for storage of waste-filled universal canisters in a storage facility located in their respective states, assuming the requirements of the permit are met

1.4.7 Defense Nuclear Facilities Safety Board

The Defense Nuclear Facilities Safety Board (DNFSB) is an independent organization within the executive branch chartered with the responsibility of providing recommendations and advice to the President and the Secretary of Energy regarding public health and safety issues at DOE defense nuclear facilities. Consistent with its jurisdiction, the Board may choose to review the design of a new facility constructed for storage of the waste at the Hanford Site or at INL.

1.4.8 The Department of Energy

The DOE owns the wastes discussed in this report and is responsible for storing, transferring, transporting, and disposing of them. In addition, the DOE is self-regulating with respect to storage of these wastes and with respect to transfer of the wastes at DOE sites. The DOE is also allowed to evaluate, approve, and certify packagings consistent with packaging standards equivalent to those specified by the NRC in 10 CFR Part 71.

1.5 Definitions

Some of the terms used in this report are defined below. Note that these terms may be defined differently in other reports cited herein.

Confinement—Systems, including ventilation, that act as barriers between areas containing radioactive substances and the environment (10 CFR Part 72).

Containment—The assembly of components of the packaging intended to retain the radioactive material during transport (10 CFR Part 71).

Disposal Overpack—A container into which one or more waste-filled universal canisters may be placed for disposal. In conjunction with the canister, it is designed to withstand the chemical conditions, the pressure, and the temperature in the disposal environment for a specified period of time.

Package—The packaging together with its radioactive contents as presented for transport (10 CFR Part 71).

Packaging—The assembly of components necessary to ensure compliance with the packaging requirements of 10 CFR Part 71. It may consist of one or more receptacles, absorbent materials, spacing structures, thermal insulation, radiation shielding, and devices for cooling or absorbing mechanical shocks. The vehicle, tie-down system, and auxiliary equipment may be designated as part of the packaging (10 CFR Part 71).

Storage Cask—A container into which one or more waste-filled universal canisters are placed for dry storage. It is designed to provide confinement, shielding from radiation, and heat removal for a specified period of time under anticipated storage conditions. It can also be referred to as a “storage overpack.”

Transfer Cask—A container that is used for on-site transfer of one or more waste-filled universal canisters into and out of storage casks, transportation casks, and disposal overpacks. It is designed to provide heat removal and shielding from radiation during the relatively short period of time required to transfer the waste-filled canister. It can also be referred to as a “transfer overpack.”

Universal Canister (or “Canister”)—A metal container that is sealed after being filled with waste that is ultimately destined for disposal. It is not to be opened again; it provides containment of the waste for a specified period of time; and it is designed to be used with storage casks, transfer casks, transportation packaging, and disposal overpacks (as needed) to store, transfer, transport, and dispose of the waste contained therein.

Universal Canister System—System of components that includes the universal canister(s) and all needed overpacks or casks (e.g., storage, transfer, transportation, disposal) as well as any needed ancillary equipment (e.g., transportation impact limiters).

Waste Package—The waste-filled assembly of components to be emplaced in a deep borehole or mined geologic repository. It may consist of a universal canister that can itself withstand the chemical, thermal, and pressure conditions in the disposal environment, or it may consist of a universal canister in a disposal overpack that can withstand the chemical, thermal, and pressure conditions in the disposal environment. It includes the waste and any container, shielding, packing, and other absorbent materials immediately surrounding an individual waste container.

2 Wastes Considered

As mentioned above, the primary wastes to be considered as candidates for the universal canister system are the capsules containing Cs and Sr at the Hanford Site. Cs extracted from elutable resins or bound to nonelutable resins from the Hanford Site as well as calcine waste from INL are also potential candidates for deep borehole disposal. All three wastes are described in more detail below.

2.1 Cs and Sr Capsules

At the Hanford Site, the process of producing material for nuclear weapons created a substantial amount of HLW. These process wastes contained Cs and Sr, which generated a significant amount of the heat associated with the process waste and were also responsible for much of the radioactivity of the waste. In 1957, a program was initiated to remove the Cs and Sr from the process wastes (Geier 1981). Removal of these two heat-producing elements provided a 90% reduction in heat generation in neutralized waste from one-year-old fuel, thus reducing the temperature in the waste tanks (Hedquist 1997). The Cs and Sr were purified, converted to cesium chloride (CsCl) and strontium fluoride (SrF_2), and placed in capsules, which are currently stored in pools in the WESF. There are a total of 1,936 capsules: 1,335 capsules of CsCl and 601 capsules of SrF_2 (Covey 2014).

Both the Cs capsules and the Sr capsules are doubly encapsulated (i.e., a capsule within a capsule). In the case of the Cs capsules, both the inner capsule and the outer capsule are made of 316L stainless steel with welded caps. In the case of the Sr capsules, the inner capsule is made of Hastelloy C-276 and, for most capsules, the outer capsule is made of 316L stainless steel. Some of the initial Sr outer capsules were made of Hastelloy C-276, rather than 316L stainless steel (Bath et al. 2003). In addition, 23 of the 1,335 Cs capsules were overpacked in Type W capsules because (1) they were suspected of failing or had failed, (2) they had undergone destructive testing, or (3) otherwise did not have both a WESF inner capsule and a WESF outer capsule (Covey 2014). Nominal dimensions for the capsules are shown in Table 2-1. Figure 2-1 shows a typical Cs capsule while Figure 2-2 shows a typical Sr capsule. Further information regarding the contents and construction of the Type W overpacks is given in Table 2-2.

The Cs capsules were filled by pouring molten CsCl into each inner capsule, welding the inner cap to the inner capsule, performing a helium leak check, decontaminating the outside surface of the inner capsule, placing it in the outer capsule, and then welding the outer cap to the outer capsule. The second weld was checked ultrasonically, and the Cs content of the capsule was determined calorimetrically. Destructive testing of some of the Cs capsules indicates that, because the salt was molten when it was poured into the inner capsule, there is a shrinkage hole at the top of the capsule (shaped like an inverted right circular cone) and the salt is very hard, making it difficult to obtain samples via scraping (Sasmor et al. 1988). The Sr capsules were filled and sealed in a similar fashion, except that the waste was in the form of a dried granular material that was mechanically compacted into the inner capsule.

The Cs capsules were limited to 8.00×10^4 Ci of ^{137}Cs upon loading, which is equivalent to a heat generation rate of about 380 W (Geier 1981). The SrF_2 capsules were limited to 1.70×10^5 Ci of ^{90}Sr at the time of loading, which is equivalent to a heat generation rate of about 1,020 W (Geier 1981). The radioactivity of each capsule varied; statistics for the radioactivity, the heat generation, and the unshielded dose rate of the capsules as of January 1, 2016 are shown in Table 2-3. The capsules contain 4.90×10^7 Ci of Cs and Sr, and another 4.70×10^7 Ci of their respective daughter products, ^{137m}Ba and ^{90}Y (as of January 1, 2016).

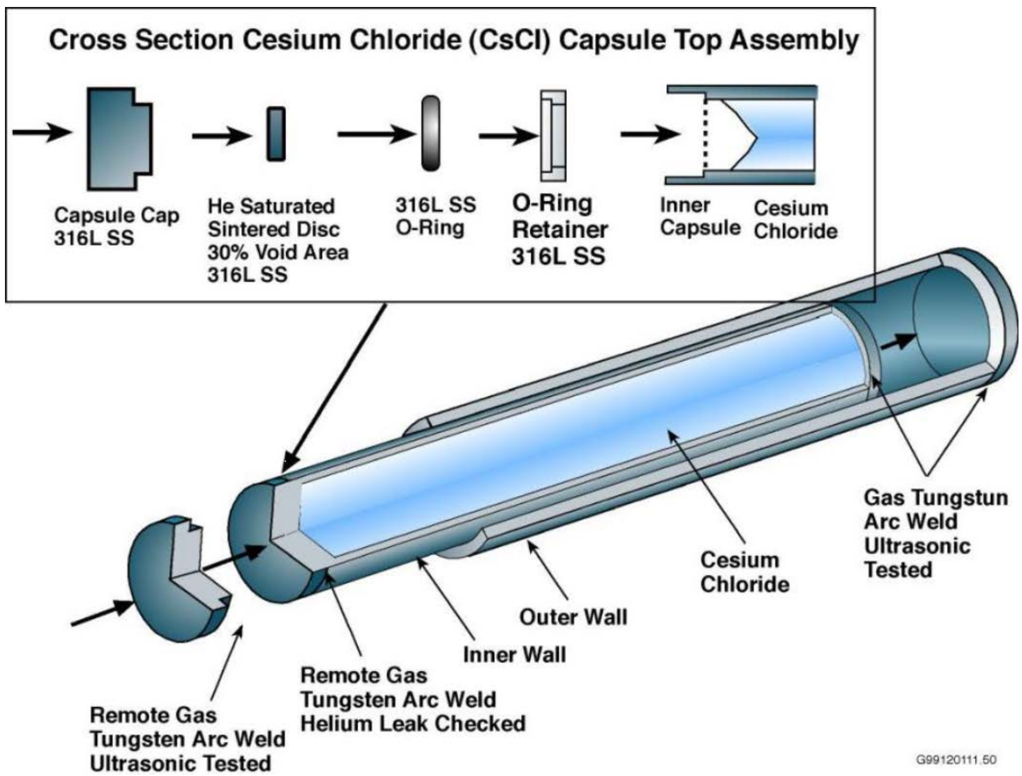
Table 2-1. Material and Nominal Dimensions for Cs and Sr Capsules

Item	Containment Boundary	Material	Wall Thickness[a] (in.)	OD (in.)	Total Length (in.)	Cap Thickness (in.)
Cs Capsule	Inner	316L Stainless Steel	0.095 0.103 0.136	2.250 2.250 2.255	19.725	0.4
	Outer	316L Stainless Steel	0.109 0.119 0.136	2.625 2.645 2.657	20.775	0.4
Cs Type W Overpack	Single	316L Stainless Steel	0.125	3.25	21.825	0.4
Sr Capsule	Inner	Hastelloy C-276	0.120 0.136	2.250	19.05	0.4
	Outer	316L Stainless Steel or Hastelloy C-276	0.109 0.119 0.120 0.136	2.625	20.1	0.4

NOTE: [a] The specified wall thickness of the Cs capsules was increased twice during production. The capsules are referred to as Type 1, Type 2, and Type 3, with Type 3 being the most numerous (Heard et al. 2003).

OD = outside diameter.

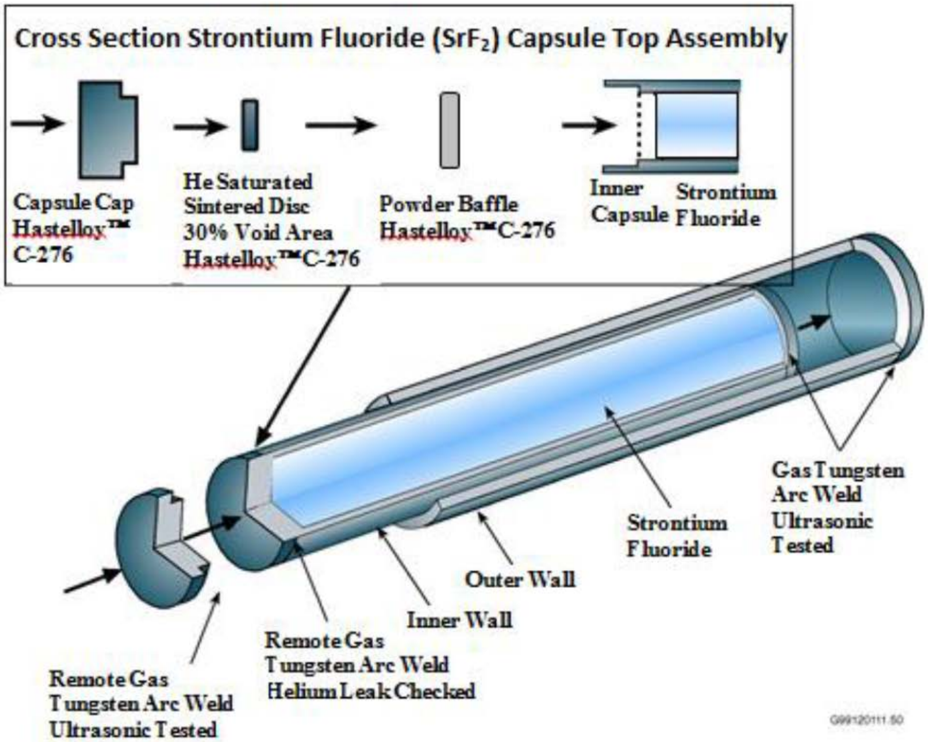
Source: Heard et al. 2003 and Fluor Hanford undated.



NOTE: SS = stainless steel.

Source: Covey 2014.

Figure 2-1. Typical Cs Capsule



Source: Covey 2014.

Figure 2-2. Typical Sr Capsule

Table 2-2. Type W Cs Capsules and Contents

Capsule Contents	Inner Container	Outer Container	Number of Type W Capsules
10 Nordian™ Pencils from ORNL, each Containing CsCl Originating from WESF	Threaded Inner	WESF Outer	1
CsCl Powder and/or Pellets from ORNL	Threaded Inner	WESF Outer	2
304L Stainless Steel Type 4 Containers from ORNL Containing CsCl Originating from WESF	ORNL Type 4 Inner	ORNL Type 4 Outer	1
Remnants from Destructive Testing of WESF Capsules	No	WESF Outer	3
Swollen WESF Capsules Returned from Commercial Irradiators	WESF Inner	WESF Outer	16
Total			23

NOTE: ORNL = Oak Ridge National Laboratory.
WESF = Waste Encapsulation and Storage Facility.

Source: Josephson 2004.

Table 2-3. Radioactivity, Heat Generation, and Dose Rate Characteristics of Cs and Sr Capsules as of January 1, 2016

Capsule Type	Number of Capsules	Statistical Parameter	Power (W)	Cs or Sr Activity (Ci)	Surface Dose Rate[b] (rem/h)	Dose Rate at 3 ft from Capsule[b] (rem/h)
CsCl	1335	Average	118.6	2.51×10^4	6.34×10^5	4.81×10^3
		Standard Deviation	11.6	2.5×10^3	6.31×10^4	4.79×10^2
		Minimum	13	2.8×10^3	7.07×10^4	5.37×10^2
		Maximum	161	3.42×10^4	8.63×10^5	6.56×10^3
SrF_2	600[a]	Average	157.1	2.35×10^4	2.92×10^4	6.50×10^2
		Standard Deviation	82.4	12.3×10^3	1.53×10^4	3.40×10^2
		Minimum	18	2.7×10^3	3.36×10^3	7.46×10^1
		Maximum	411	6.14×10^4	7.64×10^4	1.70×10^3

NOTE: [a] Does not include one SrF_2 capsule that is a tracer and contains no radioactive Sr and, thus, emits no heat.

[b] Dose rate estimations performed at ORNL.

Cs has several radioactive isotopes, but only ^{135}Cs and ^{137}Cs are of concern for storage, transfers, transportation, and disposal because the other isotopes have such short half-lives (on the order of days, or at most 2 years) that they have already decayed into stable isotopes. The half-life of ^{135}Cs is 2,300,000 years, making it of concern primarily for long-term performance of the disposal system. The half-life of ^{137}Cs is 30.17 years, making it of concern for storage, transfers, transportation, and the preclosure and handling phases of disposal. ^{135}Cs decays via beta emission to ^{135}Ba, which is stable. About 95% of the ^{137}Cs decays via beta emission to ^{137m}Ba, which has a half-life of 2.5 minutes and decays to stable ^{137}Ba via isomeric transition, thereby emitting a gamma ray. The remaining 5% of the ^{137}Cs decays directly to stable ^{137}Ba.

Sr also has several radioactive isotopes, but only ^{90}Sr is of concern for storage, transfers, transportation, and the preclosure and handling phases of disposal of radioactive waste because the other isotopes have half-lives on the order of hours or days and have already decayed into stable isotopes of other elements. ^{90}Sr has a half-life of 29.1 years and beta decays to ^{90}Y, which has a half-life of 64 hours and beta decays to stable ^{90}Zr.

The material placed into the capsules contained other chemical constituents in addition to the CsCl and SrF_2. Because of the presence of these other constituents, some of which are considered hazardous, all the capsules are considered to be mixed waste by the State of Washington (Washington Department of Ecology 2008). The composition of the contaminants in the Cs capsules is given in Table 2-4, and the composition of the contaminants in the Sr capsules is given in Table 2-5. The weight percent of ^{135}Cs (i.e., mass of ^{135}Cs/mass of all Cs isotopes) in a given capsule ranged from 11.1% to 14.15% at the time of measurement (Bryan et al. 2003; Sasmor et al. 1988), and the best estimate of the weight percent of impurities in a capsule ranges from 5.3% to 19.6% (Sasmor et al. 1988).

Table 2-4. Cs Capsule Contaminant Composition

Element	Composition Range (wt%)	Compound	Composition Range (wt%)
Al	ND–0.3	$AlCl_3$	ND–0.5
B	ND–0.4	B_2O_3	ND–0.13
Ba[a]	ND–6.5	$BaCl_2$[a]	4.7–9.9
Ca	ND–0.19	$CaCl_2$	0.03–0.53
Cd	ND–0.02	$CeCl_3$	ND–0.02
Ce	ND–0.01	$CrCl_3$	0.1–1.5
Co	0.02–0.10	$FeCl_3$	0.1–4.7
Cr	0.02–1.4	KCl	ND–1.3
Cu	ND–0.2	$LaCl_3$	ND–0.02
Fe	0.04–1.6	$MgCl_2$	ND–0.2
K	ND–0.7	$MnCl_4$	ND–0.32
La	ND–0.01	$MoCl_3$	ND–0.55
Mg	ND–0.05	NaCl	0.1–3.8
Mn	ND–0.09	$NiCl_2$	ND–3.32
Mo	ND–0.26	$SiCl_4$	ND–1.2
Na	0.04–2.8	$SrCl_2$	ND–0.01
Ni	ND–1.5	$TiCl_4$	ND–0.2
P	ND–0.1	$ZrCl_2$	ND–0.07
Pb	ND–0.14	—	—
Pd	ND–0.02	—	—
Rb	ND–0.02	—	—
S	5	—	—
Si	ND–5	—	—
Sr	ND–0.02	—	—
Ti	ND–0.07	—	—
Zr	0.07–0.08	—	—

NOTE: [a] Ba at time of analysis; it will increase with Cs decay.
ND = not detected.

Source: Elemental composition from Bryan et al. 2003; compound composition from Sasmor et al. 1988.

Table 2-5. Sr Capsule Contaminant Composition

Element	Composition Range (wt%)
Al	< 0.5
Ba	0.1–2.0
Ca	0.1–2.0
Cd	<0.1
Cr	<0.2–0.32
Cu	<0.01
F	30
Fe	<0.1–0.41
H	<0.01
K	<0.01
Mg	0.03–0.5
Mn	<0.1
Na	1.0–4.0
Ni	<0.1–0.26
O	<0.05
Pb	<0.02
Rare Earth	<2.0
Si	<0.02
Zr[a]	0.8

NOTE: [a] Zr at time of analysis; it will increase with Sr decay.

Source: Josephson 2004.

2.2 Cs To Be Processed using Elutable or Nonelutable Resins

At the Hanford Site, part of the proposed plan for retrieving and treating the tank waste is to develop the Low Activity Waste Pretreatment System (LAWPS) that will separate the HLW components from selected tank wastes and create a low activity waste (LAW) stream that will be vitrified in the WTP LAW Vitrification Facility. The LAWPS is intended to remove the solids that do not meet the WTP LAW facility waste acceptance criteria and to remove Cs from the waste. The current preferred alternative for removing Cs from the waste is to use an ion-exchange resin (Ramsey and Thorson 2010).

To separate Cs from the tank waste, ion-exchange resins are placed in a flow-through column and the waste is pumped through the column. The Cs in the flowing waste stream selectively sorbs to the ion-exchange medium, thereby removing the Cs from the waste stream. Resins are categorized as being either elutable or nonelutable. Elutable resins can be used multiple times in an ion-exchange column. When the ion-exchange sites on an elutable resin are "full" of Cs, the resin is treated with appropriate solvents to remove the Cs and regenerate the resin, and then put back in the ion-exchange column to be used again. In contrast, a nonelutable resin cannot be used again; once it is fully loaded with Cs, it is removed from the column and becomes a waste that needs to be treated and disposed.

The current preferred ion-exchange medium for the LAWPS is spherical resorcinol formaldehyde (sRF) (Ramsey and Thorson 2010), which is an elutable resin in the form of spherical beads that are about 400 μm in diameter. Once the ion-exchange sites on the sRF beads are full, the beads are washed with dilute nitric acid to remove the Cs, regenerated with a caustic solution, and then treated with a polished LAW feed (e.g., LAW waste feed that has had Cs removed via ion exchange). The beads swell and shrink as Cs is sorbed and de-sorbed, respectively. Cs that is removed from the beads is returned to the tanks with waste slated to be vitrified and disposed of as HLW. Eventually, the beads can no longer be regenerated (i.e., they are spent); the current plan for spent sRF beads is to place them in high integrity canisters and dispose of them in Hanford's Integrated Disposal Facility. Therefore, the elutable sRF resin that is currently the preferred alternative for removing Cs from the tank waste is not being considered for disposal using a universal canister system at the present time. While the treated Cs is a candidate for a universal canister system, the current preferred alternative calls for the Cs that is removed from the resin to be returned to HLW tanks to be vitrified instead. Quantities and characteristics of this possible Cs waste are not known.

Should the preferred alternative for removing Cs from the tank waste change to using a nonelutable resin, the Cs-filled spent resin would be a potential candidate for a universal canister system. The resin would sorb other radionuclides that are in the waste stream such as Sr, potassium, and rubidium, in addition to Cs, and the inventory of those other radionuclides would have to be considered. It should be noted that the Savannah River Site is planning on removing Cs from its waste stream using a nonelutable resin—crystalline silicotitanate (CST)—in an ion-exchange column. At Savannah River, the spent resin will be ground and processed later in their Defense Waste Processing Facility. Quantities and characteristics of this possible resin waste are not known.

2.3 Calcine Waste

At the Idaho Nuclear Technology and Engineering Center (INTEC), located at INL in southeastern Idaho, SNF was reprocessed to recover enriched uranium and other radionuclides. 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).

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 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 NO_x. 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. The can shrinks around the glass ceramic and the interstitial voids in the mixture collapse. A volume reduction of approximately 35% is expected. After the HIP process, the compressed cans will be placed in canisters measuring 5.5-ft diameter × 17-ft tall, presently certified for SNF (CDP 2012). With the volume of each HIP can being reduced approximately 30%, each canister could hold 10 HIP-processed cans. Voids in the canister will be filled with sand, steel shot, or glass shot before being sealed. The glass ceramic would have properties consistent with HLW borosilicate glass. The main minerals in the glass ceramic are titanates, sulfides, glass/quartz, and nepheline (CDP 2012).

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% (Hagers 2007). This alternative calcine waste form would include RCRA waste constituents and would be acceptable for disposal at a facility that accepts RCRA wastes.

The current HIP process creates waste forms that are too large for deep borehole disposal. However, should the preferred alternative for calcine waste treatment and disposal change to direct disposal in a mined repository or deep borehole, the untreated calcine would be a candidate waste form for the universal canister system.

2.3.1 Chemical Characteristics of the Calcine Waste

Chemically, there is variability in the composition of the calcine among the CSSFs, among the bins within a CSSF, and even within an individual bin because of the sequence in which the fuel was reprocessed and then calcined. According to 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 bins also contain dolomite ($CaMg(CO_3)_2$) and fluorapatite ($Ca_{10}(PO_4)_6F_2$), which are present because they were used as materials for startup beds for the calciners. Table 2-6 provides a summary of the chemical composition of the four types of calcine.

Table 2-6. Typical Compositions of the Four Different 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.

Table 2-7 provides details on the chemical inventory by element or chemical species for each CSSF. Most of the solids formed in the calcination process were nonradioactive oxides of aluminum and zirconium from the fuel cladding. A small amount of these metals formed metallic chlorides, phosphates, and sulfates with the small quantities of those anions that were present in the liquid waste. Calcium, in the form of calcium nitrate, was added to fluoride-bearing wastes as part of the calcination process to form calcium fluoride and suppress the volatility of fluorine, which would have been highly corrosive to the calciner off-gas system. Other metals are present either because they were in the cladding in trace quantities or they were introduced as an additive or as an associated trace contaminant in the fuel reprocessing or calcination processes. Details on the content and source(s) of the elements listed in Table 2-7 are available in Staiger and Swenson (2011).

Table 2-8 summarizes the calcine radioactivity by isotope for each of the six CSSFs (Staiger and Swenson 2011). As is the case with chemical composition of the calcine described above, the radionuclide concentrations and associated activities vary among the CSSFs and within individual bins. During fuel reprocessing more than 99.9% of the fission products were separated from the uranium in the first-cycle extraction system and went with the first-cycle raffinate to the Tank Farm. The fission-product activity of this raffinate was primarily a function of uranium burnup and the age of the waste. Fuels with high uranium burnup had higher fission-product activity than fuels with low burnup. Radionuclides with short half-lives (such as ^{95}Zr and ^{144}Ce) varied significantly among first-cycle raffinates depending on the fuel cooling time and age of the waste. However, even the “newest” calcine in the CSSFs came from fuel reprocessed more than 20 years ago, so the activity of short-lived fission products has decreased to insignificant levels in all calcine, and the combined activity of ^{137}Cs/^{137m}Ba and ^{90}Sr/^{90}Y accounts for more than 99% of the current fission product activity in calcine.

Table 2-9 summarizes the RCRA hazardous waste content of the calcine. The RCRA metals are found in calcine in varying concentrations. Some, such as arsenic, were not a component of any fuel or used in any fuel or waste processing system. Such species did not exist in reliably detected concentrations in either liquid or calcine waste. Cadmium, chromium, and mercury were process additives for the different flowsheets used during the fuel reprocessing mission. Nickel and chromium were components of some of the alloys used as fuel cladding. Lead is present from dissolved shielding. Other hazardous waste species (silver, arsenic, barium, and selenium) are present in trace amounts, primarily as fission products from the reprocessed fuel. The State of Idaho considers the waste to be hazardous waste and has thus issued a RCRA Part B Permit for the CSSFs (Idaho Department of Environmental Quality 1995).

Table 2-7. Chemical Inventory in Each of the Six CSSFs

1/1/2016	CSSF I	CSSF II	CSSF III	CSSF IV	CSSF V	CSSF VI	Total
Element/ Chemical Species	**(kg)**	**(kg)**	**(kg)**	**(kg)**	**(kg)**	**(kg)**	**(kg)**
Al	8.68E+04	2.28E+05	1.54E+05	6.29E+04	1.52E+05	2.77E+05	9.60E+05
B	2.29E+02	5.62E+03	1.09E+04	5.57E+03	1.17E+04	4.10E+03	3.82E+04
Ca	—[a]	1.84E+05	3.81E+05	1.88E+05	3.46E+05	6.79E+04	1.17E+06
Cd	4.91E-01	9.33E-01	9.81E-01	5.72E-01	4.06E+04	5.60E+03	4.62E+04
Cr	1.14E+02	2.00E+03	3.71E+03	1.90E+03	1.94E+03	1.12E+03	1.08E+04
Cs	5.61E+01	1.09E+02	1.15E+02	6.74E+01	1.44E+02	3.79E+01	5.30E+02
Fe	1.54E+03	2.59E+03	3.79E+03	3.00E+03	5.90E+03	5.46E+03	2.23E+04
Hg	3.43E+03	7.19E+03	1.74E+01	1.15E+01	2.78E+01	2.77E+01	1.07E+04
K	4.00E+02	1.46E+03	3.27E+03	2.50E+03	8.57E+03	1.26E+04	2.88E+04
Mg	6.02E+02	5.94E+03	1.25E+04	3.00E+03	9.84E+03	6.80E+03	3.86E+04
Mn	5.07E+01	6.27E+02	1.51E+03	6.37E+02	1.78E+03	1.58E+03	6.19E+03
Mo	1.01E+02	1.96E+02	2.11E+02	1.24E+02	2.61E+02	6.83E+01	9.61E+02
Na	2.41E+03	9.24E+03	1.41E+04	1.02E+04	3.63E+04	4.71E+04	1.19E+05
Nb	6.71E-04	7.57E+00	1.97E+01	1.23E+01	2.86E+03	6.06E+00	2.90E+03
Nd	1.26E+02	2.33E+02	2.33E+02	1.35E+02	2.92E+02	7.76E+01	1.10E+03
Ni	—	2.16E+02	6.14E+02	4.80E+02	7.81E+02	4.98E+02	2.59E+03
Sn	1.02E+00	1.75E+03	3.13E+03	1.38E+03	2.27E+03	2.55E+02	8.79E+03
Sr	1.84E+01	1.95E+03	3.90E+03	1.93E+03	3.60E+03	2.32E+02	1.16E+04
Zr	1.38E+02	1.11E+05	1.98E+05	8.73E+04	1.43E+05	1.60E+04	5.55E+05
Cl	6.88E+01	3.86E+02	9.24E+02	7.36E+02	1.95E+03	1.73E+03	5.80E+03
F	—	1.62E+05	2.77E+05	1.27E+05	2.17E+05	2.75E+04	8.11E+05
CO_3	—	1.08E+04	2.53E+04	4.56E+03	1.85E+04	1.64E+04	7.56E+04
NO_3	4.70E+03	1.80E+04	2.86E+04	2.07E+04	7.35E+04	8.43E+04	2.30E+05
PO_4	2.81E+03	9.97E+03	2.39E+04	5.07E+03	1.23E+04	2.39E+03	5.65E+04
SO_4	3.42E+03	2.10E+04	3.12E+04	1.39E+04	3.94E+04	9.01E+03	1.18E+05
Trace FP	6.15E+02	1.08E+03	1.04E+03	5.69E+02	1.37E+03	5.34E+02	5.21E+03
U	1.42E+01	2.79E+01	1.69E+01	3.42E+01	1.84E+02	2.14E+02	4.91E+02
O	7.82E+04	2.59E+05	2.63E+05	1.22E+05	2.67E+05	2.86E+05	1.28E+06
Total	1.86E+05	1.04E+06	1.44E+06	6.63E+05	1.40E+06	8.75E+05	5.61E+06

NOTE: [a] A dash within a cell indicates an insignificant quantity.

CSSF = Calcine Solids Storage Facility.

Source: Staiger and Swenson 2011.

Table 2-8. Calcine Radioactivity Decayed to January 1, 2016

1/1/2016 Radionuclide	CSSF I (Ci)	CSSF II (Ci)	CSSF III (Ci)	CSSF IV (Ci)	CSSF V (Ci)	CSSF VI (Ci)	Total (Ci)
^{60}Co	3.82E-01	2.24E+01	4.05E+01	3.48E+01	7.10E+02	1.09E+02	9.18E+02
^{63}Ni	0.00E+00	1.09E+03	2.79E+03	1.83E+03	3.19E+03	5.52E+02	9.45E+03
^{79}Se	2.72E+00	5.02E+00	5.00E+00	2.91E+00	6.27E+00	1.67E+00	2.36E+01
^{90}Sr	6.72E+05	1.49E+06	1.58E+06	9.94E+05	2.13E+06	5.35E+05	7.40E+06
^{90}Y	6.72E+05	1.49E+06	1.58E+06	9.94E+05	2.13E+06	5.35E+05	7.40E+06
^{99}Tc	4.25E+02	7.68E+02	7.41E+02	4.28E+02	9.33E+02	2.49E+02	3.54E+03
^{106}Ru	1.48E-10	9.20E-09	5.50E-07	5.16E-06	3.61E-04	1.84E-04	5.51E-04
^{125}Sb	1.11E-01	7.23E+00	3.39E+00	6.05E+00	3.48E+01	1.22E+01	6.38E+01
^{126}Sn	1.10E+01	2.02E+01	2.02E+01	1.17E+01	2.53E+01	6.76E+00	9.51E+01
^{129}I	6.88E-03	1.25E-02	1.22E-02	7.08E-03	1.54E-02	4.11E-03	5.82E-02
^{134}Cs	5.04E-03	3.73E-01	8.35E-01	2.52E+00	3.68E+01	9.10E+00	4.96E+01
^{135}Cs	1.07E+01	2.61E+01	3.51E+01	2.12E+01	4.26E+01	1.07E+01	1.46E+02
^{137m}Ba	7.66E+05	1.59E+06	1.74E+06	1.01E+06	2.18E+06	5.86E+05	7.88E+06
^{137}Cs	8.09E+05	1.68E+06	1.84E+06	1.07E+06	2.30E+06	6.19E+05	8.33E+06
^{144}Ce	4.13E-14	8.14E-12	1.53E-09	2.49E-08	9.59E-06	4.31E-06	1.39E-05
^{144}Pr	4.13E-14	8.14E-12	1.53E-09	2.49E-08	9.59E-06	4.31E-06	1.39E-05
^{147}Pm	2.96E+00	5.16E+01	7.26E+01	3.85E+01	9.59E+01	3.64E+01	2.98E+02
^{151}Sm	1.64E+04	2.43E+04	1.35E+04	6.85E+03	1.90E+04	5.94E+03	8.60E+04
^{152}Eu	6.92E+00	3.95E+01	6.62E+01	3.97E+01	8.06E+01	2.14E+01	2.54E+02
^{154}Eu	4.31E+02	2.39E+03	2.14E+03	1.99E+03	6.62E+03	1.42E+03	1.50E+04
^{155}Eu	2.13E+01	1.14E+02	1.26E+02	1.31E+02	5.41E+02	1.53E+02	1.09E+03
^{230}Th	1.01E-01	1.22E-01	6.15E-03	1.17E-03	7.24E-02	2.90E-02	3.31E-01
^{231}Th	2.06E-02	3.97E-02	1.95E-02	1.61E-02	8.72E-02	7.31E-02	2.56E-01
^{233}Pa	1.09E+00	1.76E+00	7.83E+00	1.95E+01	3.70E+01	5.43E+00	7.26E+01
^{232}U	8.02E-05	8.82E-03	9.80E-02	6.93E-02	9.68E-02	1.35E-02	2.86E-01
^{233}U	1.57E-04	2.37E-04	1.27E-03	3.12E-03	5.35E-03	7.29E-04	1.09E-02
^{234}U	2.96E+00	6.67E+00	2.00E+00	1.81E+00	7.10E+00	3.00E+00	2.35E+01
^{235}U	2.06E-02	3.97E-02	1.95E-02	1.61E-02	8.72E-02	7.31E-02	2.56E-01
^{236}U	4.78E-02	1.01E-01	5.02E-02	4.38E-02	2.70E-01	1.63E-01	6.76E-01
^{237}U	3.79E-03	1.19E-01	2.47E-01	1.50E-01	2.95E-01	7.83E-02	8.93E-01
^{238}U	1.17E-03	2.26E-03	2.37E-03	8.68E-03	4.72E-02	5.74E-02	1.19E-01
^{237}Np	1.09E+00	1.76E+00	7.83E+00	1.95E+01	3.70E+01	5.43E+00	7.26E+01
^{238}Pu	3.16E+02	8.10E+03	1.66E+04	1.65E+04	3.23E+04	4.99E+03	7.88E+04
^{239}Pu	4.27E+01	1.82E+02	4.41E+02	5.09E+02	8.87E+02	3.34E+02	2.40E+03
^{240}Pu	1.71E+01	1.44E+02	3.21E+02	3.30E+02	6.24E+02	1.80E+02	1.62E+03

Table 2-8. Calcine Radioactivity Decayed to January 1, 2016 (continued)

1/1/2016	CSSF I	CSSF II	CSSF III	CSSF IV	CSSF V	CSSF VI	Total
Radionuclide	(Ci)	(Ci)	(Ci)	(Ci)	(Ci)	(Ci)	(Ci)
^{241}Pu	1.19E+02	4.05E+03	8.57E+03	6.01E+03	1.71E+04	4.90E+03	4.07E+04
^{242}Pu	9.86E-03	3.35E-01	8.18E-01	8.69E-01	1.48E+00	3.76E-01	3.89E+00
^{241}Am	1.22E+02	1.13E+03	2.48E+03	1.54E+03	2.87E+03	4.01E+02	8.55E+03
^{243}Am	8.65E-03	8.38E-02	3.12E-01	1.96E-01	3.43E-01	1.12E-01	1.06E+00
^{242}Cm	7.22E-03	2.37E-01	5.09E-01	3.07E-01	6.06E-01	1.74E-01	1.84E+00
^{244}Cm	1.28E-02	8.87E-01	2.43E+00	1.49E+00	2.79E+00	8.44E-01	8.45E+00

NOTE: CSSF = Calcine Solids Storage Facility.

Source: Staiger and Swenson 2011.

Table 2-9. RCRA Metal Content of INL Calcines

Metal	CSSF I (kg)	CSSF II (kg)	CSSF III (kg)	CSSF IV (kg)	CSSF V (kg)	CSSF VI (kg)	Total
Ag	3	14	16	7	19	12	71
As	3	12	13	6	13	4	49
Ba	67	154	186	103	215	50	775
Cd	0.5	1	1	1	40,609	5,605	46,217
Cr	114	2,004	3,707	1,900	1,940	1,122	10,788
Hg	3,425	7,185	17	11	28	28	10,695
Ni[a]	0	216	614	480	781	498	2,588
Pb	12	23	78	79	270	571	1,033
Se	2	3	3	2	4	1	16
Metal	**CSSF I (ppm)**	**CSSF II (ppm)**	**CSSF III (ppm)**	**CSSF IV (ppm)**	**CSSF V (ppm)**	**CSSF VI (ppm)**	
Ag	17	14	11	11	13	17	
As	16	11	9	9	9	4	
Ba	360	148	130	156	154	57	
Cd	3	1	1	1	29,117	6,412	
Cr	614	1,924	2,577	2,873	1,391	1,284	
Hg	18,444	6,896	12	17	20	32	
Ni	0	207	427	725	560	570	
Pb	63	22	54	119	194	653	
Se	10	3	2	3	3	1	

NOTE: [a] Nickel is an underlying hazardous constituent in 40 CFR Part 268.
CSSF = Calcine Solids Storage Facility.

Source: Staiger and Swenson 2011.

2.3.2 Physical Characteristics of the Calcine Waste

Physically, the calcine waste also exhibits some variation in properties (Swenson 2010). The calcination process converted liquid wastes into a solid form in a high-temperature (400°C−600°C) fluidized bed. During calcination, liquid radioactive wastes were atomized with air and sprayed into a heated bed of air-fluidized, granular solids, and the constituents dissolved in the liquid wastes built up layer by layer on the fluidized-bed particles. Gases and some of the smaller solids are referred to as calcine fines. The fines exhibit a particle size of less than 150 μm in diameter and a significantly lower bulk density (Table 2-10). As a result they were swept from the vessel with the fluidizing air. The average bed particle size of the waste in the calciner was kept at the desired value, typically ranging from 0.3 to 0.7 mm in diameter, by controlled attrition of the bed particles and is referred to as the calcine product. The bulk density of the waste varies from 0.9 to 1.7 gram/cc with an average bulk density of ~1.4 g/cc.

According to Swenson (2010), the chemical content of the liquid and calcine waste also affected the particle density and size. For a given set of operating conditions, aluminum waste produced the smallest calcine particles with the lowest bulk density, blends of SBW produced the largest particles with the highest bulk density, and zirconium waste produced calcine with particles sizes and bulk densities between those of the aluminum and SBW blends (Table 2-11). The fines were captured from the fluidizing air off-gas by a cyclone separator and transferred into a pneumatic transfer system. The calcine product was periodically removed from the calciner before entering the pneumatic transfer system where the product joined the fines. Both fines and product were then transferred to the CSSFs. Calcine retrieved from the CSSFs will be a mixture of both fines and product. All calcine product is a free-flowing granular solid. However, the calcine fines do not flow readily, and the design of calcine-handling systems must take into account the fines flow characteristics for further waste processing for disposal.

2.3.3 Thermal Characteristics of the Calcine Waste

The thermal output of the calcine also varies with the type of calcine. The hottest calcine will have a heat generation rate of about 40 W/m^3, and the coldest calcine will have a heat generation rate of about 3 W/m^3 (in 2016) (SNL 2014a). The average heat generation values for the calcine in the various CSSFs are provided in Section 4.3. The combined Cs and Sr activity in 2016 contributes 99.2% of the calcine activity and 96.7% of the heat generation. The rest of the heat comes primarily from ^{238}Pu, with a small amount due to ^{241}Am, and lesser amounts due to ^{154}Eu and ^{151}Sm (Swenson 2015).

Table 2-10. Particle Size and Bulk Density of Calcine Fines

Calcine Fines Property	Description
Particle Size	1–150 µm (most particles 1–40 µm in diameter with a mass-mean particle diameter of about 10 µm)
Bulk Density	30%–50% of the bulk density of product for a given calcine
Product: Fines Mass Ratio	Varies from 4:1 to 1:4 (80 to 20 wt% calcine product)

Source: Swenson 2010.

Table 2-11. Particle Size and Bulk Density of Calcine Product

Calcine Type	Bulk Density (g/cc)	Particle Density (g/cc)	Mass-Mean Particle Diameter (mm)	
			NaK heating	IBC heating
Aluminum	0.9–1.1	1.5–1.9	0.4–0.7	0.25–0.35
Zirconium	1.3–1.6	2.2–2.7	0.6–1.0	0.3–0.5
Zirconium/SBW blend	1.5–1.7	2.5–2.9	N/A	0.4–0.5
Aluminum nitrate/SBW	1.4–1.5	2.4–2.5	N/A	0.35–0.45
Aluminum nitrate/SBW[a]	1.5–1.65	2.5–2.8	N/A	0.5–0.65
Dolomite (startup bed)	1.6–1.7	2.7–2.9	0.4–0.6	0.4–0.6

NOTE: [a] High-temperature (600°C) flowsheet demonstration in 1999 and 2000.

N/A = Not applicable; no calcine of this type generated for this heating method.

SBW = sodium-bearing waste.

IBC = in-bed combustion.

Source: Swenson 2010.

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3 Waste Management Functions

The waste-filled universal canisters are to be stored, transferred, transported, and disposed. These activities must be done in a manner that prevents dispersion of radioactive material, minimizes surface contamination, and keeps doses and risks to individuals as low as reasonably achievable (ALARA). Sufficient built-in shielding and safe remote canister handling are required for all operations. These four waste management functions are described below; requirements associated with these four functions are discussed in Section 6.

3.1 Storage

Storage consists of keeping waste at a particular location until it is moved to another location, either on site or off site, for treatment, continuing storage, or geological disposal. Currently, some wastes are stored in pools or tanks, but for long-term storage, wastes generally need to be stored dry in packagings, casks, or vaults. The storage packagings/casks must provide confinement of the waste, shielding of the radiation emitted by the waste, and thermal management as well as prevent criticality of any fissile waste. Some storage packagings/casks are designed and certified for dual use, i.e. both storage and transport.

3.2 Transfers

Transfer operations are usually of a short duration and involve the movement of the waste over short distances at a single site. The transfer of waste generally consists of moving waste from one configuration to another, such as from a storage system to a transportation system (or vice versa), or from a storage or transportation packaging/cask to its final disposal location. Examples of transfers would include loading the waste into the universal canister, transferring the universal canister from where it was loaded to on-site dry storage, transferring the universal canister from its storage system to its transportation packaging/cask (if a dual-purpose cask is not used), and transfer of the universal canister from the waste receiving and storage facility at the disposal site to a borehole or mined repository.

3.3 Transportation

Transportation consists of moving waste on public thoroughfares such as roadways, waterways, and/or rail lines. It is anticipated that the universal canister will be sized to ensure disposal flexibility such that disposal could occur in either a deep mined geologic repository or in a borehole. Based on the predicted borehole diameters, the outside diameter (OD) of the universal canister will most likely be limited to no larger than 12.5 in. Thus, options for transportation include the use of truck-sized packages that would accommodate several canisters or the use of rail-sized packages that would accommodate many canisters. Each of these options has pros and cons; each will be fully studied and a determination regarding each will be made moving forward with this project. Depending on the level of radioactivity contained within a package, different levels of rigor are placed in the design of the packaging. For the types of materials considered as transport package contents for this report, Type B packages will be used (see Section 6.2.2 for discussion on Type B packages).

During transport activities, the package must contain radioactive material content, shield radiation emitted by the radioactive material content, dissipate heat generated by the content, and ensure subcriticality of the fissile radioactive material contents.

The containment function is typically performed through the use of metal containers with bolted metal lids. The interface between the containers and lids typically employs either an elastomeric or metallic seal. Prior to transport, a preshipment leak test is performed to ensure that the interface between the body and the lid is properly sealed.

The shielding function is typically performed through use of materials that are efficient at absorbing the specific types of radiation emitted by content. Alpha and beta particles are not difficult to shield; therefore, there is no real need to design shielding for these particle types. Gamma rays must be shielded, and this function is typically performed through the use of materials such as lead, depleted uranium, or massive amounts of steel. The materials considered in this report emit only a minute quantity of neutrons, so neutron shielding is not an issue for the package to be used.

The heat dissipation function is typically passive, though fins are used in some packaging designs. The package design must be analyzed to ensure that all parts of the package, including the content, remain below their respective allowable operating temperatures as specified within the safety basis document. Additionally, no accessible surface of the package should exceed 85°C (185°F) under specified conditions (shade and 38°C (100°F) ambient temperature).

Very small quantities of fissile isotopes are present in the materials that are under consideration in this report, so ensuring subcriticality is not expected to be an issue for the transport of these materials.

Packaging functionality must be maintained both during normal conditions of transport (NCT) as well as in the event of a transportation incident or accident.

Although the terms “package” and “packaging” are similar, they have considerably different regulatory meanings. Packaging refers to the container and all of its associated components in which radioactive materials (content) are shipped, whereas a package includes the packaging and its content. This subtle difference becomes very important when interpreting the regulations. Certifications, whether provided by the NRC or DOE, are for packages, not for packagings.

3.4 Disposal

The final step in the waste management process is disposal, which consists of emplacing waste packages containing radioactive material in a repository with no foreseeable intent of recovery, whether or not such emplacement permits the recovery of such waste. The waste package is the primary container that holds and is in contact with the solidified radioactive materials (i.e., the universal canister) as well as any overpacks that are emplaced at a repository. Since the mid-1980s, the form of deep geologic disposal favored by the DOE has been a mined repository, which was envisioned as the final resting place for all HLW and SNF regardless of origin. Potential geologic media include halite (salt), clay/shale, and crystalline rock. Recently, however, the DOE has evaluated alternatives for disposing of DOE-managed waste separately from commercial waste (SNL 2014a,b; DOE 2014a) and concluded that deep borehole disposal in crystalline rock is an option that should be pursued further with respect to the smaller DOE-managed wastes that are of the appropriate dimensions for deep borehole disposal.

Numerous factors suggest that the deep borehole disposal concept is viable and safe (Brady et al. 2009; Arnold et al. 2011). Described more fully in Section 4.5, this disposal option proposes disposing of waste packages in a crystalline rock disposal zone from 3,000 to 5,000 m in depth using currently available commercial drilling technology. In the United States, there are large areas of stable, crystalline basement rock that might be suitable for deep borehole disposal. Evidence indicates that groundwater at depths of several kilometers in crystalline basement rock has (1) low velocity and long residence times, (2) density-stratified high-salinity fluids that have limited potential for vertical flow and colloidal transport of radionuclides, and (3) geochemically reducing conditions that stabilize low solubility phases and enhance the retardation of key radionuclides.

The universal canister system design must accommodate the needs of the deep borehole disposal option while maintaining the flexibility to be disposed of in a mined geologic repository should deep borehole disposal not become a viable disposal option. The associated technical requirements for the universal canister design are discussed in Section 6.1; the regulatory requirements are discussed in Section 6.2.

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4 Available and Proposed Technologies and Concepts

The following sections discuss the existing and proposed technologies and concepts for storage, transfer, transportation, and disposal of the various wastes considered in Section 2.

4.1 Existing Storage Facilities

Two of the wastes considered for universal canister systems currently exist and are stored at DOE facilities. Storage of these two wastes—the Cs and Sr capsules and the calcine waste—is discussed below.

4.1.1 Storage of Cs and Sr Capsules

As discussed in Section 2.1, the 1,936 Cs and Sr capsules are currently being stored under water at the WESF in pool cells that were constructed specifically for storage of the capsules, as shown in Figure 4-1. This facility has been granted a permit by the State of Washington (Washington Department of Ecology 2008). There are 12 concrete pool cells, each of which is lined with stainless steel and equipped with a monitoring system to detect leakage from the capsules. The water in the pools is approximately 13-ft deep and provides cooling and shielding for the capsules (Fluor Hanford 2000).

The WESF began operation in 1974 with a design life of 30 years. DOE’s Office of Inspector General (DOE 2014b) audited the WESF in 2013–2014 and found that the concrete in the WESF pools has begun to deteriorate as a result of the years of radiation exposure, thus increasing the risk that a beyond-design-basis earthquake would breach the walls, resulting in a loss of fluid and a loss of shielding for the capsules. DOE/Richland has taken actions to mitigate the consequences of a beyond-design-basis earthquake, such as moving the capsules within the pools to reduce radiation exposure to the pool cell concrete and to reduce heat, thereby increasing the time it would take for capsules to fail (should the capsules no longer be covered by water) (DOE 2014b).

The audit (DOE 2014b) also found that storage of the capsules in the pool cells resulted in a higher operating cost than did proposed dry storage alternatives that have been under consideration. Currently, wet storage operation costs at the WESF are approximately $7.2 million per year, whereas dry storage operation costs would be about $1 million per year. DOE (2015b, Table 1) reported that it would cost between $75 million and $300 million to move the capsules from the WESF into a dry storage facility. The suggested path forward (DOE 2014b) was described as follows:

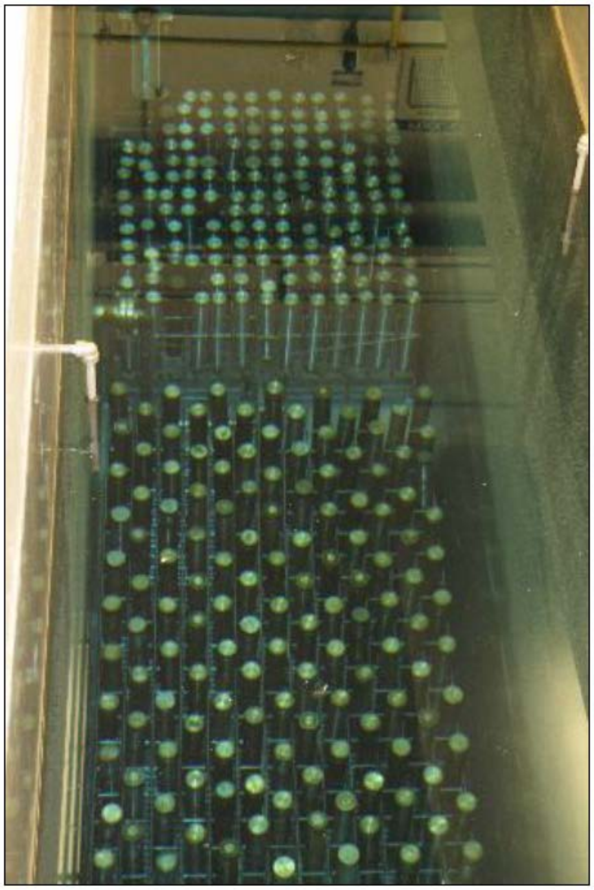


Figure 4-1. WESF Pool Cell

The Department is aware of the current safety conditions associated with the storage of cesium and strontium capsules at WESF and has taken actions to mitigate any risks associated with WESF. Furthermore, we acknowledge the budgetary challenges facing the Department, and its impact on moving the capsules into dry storage. Therefore, we are not making any formal recommendations. However, we suggest that the Manager, Richland Operations Office, expeditiously proceed with its plans to pursue a dry storage alternative to support transfer of the capsules out of WESF at the earliest possible timeframe.

Hanford is currently seeking to move the capsules into dry storage and has stated that options for disposing of the Cs and Sr capsules in a deep borehole will be incorporated into the mission to store the capsules in a dry storage facility (DOE 2015b). Hanford's mission to move the capsules into a dry storage facility and the Universal Canister Project's efforts to design a universal canister system are occurring in parallel. It is expected that these two projects will collaborate such that a universal canister can be designed and produced that meets the needs of both the Hanford Site to move the capsules to a dry storage facility in a timely fashion and the requirements for universal canister storage, transfer, transport and disposal in a deep borehole. In this report, it is assumed that the universal canister is part of Hanford's extended storage system for the capsules.

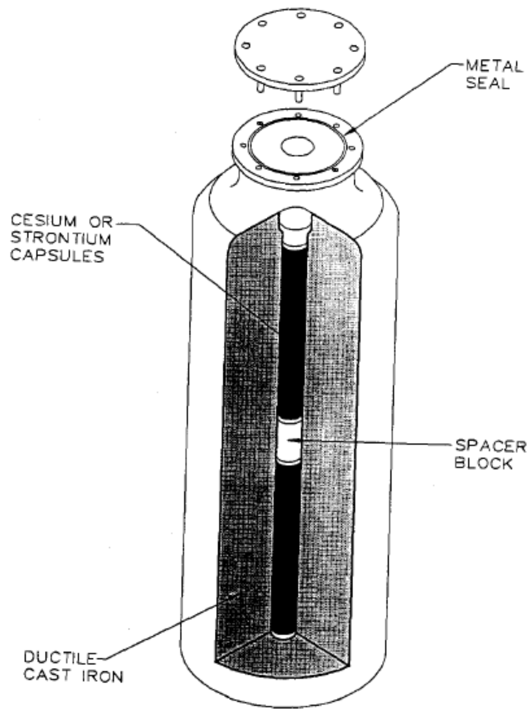
In addition, Hanford has committed to two milestones with respect to disposition of the capsules as part of the Tri-Party Agreement Action Plan (HFFACO 2015). These two milestones are shown in Table 4-1. Development of the universal canister system for possible direct disposal of the Cs and Sr capsules is one of the activities undertaken to meet these milestones.

Table 4-1. Tri-Party Milestones Applicable to the Cs and Sr Capsules

Milestone Number	Milestone	Due Date
M-092-00	Complete acquisition of new facilities, modification of existing facilities, and/or modification of planned facilities necessary for the storage, treatment/processing, and disposal of Hanford Site cesium and strontium capsules (Cs/Sr), bulk sodium (Na), and 300 Area special waste (SCW).	To Be Established By 9/30/2018
M-092-05	Determine disposition path and establish interim Agreement Milestones for Hanford Site Cs/Sr capsules. DOE will assess the viability of direct disposal of the Hanford Cs/Sr capsules at the national high-level waste repository and provide a schedule leading to its disposition. If DOE concludes that direct disposal is a viable and preferred alternative to vitrification, DOE will submit to Ecology specific documentation justifying its conclusion, with a proposed milestone change request establishing enforceable Agreement Milestones for dispositioning Hanford Cs/Sr capsules.	06/30/2017

NOTE: SCW = special case waste.

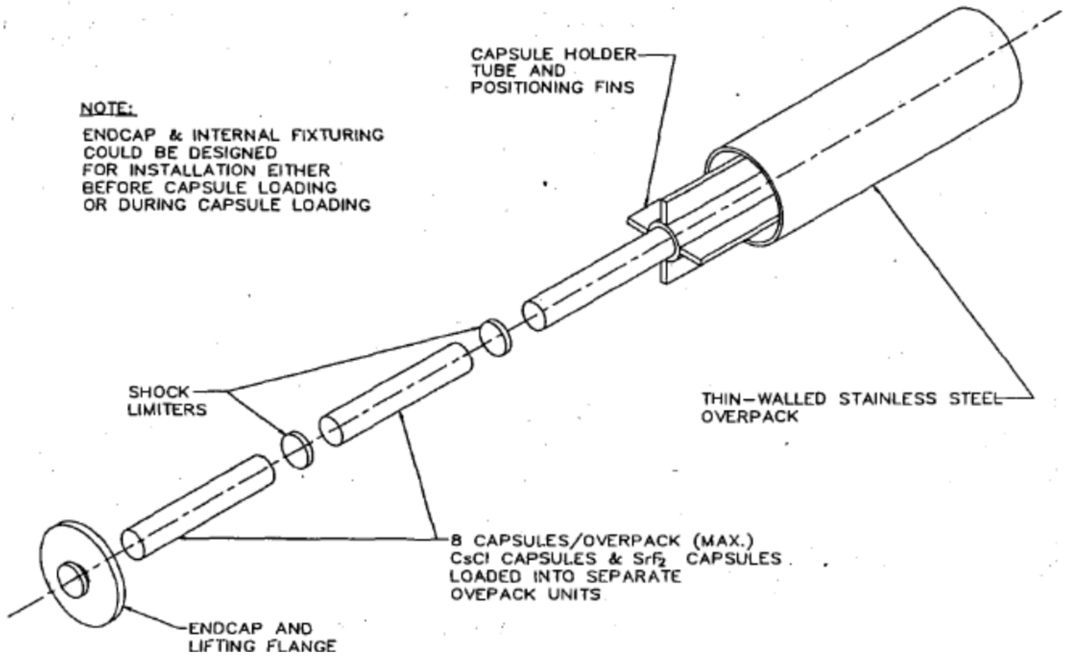
A trade study analyzing alternatives for the management of the Cs and Sr capsules at the WESF was performed in 1996 (Claghorn 1996). One of the alternatives evaluated was dry storage of the capsules in below-grade tubes in a canister storage building. The schematic of the canister design proposed (but never built) for this storage concept is shown in Figure 4-2 below (Claghorn 1996, Figure 6-1). The proposed design consists of a solid body of ductile cast iron with a centerline cylindrical cavity, which is slightly larger in diameter than the Cs and Sr capsules allowing loading. The cavity could accommodate two Cs or Sr capsules (nominally 53 cm each) separated by a 10-cm metal spacer. The bottom of the canister is 15-cm solid cast iron for shielding purposes. Above the top capsule is another 15 cm of shielding with a metal plug inserted in the centerline cavity. This two-capsule canister is 167.5-cm high with an OD of 66 cm. The internal centerline cylindrical cavity is nominally 7.2 cm in diameter to allow a 0.25-cm radial gap between the capsule and the canister.



Source: Claghorn 1996, Figure 6-1.

Figure 4-2. Proposed Storage Canister for Cs and Sr Capsules

Another conceptual design of a capsule disposal overpack was also proposed by Claghorn, as shown in Figure 4-3. This disposal overpack was designed for use in a mined geologic repository, not a deep borehole, with the intent that the waste in the disposal overpack would be accepted for disposal at the repository (i.e., the capsules would not be taken out of the overpack and placed in another one at the repository) (Claghorn 1996, Section 7.1). The outer wall of this disposal canister consisted of austenitic stainless steel. To save the costs of qualifying the package for shipping, the canister was to be made to the HLW canister specifications. A canister with a length of 4.57 m would accommodate up to eight capsules stacked end to end. A 3.0-m-long canister would accommodate five vertically stacked capsules. The canister contained an internal sleeve with fins that extended out to the inner wall of the canister in order to limit the lateral movement. Sintered metal spacers in between each of the capsules limited the axial movement of the capsules. This proposed overpack was never built.



Source: Claghorn 1996, Figure 7-3.

Figure 4-3. A Disposal Overpack Concept for Cs and Sr Capsules

In 2003, Hanford began investigating the requirements for dry storage of the Cs and Sr capsules as a part of the Hanford Capsule Dry Storage Project (CDSP), as documented by Sexton (2003). Some of the other documents that were produced as a part of the CDSP examined capsule corrosion (Bryan et al. 2003), capsule performance criteria (Bath et al. 2003), thermal analysis (Heard et al. 2003), and capsule integrity (Tingey et al. 2003). In addition, a Capsule Advisory Panel (CAP) was chartered in 2003 to ensure a sound technical basis for the CDSP; the findings of the Panel are documented by Plys and Miller (2003). These documents provide the basis for some of the storage requirements discussed in Section 6.

The final configuration of the dry storage cask is likely to differ from the conceptual designs of the CDSP because the decay heat from the capsules has decreased significantly since 2003 when the CAP analysis was performed. However, these results are relevant because they indicate that a dry storage cask with up to 16 capsules per cask could meet the thermal and corrosion acceptance criteria for conceptual designs with a 50-year lifetime.

In the conceptual design of the CDSP, the dry storage overpack had 16 capsules with a maximum decay heat of 2,540 W per overpack (Heard et al. 2003). The overpack was to be fabricated of 316-L stainless steel with a 50-year overpack design life, supported by an internal corrosion allowance of 0.318 cm (0.125 in.) (Plys and Miller 2003). Separate overpacks were proposed for Cs capsules, for Sr capsules, and for Cs capsules in Type W overpacks. The performance specifications for the salt-metal interface temperature are shown in Table 4-2.

Table 4-2. Performance Specifications for Salt-Metal Interface Temperatures for the Hanford CDSP

	Sr Capsules	Cs Capsules
Accident conditions	800°C	600°C
Processing, including process upsets	540°C	450°C
Interim storage configuration under summer storage conditions	540°C	317°C

Figure 4-4 shows the proposed conceptual design for a dry storage overpack with a monolithic stainless steel insert holding 16 capsules (Plys and Miller 2003). Important design features of the overpack include the following:

- 16 Cs capsules per overpack, with a maximum decay heat of about 2,540 W
- Overall diameter of 22 in.
- Single tier of capsules for ease of loading
- A thick upper shield plug for operations (not shown in Figure 4-4)

The design proposed by Plys and Miller (2003) met the performance specifications for the salt-metal interface temperature under normal operations, process operations, and selected accident scenarios (Heard et al. 2003). Temperatures of the overpack were predicted to be in the range of 200°C–225°C during normal operations and to decrease with decay of the ^{137}Cs and ^{90}Sr in the capsules. The use of loading strategies that kept the total decay power to less than 2,540 W per overpack, the presence of external fins or helium backfill, and consideration of axial thermal conduction all resulted in significant reductions in the predicted salt-metal interface temperature.

The dry storage facility into which the capsules are transferred must be able to safely store the capsules for an extended, and as yet undefined, period of time. The facility must be able to provide confinement of the waste, must provide shielding to keep doses within regulatory limits, and must manage temperatures to within acceptable limits. The design of the universal canister system used to store the waste must also allow transfer of the waste from the WESF to the storage area, transport of the waste from the storage facility to the disposal facility, and emplacement into a deep borehole at the disposal facility. Currently, the size of the universal canister that could be used to store Cs and Sr capsules is driven by both the operational limits of the WESF (Section 6.1.8) and borehole diameters (Section 4.5.1).

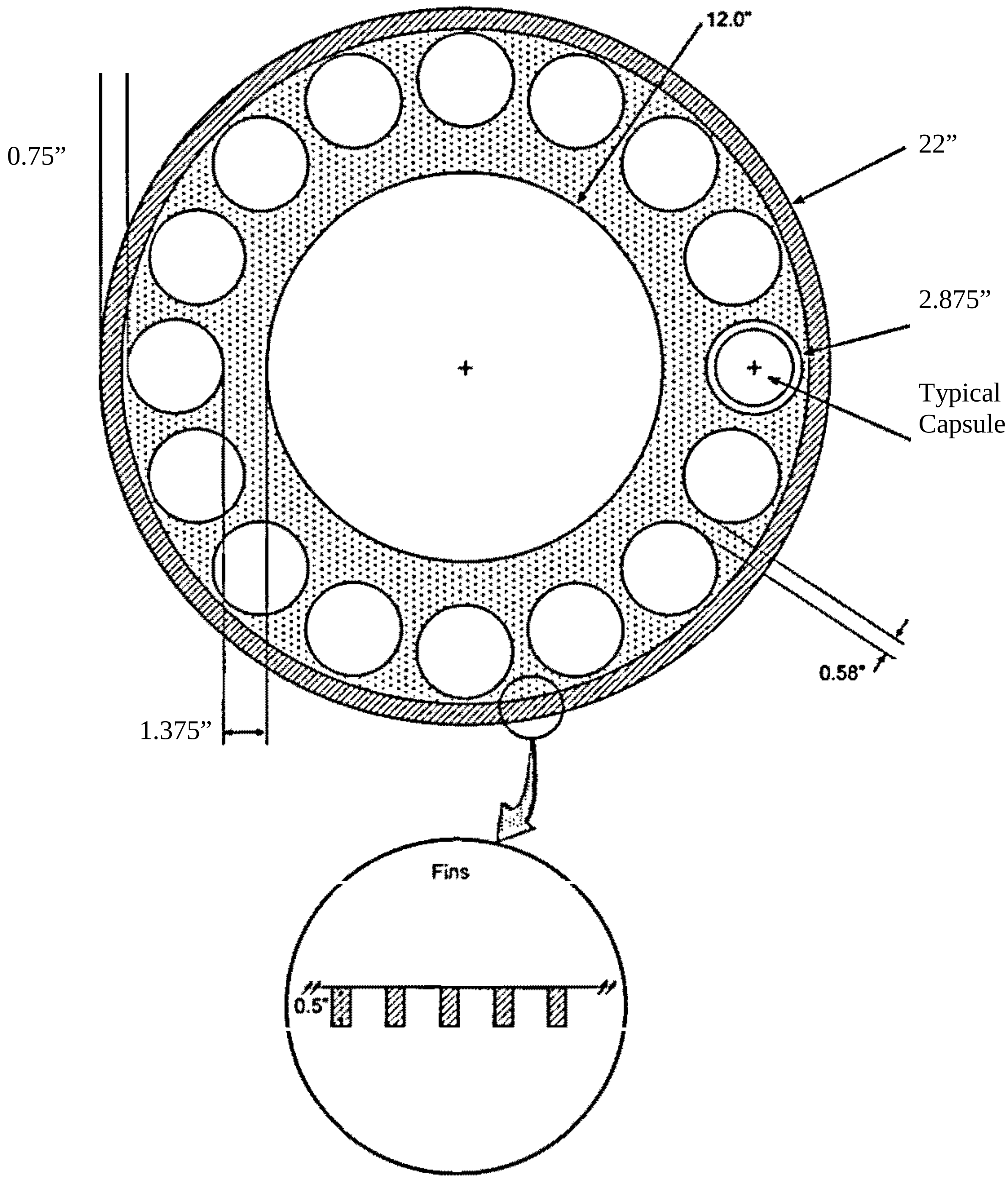


Figure 4-4. Plan View of the CAP Conceptual Design for Dry Storage Overpack with External Fins and a Monolithic Insert Holding 16 Capsules

4.1.2 Storage of Calcine Waste

According to Staiger and Swenson (2011), between December 1963 and May 2000 INTEC liquid wastes were converted into a solid, granular form called calcine at two calcining facilities: the Waste Calcining Facility (WCF) and the New Waste Calcining Facility (NWCF). The WCF converted 4,091,000 gal of aqueous radioactive waste into 77,300 ft^3 of calcined solids. The NWCF converted 3,642,000 gal of aqueous waste into 78,000 ft^3 of calcined solids. The total volume of calcine is about 155,300 ft^3 (4,400 m^3) and is stored in 6 CSSFs (Figure 4-5). There are a total of seven CSSFs, but the seventh one does not contain any calcine. Each CSSF is an underground or partially underground concrete vault containing several stainless steel storage bins.

Figure 4-5. Photograph Showing the Locations of the Six CSSFs Containing Waste

CSSFs II, IV , and V are completely full, CSSFs I and III are nearly full, CSSF VI is about half full, and CSSF VII is empty (Staiger and Swenson 2011). The first waste calcination campaign, Campaign 1, at the WCF ended just prior to filling CSSF I. This left sufficient room in CSSF I to receive the calciner bed from the WCF after shutdown and avoid the need to process the calciner bed by dissolution and return the resulting waste to the Tank Farm. Campaign 1 stopped short of filling CSSF I because CSSF II did not yet exist, so there was no place to send the additional calcine. Similarly, CSSF III is not quite full because WCF Campaign 9 ended just prior to the filling of the CSSF III bins, allowing enough room for CSSF III to receive the WCF calciner bed after shutdown. Campaign 9 was the last for the WCF. The NWCF was connected to CSSF IV so CSSF III could not be filled by NWCF. CSSFs II, IV, and V are filled to capacity because they were filled during campaigns when the next (empty) CSSF was available to receive calcine. When CSSFs II, IV, and V were filled, the next CSSF was available to be placed in service without stopping operation of the calciner. CSSF VI is about half full and CSSF VII is empty because the DOE decided to stop calcining waste in May 2000 and to treat the waste remaining in the Tank Farm with an alternative method.

Cut-away diagrams of the CSSFs are shown in Figure 4-6. Each vault also contains a cyclone cell that was used for calcine distribution and an instrument room with CSSF monitoring equipment (Idaho Department of Environmental Quality 2006, Appendix D). Additional details of the construction and specifications of each CSSF and their associated bins are provided in Staiger and Swenson (2011).

The approximate volumes and types of calcine sent to storage in CSSF I include 230 ft^3 of nonradioactive alumina used for calciner startup material and 7,530 ft^3 of radioactive aluminum calcine. CSSF II includes approximately 2,770 ft^3 of dolomite and nonradioactive alumina used for calciner startups, 12,060 ft^3 of radioactive aluminum calcine, and 15,200 ft^3 of radioactive zirconium calcine. CSSF III includes approximately 5,200 ft^3 of nonradioactive dolomite, alumina, and fluorapatite calciner startup material, 1,860 ft^3 of radioactive aluminum calcine, and 32,400 ft^3 of radioactive zirconium (and coprocessing), zirconium/SBW blend, and zirconium/stainless-steel blend calcine. CSSF IV includes approximately 730 ft^3 of nonradioactive dolomite and alumina from calciner startups, 120 ft^3 of aluminum nitrate/zirconium blend, and 16,300 ft^3 of radioactive zirconium and zirconium/stainless-steel/SBW blend calcines. CSSF V includes approximately 3,200 ft^3 of nonradioactive alumina, dolomite, and pilot plant calcine used in calciner startups, 580 ft^3 of aluminum calcine, and 31,820 ft^3 of calcine formed from blends of radioactive zirconium, SBW, aluminum, stainless-steel, and Rover wastes, and nonradioactive aluminum nitrate. CSSF VI contains approximately 2,070 ft^3 of nonradioactive pilot plant calcine, alumina, and dolomite from calciner startups, 600 ft^3 of aluminum calcine, and 22,500 ft^3 of calcine blends formed from aluminum, zirconium, stainless steel, SBW and aluminum nitrate.

The temperature within the bins continues to be monitored via thermocouples in various locations. The bin temperature has stabilized and currently shows only ambient temperature fluctuations. The ventilation system for cooling was therefore not necessary and was closed. Each CSSF is equipped with continuous air monitors to detect loss of bin containment. If a monitor alarm is activated, the filter of the continuous air monitor will be analyzed to determine whether a release occurred. However, because all vaults have been isolated from the atmosphere by mechanically closing the cooling air inlets and outlets, there is no motive force to spread contamination outside the vault (Idaho Department of Environmental Quality 2006, Appendix D).

The DOE has been issued a RCRA Part B Permit by the State of Idaho for the CSSF, Permit #ID4890008952 (Idaho Department of Environmental Quality 2006). The DOE has also agreed that it will treat all HLW currently at the Idaho National Engineering Laboratory so that the HLW is ready to be moved out of Idaho for disposal by a target date of 2035. In particular, with respect to calcine waste, the DOE expects all calcined waste to be treated by December 31, 2035, but the State of Idaho expressly reserves its right to seek appropriate relief from the Court in the event that this deadline is not met (Idaho Department of Environmental Quality 1995).

Source: modified from Staiger and Swenson 2011.

Figure 4-6. Cutaways of Each of the Seven CSSFs

4.2 Existing Transfer Mechanisms for Cs and Sr Capsules

Using existing equipment and facilities, loading of the Cs and Sr capsules into the universal canister would likely be performed in hot cell G in the WESF at the Hanford Site (see Figure 4-7 for a floor plan of the WESF). The pool is equipped with a transfer system capable of moving individual Cs and Sr capsules from their pool storage locations through pool cell #12 to hot cell G. Transfer from the storage pool to the hot cell would be performed underwater (i.e., through the underwater transfer drawer between storage pool and hot cell) to minimize radiation exposure of personnel during capsule transfer. Once in hot cell G, the capsules would be handled remotely via manipulators. If sufficient space and equipment are available, each capsule would be dried and placed into a universal canister, which would then be welded shut, perhaps after being backfilled with helium. Canisters could contain one or more capsules. The sealed canister would be leak checked and, once it was ascertained that the canister was not leaking, would be transferred to dry storage. Transfer of the canister to dry storage could be accomplished with a dedicated transfer cask. The components of the transfer system include a pool with 12 cells, a hot cell (8-ft wide × 16-ft long × 12-ft high), a capsule pushing tool, pool cell tongs, underwater camera or binoculars, television monitor and control panel, underwater lights, and a motorized catwalk (Covey 2015). More features of the current system used to transfer capsules are given in Table 4-3.

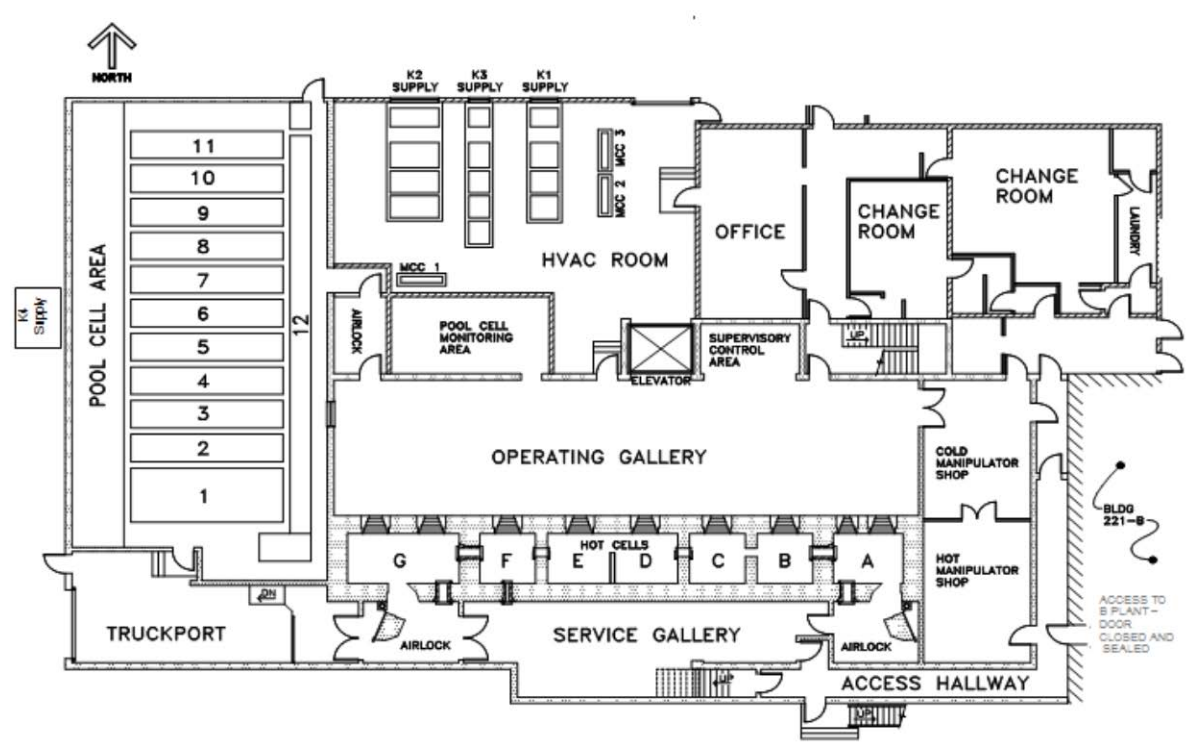


Figure 4-7. First Floor of the WESF

Table 4-3. Features of the Current Capsule Transfer System at the WESF

Component	Limit, Dimension, or Capacity
Truck Port Door (usable opening)	10-ft wide × 12-ft high
Truck Port Cover Block	12 ft × 7 ft
Hot Cell G Cover Block	8 ft × 16 ft
Overhead Canyon Crane (access to hot cell G, the truck port, and the underwater cask loading area)	15 tons
Hot Cell G Ceiling	12 ft
Hot Cell G Hoist	2 tons, 10 ft
Hot Cell G Door	2.5-ft wide × 6.5-ft high
Hot Cell G Floor	23,000 lb[a]
Hot Cell G Manipulators	100-lb vertical, 50-lb horizontal capacity

NOTE: [a] The floor has been analyzed to support this weight. It may have a higher capability, beyond the previous analysis.

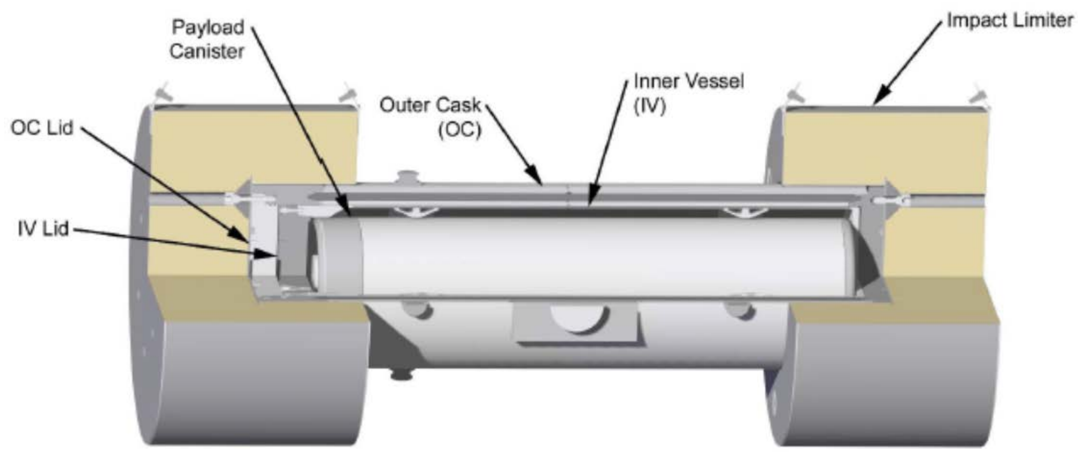
Source: Sexton 2003, Section 4.3.

It should be noted that the pool in the WESF has a cask pit at the end of pool cell #12, the pool cell used for transferring the capsules to hot cell G. The cask pit is 4-ft 5-in. wide × 7-ft 5-in. long × 18-ft deep. The overhead canyon crane can access the cask pit, hot cell G, and the truck port (Sexton 2003). However, the cask pit has never been used.

Hot cell G and the cask pit are currently the only two ways in which capsules can be removed from the WESF pool, regardless of the dry storage system into which the capsules are eventually moved.

4.3 Existing Canisters for Calcine Wastes

Should the DOE choose not to use HIP to treat the calcine, it could be disposed of in a universal canister. Several specific examples similar to this option were evaluated by Herbst (2005). In one of the seven canister options evaluated, the calcine is placed in a RH-TRU 72-B canister (Figure 4-8). The NRC certificate of compliance (CoC) for this canister is USA/9212/B(M)F-96. The RH-TRU 72-B canister is 121-in. long and 26-in. in diameter with a 0.25-in.-diameter steel wall. Internal volume is 0.9 m^3. Thus, the 4,400 m^3 of calcine would yield approximately 4,900 canisters at 98% fill or 5,400 canisters at 90% fill. Average heat loads varied in the six CSSFs and were to be between 34.6 W/canister and 8.8 W/canister (Table 4-4). All the results in Table 4-4 are based on the calcine source term as of January 1, 2016. Average dose rates from the RH-TRU 72-B canister wall were calculated at 1 cm, 30 cm, and 100 cm and are also provided in Table 4-4. Additionally, direct disposal of calcine waste could utilize other packaging configurations, including universal canisters configured for deep borehole disposal.



Source: Day and Sellmer 2009.

Figure 4-8. RH-TRU 72-B Canister (the Payload Canister) inside the Transportation Overpack with Impact Limiters Installed

Table 4-4. Average Dose Rate at Various Distances from the RH-TRU 72-B Canister Wall

Calcine	Average Dose Rate at 1 cm (R/hr)	Average Dose Rate at 30 cm (R/hr)	Average Dose Rate at 100 cm (R/hr)	Average Decay Heat (W)
CSSF I	1225.0	552.5	206.9	34.6
CSSF II	453.5	203.8	78.0	19.6
CSSF III	357.3	160.5	61.6	17.9
CSSF IV	453.0	203.4	78.2	22.5
CSSF V	446.0	200.2	77.1	23.6
CSSF VI	204.0	91.6	35.0	8.8

4.4 Existing Transportation Casks

There are many existing transportation packagings that, with some adaptation and subsequent certification, would be suitable for the transport of the universal canister loaded with radioactive materials, such as the Cs and Sr capsules at Hanford and the calcine wastes at INL. Truck and rail casks designed for the transport of used nuclear reactor fuel are particularly attractive for this use as these packagings are designed for high heat loads and provide the shielding necessary for transport of the various contents in universal canisters. Although the details of the universal canister design and materials of construction are not yet determined, the universal canister is expected to be compatible with the internal cavity diameters of the transportation casks. From a thermal standpoint, the subject materials can be accommodated as well. All of the current transportation packages for used nuclear fuel are shipped dry with an inert (helium) atmosphere, which is compatible with the waste considered in this report.

The choice of transportation packagings is related to facility design, both at the storage location and at the disposal site. It is assumed that the transport cask for the Cs and Sr capsules will be loaded at the Hanford Site. Considerations favoring transport by truck include the need for more extensive facilities to load rail cars, and possible lack of availability of direct rail access to the capsule storage facility and the borehole site (to avoid intermodal transfers). The primary consideration favoring the use of rail casks is the reduction in risk to the public that goes hand-in-hand with the reduction in the number of transportation operations. Either primary shipment mode can be readily accomplished. These issues are more readily dealt with in the future when more is known about the origin and destination sites.

With respect to transport, the cost/capacity tradeoff is generally favorable for the highest cask capacity possible. Larger capacity casks and/or per-shipment capacity would reduce the transportation operations costs (shipment preparation, cask loading and unloading, security and monitoring, trained drivers or engineers, and state oversight) and physical equipment needs (casks, trucks, locomotives).

4.4.1 Transportation Cask Heat Generation Limits

An important aspect of transportation packagings is their ability to allow for the dissipation of heat generated by the contents they carry. The waste materials considered herein generate substantial heat through radioactive decay, and this heat must be transferred through the transportation packaging and subsequently be dissipated to the surrounding environment. As required in 10 CFR Part 71, the accessible surface temperature of the package, when in use, must not get hot enough for an individual to be burned by touching it.

Some packagings include features, such as cooling fins, that are specifically designed to increase the rate at which heat is transferred to the surrounding environment. However, with the use of cooling fins comes decontamination issues that can be significant, especially if the transportation package is loaded wet (such as in a pool) and subsequently drained and dried. Generally speaking, modern US SNF casks (both rail and truck casks) do not use cooling fins due to this decontamination issue. Many older casks did employ such design features; however, those casks are generally not readily available for use.

Because the waste forms currently under consideration will be contained within the universal canister, contamination of the transport packaging during wet loading or unloading is not expected to be an issue. However, most older generation packagings are not certified to the current regulatory standards, so they cannot be fabricated (even replacement parts, other than off-the-shelf items likes bolts, cannot be fabricated) until the certification has been updated.

Because some of these older generation packagings do have higher heat generation ratings, it may be worthwhile considering their use moving forward.

4.4.2 Existing Transportation Cask Capacities and Limits

Commercial used fuel transportation casks capable of shipping radioactive source materials were evaluated to identify those that would be likely candidates for the transport of Hanford and INL waste materials considered herein. These potential casks are listed in Table 4-5. The potential casks are separated into truck casks and rail casks. Table 4-5 provides the cask cavity dimensions needed to configure various versions of the universal canister as well as the heat load limit.

The radiation source of the universal canister must be shielded by the transportation cask walls, but the arrangement of universal canister radiation sources within the cask cavity and the details of the transportation cask steel walls and shielding layers can be very complex, and it is misleading to simply list shielding thicknesses for the various casks. For design purposes of universal canister concepts, it is suggested that the heat generation limit be used to identify the total quantity of radioactive material to be shipped, and that confirmatory shielding calculations be performed at a later date. Many cask designers have encountered difficulties in local shielding designs caused by valve ports and machining for various purposes, and the detailed geometric arrangement of the universal canisters in a cask can have a very strong effect upon the cask dose rates.

Similarly, the cask weight and payload weight limits may not reflect the current gross vehicle weight limit, which was changed from 73,280 lb to 80,000 lb. The distribution of weight on the axles of the 18-wheeler tractor/trailer is also regulated, and the potential to use all or some of the weight increase is very specific to each cask and the weight limits used for its design.

The cask transport CoC expiration date is also provided in Table 4-5. The casks must be recertified on a five-year basis for transport.

The Beneficial Uses Shipping System (BUSS) cask was designed by Sandia National Laboratories (SNL) for transportation of the Hanford Cs and Sr capsules and was used for that purpose. The package identification for BUSS R-1 is USA/9511/B(U), and its CoC expired on March 31, 2008 (Ross et al. 2014), although it is in the process of being recertified as a storage-only packaging.

The BUSS cask with impact limiters attached is shown in Figure 4-9.

Table 4-5. Existing Transportation Cask Capacities and Limits

Cask Name	CoC Number	CoC Expiration Date	Cask Cavity Length (in.)	Cask Cavity Diameter (in.)	Cask Heat Load (kW)	Cask Payload Weight (lb)	Cask Gross Weight (lb)
Truck Casks							
BUSS[a]	9511	3/31/2008	23.0	20.25	4.0[b]	400	30,000
GE-2000	9228	5/31/2016	54.0	26.5	2.0[c]	5,450	33,550
NAC-LWT	9225	4/30/2020	178	13.375	2.5	4,000	52,000
TN-LC	9358	12/31/2017	182.5	18.00	3.0	7,100	51,000
GA-4	9226	10/31/2018	167	8.8 (square)	2.468	6,648	55,000
TN-FSV	9253	6/30/2019	199	18.0	0.360	5,000	47,000
RH-TRU 72-B	9212	2/28/2015[d]	130	32	0.050	8,000	45,000
Rail Casks							
NAC-STC	9235	5/31/2019	165	71.0	22.1	39,650	260,000
NAC-UMS	9270	10/31/2017	192	67	16	77,500	256,000
NuHoms MP-187	9255	11/30/2018	187	68	13.5	81,000	282,000
NuHoms MP-197	9302	8/31/2017	199.25	70.5	26	112,000	304,000
TN-68	9293	2/29/2016	178	69.5	21.2	75,600	272,000
HI-STAR 100	9261	4/30/2019	191.1	68.8	20	100,183	282,000
HI-STAR 180	9325	10/31/2019	174.37	72.83	32	—[e]	308,000
FuelSolutions TS125	9276	10/31/2017	193	66.88	20.35	85,000	285,000

NOTE: [a] The BUSS (Beneficial Uses Shipping System) cask is currently under review to recertify it as a storage-only packaging.

[b] 850,000 Ci of ^{137}Cs, or 650,000 Ci of ^{90}Sr.

[c] 422,000 Ci of ^{137}Cs divided into two sources in a “two-tier” arrangement of 211,000 Ci, or 596,000 Ci of ^{90}Sr.

[d] Currently under timely renewal.

[e] Payload weight is not available for the HI-STAR 180.

CoC = certificate of compliance.

Figure 4-9. Picture of the BUSS Cask with Impact Limiters Attached

The BUSS cask with impact limiters and transport skid weighs ~16 metric tons. The cask body is a cylindrical stainless steel forging with an OD of 137.8 cm and height of 124.5 cm, and it provides approximately 40 cm (16 in.) of steel shielding for radioactive contents. The cask cavity has a diameter of 51.4 cm and height of 58.4 cm. The cask cavity accommodates a solid stainless steel basket. The cask main components are shown in Figure 4-10, which was taken from Yoshimura and Bronowski (1996). Basket design configurations for different capsule loading exist, as summarized in Table 4-6 (Yoshimura and Bronowski 1996). The basket configuration for a capacity of 16 capsules, taken from Yoshimura et al. (1985), is illustrated in Figure 4-11, where dimensions are shown in inches.

Table 4-6. BUSS Cask Radioactive Material Limits

Basket Configuration	Allowable Capsule Type	Maximum Thermal Power per Capsule (W)	Maximum Total Cask Thermal Power (kW)	Maximum Total Cask Activity (millions of Ci)
16 holes	Cs	250	4.0	0.85
12 holes	Cs	333	4.0	0.85
6 holes	Sr	640	3.9	0.65
4 holes	Sr	850	3.4	0.56

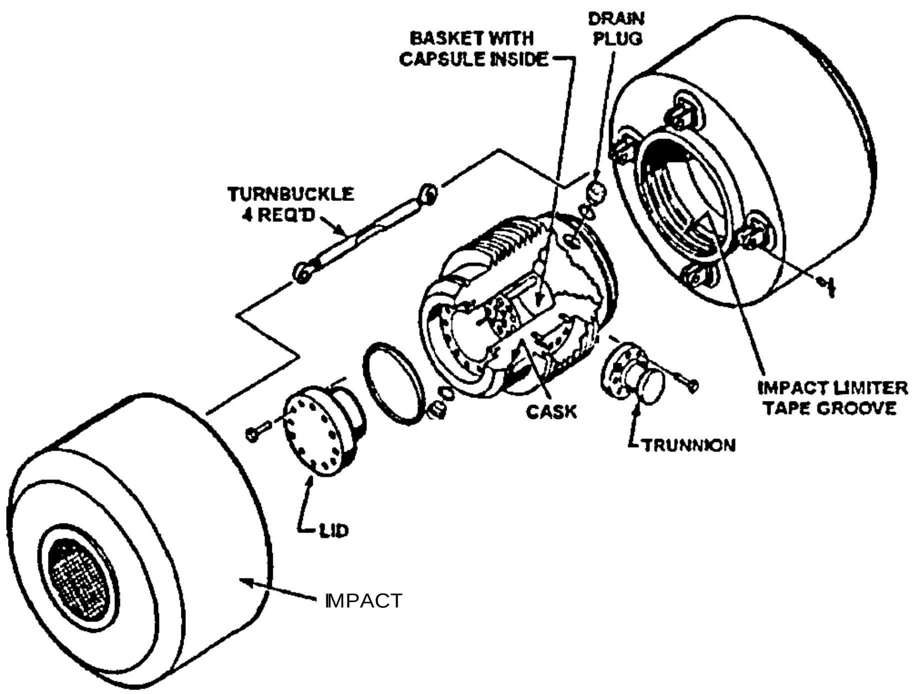
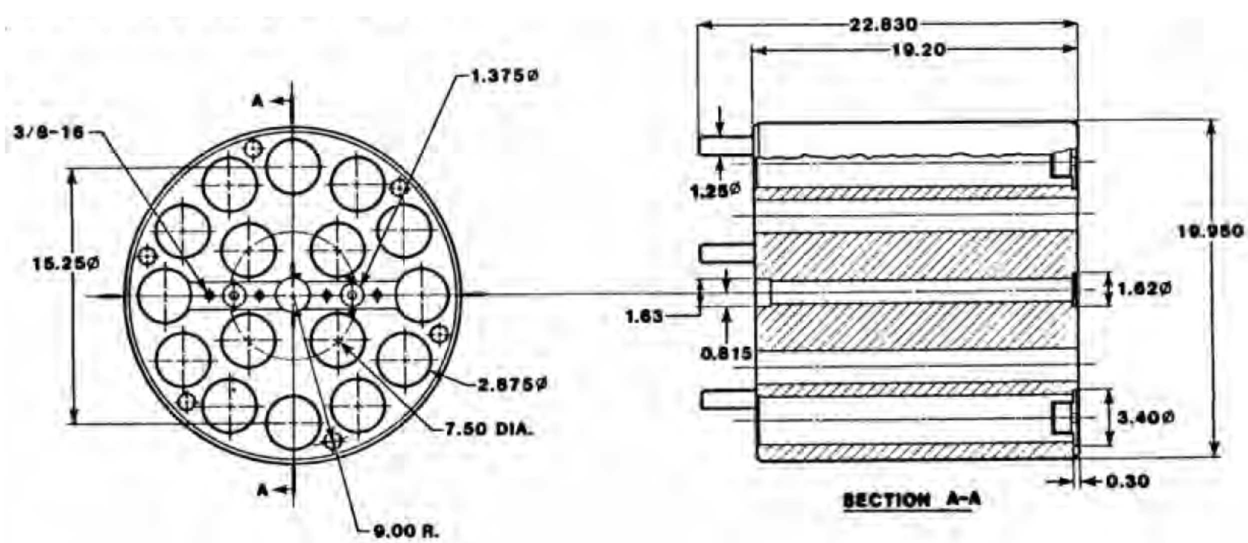


Figure 4-10. Schematic Diagram of the BUSS Cask



NOTE: Dimensions are in inches.

Figure 4-11. Schematic for a Basket Configuration with Capacity for 16 Capsules

4.5 Deep Borehole Disposal Concept

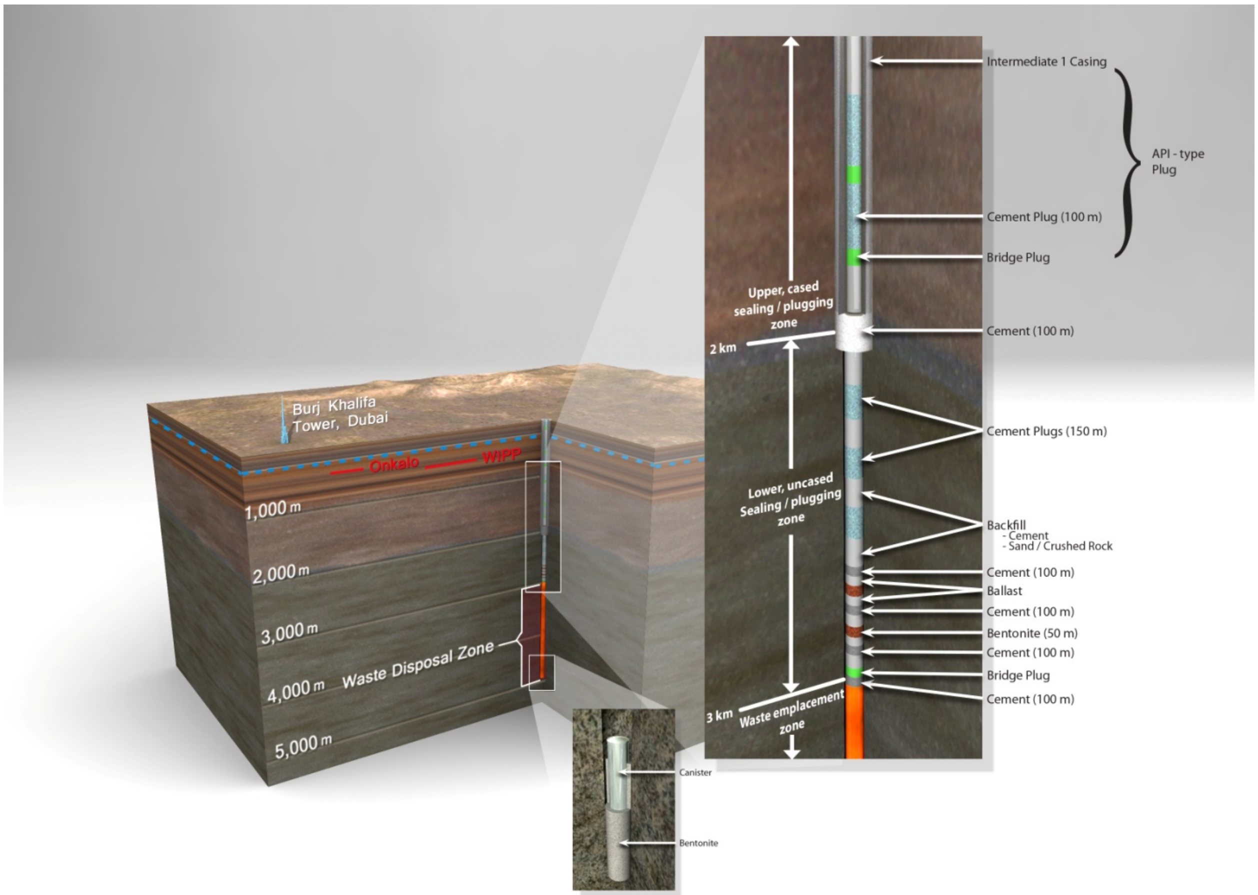
The idea of using deep boreholes to dispose of SNF and/or HLW has been studied by researchers in the United States and the international community for over fifty years (e.g., Hess et al. 1957, O'Brien et al. 1979; Woodward-Clyde Consultants 1983; Juhlin and Sandstedt 1989; NIREX 2004). While more recent efforts by the DOE initially focused on commercial SNF (Brady et al. 2009; Arnold et al. 2011 and 2013), the emphasis has since shifted to consideration of using deep borehole disposal for smaller, DOE-managed waste forms (DOE 2014a). Arnold et al. (2014) developed preliminary reference designs exploring potential combinations of borehole diameter, casing configuration, and waste package design for Cs and Sr capsules as well as untreated calcine waste, described in Sections 4.5.1 and 4.5.2 respectively. Plans for the DBFT are discussed in Section 4.6.

4.5.1 Disposal of Cs and Sr Capsules

At its most basic, the deep borehole disposal concept consists of drilling a large diameter borehole to a depth of 5,000 m in crystalline basement rock, emplacing waste packages in the lower 2,000 m of the borehole, and then sealing the upper 3,000 m of the borehole with a combination of bentonite, cement plugs, and cement/crushed rock backfill. As shown in Figure 4-12, the deep borehole disposal system is intended to be several times deeper than typical mined repositories. For reference, the dashed blue line shows the typical maximum depth of fresh groundwater resources.

Important factors pertaining to waste package design that stem from the use of the deep borehole disposal method include the following:

- **In-situ Temperature**—At a depth of 5,000 m, the natural geothermal gradient may lead to temperatures as high as 170°C, assuming a mean annual surface temperature of 20°C and a gradient of 30°C/km (Hardin 2015a, Section 2.10). This represents the maximum temperature for waste packages that produce little heat. For wastes that generate heat, such as the Cs and Sr capsules, in-situ temperatures would be higher. If the Cs and Sr capsules were stacked end-to-end in a borehole in 2026, the estimated temperature rise would be about 80°C, such that the maximum package surface temperature would be around 250°C (Hardin 2015a, Section 2.10).
- **In-situ Pressure—**The hydrostatic pressure at a depth of 5,000 m depends on the properties of the emplacement mud, but would probably be between 50 MPa (490 atm) and 65 MPa (640 atm). The maximum loading on a waste package consists of the hydrostatic pressure plus axial tensile or compressive loads. Axial tensile loads may be experienced during emplacement or retrieval, and compression loads will be experienced once the waste packages are stacked on each other in the borehole. The design of the waste package must account for the maximum loading with a safety factor of 2.0 (Hardin 2015a, Section 2.10).



NOTE: The dashed blue line shows the typical maximum depth of fresh groundwater resources.

Source: SNL 2015, Figure 1-1.

Figure 4-12. Generalized Schematic of the Deep Borehole Disposal Concept

- **In-situ Chemical Environment—**Waste packages must be able to withstand the corrosive effects of the in-situ chemical environment. Fluids at depth are expected to consist of high ionic strength chloride brines, and reducing conditions are also expected to prevail (Brady et al. 2009, Section 3). In addition, materials introduced into the system because of the disposal process can affect the system chemistry.
- **Emplacement Method—**The design of the waste packages must be able to accommodate whatever emplacement method is selected for use. A recent study (SNL 2015) examined various emplacement methods and narrowed the choices to (1) emplacing packages one-by-one on an electric wireline, or (2) assembling strings of packages threaded together which are then emplaced by lowering on a string of drill pipe. The study recommended that the wireline method be used in the DBFT on the basis that,

if the method is used for actual disposal, it would result in lower cost and less likelihood of a breached waste package and contamination of the borehole.

- **Borehole Dimensions—**Borehole total depth and diameter are related in that the deeper the borehole, the smaller the associated diameter. The borehole diameter affects the casing diameter, which will in turn affect the space available for the waste package.

For the Cs and Sr capsules, the simplest configuration is to put one or more capsules end to end, axially aligned, inside a cylindrical canister. The waste package used for disposal may include the canister alone or the canister plus a disposal overpack depending on the situation. As seen in Table 4-7, Arnold et al. (2014) studied several possible package designs including a two-capsule package, a six-capsule package, the reference package developed in 2011 for SNF (Arnold et al. 2011), and a large diameter package for SNF. It should be noted that most of the proposed design alternatives include waste packages on the order of 5 m (16.5 ft) in internal length (Arnold et al. 2014; SNL 2015). In developing the specifications for the universal canister system, the use of shorter waste packages will also be examined.

Table 4-7. Possible Alternative Deep Borehole Disposal Concepts for Cs and Sr Capsules

	Two-Capsule Package	Six-Capsule Package	Reference SNF Package[a]		Large Diameter SNF Package	
Borehole Diameter (in.)	8.5	12.25	17		22	
Disposal Zone Casing OD (in.)	7.00	10.75	13.40		17.90	
Disposal Zone Casing ID (in.)	6.40	9.65	12.65		16.50	
Package OD (in.)	4.50	7.50	10.75		14.20	
Package ID (in.)	3.50	6.50	8.35		12.60	
Capsules per Layer	1	3	7		14	
Number of Layers	2	2	2	8[b]	2	8[b]
Capsules per Package	2	6	14	56	28	112

NOTE: [a] From Arnold et al. 2011.

[b] Assumes disposal in a SNF canister with a height of 4.235 m.

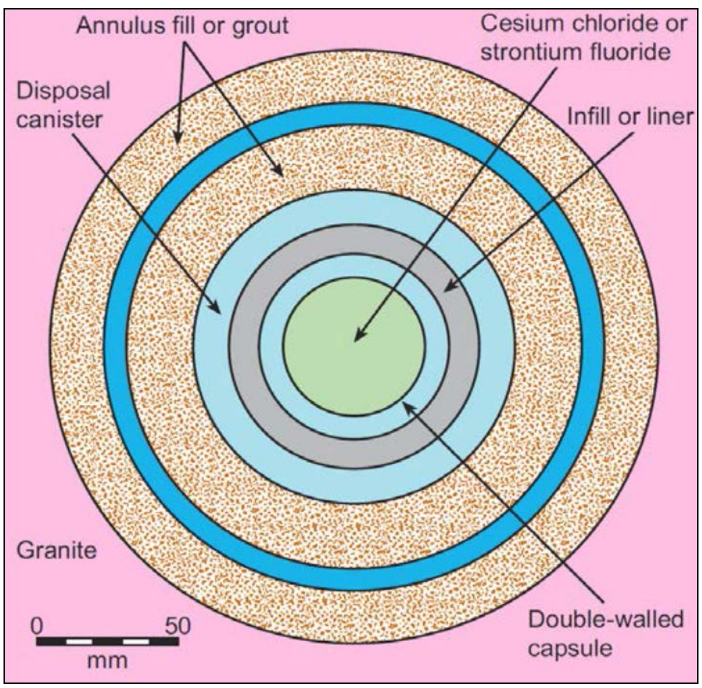
All dimensions for the borehole, casing, and package are for study purposes only. The dimensions to be used for actual deep borehole disposal have yet to be determined.

ID = inside diameter.

OD = outside diameter.

Source: modified from Arnold et al. 2014, Table 3-4.

Two-Capsule Package, 8.5-in.-Diameter Borehole—The two-capsule package concept (Figure 4-13) was developed as part of a baseline design to facilitate modeling and analysis efforts. This baseline design features two capsules placed end to end in a 42.64-in.-long stainless steel waste package with an OD of 4.5 in. and a wall thickness of 0.50 in. The wall thickness was determined assuming a maximum hydrostatic pressure of 57 MPa and standard tubing collapse relationships, as used in the design of the SNL reference waste packages for SNF in Arnold et al. (2011). With this design, the entire inventory of Cs and Sr capsules could be disposed of in one 8.5-in.-diameter borehole. The effort would require 968 two-capsule waste packages emplaced in a disposal zone of ~1,300 m. If the borehole is drilled to 5,000 m, the main seals could be emplaced just above 4,000-m depth, providing more than the 3,000 m of isolation generally regarded as appropriate for deep borehole disposal (Gibb et al. 2012; Brady et al. 2009; Beswick et al. 2014). Alternatively, the borehole could be drilled to only a little over 4,000 m, which would still provide 3,000 m of isolation for the waste packages.



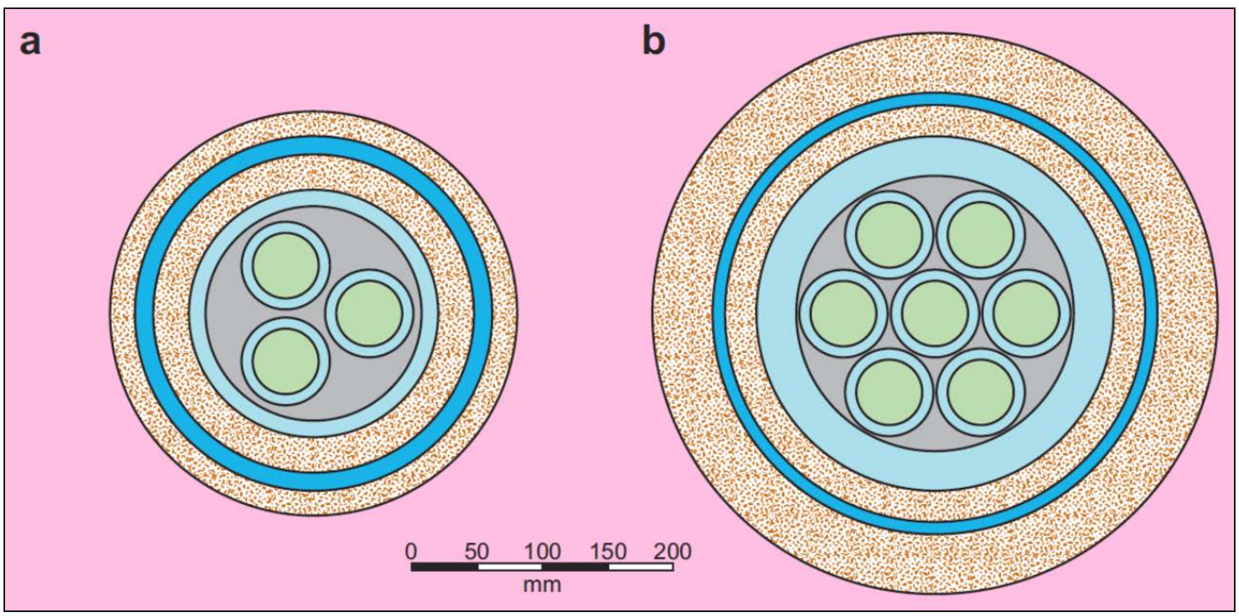
NOTE: There are two layers, each with one capsule.
The dark blue ring is the drill casing.

Source: Arnold et al. 2014, Figure 3-3.

Figure 4-13. Horizontal Cross Section of Concept for Axially Aligned Cs and Sr Capsules Used as Baseline Design for Study Purposes

Six-Capsule Package, 12.25-in.-Diameter Borehole—Besides the two-capsule waste package, Arnold et al. (2014) considered a slightly larger diameter package (OD of 7.5 in. with a wall thickness of 0.50 in.) that could take six capsules in two layers of three (Figure 4-14(a) and Table 4-7). This configuration would require a 12.25-in.-diameter borehole and 10.75-in. OD casing.

Reference SNF Package, 17-in.-Diameter Borehole—Another concept investigated in the 2014 study is based on the reference design for SNF disposal proposed by Arnold et al. (2011). The reference SNF design uses a 13.40-in. (OD) steel waste package and a 17-in.-diameter borehole. Depending on any constraints imposed by heat flow, this could be either a purpose-made waste package designed to take one or two seven-capsule layers (Figure 4-14(b)) or, depending on the length of the SNF waste package, up to eight layers. In the absence of any constraints arising from the temperatures generated in and around the waste packages, the entire capsule inventory could fit into approximately 35 reference SNF packages.



NOTE: a. Six-capsule design having two layers of three capsules each
b. SNL 2011 reference design having one or two layers of seven capsules each
Colors represent components/materials as in Figure 4-13.

Source: Arnold et al. 2014, Figure 3-4.

Figure 4-14. Horizontal Cross Sections of Possible Package Geometries Studied for Cs and Sr Capsules

Large Diameter SNF Package, 22-in.-Diameter Borehole—The largest borehole concept considered in the 2014 study was a 22-in. borehole. Large diameter waste packages that would need a correspondingly large diameter borehole have been proposed for SNF, specifically to take a complete pressurized water reactor fuel assembly (e.g., Gibb et al. 2012) or for vitrified reprocessing waste (Beswick et al. 2014). If used, this type of design could entail the disposal of the entire Cs and Sr capsule inventory in a few tens of packages deployed over a few hundred meters of borehole, with the higher costs of the larger hole being offset by savings on the length of the disposal zone required, 20 fewer waste packages, and reduced disposal operations.

4.5.2 Disposal of Untreated Calcine

Besides examining Cs and Sr capsules, the 2014 study considered the disposal of untreated calcine waste (Arnold et al. 2014, Section 3.3) using the SNL reference design described in Arnold et al. (2011). The 2014 study assumes a low-temperature waste package design because of the relatively low thermal output from the calcine waste. Each low-temperature disposal waste package has an internal volume of about 0.149 m^3. Disposal of all 4,400 m^3 of untreated calcine would thus require about 29,550 waste packages, assuming 100% filling. The reference design for borehole disposal includes 400 disposal packages, so the total number of disposal boreholes would be 74 boreholes. If the packages could only be filled to 90% of capacity, then about 82 disposal boreholes would be required.

A possible alternative for borehole disposal of calcine waste would be in larger-diameter boreholes that are not as deep as the reference design for deep borehole disposal. As described earlier, the bulk of the activity in the calcine waste is from relatively short-lived radionuclides, with considerably smaller contributions from other fission products and actinides (DOE 2014a). Boreholes with diameters of up to about 30 in. have been drilled to depths of 3,000 m (Beswick 2008). It is possible that disposal depths of 1,000 m to 3,000 m would provide adequate isolation for the calcine waste in a borehole disposal system. Long-term performance of this alternative, with disposal of calcine in shallower holes at depths of perhaps 3,000 m or less, remains to be evaluated. However, analyses of disposal of transuranic wastes in very large diameter (3 m) boreholes at significantly shallower depths (bottomhole depth of 36 m) have demonstrated safety for the particular environmental conditions of the Greater Confinement Disposal system at the Nevada Test Site (Cochran et al. 2001).

4.6 DBFT

While deep borehole disposal has been researched for many years, to date there has been no field testing of the concept. In 2014, the DOE issued a plan (SNL 2014c) for a field test designed to develop the logistics and advance the technical basis for the siting and implementation of a deep borehole disposal facility. The DBFT will be used to validate proof of concept, but will not involve the disposal of actual waste. There are three specific purposes: (1) to evaluate the capability for drilling and construction of deep, large-diameter boreholes, (2) to conduct downhole scientific analyses that assess the hydrogeochemical conditions pertinent to control waste stability and containment, and (3) to conduct an engineering analysis to assess the viability and safety of deep borehole waste package emplacement.

For the DBFT, two boreholes will be drilled up to 5,000 m into crystalline basement rock in a geologically stable continental location. The Characterization Borehole, with a planned diameter at total depth of approximately 8.5 in., will be drilled and completed first to facilitate downhole scientific testing (e.g., examination of hydrogeologic, geochemical, and geomechanical characteristics of the near-borehole host rock). The scientific testing and analysis activities will identify the critical downhole measurements that must be made to determine if conditions favorable to long-term isolation of high-activity waste exist at depth. The second borehole will be the Field Test Borehole. With a larger bottomhole diameter of 17 in., this borehole is designed to facilitate proof-of-concept for the disposal system using surrogate test packages. Engineering activities will evaluate the safety and efficacy of borehole drilling and construction; test packages; and the system for handling, emplacing, and retrieving packages in the borehole. In addition, borehole sealing materials and designs will be examined through above-ground testing.

To support the DBFT, one or more test packages will be instrumented for the purposes of collecting data at depth. Measurements that are of interest include temperatures, borehole fluid pressures, acceleration, mechanical strain, and emplacement fluid radioactivity.

Experience from the DBFT will further the understanding of instrumentation needed to facilitate actual deep borehole disposal. Because of the permanent and inaccessible nature of borehole disposal, instrumentation development will focus on the preclosure, rather than the postclosure, timeframe. During emplacement operations, the instrumentation should enable the surface crew working at an actual borehole disposal site to do the following: (1) monitor the condition of the instrumented waste packages as they are lowered in the borehole, (2) detect any radioactive contamination caused by leaking waste packages, and (3) detect any adverse conditions in the borehole during emplacement, such as collapsed casing.

Transmitting data collected at depth (temperature, fluid pressure, acceleration, etc.) to the surface in real time and in a reliable manner will require either a data link to the wireline cable head (for wireline emplacement) or a wireless telemetry string (for drill-string emplacement). Due to the small radial clearance between waste packages and guidance casing, it would be impractical to run cable there externally. For drill-string emplacement, internally wired waste packages could be developed, but would require high-pressure electrical connections that can be made up remotely in the radiological environment of the rig basement. A commercial wireless solution that has been developed for use in the oilfield is battery-powered electromagnetic telemetry. The DBFT project is currently coordinating with the developers of the universal canister system to develop test package instrumentation and telemetry with the emplacement method selected for design.

The DBFT comprises the following major activities: site selection and characterization, the drilling and construction of the Characterization Borehole and the Field Test Borehole, design of the engineering demonstration (e.g., packages, package handling, and package emplacement and retrieval), fabrication of packaging and emplacement equipment, and related scientific research and development activities to evaluate the deep borehole disposal concept. Key milestones as of July 2015 are shown in Figure 4-15. The Universal Canister Project will collaborate with the DBFT project to develop the surrogate waste packages for the field test, and to develop and test the system for handling and emplacing waste packages at the field test site.

	FY15	FY16	FY17	FY18	FY19
Site Management and Drilling Integration Services (SM&D) Draft RFP – Issued	◆ 04/07/15				
Field Test – Award Engineering Services Contract	◆ 06/25/15				
SM&D Final RFP – Issue	◆ 07/09/15				
SM&D RFP – Proposals Due	◆ 09/09/15				
SM&D – Award Contract		◆ 02/05/16			
Field Test Borehole Services – RFP Issued		◆ 06/03/16			
Field Test Borehole Services – Proposals Due		◆ 08/05/16			
Characterization Borehole – Start Drilling		◆ 09/01/16			
Field Test Borehole Services – Award Contract			◆ 01/13/17		
Characterization Borehole – Completed			◆ 02/27/17		
Field Test Borehole – Start Drilling			◆ 07/07/17		
Field Test Borehole – Completed				◆ 01/05/18	
Field Test – Start Emplacement Demonstration				◆ 01/17/18	
Field Test – Complete Emplacement Demonstration					01/17/19 ◆
Documentation – Field Test Analyses and Evaluation					09/30/19 ◆

Source: MacKinnon 2015.

Figure 4-15. DBFT Key Milestones

A detailed five-year schedule for all project work breakdown structure (WBS) elements supporting these major activities and milestones is provided in the DBFT plan (SNL 2014c, Appendix C). Of particular interest to the Universal Canister Project is WBS 1.7 Engineering and Demonstration, which is outlined with the schedule in Table 4-8. As work progresses on the DBFT and the Universal Canister Project, refinements to the schedule and WBS elements are expected. For example, the 2014 schedule includes the concept of drill string emplacement (WBS 1.7.1.1.10), but does not reflect the fact that a study considering various emplacement methods recommended the use of a wireline (SNL 2015).

Table 4-8. DBFT Project WBS Elements Pertaining to Developing and Testing Packages

DBFT Project WBS	Activity	Scheduled Fiscal Years (FY)
1.7	Engineering and Demonstration	FY15 – FY19
1.7.1	Package and Package Handling Equipment	FY15 – FY17
1.7.1.1	Packages	FY15 – FY17
1.7.1.1.1	Design Requirements	FY15
1.7.1.1.2	Conceptual Design	FY15
1.7.1.1.3	DBFT Specification Report	FY15
1.7.1.1.4	Package Specifications	FY16
1.7.1.1.5	Procure Single Test Package	FY16
1.7.1.1.6	Package Testing	FY16 – FY17
1.7.1.1.7	Package Testing Complete	FY17
1.7.1.1.8	Revise Package Specs If Needed	FY17
1.7.1.1.9	Specification Finalized	FY17
1.7.1.1.10	Procure Packages for String Testing	FY17
1.7.1.1.11	Test Packages Procured	FY17
1.7.1.2	Package Handling Equipment	FY15 – FY17
1.7.1.2.1	Design Requirements	FY15
1.7.1.2.2	Conceptual Design	FY15
1.7.1.2.3	Handling Design Requirements	FY16
1.7.1.2.4	Procure Handling Equipment	FY16
1.7.1.2.5	Evaluation and Testing Prototype	FY16
1.7.1.2.6	Revise Handling Equipment Specs If Needed	FY16 – FY17
1.7.1.2.7	Specifications Finalized	FY17
1.7.1.2.8	Procure Equipment	FY17
1.7.1.2.9	Handling Equipment Procured	FY17
1.7.2	Package Emplacement Demonstration	FY18 – FY19
1.7.2.1	Package Handling	FY18 – FY19
1.7.2.2	Package Lowering and Retrieval	FY19
1.7.2.3	Engineering Demonstrations Activities Completed	FY19

NOTE: The WBS activities in the DBFT plan refer to "canister" rather than "package." The wording was changed above to maintain consistency with the definitions provided in Section 1.5.

Source: modified from SNL 2014c, Appendix C.

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5 System Concept Description

This section presents preliminary system-level descriptions for the universal canister system. Unique design features must be developed and incorporated to enable a seamless transition between the different operational phases, with a primary focus on the borehole disposal emplacement phase from which requirements can be propagated back to the transportation package and the initial dry storage configuration to minimize financial and radiological liabilities. The universal canister system will handle individual modular canisters separately or collectively within the system's larger container (referred to as the cask). The cask is primarily used to move, store, and transport multiple canisters at a time. Initial concept focus will be on permanent disposal of Cs and Sr capsules from the WESF. The capsules are a well-defined waste form (i.e., encapsulated with known dimensions) that will provide an initial set of constraints to focus integrated system concept development. The calcine waste itself is well defined, although it has fewer constraints with respect to possible configurations of a universal canister system. A system concept for calcine waste is also discussed below. The Cs to be processed using elutable or nonelutable resins at the LAWPS is also being considered for the universal canister, but this waste is not yet well enough defined to discuss system concepts for a universal canister.

System-level requirements for the universal canister will be developed based on the following assumptions:

- Permanent disposal of these waste forms will be in deep boreholes or in a mined geologic repository in a crystalline rock, in halite (salt), or in a clay-shale formation.
- A modular universal canister system will be utilized for dry storage, transfers, transportation, and disposal. A universal canister may be directly loaded with a granular material, such as calcine waste, or may function as a container for an existing waste form, such as the Cs and Sr capsules at Hanford. The OD, wall thickness, and internal basket/support configuration of the universal canisters will be developed for efficient disposal of the candidate waste forms in a deep borehole. A modular-based canister system allows synergies between the different operational overpacks, promoting efficient handling and operations. A modular-based system also provides for flexibility in defining waste form specifications for the calcine, the Cs extracted from the elutable resin, or the Cs sorbed onto the nonelutable resin. For example, smaller diameter canisters can be efficient for deep borehole disposal of the Cs and Sr capsules, which is a small volume waste stream. Larger diameter canisters may be efficient for disposal of the larger volumes of calcine waste in a deep borehole or in a mined geologic repository. The specific operational overpacks for storage, transportation and disposal will be determined by a systems engineering effort during the next phase of this work.
- The Cs and Sr capsules will not be reopened during storage, transfer, transportation, or disposal operations. Similarly, a universal canister may be placed in a cask (i.e.,

overpacked) for storage, transfers, transportation, or disposal, but the canister will not be reopened once it is filled with waste and sealed.

- The Cs and Sr capsules that are currently stored in pool cells at the WESF will be moved into dry storage on the Hanford Site before being transported to the permanent disposal site.

The system concept for disposal of the Cs and Sr capsules is discussed separately from the disposal system concept for the calcine waste. The predicted activity of the capsules in 2048, 4.39×10^{7} Ci, is approximately three times greater than the predicted activity of the calcine waste in 2048, 1.48×10^{7} Ci (SNL 2014b, Table F-2). The disposal volume of untreated calcine waste, 4,400 m^3, is more than 1,000 times greater than the volume of the capsules, about 4 m^3. The capsules represent a small volume waste stream with high levels of decay heat and activity, while the calcine waste has a much larger volume with lower levels of decay heat. Given these differences, it is appropriate to discuss the disposal system concepts for these waste streams separately.

5.1 System Concept for Disposal of Cs and Sr Capsules

Figure 5-1 illustrates the major steps in permanent disposal of the Cs and Sr capsules currently at the WESF. The pictures in Figure 5-1 are only for illustrative purposes and do not represent proposed designs for casks/overpacks for storage, transfers, transportation and disposal. The individual steps in Figure 5-1 are as follows:

(a) Remove Capsules from Pool Cell(s)

Capsules will be removed from their pool cells.

(b) Load and Seal Capsules in Universal Canister(s)

After removal, capsules will be dried, placed in a universal canister and the canister will be sealed shut. This activity may take place in hot cell G of the WESF. There is also a cask pit area that could possibly support a removal strategy that avoids the use of the hot cell. If existing facilities are deemed not suitable for this activity, then a new facility may need to be constructed. The universal canister specifications will be based on the borehole disposal requirements as well as on storage and transportation requirements. Design specifications would need to consider the use of corrosion-resistant materials as well as the thermal, chemical, and pressure conditions in a deep borehole (Section 4.5.1) for a prescribed period of time. Preliminary deep borehole concepts are described in Table 5-1. All options would require casks for transportation and storage; however, a separate disposal overpack may or may not be used. The design of the universal canister will be developed in collaboration with the DBFT project.

NOTE: The pictures are only for illustrative purposes; actual designs have yet to be determined.

Figure 5-1. System Concept for Disposal of Cs and Sr Capsules in a Deep Borehole or Mined Geologic Repository

(c) Transfer Universal Canisters to Dry Storage Cask

Assuming hot cell G is used to transfer the canisters to dry storage, at the start of loading operations (i.e., moving the capsules into a canister), a transfer cask will be loaded into the hot cell. The transfer cask should be selected/developed considering (1) the facility constraints, (2) the process and operational requirements for moving the capsules from wet storage to the hot cell, drying them, placing them into the canisters, and sealing the canisters, and (3) transferring the waste-filled universal canisters from the hot cell to a dry storage configuration. The selection/development of the transfer cask will consider DOE–EM-certified transportation packagings and also account for (1) transfer from the storage cask to the transportation overpack if two separate overpacks are used, and/or (2) transfer to the waste package delivery system at the borehole or mined geologic repository. If, rather than using hot cell G, the cask pit in the WESF pool is used to remove the capsules from the pool and load them into universal canisters, the steps for transferring the loaded universal canisters to dry storage would require further evaluation. In addition, developments in the WESF extended storage project will be considered in the design of the universal canister system.

A preliminary conceptual design for a dry storage cask was developed and evaluated by the CAP during the Hanford Site CDSP (Heard et al. 2003). This information is relevant because detailed thermal, corrosion, and accident analyses demonstrated that up to 16 Cs capsules could be stored in a single dry storage overpack and still meet the performance requirement for a 50-year lifetime (Plys and Miller 2003; Heard et al. 2003). Additional details are provided in Section 4.1.1.

(d) Store Universal Canisters

As noted in Section 4.1.1, a newly constructed area adjacent to the Hanford 200 Area Interim Storage Area is likely to be the dry storage facility at which the capsules will be stored initially. The canisters will remain in dry storage until the disposal site is ready for receipt. While in storage, considerations regarding canister system integrity and safeguarding (e.g., seal monitors) need to be addressed.

(e) Transport Canister(s) to Disposal Site

When the disposal site is ready to receive the canister(s), the canister(s) will be transferred from the dry storage cask, packaging, or vault and placed into a transportation cask for shipment to the repository. If the storage cask or packaging is also rated for transport, then it would be loaded onto the transport vehicle for shipment to the disposal site. Transportation by truck, railcar, or barge is envisioned; Figure 5-1(e) illustrates the option to transport via truck.

(f) Emplace Waste Packages: Deep Borehole Disposal or Mined Geologic Repository

The universal canister(s) will be removed from the transport cask, placed in a disposal overpack (if required), and emplaced in the deep borehole. The need for and design of the disposal overpack will be determined by a systems engineering effort during the next phase of this work. Deployment of the final waste packages (consisting of the universal canister plus disposal overpack, if needed) could be done singly, in small batches, or in longer strings. Emplacement could be by wireline (for a single package), by coiled tubing, or by the drill string (for multiple packages) (Cochran and Hardin 2015). Figure 5-1(f1) shows permanent disposal in a deep borehole because the 2,000-m-long disposal zone of a single deep borehole is sufficient to hold the universal canisters containing all the capsules from the WESF. Figure 5-1(f2) shows disposal in a mined geologic repository as an alternative possibility.

Table 5-1 shows possible design parameters for deep borehole disposal using an 8.5-in. diameter and a 12.3-in.-diameter borehole at depth as proposed by Hardin (2015b). The configuration for the waste package considers two representative options: one capsule per layer and three capsules per layer. The disposal zone is 2,000-m long, and extends from 3,000 to 5,000 m below the surface. In both concepts, all of the waste packages only partly fill a single borehole, as shown by the fill fraction on the last line in Table 5-1.

Table 5-1. Possible Design Parameters[a] for Capsule Disposal in a Deep Borehole

Disposal Zone Borehole Diameter	8.5 in.	12.3 in.
Waste Package[b] OD	5 in.	8.5 in.
Waste Package[b] ID	4 in.	6.5 in.
Canister Arrangement inside Waste Package	1 per layer with up to 8 layers	3 per layer with up to 8 layers
Total Height of Waste Packages	~4,500 ft (~1,400 m)	~1,500 ft (~460 m)
Fill Fraction of Disposal Zone	69%	23%

NOTE: [a] The design parameters in this table differ from those in Table 4-7 (from Arnold et al. 2014) because of different design assumptions. These changes are indicative of the evolution of the deep borehole concept over time. The dimensions of the borehole and waste package are for study purposes only. The dimensions to be used for actual deep borehole disposal have yet to be determined.

[b] Waste package refers to the universal canister and its disposal overpack (if needed).

ID = inside diameter.

OD = outside diameter.

Source: Hardin 2015b.

Although the radiation dose received by the public is expected to be inconsequential, ALARA practices must be followed by workers during processing and emplacement of the waste packages. The design of the handling and emplacement systems at the disposal site will be developed in collaboration with the DBFT project.

The Cs and Sr capsules could also be sent to a mined geologic repository for disposal.

5.2 System Concept for Disposal of Calcine Waste

This section presents a preliminary system-level description for permanent disposal of calcine from the CSSFs at INTEC (Section 2.3). The disposal concept is based on similar assumptions to those for disposal of Cs and Sr capsules: (1) permanent disposal of calcine will be in deep boreholes or in a mined geologic repository; (2) a modular canister-based system will be utilized for dry storage, transfers, transportation, and disposal; and (3) canisters with calcine waste will not be reopened once they are sealed.

Figure 5-2 depicts the major steps in permanent disposal of the calcine waste from INL. The pictures in Figure 5-2 are only for illustrative purposes and do not represent proposed designs for casks/overpacks for storage, transfers, transportation, and disposal. The individual steps in Figure 5-2 are as follows:

(a) Retrieve Calcine from Bin Sets

Calcine will be removed from the stainless steel bins and loaded into a universal canister. The process for transferring the calcine waste from the 43 storage bins and ensuring a specified fill level in the universal canisters has not yet been defined. The calcine will not be otherwise processed or treated before being loaded into the universal canister.

(b) Compact Waste and Seal the Canister(s)

The canister used for the calcine would likely be larger than the one used for the Cs and Sr capsules to facilitate the removal operation; the canister could be either thick-walled (i.e., designed to withstand environmental conditions in a deep borehole without a disposal overpack) or thin-walled.

(c) Place in Interim Storage

The waste-filled canister would be transferred to a storage cask or vault and placed in interim storage pending shipment to the disposal site. Because the calcine is already safely stored, there is no pressing need for an alternative calcine dry storage facility. Therefore, a just-in-time approach will likely be used to retrieve, store, transport, and dispose of the calcine waste once a disposal facility exists and is ready to accept the calcine waste. The interim storage is thus expected to be lag or buffer storage, rather than extended storage. The location of the storage site has not been identified.

NOTE: The pictures above are only for illustrative purposes; actual designs have yet to be determined.

Figure 5-2. System Concept for Disposal of Calcine Waste in a Deep Borehole or Mined Geologic Repository

(d) Prepare Canisters for Transportation to the Disposal Site

When the disposal site is ready to receive the waste, each canister will be transferred from the dry storage cask or vault and placed in a transportation cask for shipment to the repository. If the storage cask is also rated to serve as a transport cask, then this single cask will be loaded onto the transport vehicle. The transport vehicle could be a truck, a railcar, or a barge.

(e) Transport to Disposal Site

The waste-filled transportation cask will be transported to the disposal site by truck (as shown) or possibly by rail or by barge.

(f) Emplace Waste Packages: Deep Borehole Disposal or Mined Geologic Repository

If necessary, the universal canister will be placed in a disposal overpack for permanent disposal. Figures 5-2(f1) and 5-2(f2) show disposal in a deep borehole or a mined geologic repository.

A key feature of calcine pertinent to repository design is that calcine generates low levels of decay heat. The four types of calcine will produce between 3 and 40 W/m^3 (as of 2016) (Section 2.3.3). If calcine fills a cylindrical container with an inside diameter (ID) of 1 ft and an inner length of 15 ft, the total volume of calcine is about 12 ft^3 (0.33 m^3) and produces between 1 and 13 W of decay heat. As a comparison, an average Cs capsule produces about 119 W of decay heat and an average Sr capsule produces about 157 W of decay heat.

Table 5-2 shows possible design parameters for deep borehole disposal of untreated calcine in waste packages using 17-in.- and 22-in.-diameter boreholes (Hardin 2015b). The disposal zone extends from 3,000 m to 5,000 meters in depth, and the packages are stacked end-to-end in a borehole. The packages require a substantial wall thickness, 2.5 in. or 4 in. for a 17-in.- or 22-in.-diameter borehole (respectively) at depth, because the packages must withstand the hydrostatic pressure at the maximum depth of 5,000 m and required wall thickness increases as the package diameter increases. Disposal of the total untreated volume of calcine, 4,400 m^3, requires about 74 boreholes for a 17-in.-diameter borehole and about 38 boreholes for a 22-in.-diameter borehole.

At the present time, permanent disposal of calcine waste in either a deep borehole or a mined geologic repository are reasonable options. In the future, systems engineering studies will evaluate the relative merits of disposal in a deep borehole disposal versus mined geologic repository for calcine waste.

Table 5-2. Possible Design Parameters[a] for Disposal of Untreated Calcine in a Deep Borehole with Large Diameter at Depth

Disposal Zone Borehole Diameter	17 in.	22 in.
Waste Package[b] OD	11 in.	16 in.
Waste Package[b] ID	8.5 in.	12 in.
Length of Waste Package	16.7 ft	16.7 ft
Disposal Volume/Borehole	2,090 ft^3	4,170 ft^3
Total Number of Boreholes	~74	~38

NOTE: [a] The design parameters in this table differ from those in Table 4-7 (from Arnold et al. 2014) because of different design assumptions. These changes are indicative of the evolution of the deep borehole concept over time. All dimensions for the borehole and waste package are for study purposes only. The dimensions to be used for actual deep borehole disposal have yet to be determined.

[b] Waste package refers to the universal canister and its disposal overpack (if needed).

ID = inside diameter.

OD = outside diameter.

Source: Hardin 2015b.

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6 Key Requirements, Parameters, and Components

6.1 Universal Canister

This section defines preliminary performance requirements and preliminary interface requirements for a family of universal canisters that can be used for disposal of waste in deep boreholes or in a mined geologic repository. Three types of waste are currently being considered for the universal canister: (1) the Cs and Sr capsules at the Hanford Site, (2) calcine waste currently being stored at the INL, and (3) Cs extracted from elutable resins or Cs bound to nonelutable resins that may be generated at the LAWPS at the Hanford Site. The design features of the disposal medium (deep borehole or mined geologic repository) and the properties of these three waste streams have a direct impact on the performance requirements and operational requirements. It follows that the requirements in this section are preliminary and likely to evolve as the designs of the disposal systems are refined due to new design studies and refined due to systems engineering evaluations and tradeoffs.

6.1.1 General

- The family of universal canisters shall be capable of being loaded with capsules containing Cs or Sr from the WESF at Hanford, bulk calcine waste from the CSSF at INL, and Cs removed from elutable resins or bound to nonelutable resins at the LAWPS.
- The universal canister shall be a family of right circular cylinders compatible with deep borehole disposal. The OD, wall thickness, length, and material requirements shall be determined in conjunction with the DBFT project.
- The designed maintainable service lifetime of the universal canister shall be consistent with plans for storage and aging of the Cs and Sr capsules and with the need to isolate long-lived radionuclides in the geologic environment.
- Waste acceptance criteria shall specify the surface service lifetime of the universal canister as part of the universal canister system at the disposal facility. A final determination of the length of time will be based on the need for storage or aging at the disposal facility.
- The universal canister shall be designed for storage in a vertical or horizontal orientation. The preferred orientation has not yet been determined.
- The universal canister shall be designed for transportation between sites in a horizontal or vertical configuration. The preferred orientation has not yet been determined.
- A canister-lifting feature shall be incorporated into the top lid of the universal canister and shall not protrude beyond the canister sidewalls. The lifting feature may be integral with the lid or mechanically attached.

- Connections for drill-string emplacement of universal canisters in a deep borehole may include (1) a threaded connection to the canisters below, and (2) a threaded connection to drill pipe above for emplacement or fishing. Connections for wireline emplacement may include a releasable cable head and a fishing neck, both located on top of the canister. The bottom of the canister may include a threaded connection for attaching additional hardware such as instrumentation, centralizers, or shock-absorbing materials. This requirement applies to canisters that can be disposed of without a disposal overpack.
- All external edges of the universal canister shall have a radius of curvature sufficient to protect against gouging of the internal surfaces of overpacks for storage and transportation.
- Projections or protuberances on the lateral surface of the canister shall be minimized to facilitate loading into a storage or transportation overpack with a low potential for damage to the interior of the overpack.

6.1.2 Structural

- The family of universal canisters shall be designed to maintain its mechanical integrity under external loads, which may include hydrostatic load, axial tension, axial compression, and possibly bending loads.
- The universal canisters for deep borehole disposal that are to be disposed of without a disposal overpack must withstand the hydrostatic pressure in the underground environment. Pending the results of the DBFT, the maximum design hydrostatic pressure for waste packages at the bottom of the 5,000-m borehole is estimated to be 65 MPa (640 atm), based on an assumed fluid density in a 5,000-m column. The minimum hydrostatic pressure is 50 MPa (490 atm), based on the density of pure water (Hardin 2015a, Section 2.10).
- The structural analysis for the canister designed to be disposed of without a disposal overpack shall incorporate a minimum safety factor of 2.0 with respect to elastic/plastic failure calculations for canister wall thickness (Hardin 2015a, Section 2.10).
- Connections for drill-string emplacement or wireline emplacement of universal canisters designed to be disposed of without a disposal overpack in a deep borehole shall have sufficient strength to withstand mechanical loads during emplacement, retrieval, and fishing of stuck canisters (Hardin 2015a, Section 2.10).

6.1.3 Thermal

- The maximum surface temperature of the universal canister shall be limited to maintain the integrity of the canister and its contents during storage, on-site transfers, off-site transportation, and disposal operations.

6.1.4 Criticality

- No criticality requirements are anticipated for universal canisters filled with Cs capsules or Sr capsules. The elemental composition of the CsCl and SrF_2 waste does not include uranium or plutonium, and the compound compositions of the CsCl and SrF_2 waste do not include any uranium-based or plutonium-based compounds (Tables 2-4 and 2-5 of this report). The conclusion is that fissile radioisotopes, such as ^{239}Pu, ^{241}Pu, ^{233}U, or ^{235}U, are not present in the capsules.

- The need for criticality requirements for calcine waste and for Cs treated with resins shall be considered in the future.

6.1.5 Containment

- Containment of the waste within the universal canister and its overpacks is required during all phases of waste management operations until the borehole is sealed. Additional containment longevity may be required depending on the disposal environment, the half-life of radionuclides in the waste, and other characteristics of disposal operations and the disposal system.
- If the universal canister is identified as the containment barrier, then it shall be designed to be "leak tight," as defined in ANSI N14.5-2014 (ANSI 2014).
- An inert gas, such as helium, shall be the universal canister fill gas.

6.1.6 Materials

- The shells and lids for the universal canister shall be designed and fabricated in accordance with the applicable industry standards.
- The materials for fabrication of the universal canister are to be determined. Type 300-series stainless steel will be considered, as well as more corrosion-resistant materials if required to meet the service lifetime of the universal canister.
- Selection of the universal canister system materials shall consider the effects of uniform corrosion, pitting, stress corrosion cracking (SCC), or other types of corrosion under the environmental conditions and dynamic loading effects relevant to storage, transfers, transportation, and disposal in a deep borehole or mined geologic repository.
- All external welds on the universal canister except the closure welds shall be treated (e.g., stress relieved) prior to loading to mitigate the potential for SCC. The final closure welds shall be capable of being treated after loading.
- The following is a list of prohibited or restricted materials:

  - The universal canister shall not use organic, hydrocarbon-based materials of construction.
  - The universal canister shall not be constructed of pyrophoric materials.

- The following is a list of marking requirements:
  - The universal canister shall be capable of being marked on the lid and body with an identical unique (vendor independent) identifier prior to delivery for loading.
  - The markings shall remain legible for the service life of the canister without intervention or maintenance during normal operations and off-normal conditions associated with loading, closure, storage, transportation, aging, and permanent disposal.

6.1.7 Security

- Security requirements will be met consistent with the waste classification and attractiveness of the waste during storage, on-site transfers, transportation, and disposal.

6.1.8 Interfaces with Hot Cell G at the WESF

Universal canisters may be loaded with capsules in hot cell G at the WESF or may be loaded after the capsules are transferred to a specifically designed transfer facility at Hanford. If the canisters are loaded in hot cell G, then the canister must be compatible with the physical dimensions and operational constraints of hot cell G. If canisters are not loaded in hot cell G, then these performance requirements will be applicable to the transfer cask, rather than the canister.

- The WESF has a single working hot cell, namely hot cell G. Hot cell G may be used to transfer capsules from pool cells to transfer casks, such as the BUSS cask (Section 4.4.2), or may be used to load capsules directly into universal canisters. The physical dimensions of hot cell G and the operational limits when using hot cell G must be considered in developing a disposal system using a family of universal canisters.
- The system for loading universal canisters or the transfer cask shall be compatible with the physical dimensions of hot cell G and with the equipment (e.g., hoist and crane) in hot cell G, as presented in Table 4-3.
- During operations, the heat load limit in hot cell G, 1,800 W (Covey 2014, Section 3.1.3), shall not be exceeded.
- The capsule inventory limit in hot cell G shall not exceed 1.50×10^5 Ci ^{90} of Sr and 1.50×10^5 Ci ^{137} of Cs (Covey 2014, Section 3.1.3).

6.2 Regulatory Requirements and DOE Orders

The regulatory requirements and DOE orders that apply to storage, transport, and disposal are discussed below in Sections 6.2.1, 6.2.2, and 6.2.3, respectively. Regulatory requirements for transfers are not discussed separately. Transfers typically occur at a storage facility or a disposal facility, and the regulatory requirements for the storage facility or the disposal facility would apply to the transfer as well.

6.2.1 Storage

The requirements for storage of the universal canisters containing Cs and Sr capsules, Cs extracted from elutable resins, Cs sorbed onto nonelutable resins, and calcine wastes are derived primarily from DOE orders and manuals. The regulatory requirements that are related to the storage of universal canisters are summarized below. These requirements apply to storage of waste at a facility that is not part of a geologic repository; requirements that apply to storage of waste at a facility that is part of a geologic repository are discussed in the context of disposal requirements (Section 6.2.3). It is assumed that the universal canisters at the storage facility are already loaded with the waste material and sealed, and that they remain sealed throughout their storage lifetime and upon either transfer on-site or transportation from the storage facility to the disposal site (i.e. either a mined geological repository or deep borehole). Although the NRC does not have regulatory authority over storage of these wastes, the DOE may choose to require that the storage system meet certain functional requirements found in NRC regulations (e.g., 10 CFR Part 72).

DOE Order (O) 435.1, Radioactive Waste Management

This order, DOE O 435.1, applies to the radioactive waste for which the DOE is responsible, including HLW, transuranic waste, low-level waste, and the radioactive component of mixed waste. DOE O 435.1 requires that the management of the storage of the canistered waste, including during storage and handling at a storage facility, be systematically planned, documented, executed, and evaluated in order to do the following:

- Protect the public from radiation exposure
- Protect the environment in accordance with the requirements of DOE O 5400.1 and DOE O 5400.5
- Protect workers in accordance with 10 CFR Part 835 and DOE O 440.1A
- Comply with all applicable Federal, State, and local laws and regulations

It also specifies that the waste shall be managed in accordance with DOE Manual (M) 435.1-1, *Radioactive Waste Management Manual*.

Since DOE O 435.1 was issued, DOE O 5400.1 has been cancelled and the orders that replaced it have been cancelled. DOE O 5400.5 has been replaced by DOE O 458.1, and DOE O 440.1A has been replaced by DOE O 440.1B Chg 2, which is addressed below.

DOE M 435.1-1, Radioactive Waste Management

Chapter II of this manual is specific to the management of HLW. Of the many requirements that apply, those of interest to specifying the requirements for storage of waste in a universal canister are the following:

- The storage facility shall have waste acceptance requirements and a waste certification program as part of its radioactive waste management basis (II.F.(3)). Further requirements for waste acceptance are given in II.J, and further requirements for waste certification are given in II.M. Guidance regarding development of the waste acceptance requirements and the waste certification program for HLW can be found in DOE Guide 435.1-1 Chapter II.

- Waste transfers shall be authorized prior to the transfer occurring, and data and records of the transfer shall be kept (II.N).

- The design of the storage facility must meet requirements for safety, confinement, lifting devices, ventilation, decontamination and decommissioning, exposure reduction, receipt and retrieval, structural integrity, instrumentation and control systems, and leak detection systems (II.P.(2) and II.Q).

- Storage facilities shall be monitored to detect failure of system confinement, integrity, or safety, which could lead to abnormal events or accidents (II.T).

DOE O 458.1, Radiation Protection of the Public and the Environment

This order applies to all DOE sites that manage radioactive waste and is intended to protect the public and the environment against undue risk from radiation associated with the radiological activities conducted by the DOE. In particular, storage of radioactive waste shall comply with an ALARA process and shall not result in a total effective dose greater than 25 mrem to a member of the public.

DOE O 470.4B, Safeguards and Security Program

This order establishes requirements for preventing unacceptable adverse impacts by specifying safeguards and security programs that are designed to protect assets and activities against the consequences of actions that include attempted theft, diversion, terrorist attack, radiological sabotage, unauthorized access, compromise, and other acts that may have an adverse effect on national security or the environment or that may pose significant danger to the health and safety of DOE federal and contractor employees or the public (§§470.4B.1, 4.f., and Attachment 1.1.f.).

10 CFR Part 835, Occupational Radiation Protection

This regulation, 10 CFR Part 835, establishes DOE requirements for “radiation protection standards, limits and program requirements for protecting individuals from ionizing radiation resulting from the conduct of DOE activities” (§ 835.1(a)), which would include providing radiation protection for all operations associated with the storage of universal canisters in a storage facility. The regulation sets limits on occupational doses and on the dose to a member of the public entering a controlled area. The storage facility is to be designed to maintain radiation exposure ALARA.

DOE O 440.1B, Worker Protection Program for DOE (Including the National Nuclear Security Administration) Federal Employees

DOE O 440.1B specifies a Worker Protection Program for DOE (including the National Nuclear Security Administration) federal employees. The objective of this order is to establish the framework for an effective worker protection program that will reduce or prevent injuries, illnesses, and accidental losses. These requirements are generally applicable to all DOE elements; there are no requirements that are specific to a universal canister storage facility.

Washington Administrative Code (WAC) 173–303, Dangerous Waste Regulations

The State of Washington has promulgated regulations regarding the management of dangerous waste in the state. The State of Washington is authorized to implement RCRA requirements; these requirements are in WAC 173–303. The State of Washington considers the Cs and Sr capsules to be dangerous waste and has issued a permit to the DOE to operate the WESF at Hanford (Washington Department of Ecology 2008). If the capsules are packaged into universal canisters and stored at another location at Hanford, another permit for such storage would need to be obtained.

Idaho Administrative Code (IAC) Chapter 58.01.05, Rules & Standards for Hazardous Waste

The State of Idaho has promulgated rules regarding the management of hazardous waste in the state. The State of Idaho is authorized to implement RCRA requirements; these requirements are incorporated by reference in IAC Chapter 58.01.05. The State of Idaho has issued a permit to the DOE to operate the CSSF at INL (Idaho Department of Environmental Quality 2006). If the calcine waste were to be repackaged into universal canisters and stored at another location at INL, another permit for such storage would need to be obtained.

DOE O 460.1C, Packaging and Transportation Safety

DOE O 460.1C establishes requirements for the packaging and transportation of both off-site shipments and on-site transfers of radioactive material. On-site transfers are defined to occur within the boundaries of a DOE site or facility to which access is controlled. Any movement of the loaded universal canisters within the boundaries of a DOE site would be deemed to be on-site transfer. An example would be transferring waste-filled universal canisters from the WESF to a storage facility on the Hanford Site.

On-site transfers will be subject to the applicable requirements specified in DOE O 460.1C. Specifically, such transfers must be conducted in accordance with one of the following requirements:

- The *Hazardous Materials Regulations* (49 CFR Parts 171–180) and the *Federal Motor Carrier Safety Regulations* (49 CFR Parts 350–399)
- A Transportation Safety Document that has been approved by the relevant Head of the Operations Office or the Field Office/Site Office Manager, as appropriate

If an on-site transfer is to be undertaken in accordance with a Transportation Safety Document, it must have a description of the methodology and compliance process used that meets a standard of safety equivalent to that which one would obtain from application of 49 CFR Parts 171–180 and 49 CFR Parts 350–399 (§O 460.1C 4.b(2)(a)).

6.2.2 Transportation

All regulatory matters for the transportation of hazardous materials, including radioactive materials, within the United States fall under the authority of the Department of Transportation (DOT). The regulations for which DOT is the Competent Authority are found within Title 49 of the CFR. Because the waste materials may or may not be considered special form, different quantity limitations will apply to determining the package classification. If the capsules are considered to be special form, the determination of packaging will be dependent on the activity level of the waste and the A_1 values given in 49 CFR Part 173.435 and Appendix A of 10 CFR Part 71. If the capsules are considered to not be special form, the determination of packaging will be dependent on the activity level of the waste and the A_2 values given in 49 CFR Part 173.435 and Appendix A of 10 CFR Part 71. The A_1 value for ^{137}Cs is 54 Ci, and the A_2 value for ^{137}Cs is 16 Ci. The A_1 value and the A_2 value for ^{90}Sr are 8.1 Ci. Because the activity of the waste capsules is in excess of all of these values, the waste capsules will be transported in Type B packages. 49 CFR Part 173.413 states that Type B packages must be designed and constructed to meet the applicable requirements in 10 CFR Part 71, *Packaging and Transportation of Radioactive Material*. In general, Type B packages certified under 10 CFR Part 71 are certified by the NRC, however, 49 CFR Part 173.7(d) allows DOE to evaluate, approve, and certify "packagings made by or under the direction of DOE…against packaging standards equivalent to those specified in 10 CFR Part 71. Packagings shipped in accordance with [49 CFR Part 173.7(d)] shall be marked and otherwise prepared for shipment in a manner equivalent to that required…for packagings approved by the NRC." Thus, regardless of the certifying authority (DOE or NRC), the package must comply with 10 CFR Part 71.

The regulations and DOE Orders regarding the package for transportation are detailed in this section. Regulations that do not affect the design, functions, or requirements of the package (i.e., DOT requirements such as labeling and placarding) are outside the scope of this document.

Due to the expected high radiation dose rates from this waste material, it is likely that these packages would be transported as exclusive-use shipments, where exclusive use is defined as "the sole use by a single consignor of a conveyance for which all initial, intermediate, and final loading and unloading are carried out in accordance with the direction of the consignor or consignee. The consignor and the carrier must ensure that any loading or unloading is performed by personnel having radiological training and resources appropriate for safe handling of the consignment. The consignor must issue specific instructions, in writing, for maintenance of exclusive use shipment controls, and include them with the shipping paper information provided to the carrier by the consignor."

By 10 CFR Part 71.41, it must be demonstrated by using a package specimen or a scale model that the package will pass (1) all of the tests for NCT in 10 CFR Part 71.71, (2) all of the tests for hypothetical accident conditions (HAC) in 10 CFR Part 71.73, and (3) if more than 10^5 A_2 are contained in a single package, the requirements in 10 CFR Part 71.61.

By 10 CFR Part 71.43, the outside of the "package must incorporate a feature, such as a seal, that is not readily breakable and that, while intact, would be evidence that the package has not been opened by unauthorized persons." It also "must include a containment system securely closed by a positive fastening device that cannot be opened unintentionally or by a pressure that may arise within the package." It must be ensured that no chemical reaction can take place within the package. When subjected to the NCT tests, the package must contain all radioactive contents, to a sensitivity of 10^{-6} A_2 per hour (10 CFR Part 71.51(a)(1)), and the external radiation levels must not be significantly increased. No accessible surface of the cask may exceed 185°F or 85°C when this package is transported as an exclusive use shipment in still air at 100°F (38°C) in the shade. Finally, continuous venting will not be allowed during transport.

For exclusive use shipments, by 10 CFR Part 71.47, the following radiation levels must not be exceeded:

- 2 mSv/hr (200 mrem/hr) on the outside surface, unless "the shipment is made in a closed transport vehicle, the package is secured within the vehicle so that its position remains fixed during transportation, and there are no loading or unloading operations between the beginning and end of the transportation," in which case the radiation level on the outside surface must be below 10 mSv/hr (1000 mrem/hr)
- 2 mSv/hr (200 mrem/hr) on the outside of the vehicle. If a flatbed trailer is used to transport the waste material, this radiation limit applies to "any point on the vertical planes projected from the outer edges of the vehicle."

- 0.1 mSv/hr (10 mrem/hr) "at any point 2 m (80 in.) from the outer lateral surfaces of the vehicle (excluding the top and underside of the vehicle); or in the case of a flat-bed style vehicle, at any point 2 m (6.6 ft) from the vertical planes projected by the outer edges of the vehicle"
- 0.02 mSv/hr (2 mrem/hr) "in any normally occupied space" unless the transport crew wears dosimeters

Under the HAC tests, the package cannot release a total of more than A_2 in one week; the requirements involving ^{85}Kr will not apply to this content. The radiation levels may not exceed 10 mSv/hr (1 rem/hr) at 1 m from the package after the HAC sequence of tests is complete.

If more than 10^5 A_2 are contained in one package, 10 CFR Part 71.61 applies, which states that "its undamaged containment system can withstand an external water pressure of 2 MPa (290 psi) for a period of not less than 1 hour without collapse, buckling, or in leakage of water."

Requirements specific to transportation of fissile material and plutonium do not apply to this package when transporting Cs or Sr capsules; the applicability of fissile material transportation regulations for shipments of calcine waste and for Cs treated with resins shall be considered in the future.

6.2.2.1 NCT Tests

Packages must withstand tests designed to evaluate performance under NCT. A battery of 10 tests is prescribed by 10 CFR Part 71.71, such that the package must not release more than 10^{-6} A_2 per hour as a result of any of these tests. Separate package specimens may be used for the free drop, compression, and penetration tests as long as in each case they are subjected to the water spray test first. The tests are to take place at a constant temperature between –20°F and 100°F (–29°C and 38°C), at whichever temperature will cause the most unfavorable conditions for the package's performance. The internal pressure of the package must be the maximum normal operating pressure, unless a lower pressure is expected to be more unfavorable to the performance of the package. The test descriptions are as follows:

- **Heat—**The package is placed in still air at 100°F (38°C), and subjected to insolation of an average heat flux of 800 W/m^2 for 12 hours on flat upward-facing surfaces when the package is transported horizontally (e.g., the lid of a drum-type package), 200 W/m^2 on flat surfaces that are not horizontally facing upwards, and 400 W/m^2 on curved surfaces.
- **Cold—**The package is placed in still air without insolation at –40°F (–40°C).
- **Reduced External Pressure—**The pressure outside the package is reduced to 25 kPa (0.247 atm).
- **Increased External Pressure—**The pressure outside the package is increased to 140 kPa (1.382 atm).

- **Vibration—**The package is subjected to vibration representative of NCT.
- **Water Spray—**The package is subjected to a spray of water that simulates rain falling at 2 in./hr (5 cm/hr) for at least 1 hour.
- **Free Drop—**At least 1.5 hours after the water spray test concludes, but no more than 2.5 hours later, the package is dropped onto a flat unyielding horizontal surface. The drop should cause the package to fall in the position for which the maximum damage is expected. The drop height is dependent on the weight of the package; packages weighing over 15,000 kg must be dropped from 1 ft (0.3 m). Lighter weight packages must be dropped from greater heights, as given by 10 CFR Part 71.71 (7).
- **Corner Drop—**This test is for packages of lesser weight than that expected to transport this waste material, and so will not apply.
- **Compression—**The compression test is applied to packages weighing up to 5,000 kg (11,000 lb); it is expected that this test will not apply to the transportation package for this waste material.
- **Penetration—**A steel cylinder 3.2 cm (1.25 in.) in diameter and 6 kg (13 lb) in mass is dropped from 1 m (40 in.) onto the surface of the package expected to be most vulnerable to puncture. The part of the steel cylinder striking the package is to be hemispherical in shape, and the long axis of the cylinder is to be perpendicular to the package.

6.2.2.2 HAC Tests

The transportation package for this waste material will be subject to four tests, the first three of which must be performed in sequence on the same package. A separate package may be used for the fourth test (immersion test). These tests are described by 10 CFR Part 71.73. Due to the expected large mass of the package, and the fact that the contents are not fissile, the crush test and the immersion test specific for fissile material described in 10 CFR Part 71.73(c)(2) and 10 CFR Part 71.73(c)(5) will not apply for the transport of Cs and Sr capsules. As with the NCT tests, the HAC tests are to take place at a constant temperature between -20^{o}F and 100^{o}F (-29^{o}C and 38^{o}C), at whichever temperature will cause the most unfavorable conditions for the package's performance. The internal pressure of the package must be the maximum normal operating pressure, unless a lower pressure is expected to be more unfavorable to the performance of the package. The test descriptions are as follows:

- **Free Drop—**The package is dropped 9 m (30 ft) onto a horizontal unyielding flat surface in a position such that the maximum damage is expected.
- **Puncture—**The package is dropped 1 m (40 in.) onto a vertical cylindrical steel bar mounted on a flat unyielding surface. The package must be dropped so that the most vulnerable surface strikes the steel bar. The steel bar must be 15 cm (6 in.) in diameter and at least 20 cm (8 in.) in length, but the length should be expected to cause maximum

damage to the package. The top of the steel bar should be horizontal, with the edge rounded to a radius of at most 6 mm (0.25 in.).

- **Thermal—**The package must be engulfed in a fire of hydrocarbon fuel and air, which will provide an average emissivity coefficient of 0.9 and an average flame temperature of 800^{o}C (1475^{o}F) for 30 minutes. The package is placed 1 m (40 in.) above the fuel source, and the fuel source must extend between 1 m (40 in.) and 3 m (10 ft) horizontally past the package. After the 30-minute fire, the package cannot be cooled artificially, and any part of the package that remains on fire cannot be put out, but must be let to burn out on its own.
- **Immersion—**An undamaged package must be immersed in water pressure corresponding to a depth of at least 15 m (50 ft). This is equivalent to 150 kPa gauge pressure in water.

Note that a fifth test, the crush test, is assumed to not apply to the packages used for transport of the subject materials as this test only applies to packages with mass not greater than 500 kg (1,100 lb).

6.2.2.3 Quality Assurance Requirements

All aspects of the development and use of the package (design, purchase, fabrication, handling, shipping, storing, cleaning, assembly, inspection, testing, operation, maintenance, repair, and modification of components of the package that are important to safety) must be done under the auspices of a quality assurance program that meets all of the requirements of 10 CFR Part 71 Subpart H – Quality Assurance. Subpart H is consistent with a conventional 18-point quality assurance program and addresses the following topics:

1. Quality assurance requirements
2. Quality assurance organization
3. Quality assurance program
4. Changes to quality assurance program
5. Package design control
6. Procurement document control
7. Instructions, procedures, and drawings
8. Document control
9. Control of purchased material, equipment, and services
10. Identification and control of materials, parts, and components
11. Control of special processes
12. Internal inspection

13. Test control
14. Control of measuring and test equipment
15. Handling, storage, and shipping control
16. Inspection, test, and operating status
17. Nonconforming materials, parts, or components
18. Corrective action
19. Quality assurance records
20. Audits

6.2.2.4 DOE Orders Applicable to Transportation

In addition to satisfying the above-specified NRC regulatory requirements, the requirements from relevant DOE Orders may also apply.

DOE Order 460.1C, Packaging and Transportation Safety

Key requirements from O 460.1C that will apply include Section C 4.a(1), which requires compliance with 49 CFR Parts 171–180. Key issues that are addressed include (1) 49 CFR Part 171.15 and 16, *Incident Reporting*; (2) 49 CFR Part 172, Subpart H, *Training*; (3) 49 CFR Part 172, Subpart I and 49 CFR Part 174.9, *Security Plans*; (4) 49 CFR Part 173.22, *Advance Notifications*; and (e) 49 CFR Part 397.101, *Routing for Road Shipments*.

DOE O 460.1C Section 4.a(2) requires DOE shippers to be in full compliance with all conditions specified in the relevant packaging CoC as issued by the NRC, DOE Headquarters Certifying Official, or National Nuclear Security Administration Certifying Official.

DOE O 460.1C Section 4.a(3) requires transporting entities to have a quality assurance program approved and audited by the DOE Headquarters Certifying Official, or National Nuclear Security Administration Certifying Official, demonstrating that the packagings to be used satisfy the requirements of 10 CFR Part 71. In addition, all aspects of the development and use of the package (design, purchase, fabrication, handling, shipping, storing, cleaning, assembly, inspection, testing, operation, maintenance, repair, and modification of components of the package that are important to safety) must be performed under the auspices of a quality assurance program that meets all of the requirements of 10 CFR Part 71 Subpart H – Quality Assurance as described above.

DOE Order 460.2A, Departmental Materials Transportation and Packaging Management

DOE O 460.2A, Section 4.1.(a) states that “DOE organizations must conduct operations in compliance with all applicable international, Federal, State, local, and Tribal laws, rules, and regulations governing materials transportation that are not inconsistent with Federal regulations.”

It also states that “shipments under this provision will comply with the requirements of 49 CFR, [P]arts 100−185, except those that infringe on maintenance of classified information.”

Finally, it is noted that DOE O 460.3, which is an order currently under development, is expected to guide the transportation operations and security requirements that must be satisfied during off-site shipments. When issued, this order could require compliance during off-site transportation with DOE O 470.4B and/or 10 CFR Part 37.

6.2.3 Disposal

The Nuclear Waste Policy Act (NWPA) provides for two pathways to dispose of HLW or SNF stemming from atomic energy defense activities, DOE research and development activities, or both. One is to use a common repository to dispose of all HLW and SNF, regardless of origin. The other is to dispose of commercial waste in one repository and the DOE-managed waste in a separate repository.

Section 8(b) of the NWPA calls on the President to make a determination based on six statutory factors regarding whether a separate repository for the DOE-managed waste is required. After an evaluation by the DOE, President Reagan concluded in 1985 that a separate repository was not required. Over time, however, the circumstances have changed materially such that a recent evaluation by DOE (DOE 2015a) came to the opposite conclusion. Based on this updated evaluation, President Obama issued a memorandum on March 24, 2015 documenting his finding that “the development of a repository for the disposal of high-level radioactive waste resulting from atomic energy defense activities only is required” (Obama 2015).

Given the 2015 presidential finding, development of a defense-related repository will not be subject to NWPA’s siting provisions, except for the state and tribal participation provisions specified in Section 101 of the NWPA. That said, the repository will be subject to licensing by the NRC, and it will be subject to NRC requirements for siting development, construction, and operation per Section 8 of the NWPA. Under the current regulatory structure, the repository (which could include deep boreholes) will be subject to the Environmental Protection Agency’s (EPA) 40 CFR Part 191 and may be subject to the NRC’s 10 CFR Part 60.

The EPA’s 40 CFR Part 191 establishes standards in the form of dose limits for management and storage of waste at the disposal site and in the form of long-term performance objectives. The long-term performance objectives limit the likelihood of cumulative releases of certain radionuclides into the accessible environment over 10,000 years, limit doses to an individual for 10,000 years, and limit the levels of radioactivity in underground sources of drinking water for 10,000 years. The probability and consequences of human intrusion are to be included in the calculation of the likelihood of cumulative releases.

The NRC’s 10 CFR Part 60 contains requirements for the license application including site characterization; requirements for participation by state governments and affected Indian tribes;

requirements for records, reports, tests, and inspection; technical criteria, including siting criteria, waste package design criteria, and long-term performance objectives; requirements for a performance confirmation program; and quality assurance requirements. The long-term performance objectives require compliance with generally applicable environmental standards for radioactivity that have been set by the EPA; set performance standards for barriers, both engineered and natural; and require that waste be retrievable during the period of emplacement and perhaps for another 50 years after emplacement.

However, it is not clear whether 10 CFR Part 60 applies to disposal of radioactive waste in any type of geologic repository or only to disposal of radioactive waste in a geologic repository that has been mined. The NRC's 10 CFR Part 60 defines "geologic repository" as "a system which is intended to be used for…the disposal of radioactive waste in excavated geologic media." Thus, 10 CFR Part 60 would not apply to deep borehole disposal if drilling a borehole is not considered a form of excavation.

The Center for Nuclear Waste Regulatory Analyses (CNWRA) evaluated the concept of deep borehole disposal from a regulatory perspective for the NRC (Winterle, Pauline, and Ofoegbu 2011), but did not take the position that 10 CFR Part 60 did not apply to deep borehole disposal. On the other hand, the Blue Ribbon Commission on America's Nuclear Future (BRC 2012, Section 4.3) recommended that the NRC develop a regulatory framework for borehole disposal in parallel with their development of a site-independent safety standard for mined geologic repositories, implying that a separate regulatory framework was needed for disposal of radioactive waste in deep boreholes and, thus, that 10 CFR Part 60 did not apply to deep borehole disposal. For the purposes of this report, it is not necessary to resolve the question of whether 10 CFR Part 60 applies to disposal of radioactive waste in deep boreholes; it is sufficient to recognize that the disposal system will be licensed by the NRC and that the complete regulatory framework for such licensing does not yet exist.

The CNWRA study indicates that, to effectively support a deep borehole disposal repository, changes would be needed to EPA regulations (i.e., 40 CFR Part 191) and NRC regulations (i.e., 10 CFR Part 60). The study suggests that much of the regulatory framework using performance assessments established for Yucca Mountain in 40 CFR Part 197 and 10 CFR Part 63 could be adapted for use in regulating deep borehole disposal. One aspect of 10 CFR Part 63 that would most likely not be incorporated into a new regulatory framework is the concept of retrievability. Retrieval after closure in a deep borehole is not generally considered a viable option. It is assumed that new regulations would need to recognize the more permanent nature of deep borehole disposal.

It is uncertain at this time how a new regulatory framework would impact the universal canister design in terms disposal requirements. According to the CNWRA study, any rulemaking effort regarding deep borehole disposal would likely include consideration of certain preclosure and operational issues as well as development of performance-based regulations pertaining to canister construction specifications, surface transport to the disposal site, handling of waste on site, filling and sealing of canisters, lowering of waste packages, and backfilling and sealing of boreholes. The study also envisions the development of specific performance criteria for waste packages, shafts, and boreholes.

All federal projects are subject to compliance with the National Environmental Policy Act (NEPA). The type of NEPA assessment (e.g., categorical exclusion or Environmental Impact Statement) needed for the DBFT will be determined and implemented prior to initiating field activities (Hardin 2015a). For the actual disposal of DOE-managed waste in a deep borehole or a mined repository, it is expected that an Environmental Impact Statement will be required.

Both the Cs and Sr capsules and the calcine waste are regulated by the State of Washington and the State of Idaho, respectively, as RCRA waste. If the state in which the disposal site is located has been authorized by the EPA, the state will implement the RCRA-related requirements for disposal of the waste and serve as the regulator. If the state is not authorized by the EPA, the EPA itself will implement the RCRA-related requirements for disposal of the waste and serve as the regulator.

Various types of state and local permits (e.g., for drilling, air quality, land use, or water use) will be required for the DBFT and any future waste disposal activities. While specific permitting requirements vary by location, the focus is expected to be on various elements in the life cycle of the disposal well rather than on the universal canister or waste package.

7 Risks and Technical Challenges

Several risks and technical challenges are associated with developing a universal canister system in which the wastes discussed in Section 2 can be managed. The risks and technical challenges associated with the waste forms, with storage, with transportation, and with disposal are discussed below, as are potential regulatory and programmatic risks.

7.1 Waste Form

The risks and technical challenges associated with the Cs and Sr capsules, with the calcine waste, and with the Cs that might be removed from elutable resins or Cs-filled nonelutable resins are discussed below.

7.1.1 Cs and Sr Capsule Waste Form

The principal challenge with managing and dispositioning of the Cs and Sr capsules results from the radiation and heat they produce. Unshielded dose rates range from 5.37×10^2 to 6.56×10^3 rem/hr for the bare Cs capsules and 7.46×10^1 to 1.70×10^3 rem/hr for the bare Sr capsules at a distance of 3 ft based upon the contents of individual capsules (Table 2-3). The Cs capsules produce on average 119 W of heat per capsule while the Sr capsules produce an average of 157 W of heat per capsule.

In addition, there are corrosion considerations for Cs and Sr capsules as they await final dispositioning (DOE 2014b). The capsule materials can be susceptible to intergranular corrosion as a result of elevated carbon content in the steel and sensitizing heat treatments (exposures). Also, because the capsules were welded, there is the potential for SCC to occur. The DNFSB (1996) noted that some Cs capsules stored in the pool may have experienced chloride-induced SCC near the outer capsule welds due to lack of water chemistry requirements and control. One capsule has suffered a through-wall crack while another leak was attributed to a fabrication defect in the weld (DNFSB 1996). Although there have been issues with individual capsules, a corrosion study completed in 2003 (Bryan et al. 2003) concluded that corrosion is not expected to result in the release of radioactive material in dry storage either at design or upset temperatures. The Cs and Sr capsules are free of fissionable isotopes so criticality is not a consideration.

There are both accident and security considerations during storage, transfers, and transportation of SrF_2 because it is granular, and it can potentially be dispersed should a canister be unintentionally or intentionally breached at any point after loading.

7.1.2 Nonelutable Resin Waste Forms

Should the preferred alternative for removing Cs from the tank waste change to using a nonelutable resin, the Cs-filled spent resin would be a candidate for a universal canister system.

The resin would potentially sorb other radionuclides that are in the waste stream, in addition to Cs, and the inventory of those other radionuclides would have to be considered in designing the universal canister and associated universal canister system.

Many of the physical and chemical properties of CSTs (thermal properties, radioactivity, solubility, corrosivity, etc.) needed to assess the performance of the CST as a waste form have not been determined. CST is an inorganic zeolite-like material engineered to be highly selective for Cs and is chemically, thermally, and radiation stable (Miller and Brown 1997). One particular CST, known as TAM-5, is remarkable for its ability to separate parts-per-million concentrations of Cs from highly alkaline solutions (pH>14) containing high sodium concentrations (>5M). It is also highly effective for removing Cs from neutral and acidic solutions, and for removing Sr from basic and neutral solutions. CSTs may also capture small amounts of other metals in column I of the periodic table (alkali metals) if they are present with Cs such as potassium and rubidium.

In addition, a treatment system will need to be designed to load the Cs onto the CSTs, possibly using a column or fluidized bed technology. There have been observations of agglomeration and clumping of CSTs during small-scale column testing (Taylor and Mattus 2001). This presents a performance challenge for the use of CSTs in large-scale column applications for Cs removal from tank waste streams. Clumping or agglomeration would also potentially affect waste transfer into the universal canisters. After loading, the CSTs will need to be retrieved from the columns and loaded into appropriate canisters (and potentially overpacks) for storage, transportation, and disposal. Because the waste would consist predominately of Cs isotopes with some Sr and minor amounts of other alkali metals (rubidium and potassium) criticality will not be a concern for this waste form. Similar to the Cs and Sr capsules, the use of highly selective materials such as CSTs will require careful consideration of Cs and Sr loadings onto the CST for thermal and radioactive management of the material as a waste form during storage and transportation.

As with the SrF_2 waste, there are both accident and security considerations during storage, transfers, and transportation because CSTs are granular, and they can potentially be dispersed should a canister be unintentionally or intentionally breached at any point after loading.

7.1.3 Granular Calcine Waste Form

A retrieval system is being designed that will allow evacuation of the calcine from the storage bins and pneumatic transfer to the treatment and packaging facility. The challenges associated with calcine retrieval and transport, calcine fluidization, and waste confinement are expected to be resolved with this retrieval system. Using this system, the untreated calcine waste would be transferred dry and at ambient air temperatures into canisters. This will also mitigate the release of the volatile components in the waste including volatile metals (mercury and cadmium), organic volatiles, and volatile radionuclides (technetium and iodine). These volatiles would otherwise need to be captured and sequestered as part of the HIP treatment process. It is uncertain whether or not this retrieval system will be amenable to calcine transfer into a universal canister system. However, integrating the design of the calcine waste retrieval system with the design of the universal canister system will mitigate this risk.

With respect to corrosion, long-term laboratory corrosion testing using synthetic zirconium calcine was undertaken in 1966 to evaluate candidate materials for future construction of calcine storage bins (Dirk 1994). These corrosion tests were run for 17 years at 350°C. The candidate materials tested were Type 1025 carbon steel, Type 405, 304, 304L, and 316L stainless steels, and 606 I-T6 aluminum. The first calcine storage bin set was constructed with Type 405 stainless steel. A decision was made during the late 1960s to construct calcine storage bin sets 2–4 out of Type 304 stainless steel. Storage bin sets 5–7 were constructed at a later date out of Type 304L stainless steel. The highest corrosion rate was observed for Type 304L stainless steel with a measured rate of 8.1×10^{-7} in./mon. While these rates indicate that corrosion is minimal with respect to bin materials, careful consideration of the materials chosen for the universal canister system must be given to minimize potential canister corrosion.

With respect to criticality, the waste does contain small amounts of fissionable isotopes. However, over 99% of the fissionable isotopes have been removed from the waste. A criticality analysis by Hoffman (2010) concluded that criticality was beyond extremely unlikely in the calcine waste and the HIP waste form in both normal and abnormal operations at INTEC.

As with the nonelutable resins and SrF_2, there are both accident and security concerns during storage, transfer, and transportation as calcine waste is granular, and it is potentially dispersed should a canister be unintentionally or intentionally breached at any point after loading.

7.2 Storage

The risks and challenges with continuing storage of these waste forms appear to be primarily associated with the Cs and Sr capsules currently stored in the pool cells at the WESF. As of 2015 the pool cells in WESF were more than 11 years past their design life. The sooner these capsules can be removed from wet storage at WESF to an on-site facility for dry storage, the sooner operational costs can be reduced and safety can be enhanced.

One option for removal of capsules from wet storage at WESF is to dry the capsules and load them into a transfer cask or a universal canister in hot cell G. However, the size and radioactivity limits in hot cell G may constrain loading operations. Hot cell G is 8-ft wide × 16-ft long × 12-ft high, which would prohibit loading of a 15-ft-long canister or cask. Another option for removal of capsules from wet storage at the WESF is to use the cask pit at the end of pool cell #12, which is the pool cell used for transferring capsules into hot cell G. The cask pit is 4-ft 5-in. wide × 7-ft 5-in. long × 18-ft deep, and could be used for loading a 15-ft-long universal canister or transfer cask. In addition, the overhead canyon crane, with a 15-ton capacity, has access to the cask pit, to hot cell G, and to the truck port (Sexton 2003). The cask pit in the WESF pool has never been used, so further evaluation is required to determine if this is a workable option for loading capsules into transfer casks or universal canisters.

The radioactivity limits in hot cell G also constrain loading operations. The capsule inventory limit for hot cell G cannot exceed 150,000 Ci of Sr and cannot exceed 150,000 Ci of Cs (Covey 2014, Section 3.1.3). Since the average activities of the Cs and Sr capsules are 25,100 Ci and 23,500 Ci, respectively (as of January 1, 2016, see Table 2-3), it follows that hot cell G can hold 5–6 Cs capsules and 6 Sr capsules with mean activity levels. Further evaluation of the capacities and technical safety requirements for hot cell G is required to determine if loading operations in hot cell G are practical. If not, loading operations in a portable hot cell near the proposed dry storage area at the Hanford Site may be considered.

Risk introduced during storage could impact universal canister transportability and disposability. The risk is associated with storage periods beyond the initial licensing basis and development of an adequate aging management program for management of issues associated with aging that could adversely affect structures, systems, and components important to safety during transportation and disposal emplacement operations. Managing aging effects on the universal canister system for extended long-term storage requires knowledge and understanding of the various aging degradation mechanisms for materials of the structures, systems, and components and their environmental exposure conditions for the intended period of operation. Managing aging effects depends on aging management programs to prevent, mitigate, and detect aging effects on the structures, systems, and components early, by condition and/or performance monitoring. The challenges in the detection of aging effects will always be the inaccessible areas for inspection and monitoring and the frequency of inspection and monitoring (i.e., periodic versus continuous). Applicable issues for consideration are described in NUREG-1927 (NRC 2015).

7.3 Transportation

Risks and technical challenges associated with transportation of the candidate waste forms are believed to be low. The transportation of radioactive materials, including HLW, is a safe activity with a history spanning many decades. Due primarily to the rigorous regulatory requirements, there has never been a transportation accident worldwide involving SNF or HLW in which harm was done to a person or to the environment from the radiological effects of the accident. It is expected that the transportation of this waste material will not be subject to any technical risks or challenges that have not already been successfully met when transporting other highly radioactive waste materials such as SNF and HLW.

The risk associated with the transport of commercial SNF has been the subject of several studies by the NRC, the last of which was documented in NUREG-2125, *Spent Fuel Transportation Risk Assessment* (NRC 2014). This NRC study found that the risk associated with the transport of this material is very low due to the high degree of regulatory rigor associated with such shipments.

While the Universal Canister Project does not address SNF, the packagings used for transport of Cs and Sr capsules are expected to be very similar to the packagings used to transport commercial SNF. As a result, it is believed that the findings of NUREG-2125 are very much applicable to the work on the universal canister system as well.

The process for developing and certifying a new package (package = packaging + content), whether using an existing packaging design or developing a completely new packaging design, is a known process with much precedent. The key, and greatest technical challenge, in this process is the development at project initiation of a complete and comprehensive systems requirements document. If developed properly, the systems requirements document will guide the entire package development process and ensure that the certification effort is successful. Assuming that the appropriate emphasis is placed on the development of the systems requirements document, then it is not anticipated that there would be any significant challenge in receiving certification for this package.

The most significant operational challenge currently identified would be if different overpacks were to be used for storage, transportation, and disposal. If this is the case, then there would be some operational obstacles to overcome in transferring the waste, but this would not be insurmountable. If the need to address such interfaces is recognized during development of the systems requirements document, then resolution of these issues will be a natural result of the development process.

7.4 Disposal

A deep borehole emplacement mode hazard analysis (Sevougian 2015) identified two major types of hazardous events: (1) uncontrolled drop of waste packages or equipment into the borehole, and (2) waste packages getting stuck in a borehole in the guidance casing. The first type of event could cause one or more waste packages to be breached and result in radionuclide release into the drilling fluid in the unsealed borehole. The second type of event could also cause one or more waste packages to be breached, depending on the outcome of the retrieval effort, and result in radionuclide release into the drilling fluid in the unsealed borehole. Either type of event could cause total loss of operational capability of the borehole such that the borehole could not continue to be used for disposal and would have to be sealed and closed as-is. Techniques for addressing the technical challenges associated with these types of events will be investigated as part of the DBFT.

In addition, new technology will need to be developed to handle and emplace waste that requires significant shielding, such as the Sr and Cs capsules, representing a technical challenge that will be addressed as part of the DBFT. Other waste-specific risks and technical challenges associated with disposal are discussed below.

7.4.1 Disposal of Cs and Sr Capsules

Conditions downhole for the Cs and Sr capsules are expected to be anoxic with elevated temperatures and brine concentrations (Arnold et al. 2014). Under these conditions, the steel Cs and Sr capsules will most likely be at risk from chloride-induced SCC. If the waste package and capsule walls are breached, the CsCl and SrF_2 salts will be available to concentrated brines for subsequent dissolution. The room temperature solubility of CsCl is very high at 1,910 g/L (Haynes 2015), while that of SrF_2 is relatively low at 0.21 g/L. Simulated SrF_2 from the WESF has a low solubility at room temperature (0.135 g/L), increasing slightly with temperature to 0.157g/L at 50°C (Fullam 1976), but the dissolution rate of SrF_2 from WESF was dependent on surface area, impurity content, thermal history and temperature (amongst other factors). Given the saturated nature of the brine and its stagnant nature, it is feasible that these soluble salt waste forms may exhibit slower dissolution kinetics. If the solubility of the CsCl under expected repository conditions is of concern, then one option is to design a waste package that is resistant to corrosion under expected repository chemical conditions. For example, Arnold et al. (2014) propose the use of copper or copper-coated waste packages that are resistant to corrosion under expected downhole chemical conditions.

The half-life of ^{137}Cs is 30.17 years, making it of concern for the handling phases of disposal. However, the half-life of ^{135}Cs is 2,300,000 years, and its solubility and long half-life make it of concern for the long-term performance of the disposal system (Arnold et al. 2014). Sr also has several radioactive isotopes, but only ^{90}Sr is of concern for the handling phases of disposal of radioactive waste because the other isotopes have half-lives on the order of hours or days and have already decayed into stable isotopes of other elements. Unlike CsCl, SrF_2 has moderately low solubility; its moderately low solubility and lack of long-lived isotopes make it of little concern as compared to CsCl for the long-term performance of the disposal system.

The heat generated by the radioactive decay of the Cs and Sr will potentially limit the number of capsules within a waste package (Arnold et al. 2014). In addition, the limits on temperatures in a deep borehole are those imposed by the capsules themselves and the materials used as fill. While CsCl melts at 645ºC and SrF_2 at 1,477ºC, the point at which corrosive reactions between the salt and the metal capsule become significant occurs at much lower temperatures. A previous study for long-term dry storage of the capsules (Heard et al. 2003) identified maximum temperatures for the salt-metal interface of 317ºC and 540ºC for the CsCl and SrF_2 respectively (Table 4-2). Yet another thermal constraint is the temperature dependence of waste package material properties such as yield strength. Before final designs for the universal canister and the waste package are selected, heat flow modeling will be required.

Waste acceptance criteria have not yet been developed for deep borehole disposal; however, it is assumed here that any criteria eventually developed will allow for disposal of granular waste, such as SrF_2. The situation is not as straightforward should a mined geologic repository be used instead of deep borehole disposal. It is not clear whether the waste acceptance criteria for a mined geologic repository will permit the disposal of granular waste. If it turns out that granular waste is not permitted, then the granular SrF_2 contained in universal canisters could not be disposed in a mined geologic repository without a waiver.

7.4.2 Disposal of Cs Bound to Nonelutable Resins

The nonelutable resins, assumed to be CSTs or a CST-like material, will have some characteristics of both the Cs and Sr capsules and the calcine waste. Similar to the Cs and Sr capsules, Cs-loaded CST will likely be the primary constituent of the waste along with some Sr and small amounts of potassium and rubidium. Given that the waste is not composed of fissile materials, criticality will not be an issue in the deep borehole environment. It is important to note that this waste has not been produced yet, and there is not enough known about the characteristics of this waste form (radioactivity, thermal properties, solubility, corrosivity etc.) to assess the risks and challenges associated with its disposal using the universal canister system and performance in a deep borehole system.

7.4.3 Disposal of Calcine Waste

As discussed in Table 5-2, for the 17-in.-diameter case a total of 74 disposal boreholes would be needed to dispose of untreated calcine waste assuming the canisters are filled to 100% capacity. If the canisters were to be filled to 90% of capacity, then about 82 disposal boreholes would be required. This number of boreholes will require a large site for disposal. Furthermore, at an estimated system cost of $40 million per borehole, the disposal costs will require a substantial investment, on the order of $3 billion or more. This may drive the design towards larger diameter boreholes and waste packages, but there are significant limitations on diameter at the depths required for borehole disposal, and potential limitations on the size and weight of the waste packages will also need to be assessed.

The bulk of the calcine matrix is aluminum and zirconium oxides and calcium fluoride, all of which are relatively insoluble in water; only a small fraction of the material (such as nitrates) is soluble. Cs, for example, tended to form nitrates in the calcine, and cesium nitrate would be soluble. Thus, there is concern for leaching of Cs. However, the radionuclide portion of the calcine is much less than 1% of the total calcine, so the soluble Cs is a very small fraction of the calcine. Again because the half-life of ^{135}Cs is 2,300,000 years, its solubility and long half-life make it of concern for the long-term performance of the disposal system. In addition to ^{135}Cs, other long lived radionuclides including ^{99}Tc and ^{129}I are present in the calcine and will need to be considered in performance assessment. There are also metals such as cadmium in the calcine. While the percentage of such metals is relatively low, the Toxicity Characteristic Leaching Procedure regulatory limit established by the EPA is so low that even if very small amount of metal leached out of the matrix, it could be a RCRA concern. However, with the deep borehole disposal option, the highly reducing conditions expected in the deep subsurface will limit the solubility and enhance sorption of many radionuclides and metals in the waste, leading to limited mobility in groundwater (Brady et al. 2009).

Calcine was generated from SNF from which typically 99.99% of the uranium has been extracted, thus making a criticality event during disposal operations beyond extremely unlikely. However, because small amounts of fissionable isotopes remain, a criticality assessment of calcine waste in a borehole configuration should be considered as part of the long-term performance assessment of the waste in deep borehole disposal. Finally, without further processing to remove or treat RCRA metals present in the calcine waste, any future disposal option and associated facility (regardless of canister design) will need to be licensed to accept both radioactive and RCRA waste.

Waste acceptance criteria have not yet been developed for deep borehole disposal; however, it is assumed here that any criteria eventually developed will allow for disposal of granular waste, such as calcine. The situation is not as straightforward should a mined geologic repository be used instead of deep borehole disposal. It is not clear whether the waste acceptance criteria for a mined geologic repository will permit the disposal of granular waste. If it turns out that granular waste is not permitted, then the granular calcine contained in universal canisters could not be disposed in a mined geologic repository without a waiver.

7.5 Regulations and DOE Orders

As discussed in Section 6.2.3, the NRC's 10 CFR Part 60 and the EPA's 40 CFR Part 191 are the regulatory requirements that may or currently do apply to disposal of either the Cs and Sr capsules or the calcine waste in deep boreholes. To date, 10 CFR Part 60 has not been used to license the DOE to dispose of radioactive waste in a geologic repository. 40 CFR Part 191 has been used to license the DOE to dispose of defense-related transuranic waste in the Waste Isolation Pilot Plant (DOE 1996) in New Mexico and in Greater Confinement Disposal boreholes in Nevada (Cochran et al. 2001).

As part of the NRC's risk-informed regulation implementation plan (NRC 2006) and for reasons discussed in Section 6.2.3, the NRC's 10 CFR Part 60 is likely to be revised or a new borehole-specific regulation is to be created, thereby representing some measure of regulatory risk. The NRC's plan is designed to help the NRC develop a holistic, risk-informed, and performance-based regulatory structure. In particular, the SA-7 Safety Strategic Plan Goal seeks to incorporate risk information into the HLW regulatory framework (NRC 2006). The current version of 10 CFR Part 60 is not risk-informed and so is potentially a target for revision before it could be used to license disposal of either the Cs and Sr capsules or the calcine waste in deep boreholes or a mined geologic repository.

The current version of 10 CFR Part 60 has a requirement regarding retrievability of waste (10 CFR Part 60.111(b)); demonstrating compliance with this retrievability requirement could be difficult for deep borehole disposal, but what that new requirement would be, should 10 CFR Part 60 be revised or a new regulation written, is unknown. In addition, in 10 CFR Part 60.113(a) the NRC places limits on subsystem performance, such as the engineered barrier system, that are largely inappropriate for the deep borehole disposal concept; demonstrating compliance with them may pose unnecessary costs and complexities if 10 CFR Part 60 was to apply to deep borehole disposal. Finally, 10 CFR Part 60 and 40 CFR Part 191 are based on an implicit assumption that the disposal system is a single unit, and it is not clear how this approach would apply to a disposal system consisting of multiple boreholes at a single site. Would each borehole need to be licensed separately? Could multiple boreholes be licensed simultaneously? These issues represent some measure of regulatory risk with respect to disposal of waste-filled universal canisters in deep boreholes.

DOE is currently developing a new order (DOE O 460.3), which may modify the transportation operations and security requirements that must be satisfied during off-site shipments. For example, when issued this order could require compliance during off-site transportation with DOE O 470.4B and/or 10 CFR Part 37.

7.6 Programmatic Risks and Technical Challenges

The Universal Canister System Project will need to be coordinated with and integrated with several activities being conducted in parallel by various organizations: the DBFT project and ongoing efforts of the parties involved in the Tri-Party Agreement (HFFACO 2015), namely the Washington Department of Ecology, the EPA, and the DOE. The DBFT project timeline indicates that drilling of the Characterization Borehole is to begin September 1, 2016 and conclude February 27, 2017 (Figure 4-15). However, the DBFT project has already started specifying requirements for the waste package that is to be used in their Field Test Borehole and has already started specifying options for handling and emplacement of the waste package (Cochran and Hardin 2015). Ensuring consistency between the specifications identified by the DBFT project for the waste package and for the handling and emplacement system and the specifications identified by the Universal Canister Project could present a technical challenge. In addition, the Tri-Party Agreement has milestones with due dates in the near future (Table 4-1). Developing the universal canister system quickly enough to meet those milestones, particularly while the DBFT project is still researching the feasibility of deep borehole disposal, could also present technical challenges.

There is also a programmatic risk associated with the calcine waste form. As noted in Section 2.3, the preferred alternative for calcine waste is to treat it by HIP. After the HIP process, the compressed cans will be placed in canisters 5.5-ft diameter × 17-ft tall, presently certified for SNF (CDP 2012). While a universal canister system could be designed to accommodate compressed HIP cans, this canister would again be too large in diameter for deep borehole disposal.

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8 Summary

Some of the radioactive wastes that DOE–EM manages are candidates for disposal in a deep borehole. A canister that could be used for storing, transferring, transporting, and disposing of these wastes could facilitate the eventual disposal of these wastes. The purpose of this report is to lay the groundwork for developing specifications for a universal canister system for small waste forms. Ultimately, these specifications could be used to procure canisters and associated overpacks and casks from qualified suppliers.

The wastes considered as possible candidates for the universal canister system in this report include Cs and Sr capsules at the WESF at the Hanford Site, Cs extracted from elutable resins or Cs bound to nonelutable resins from the WTP at Hanford, and calcine waste from INL. The universal canister will be designed to store, transfer, transport, and dispose of these wastes without the need to re-open the canister. With respect to disposal concepts considered, the initial emphasis will be on disposal of these wastes in a deep borehole drilled into crystalline rock. However, the design of the universal canisters or the universal canister system should not preclude disposal in a mined geologic repository.

The 1,936 Cs and Sr capsules currently stored in pool cells at the WESF are good candidates for disposal in deep boreholes using the universal canister. Their size and shape are compatible with deep borehole disposal, and the entire inventory of capsules could be disposed of in a single deep borehole, thereby disposing of approximately one third of the radioactivity of the wastes at the Hanford Site. The elutable resin (sRF) that is the current preferred alternative for removing Cs from the existing tank wastes at Hanford is not a candidate for deep borehole disposal. However, should the preferred alternative change to a nonelutable resin, the Cs-filled resin would be a candidate for a universal canister system. In addition, should the current preferred alternative be modified such that the Cs extracted from the elutable resin is not returned to the HLW tanks for vitrification but is instead stored for separate disposal, that Cs would be a candidate for the universal canister system. The 4,400 m^3 of calcine waste at INTEC is also a candidate for a universal canister system, should disposal of the untreated calcine become the preferred alternative for disposal, rather than treatment with HIP followed by disposal in a mined geologic repository.

There are plans to move the Cs and Sr capsules to a dry storage facility at Hanford. Development of the universal canister system will need to be coordinated with development of these plans and will need to consider the current equipment and facilities available for this transfer. The BUSS cask has been used to transport the Cs and Sr capsules in the past and could potentially be used again for that purpose.

Development of the universal canister system and of the waste package handling and emplacement system will be coordinated with the DBFT project to ensure consistency between the two projects. The concept for the universal canister system involves placing the waste into the universal canister, welding it shut, and then using a series of overpacks or packagings for storage, transfers, transportation, and, in some cases, disposal. Initially, to be consistent with the DBFT project, waste packages of two sizes will be considered: one that would fit in a borehole with a 8.5-in. diameter and one that would fit in a borehole with a 17-in. diameter.

There are multiple performance, operational, and regulatory requirements for storage, transfers, transportation, and disposal of radioactive waste. These requirements were identified, and future work will enumerate specifications for the universal canister system that are based on these requirements.

Some of the risks and technical challenges associated with the waste form, storage, transfers, transportation, and disposal were identified, along with regulatory and programmatic risks.

Future tasks for the near term are listed below (not in order of priority):

1. For the WESF, evaluate current hot cell and cask pit capabilities and equipment. Perform a gap analysis for capsule removal strategy by either hot cell or cask pit removal methods. Identify pros and cons of each method, define any operational constraints such as temperature limits, and recommend a preferred alternative.
2. Develop universal canister design options for Cs and Sr capsules and assess feasibility for application to calcine waste
3. Develop preliminary concepts of operations that integrate the storage, transfers, transportation, and disposal management functions discussed in Section 3
4. Develop systems analysis tools
5. Perform trade studies on concepts of operations from Task 3 using systems analysis tools from Task 4
6. Evaluate and identify the technologies for monitoring selected components of the universal canister system (i.e., identify requirements, options, and technologies available for implementation)
7. Collaborate with the DBFT project
    a. Evaluate options for surface handling and emplacing waste packages in a deep borehole
    b. Establish universal canister system and waste package design requirements
    c. Set up infrastructure needed to support collaboration and ensure version control of selected documents

8. Collaborate with the extended storage project at the WESF to develop universal canisters that meet the needs of both the extended storage project and the Universal Canister Project

9. Develop plan for Universal Canister Project

Research and development needs associated with designing and creating a universal canister system include the following (not in order of priority):

- **Material Properties and Structural Integrity—**The hydrologic and chemical environment in a deep borehole and the radiation levels and decay heat from Cs/Sr capsules will impose significant requirements on the material specifications for the universal canister system. At a maximum depth of 5,000 m, the hydrostatic stress is estimated as 50 MPa (490 atm) to 65 MPa (640 atm) and the maximum package surface temperature would be around 250°C (Section 4.5.1). In addition, chemical conditions are expected to be reducing and may be highly corrosive for low carbon steels or stainless steels because of the high chloride content of deep groundwaters. Selection of materials must consider the mechanical and chemical response of the waste package in the high temperature and high stress environment of a deep borehole. For example, high temperature will reduce the yield strength of metals and may increase corrosion rates, thereby reducing the thicknesses of key structural components and their structural integrity; high radiation levels may accelerate the deterioration of materials. These processes are important to the design of the universal canister system because they will, in part, determine failure modes and the lifetime of the waste package under downhole conditions. Similar considerations also apply during storage and transportation of the universal canister system, although the environmental conditions during storage and transport may be less severe than in a deep borehole.

  Extensive testing and analysis will be required to determine the performance of candidate materials for the universal canister system under relevant conditions for storage, transportation, and deep borehole disposal. Specific concerns include (1) material properties at elevated temperatures and after radiation exposure for extended periods of time, (2) corrosion rates as a function of temperature, particularly in the deep borehole environment, (3) failure mechanisms for welds and requirements for their stress relief, (4) the structural response and potential failure modes of a string of degraded waste packages in the deep borehole environment, and (5) interactions of CsCl and SrF_2 with canister internals and canister materials after failure of the capsules. Ultimately, design of the waste package and universal canister system requires a systems engineering approach that considers the performance and failure modes of a waste package and its contents versus its longevity requirements in the underground environment. In other words, testing/analysis for thermal response, corrosion rates, material properties, and radiation levels will provide inputs to design and to an integrated systems analysis for the universal canister system. Note that item (3), failure mechanisms for welds, is discussed in more detail below.

- **In-package Sorbents and Fillers—**The use and effectiveness of in-package fillers such as sorbents to mitigate the migration of potentially mobile and/or long-lived radionuclides under expected downhole conditions after closure of the borehole should be investigated. Further, it may be possible to engineer multifunctional materials for use in-package. For example, materials may be engineered to serve as sorbents for radionuclides and heat conductors (or insulators) for thermal management, all the while mitigating canister compaction under the compressive stresses expected downhole.

- **Waste Form Tolerance to Heat and Postweld Stress Relief—** Sensitization of stainless steel can occur during rapid heating and cooling, such as during welding. During sensitization, chemical compositions in the vicinity of the grain boundaries can be altered causing increased corrosion rates. The SCC produced by the combined action of corrosion and tensile stress (applied or residual) is a concern that needs to be addressed for the universal canisters. As such, postweld stress relief is highly desirable, and is a requirement in the performance specification for Standardized Transportation, Aging, and Disposal Canisters (ORNL 2015). Note that the universal canister may provide the confinement barrier in dry storage and may provide secondary containment during transportation and during disposal emplacement operations, so ensuring an intact canister is very important.

  One option for providing stress relief is the use of annealing, which is a heating and cooling operation with a relatively slow cooling. The peak temperature and rate of cooling depend on the material being treated and the purpose of the treatment. In other words, the peak temperature for heat treatment may affect corrosion processes of the waste inside the universal canister, so the temperatures and duration of the cooling period must be carefully evaluated. In addition, other surface stress relief techniques, such as laser peening, will be considered because they may produce lower overall temperature change of the canister and its contents relative to a classic annealing process.

- **Waste Package Impact Limiter—**The DBFT project has proposed attaching impact limiters to the bottoms of waste packages as they are emplaced in the borehole to mitigate the consequences of the waste package being dropped in an uncontrolled fashion (SNL 2015). Such impact limiters do not currently exist; research and development would need to be done to determine appropriate materials and generate appropriate designs for impact limiters that could be used to dispose of waste packages in a deep borehole.

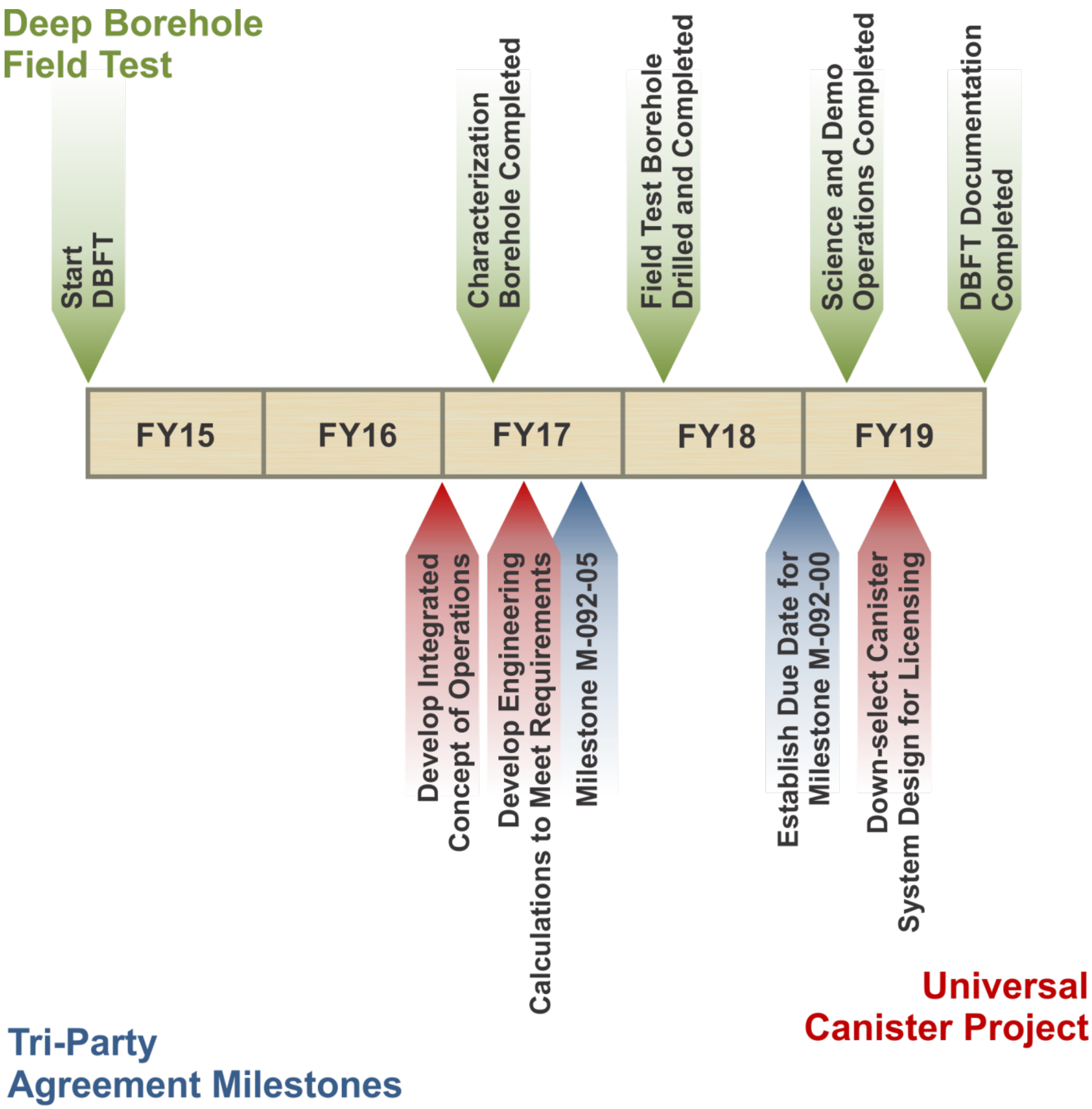
- **Sensors—**During waste emplacement operations, downhole instrumentation and telemetry needs to enable the surface crew to do the following: (1) monitor the condition of the instrumented waste packages as they are lowered in the borehole, (2) detect any radioactive contamination caused by leaking waste packages, and (3) detect any adverse conditions in the borehole during emplacement, such as collapsed casing. Commercial capabilities and new technologies need to be adapted and developed to meet these objectives, in collaboration with the DBFT project, and they need to be demonstrated in one or both of the DBFT boreholes.

- **Cs Mobility under Downhole Conditions—**Because of the high solubility of CsCl and because of the very long half-life of one of its isotopes (^{135}Cs at 2.3 million years), the mobility of Cs should be investigated under expected downhole conditions. For example, Cs incorporation into the analcime structure (an expected alteration product of bentonite at expected borehole conditions) should be investigated as a means of sequestering Cs potential releases from waste packages in a deep borehole.

- **Calcine Solubility under Downhole Conditions—**Calcine contains several long-lived isotopes including ^{135}C, ^{99}Tc, and ^{129}I as well as very small quantities of fissionable isotopes. While the solubility of calcine wastes in water is low, the solubility of the calcine wastes and the potential mobility of these isotopes under expected downhole geochemical conditions should be evaluated experimentally to assess the long-term performance of the calcine waste form.

- **Impact of High Pressure and High Temperature Environment on Seals Design—** Deep borehole high pressure and temperature experimental work should be done to characterize and bound seal degradation in deep borehole crystalline rock-based repositories (at anticipated pressure and temperature conditions), including the use of both mafic (amphibolites) and silicic (granitic gneiss) end members. The experiments should systematically add components to capture discrete changes in seal component chemistries, mineral phase changes, and kinetic effects at borehole pressure and temperature conditions. This effort will provide essential data that will be integrated with geochemical modeling efforts for refinement of deep borehole seals design.

Developing specifications for the universal canister system will require integration across parallel activities: the DBFT, the Universal Canister Project, and DOE’s responsibilities with respect to the Tri-Party Agreement (HFFACO 2015). A notional timeline for developing the specifications for the universal canister system in parallel with these other activities is shown in Figure 8-1.



M-092-00 Complete acquisition of new facilities, modification of existing facilities, and/or modification of planned facilities necessary for the storage, treatment/processing, and disposal of Hanford site cesium and strontium capsules (Cs/Sr), bulk sodium, and 300 Area special waste.

M-092-05 Determine disposition path and establish interim Agreement Milestones for Hanford Site Cs/Sr capsules DOE will assess the viability of direct disposal of the Hanford Cs/Sr capsules at the national high-level waste repository and provide a schedule leading to its disposition. If DOE concludes that direct disposal is a viable and preferred alternative to vitrification, DOE will submit to Ecology specific documentation justifying its conclusion, with a proposed milestone change request establishing enforceable Agreement Milestones for dispositioning Hanford Cs/Sr capsules.

NOTE: DBFT = Deep Borehole Field Test.
DOE = the US Department of Energy.
FY = fiscal year.

Figure 8-1. Notional Timeline for Moving Forward with the Universal Canister System Concepts

9 References

9.1 Regulations and Orders

10 CFR Part 37. *Physical Protection of Category 1 and Category 2 Quantities of Radioactive Material*.

10 CFR Part 60. *Disposal of High-Level Radioactive Wastes in Geologic Repositories*.

10 CFR Part 63. *Disposal of High-Level Radioactive Wastes in a Geologic Repository at Yucca Mountain, Nevada*.

10 CFR Part 71. *Packaging and Transportation of Radioactive Material*.

10 CFR Part 72. *Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-level Radioactive Waste, and Reactor-related Greater than Class C Waste*.

10 CFR Part 835. *Occupational Radiation Protection*.

40 CFR Part 191. *Environmental Radiation Protection Standards for Management and Disposal of Spent Nuclear Fuel, High-Level and Transuranic Radioactive Wastes*.

40 CFR Part 197. *Public Health and Environmental Radiation Protection Standards for Yucca Mountain, Nevada*.

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DOE O 440.1B. *Worker Protection Program for DOE (Including the National Nuclear Security Administration) Federal Employees*.

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DOE O 460.1C. *Packaging and Transportation Safety*.

DOE O 460.3. *Physical Protection of Unclassified Irradiated Fuel in Transit.* Currently under development.

DOE O 470.4B. *Safeguards and Security Program.*

DOE O 5400.1. *General Environmental Protection Program Requirements.*

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WAC (Washington Administrative Code) Chapter 173–303. *Dangerous Waste Regulations*.

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