Evaluation of Options for Permanent Geologic Disposal of Spent Nuclear Fuel and High-Level Radioactive Waste

in Support of a Comprehensive National Nuclear Fuel Cycle Strategy

Volume I

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Revision 01	Revised for minor corrections. Changes to correct cesium/strontium capsule radioactivity description as fraction of Hanford wastes. Additional changes to incorporate System Plan 18 information for Savannah River Site.

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EXECUTIVE SUMMARY

Background

This study provides a technical basis for informing policy decisions regarding strategies for the management and permanent disposal of spent nuclear fuel (SNF) and high-level radioactive waste (HLW) in the United States requiring geologic isolation. Relevant policy questions this study can help inform include the following: Is a “one-size-fits–all” repository a good strategic option for disposal? Do different waste types and forms perform differently enough in different disposal concepts that they warrant different treatment? Do some disposal concepts perform significantly better with or without specific waste types or forms? The study provides this basis by evaluating potential impacts of waste forms on the feasibility and performance of representative generic concepts for geologic disposal.

Participants in the study include representatives from the U.S. Department of Energy (DOE), the U.S. Navy, several national laboratories, universities and private sector firms with expertise in a broad range of fields, including nuclear engineering, earth sciences, materials science, chemical engineering, and materials safeguards and security, as well as regulatory considerations. Criteria and metrics for the study are necessarily qualitative, because of the complexity of the problem and the difficulty of quantifying estimates of the behavior of specific waste forms in generic disposal environments. Criteria include long-term repository performance, confidence in the bases for expected performance, operational considerations, technical readiness, the production of associated secondary wastes, and topics related to safeguarding and securing the wastes. In lieu of quantitative information about specific disposal sites and design concepts, insights are developed based on the full range of information available to the group, including detailed assessments done by previous repository programs in the U.S. and elsewhere in the world.

The scope of the waste in this study includes all existing SNF from commercial, defense, and research reactors, and SNF from reasonably foreseeable operations of existing reactors (projected to 2048). The study also includes existing HLW (e.g., vitrified HLW at Savannah River and West Valley) and waste forms projected to be generated in the future from existing process waste (e.g., projected vitrified HLW from HLW at Hanford, Savannah River and the Idaho National Laboratory). In addition, the study includes consideration of both direct disposal of waste forms that are not currently planned for disposal without further treatment (e.g., calcine waste at the Idaho National Laboratory) and alternatives to planned treatments. The study acknowledges existing plans, commitments, and requirements where applicable, but the study evaluates options for disposal based primarily on technical, rather than programmatic or regulatory constraints.

Major assumptions and considerations used in this study include the following:

- HLW and SNF considered in this study are restricted to existing materials and those materials that can be reasonably expected to be generated by existing or currently planned facilities and processes.
- The inventory of HLW and SNF is intended to include all existing materials in the U.S. requiring deep geologic isolation, and is based on the best available information.
- Technologies under consideration, including both for waste treatments and disposal concepts, are limited to those that can be deployed in the near future.

- Programmatic constraints, including legal, regulatory, and contractual requirements, are acknowledged where applicable, but are not considered in the technical evaluations, consistent with the goal of the study to provide technical input to strategic decisions. For example, the identification of wastes requiring deep geologic isolation is based on consideration of overall risk, rather than on specific U.S. legal and regulatory requirements.
- Evaluations are primarily qualitative, and are based in large part on insights from past experience in waste management and disposal programs in both the U.S. and other nations.

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

Disposal Concepts

Four representative disposal concepts are included in this study; these are mined repositories in three geologic media—salt, clay/shale rocks, and crystalline (e.g., granitic) rocks—and deep borehole disposal in crystalline rocks (Figure ES-1). These are the four disposal concepts selected for further research and development activities by the DOE Office of Nuclear Energy's Used Fuel Disposition Campaign. Selection of these four concepts begins with the observation that options for disposal of SNF and HLW have been evaluated in multiple nations for decades, and deep geologic disposal was recognized as early as the late 1950s to be the most promising approach. By the 1980s, the U.S. waste management program had concluded that multiple geologic media had the potential to provide robust isolation, and that conclusion remains valid today. Experience gained in waste management programs in other nations reinforces that conclusion: for example, Sweden and Finland both have license applications pending for proposed mined repositories for SNF in crystalline rock; the U.S. has an operating repository in salt for transuranic waste at the Waste Isolation Pilot Plant and Germany has extensive experience with the design of a mine for a SNF and HLW repository in salt; and France, Switzerland, and Belgium have completed detailed safety assessments for proposed SNF and HLW repositories in clay and shale media. No nations are currently planning deep borehole repositories, but the concept has been evaluated in multiple programs since the 1970s, and remains viable for waste forms small enough for emplacement.

Waste Inventory (Waste Types, Waste Forms, and Waste Groups)

This study evaluates existing and "reasonably foreseeable" SNF and HLW inventories. The existing wastes are those that can be inventoried, and those scheduled to be generated by currently operating reactors (or reactors under construction) and HLW-generating activities through 2048. The boundary of "reasonably foreseeable" is selected to include wastes that can be forecast from current actions by industry or government, but is not intended to include potential waste streams from advanced fuel cycle technologies that may be—or may not be—deployed in the future. This enables the physical and radiological characteristics of both existing and reasonably foreseeable wastes to be sufficiently well defined for evaluation in disposal options. For example, the spent fuels from several prototypical high-temperature gas-cooled reactors that have been built and operated in the U.S. are clearly included in the scope of this study, whereas proposed fuel types for developing reactor technologies that have no firm construction commitment are not considered in the scope of this study.

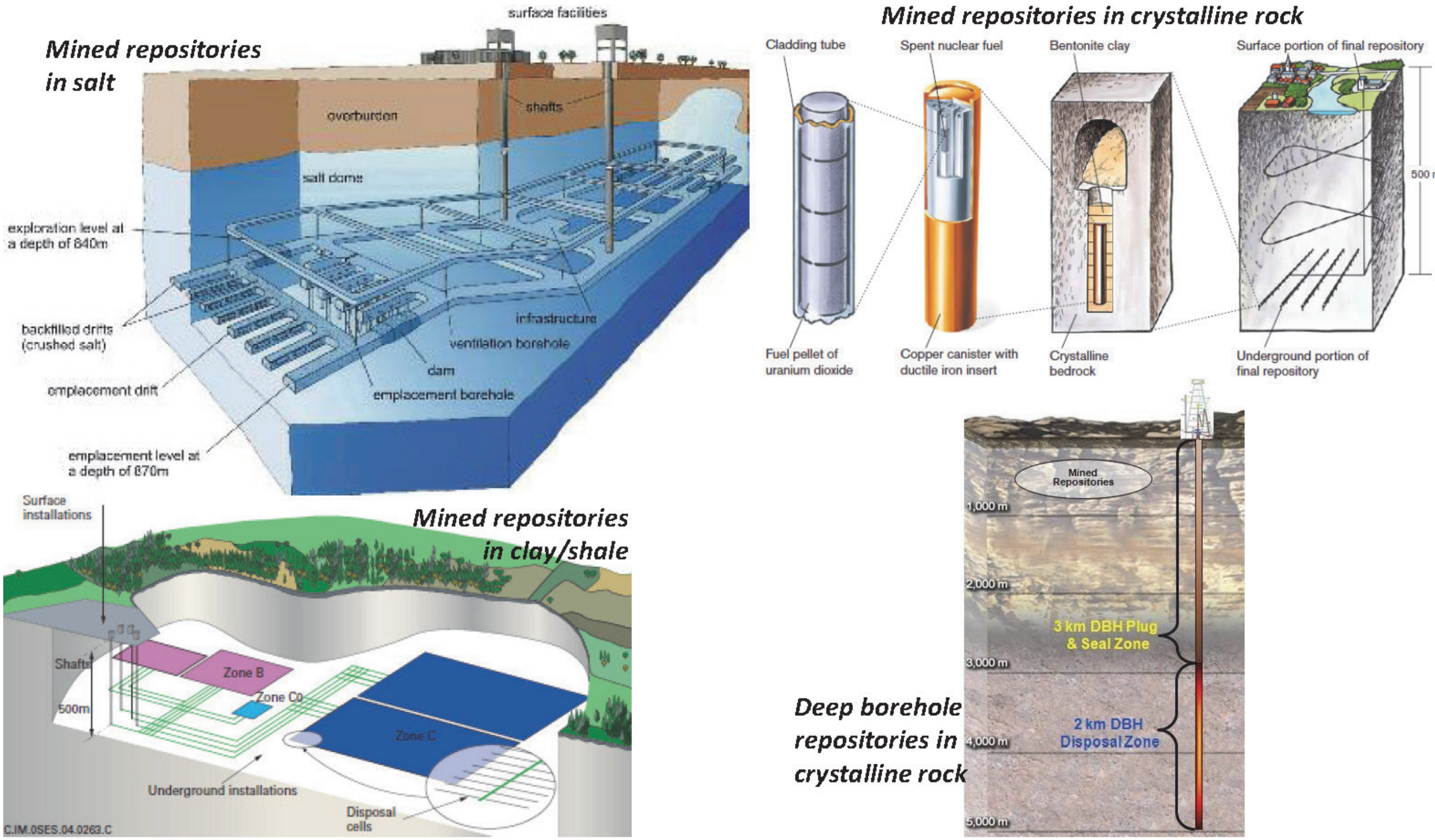
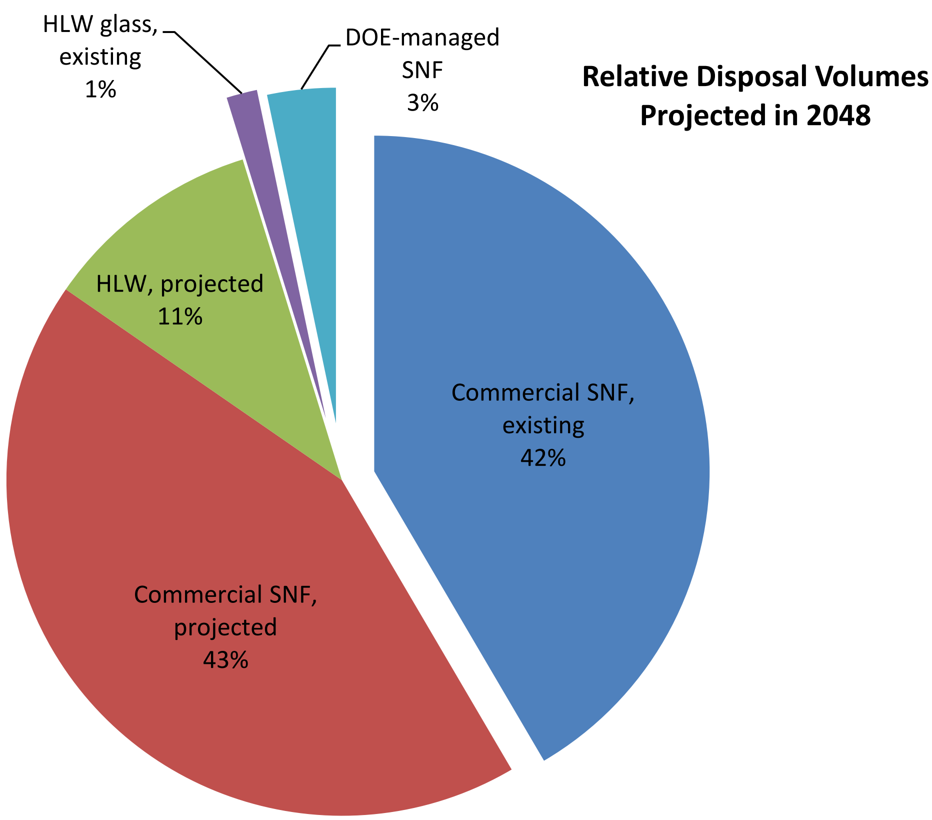


Figure ES-1. Four representative disposal concepts considered in this study (for details and sources see Sections 3.1.1 through 3.1.4)

Waste Volumes

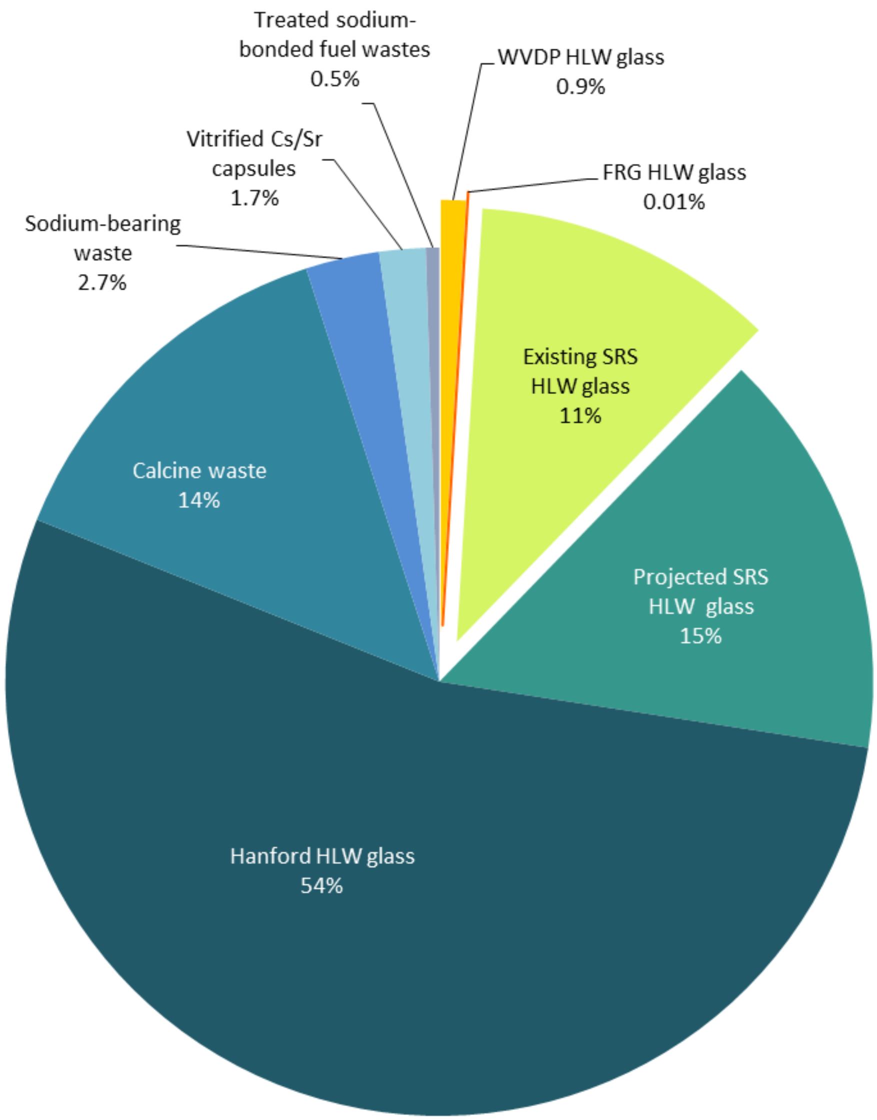
Figure ES-2 shows the relative amounts of SNF and HLW as projected through 2048. This figure shows that most of the total projected waste inventory by volume is SNF, and most of the SNF is commercial, rather than DOE-managed. The volume of commercial SNF is estimated to double by 2048 as the U.S. continues to generate SNF from civilian nuclear power generation. The existing volume of HLW represents the volume that has already been treated, while the projected volume of HLW represents estimated volumes of various existing wastes after expected treatment.



NOTE: Volume estimates assume (1) constant nuclear power generation in commercial reactors and disposal of all commercial SNF in dual-purpose canisters, (2) relative volumes of various HLW based on calcine processed by hot isostatic pressing with additives, (3) sodium-bearing waste treated by fluidized bed steam reforming, (4) sodium-bonded fuels undergo electrometallurgical treatment, and (5) all other waste forms are vitrified. For simplicity, all DOE-managed SNF is shown as "existing"; approximately 3,500 m^3 of naval SNF remains to be generated.

Figure ES-2. Relative volumes of U.S. SNF and HLW, existing and projected in 2048

Figure ES-3 shows the relative volumes of the various types of HLW. Most of the total projected HLW inventory by volume is the waste that is expected to result from vitrifying the Hanford tank waste. The vitrified waste from the tanks at the Savannah River Site represents the second largest percentage of the total projected HLW inventory by volume. The existing vitrified waste at the Savannah River Site represents about 45% of the projected total volume of vitrified waste expected to be produced at that site. Less than 13% of the projected volume of HLW currently exists in its final form; most of the waste exists but has not yet been treated so that it is in a form suitable for disposal.



NOTE: Volume estimates assume calcine processed by hot isostatic pressing with additives, sodium-bearing waste treated by fluidized bed steam reforming, sodium-bonded fuels undergo electrometallurgical treatment, and all other waste forms are vitrified.
FRG = Federal Republic of Germany; SRS = Savannah River Site; WVDP = West Valley Demonstration Project.

Figure ES-3. Relative disposal volumes of HLW and other waste, existing and projected

Table ES-1 gives disposal volumes of the various wastes shown in Figures ES-2 and ES-3, both currently existing and projected to exist in 2048. As this table shows, commercial SNF comprises the largest fraction of the volume of waste to be disposed of.

Table ES-1. Disposal volumes of U.S. SNF and HLW

Waste	Present Volume (m^3)	Additional Projected Volume in 2048 (m^3)	Total Volume (m^3)
Commercial SNF disposed of in dual-purpose canisters	90,299	93,597	183, 896
DOE-managed SNF	7,165	0	7,165
Savannah River Site vitrified HLW	2,969 (through macrobatch 8)	3,988	6,957
Hanford site vitrified HLW	0	14, 089	14, 089
Calcine waste after treatment by hot isostatic pressing	0	3,661	3,661
Sodium-bearing waste after treatment by fluidized bed steam reforming	0	721	721
Vitrified Cs/Sr capsules	0	453	453
West Valley Demonstration Project vitrified HLW	245	0	245
Treated sodium-bonded fuel (electrometallurgical treatment)	0	132	132
Federal Republic of Germany HLW glass	3	0	3
Total	100,681	116,641	217,332

Note: Table assumes constant nuclear power generation in commercial reactors. For simplicity, all DOE SNF is shown as “existing,” although approximately 3,500 m^3 of naval SNF remains to be generated. In addition, all the waste from electrometallurgical treatment of Na-bonded fuel is shown as “projected” even though a small quantity was generated during demonstration of the treatment process.

The waste inventory is classified into 43 different “waste types.” For the purposes of this study, a “waste type” is defined as the currently existing materials (in whatever form, abundance, and location they occupy) that are to be disposed of as at least one, and possibly more than one, waste form in a deep geologic disposal concept (e.g., Hanford tank wastes; commercial spent fuels, HLW glass). A “waste form” is the end-state material as packaged that is to be disposed of in a deep geologic disposal concept. Some “waste types” may have more than one possible alternative “waste form” depending on the processing needed, whereas “waste types” that require no processing other than packaging may equate to a single “waste form.”

Considering the alternative treatment options for some of the 43 waste types results in 50 waste forms. To facilitate the analysis, these 50 waste forms are aggregated into the ten “waste groups” listed in Table ES-2, with similar disposal characteristics such as radionuclide inventory, thermal output, physical dimensions, chemical reactivity, packaging of the waste form, and safeguards and security needed for handling, transporting, and disposing of the waste form in the context of the disposal concepts in this study. The aggregation into waste groups allows a high-level identification of any waste forms that may need to be considered as a separate group due to outstanding qualities in any one of these characteristics. In general, any waste forms that are largely similar in these characteristics are included in a single waste group. Figure ES-4 is an illustrative example of waste type, waste form, and waste group for HLW glass.

Table ES-2. Waste group descriptions

Waste group	Description
WG1	All commercial SNF packaged in purpose-built disposal containers
WG2	All commercial SNF packaged in dual-purpose canisters of existing design
WG3	All vitrified HLW (all types of HLW glass, existing and projected, canistered)
WG4	Other engineered waste forms
WG5	Metallic and non-oxide DOE spent fuels
WG6	Sodium-bonded fuels (driver and blanket), direct disposed[1]
WG7	DOE oxide fuels
WG8	Salt, granular solids, and powders
WG9	Coated-particle spent fuel
WG10	Naval fuel

Note: it was concluded that insufficient data exist to evaluate direct disposal of sodium-bonded fuels.

Criteria, Metrics, and Results

Each waste group is evaluated against six primary criteria for potential disposal in each of the four disposal concepts. The criteria, and their associated metrics used in the evaluation, are summarized in Table ES-3. Note that in practice not all criteria are equally relevant to all components of a disposal option. For example, secondary waste generation and safeguards and security concerns are primarily relevant to the characteristics of the waste group, whereas confidence in expected performance bases and operational feasibility are more strongly influenced by the disposal concept. Scoring is done qualitatively, using the informed and consensus judgment of a subset of the multidisciplinary team contributing to this study. A summary of the results follows.

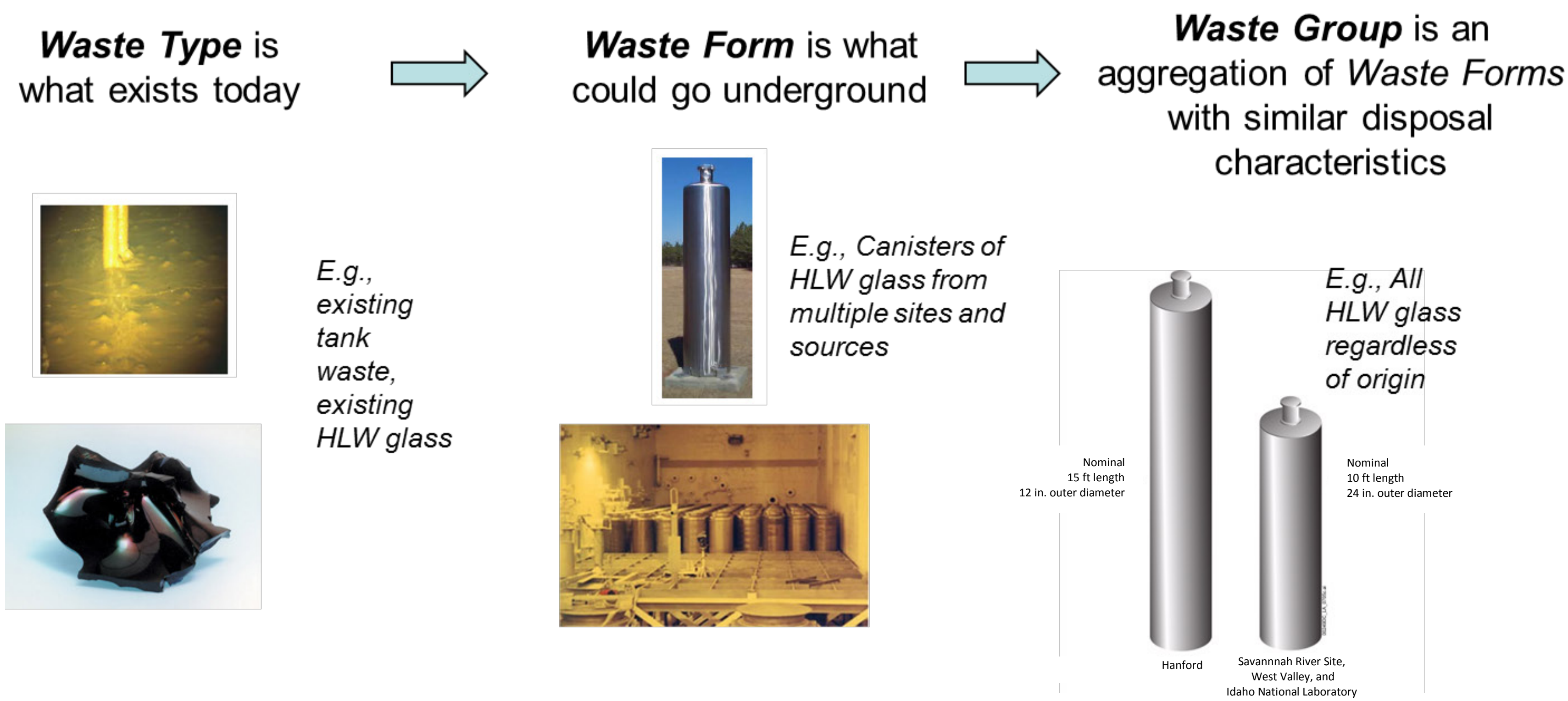


Figure ES-4. Terminology example using high-level radioactive waste glass

Table ES-3. Evaluation criteria and associated metrics

Evaluation Criteria	Metrics
Disposal Option Performance	Likely to comply with long-term standards?
Confidence in Expected Performance Bases	Additional engineered barrier system components needed above baseline for each design concept Robustness of information bases; simplicity vs. complexity; knowledge gaps
Operational Feasibility	Ease in ensuring worker health and safety at all stages Special physical considerations at any stages based on physical characteristics
Secondary Waste Generation	Amount of low-level waste generated during handling and treatment Amount of mixed waste generated
Technical Readiness	Status of waste form technologies Status of transportation and handling systems Status of disposal technologies
Safeguards and Security	National security implementation difficulty Radiological dispersion device prevention implementation difficulty

Disposal in Salt

Overall, mined repositories in salt showed strong results for most waste groups with respect to most metrics. At the level of resolution provided by this evaluation, scores for salt and clay/shale repositories are equivalent. The following pros and cons of disposal in a salt repository are noted:

- Pros:
  - The high thermal conductivity and high temperature limit of salt provide greater flexibility (i.e., larger packages and closer spacing) for disposal of heat generating wastes than other geologic media.
  - The limited far-field radionuclide transport (low permeability) of salt reduces the reliance on the waste form and waste package lifetimes, providing greater confidence in estimates of long-term performance.
  - The low permeability and reducing (oxygen poor) environment make it easier to keep specific waste packages isolated from each other, should that be necessary.
  - Some untreated waste types may be appropriate for direct disposal in salt, potentially reducing costs and risks associated with waste treatment.
  - The operational experience at the Waste Isolation Pilot Plant provides additional confidence.
  - The relative lack of water and the high cross-section of chlorine for capture of thermal neutrons make it easier to address criticality concerns (for waste forms for which criticality is relevant).
- Cons:
  - For very large waste packages (e.g., dual-purpose canisters), keeping the large shafts and ramps open during the operational period will present a challenge, as will sealing these large shafts and ramps upon closure.

  - For very large waste packages, technologies for moving and emplacing the waste packages in salt have yet to be developed.
  - Knowledge gaps exist concerning the response of salt to high thermal loads.
  - There may be a greater need for site-specific information regarding this type of disposal media because of the high reliance on the integrity of the host rock.

Disposal in Crystalline Rock

Overall, mined repositories in crystalline rock showed strong results for most waste groups with respect to most metrics; however, scores in several areas were lower than for salt or clay/shale disposal concepts suggesting that future R&D needs could be greater for a crystalline repository. The following pros and cons of disposal in a crystalline rock repository are noted:

- Pros:
  - There is significant world-wide experience with this medium.
  - If waste package retrieval is necessary, it is easiest in this disposal concept.
  - Stable rock properties enhance operational feasibility for very large packages (e.g., dual-purpose canisters).
  - Would be relatively easy to achieve separation distances between waste forms, if needed.
- Cons
  - The lower thermal conductivity of crystalline rock complicates the reliance on bentonite as a buffer for those wastes that generate a significant amount of heat.
  - Management of high heat loads could delay emplacing backfill until after a long period of ventilation.
  - Strong reliance on waste package lifetime results in lower confidence for high-heat waste forms and readily mobilized waste forms
  - Some wastes may need to be segregated from other wastes because of possible corrosive chemical reactions.
  - Some of the waste forms, particularly dual-purpose canisters, would need robust overpacks that may pose design challenges.
  - Colloids pose a potential for transport in fracture networks
  - For very large waste packages, technologies for moving and emplacing the waste packages in crystalline rock have yet to be developed.
  - May need to consider adding engineered barrier system (EBS) component to address criticality concerns for some wastes.

Disposal in Clay/Shale

Overall, mined repositories in clay/shale showed strong results for most waste groups with respect to most metrics. At the level of resolution provided by this evaluation, scores for clay/shale and salt repositories are equivalent. The following pros and cons of disposal in clay/shale repository are noted:

- Pros:
  - There is a significant amount of world-wide experience with this disposal medium.

  - The limited far-field radionuclide transport in clay/shale (low permeability and high sorption) reduces the reliance on the waste form and waste package lifetimes
  - Would be relatively easy to achieve necessary separation distances between wastes, if needed.
- Cons:
  - For very large waste packages, keeping the large shafts and ramps open during the ventilation period will present a challenge, as will retrieval during preclosure operations.
  - For very large waste packages, technologies for moving and emplacing the waste packages in clay/shale have yet to be developed.
  - May need to add EBS components to address criticality control.
  - Some wastes may need to be segregated from other wastes because of possible corrosive chemical reactions.

Disposal in Deep Boreholes

Overall, deep borehole disposal options received mixed scores. For wastes that currently exist either as unpackaged materials (e.g., untreated calcine waste and some DOE-managed SNF) or in small packages (e.g., cesium/strontium capsules), deep borehole disposal may be a feasible and potentially attractive option. For example, untreated cesium/strontium capsules, which represent about 40% of the total radioactivity originally in HLW at the Hanford site, could be disposed of in a single borehole. However, for larger waste forms (e.g., commercial SNF disposed of in dual-purpose canisters and the existing canisters of HLW glass), the option is simply not feasible because the wastes forms are larger than current drilling technology could accommodate. For many waste forms, engineering challenges associated with preparing the waste form in a small enough package for emplacement in boreholes using current standard drilling technology are sufficient to make it a less attractive option. For example, deep borehole disposal of PWR SNF could require removal of assembly hardware and consolidation of the rods. Deep borehole disposal of future HLW glass could require redesign of existing vitrification facilities to make smaller canister pours. The following pros and cons of geologic disposal in deep boreholes are noted:

- Pros:
  - Thermal load management concerns are minimized by the depth of the disposal concept and the relatively small size of the waste packages.
  - Because there is less reliance on waste form and waste package performance, it is easier to have confidence in the performance bases.
  - Smaller waste types are good candidates for this disposal concept (e.g., cesium/strontium capsules), allowing for efficient disposal such waste forms.
  - Some untreated waste types may be candidates for direct disposal in boreholes (e.g., untreated calcine waste), potentially reducing costs and risks associated with waste treatment
- Cons:
  - Currently limited to disposal of very small packages (around 1 ft (30 cm) diameter or less).
  - Lack of detailed design or demonstration of this disposal concept limits confidence.
  - If considered for disposing of commercial SNF, it would require repackaging and, in some cases, consolidating the spent fuel rods.

- Retrieving waste disposed of in deep boreholes could be difficult.
- The transportation capacity and logistics for small volumes of waste likely would limit disposal operations, so surface handling and storage concepts would need further consideration.
- For a given volume of waste, disposing of only very small-diameter packages results in handling more waste packages compared to the other three disposal concepts.

Although some of the criteria provided relatively little discrimination among disposal concepts, the evaluation identified potentially useful insights relevant to waste form properties independent of disposal concept. Specifically:

- Enough information does not current exist to evaluate the performance of direct disposal of sodium-bonded SNF in any geologic disposal concept. This waste type may require treatment regardless of the disposal concept.
- None of the disposal concepts considered posed significantly different concerns related to safeguards and security. Only those containing salts, granular solids, and powders, raised moderate security concerns associated with the potential for diversion for radioactive dispersal devices.
- All waste-form treatment options that involve handling or processing waste carry the potential for increased generation of secondary waste. This is most significant, perhaps, in comparing WG-1 (repackaging of all commercial SNF for disposal in PBCs) and WG-2 (disposal of all commercial SNF in existing dual-purpose canisters without repackaging).
- In general, demonstration of the technical readiness of a specific waste form is independent of the disposal concept being considered.

Conclusions

Technical conclusions relevant to each of the three questions posed at the beginning of the study are presented here.

Is a "one-size-fits-all" repository a good strategic option?

The study concludes that, from a technical perspective, any of the mined repository concepts could accommodate all of the waste forms with the exception of untreated sodium-bonded SNF, for which available information was insufficient to support an evaluation. The study concludes that the deep borehole disposal option is a good option for small waste packages and provides flexibility to a disposal strategy. The study also notes that disposal options that utilize multiple repositories are also technically viable.

Do different waste forms perform differently enough in different disposal environments to warrant different approaches?

The study did not identify any waste forms that required a specific disposal environment/concept. Other relevant observations include:

- With the exception of the untreated sodium-bonded SNF discussed above, all waste forms could be accommodated in multiple disposal concepts, although with varying degrees of confidence.

- Some disposal concepts may require segregating some waste forms from each other within a single repository. Specifically, halide-bearing wastes (including salt waste forms and the cesium/strontium capsules) may be corrosive, and if they are disposed of without treatment they should be isolated from other wastes in disposal concepts that rely on long-lived waste packages.
- Small waste forms are potentially attractive candidates for deep borehole disposal. Those wastes forms include salt wastes from electrometallurgical treatment of sodium-bonded SNF, untreated calcine waste, cesium/strontium capsules, and some DOE-managed SNF that has not yet been packaged.
- Salt allows for more flexibility in managing high-heat waste in mined repositories than other media.
- The study did not identify technical issues associated with disposing of mixed waste (i.e., waste containing both radioactive materials and constituents regulated under the Resource Conservation and Recovery Act).
- The study concluded that direct disposal of commercial SNF in existing dual-purpose canisters was potentially feasible but could pose significant challenges both in repository operations and demonstrating confidence in long-term performance.

Do some disposal concepts perform better with or without specific waste forms?

The study concludes that all of the disposal options evaluated have the potential to comply with applicable regulatory requirements that protect both worker and public health and safety, and protect the environment. There were a few disposal options considered that were not evaluated for the full range of criteria. These exceptions are those deep borehole options that are physically infeasible due to size constraints, and the disposal of untreated sodium-bonded SNF, for which information is insufficient to support an evaluation. All other disposal options identified in this study could be designed, constructed, and operated to provide safe and robust isolation of the waste forms. However, the evaluation results summarized above show that implementation and demonstration of robust performance may be simpler for some disposal concepts than for others.

Recommendations

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

All of the disposal concepts evaluated in this study have the potential to provide robust long-term isolation for specific wastes. Each of the three mined repository concepts could accommodate essentially all of the identified waste groups, with the exception of untreated sodium-bonded SNF, for which information is insufficient to support evaluation for disposal in any geologic disposal concepts. In addition deep boreholes are feasible for disposal of small waste packages and provide flexibility to the disposal strategy. Additional generic and site-specific R&D is needed before any disposal options can be implemented, although no recommendations were made with respect to specific R&D activities. The results of this study indicate that some options may provide greater flexibility or fewer challenges than others. Specifically:

1. For mined repository concepts:
    a. Salt provides greater flexibility for disposal of heat generating wastes because of the high thermal conductivity and high temperature limit. Disposal in this media provides greater confidence in estimates of long-term performance because it limits radionuclide transport (low permeability) and reduces the reliance on the waste form and waste package lifetimes. The relative lack of water and the high cross-section of chlorine for capture of thermal neutrons make it easier to address criticality concerns. In some cases, it may be appropriate to directly dispose of some untreated waste types, potentially reducing cost and risks associated with waste treatment. The operational experience at the Waste Isolation Pilot Plant provides additional confidence in this disposal concept.
    b. Clay/Shale is a disposal media with a significant amount of world-wide experience and it showed strong results as a disposal option for most waste groups with respect to most metrics. It is an attractive disposal option because it limits far-field radionuclide transport (low permeability and high sorption) and, therefore, reduces the reliance on the waste form and waste package lifetimes, compared to a crystalline disposal concept. However, compared to salt, there is more reliance on source-term performance and thermal constraints are greater.
    c. Mined repositories in crystalline rocks may offer operational advantages because of the rock strength, which allows robust openings to be easily maintained providing the potential flexibility of possible ramp access. However, for fractured crystalline systems, high reliance on clay barriers immediately surrounding the waste package poses additional challenges for high thermal loads that may degrade such barriers. Because of the need for robust performance of the source-term, confidence in system performance may be directly dependent on very conservative thermal management.
2. Deep borehole disposal options offer added flexibility for small waste forms and some wastes that have not yet been packaged. Attractive candidates for disposal in deep boreholes include cesium/strontium capsules, some DOE-managed spent fuels that have not yet been packaged, and the salt waste from electrometallurgical treatment of sodium-bonded SNF. Further, disposal of some untreated waste, such as direct disposal of calcine, may be an option that could reduce health risks and costs from further treatment of that waste. Because deep borehole disposal relies less on waste form and waste package lifetimes, it is easier to have confidence in the performance bases. Lastly, thermal load management issues are fewer because of the nature of the deep borehole disposal concept.

In addition, baseline cost considerations indicate that costs could be lowest for a salt disposal concept and highest for a crystalline disposal concept with cost for clay/shale systems spanning the range. Deep borehole disposal concept costs would be comparable (in this range) if all waste forms are included that can feasibly be disposed in deep boreholes. However targeted disposal of select wastes in deep boreholes could provide flexibility with only modest cost increases in conjunction with a mined disposal concept. For both salt and clay/shale disposal concepts, one primary challenge would be to address the ability to lower and lift very large packages in shaft emplacement systems.

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The complete draft report benefitted from internal technical review by Rob Rechard (Sandia National Laboratories).

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FIGURES

TABLES

ACRONYMS and ABBREVIATIONS

BWR	boiling water reactor
DOE	U.S. Department of Energy
DOE-NE	Department of Energy Office of Nuclear Energy
DPC	dual-purpose canister
DWPF	Defense Waste Processing Facility
EBR-II	Experimental Breeder Reactor II
EMT	electrometallurgical treatment
FFTF	Fast Flux Test Facility
FRG	Federal Republic of Germany
HEU	highly enriched uranium
HIP	hot isostatic pressing
HLW	high-level radioactive waste
INL	Idaho National Laboratory
INTEC	Idaho Nuclear Technology and Engineering Center
LEU	low-enriched uranium
LLW	low-level waste
MCO	multicanister overpack
MEU	medium enriched uranium
MOX	mixed oxide (fuel)
PBC	purpose-built canister
PWR	pressurized water reactor
RCRA	Resource Conservation and Recovery Act
SBW	sodium-bearing waste
SMR	small modular reactor
SNF	spent nuclear fuel
SRS	Savannah River Site
U.S.	United States
WIPP	Waste Isolation Pilot Plant
WVDP	West Valley Demonstration Project
WTP	Waste Treatment and Immobilization Plant

Units

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

EVALUATION OF OPTIONS FOR PERMANENT GEOLOGIC DISPOSAL OF SPENT NUCLEAR FUEL AND HIGH-LEVEL RADIOACTIVE WASTE

1 INTRODUCTION

1.1 Purpose and Scope

This study provides a technical basis for informing policy decisions regarding strategies for the management and permanent disposal of spent nuclear fuel (SNF) and high-level radioactive waste (HLW) in the United States (U.S.) requiring geologic isolation. Relevant policy questions this study can help inform include the following:

- Is a “one-size-fits–all” repository a good strategic option for disposal?
- Do different waste types and forms perform differently enough in different disposal concepts that they warrant different treatment?
- Do some disposal concepts perform significantly better with or without specific waste types or forms?

The study provides this basis by evaluating potential impacts of waste forms on the feasibility and performance of representative generic concepts for geologic disposal.

Participants in the study include representatives with expertise in a broad range of fields, including nuclear engineering, earth sciences, materials science, chemical engineering, and materials safeguards and security. Criteria and metrics for the study are necessarily qualitative, because of the complexity of the problem and the difficulty of quantifying estimates of the behavior of specific waste forms in generic disposal environments. Criteria include long-term repository performance, confidence in the bases for expected performance, operational considerations, technical readiness, the production of associated secondary wastes, and topics related to safeguarding and securing the wastes. In lieu of quantitative information about specific disposal sites and design concepts, insights are developed based on the full range of information available to the group, including detailed assessments done by previous repository programs in the U.S. and elsewhere in the world.

The scope of the waste in this study includes all existing SNF from commercial, defense, and research reactors, and SNF from reasonably foreseeable operations of existing reactors (projected to 2048). The study also includes existing HLW (e.g., vitrified HLW at the Savannah River Site (SRS) and the West Valley Demonstration Project (WVDP)) and waste forms projected to be generated in the future from existing process waste (e.g., projected vitrified HLW from HLW at Hanford, SRS, and the Idaho National Laboratory (INL)). In addition, the study includes consideration of both direct disposal of waste forms that are not currently planned for disposal without further treatment (e.g., calcine waste at INL) and other alternatives to planned treatments. The study acknowledges existing plans, commitments, and requirements where applicable, but the study evaluates options for disposal based primarily on technical, rather than programmatic or regulatory constraints.

The four representative disposal concepts included in this study are mined repositories in three geologic media—salt, clay/shale rocks, and crystalline (e.g., granitic) rocks—and deep borehole disposal in crystalline rocks. These are the four disposal concepts selected for further research and development activities by the U.S. Department of Energy (DOE) Office of Nuclear Energy’s Used Fuel Disposition Campaign (Rechard et al. 2011). Selection of these four concepts begins with the observation that options for disposal of SNF and HLW have been evaluated in multiple nations for decades, and deep geologic disposal was recognized as early as the late 1950s to be the most promising approach. By the 1980s, the U.S. waste management program had concluded that multiple geologic media had the potential to provide robust isolation, and that conclusion remains valid today. Experience gained in waste management programs in other nations reinforces that conclusion. For example, Sweden and Finland both have license applications pending for proposed mined repositories for SNF in crystalline rock; the U.S. has an operating repository in salt for transuranic waste at the Waste Isolation Pilot Plant (WIPP); Germany has extensive experience with the design of a mine for a SNF and HLW repository in salt; and France, Switzerland, and Belgium have completed detailed safety assessments for proposed SNF and HLW repositories in clay and shale media. No nations are currently planning deep borehole repositories, but the concept has been evaluated in multiple programs since the 1970s, and remains viable for waste forms small enough for emplacement.

Variants of the four primary concepts are also considered where appropriate. For example, as described by Hardin et al. (2012a), each of the mined repository concepts can, in principle, be implemented in both open (i.e., with active ventilation during the operational period) and closed (i.e., with early emplacement of backfill), depending on thermal load management needs. As discussed in the body of the report, some geologic media provide greater flexibility for thermal load management.

Other geologic disposal concepts have been proposed and are potentially viable. For example, Canada is currently evaluating a mined repository for intermediate-level radioactive waste in carbonate rocks (NWMO 2011) and the U.S. has evaluated a potential mined repository concept in volcanic tuff (DOE 2008). Although these concepts have unique features that distinguish them from the four selected for consideration in this report, attributes of the four concepts discussed here are representative of a broad range of other disposal concepts.

1.2 Method

The approach taken in this analysis begins with the recognition that options for permanent disposal of SNF and HLW must take into account potential interactions between the disposal concept (including features of both the geologic setting and the engineered repository) and the form in which the waste is disposed. This study begins with an assessment of the existing and reasonably foreseeable wastes that may require deep geologic disposal (i.e., the waste types), and then evaluates the various physical and chemical forms in which these waste types might be disposed (i.e., the waste forms). It should be noted here that given the wide range of waste types and forms considered in this report, there is no standard set of units (i.e., English versus metric) adopted within this discussion. Rather, the units used within the large range of literature referenced herein are used throughout this report.

Waste groups, consisting of waste forms with similar characteristics, are then defined and used to create disposal options. A disposal option consists of a waste group paired with one of the four geologic disposal concepts. These disposal options are evaluated against several different criteria that consider factors such as long-term safety of the disposal option, operational issues, the robustness of (or confidence in) available information, technical readiness, system-level cost, secondary waste production, and safeguards and security. Additional detail on the approach is provided in the following sections and in the body of the report.

1.2.1 Assumptions

Major assumptions and considerations used in this study include the following:

- HLW and SNF considered in this study are restricted to existing materials and those materials that can be reasonably expected to be generated by existing or currently planned facilities and processes

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

The set of disposal concepts used in this evaluation is the same as that identified by DOE's Used Fuel Disposition Campaign as a primary target for further research and development (Rechard et al. 2011), and is presented as a useful and representative, rather than comprehensive, set of concepts.

1.2.2 Terminology

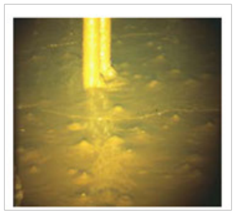
For the purposes of conducting this study, several terms relating to "waste" and "disposal" were defined and are used throughout this report. The three terms related to waste are waste type, waste form, and waste group. These are defined as follows and illustrated in Figure 1-1:

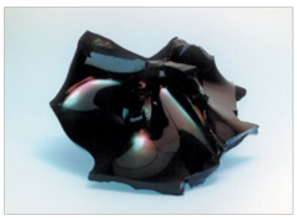
- ***Waste Type***—the currently existing materials (in whatever form, abundance, and location they occupy) that are to be disposed of as at least one, and possibly more than one, *Waste Form* in a deep geologic disposal concept (e.g., Hanford tank wastes; commercial spent fuels, HLW glass). Note some *Waste Types* that require no processing other than packaging may equate to a single *Waste Form* (Figure 1-1).
- ***Waste Form***—the end-state material as packaged that is to be disposed of in a deep geologic disposal concept. Note that some *Waste Types* may have more than one possible alternative *Waste Form* depending on the processing needed to modify the *Waste Type* to a *Waste Form*. The details of the expected *Waste Forms* are needed for evaluating their disposal.
- ***Waste Groups***—categorization of the suite of *Waste Forms* into groups that are similar enough in their thermal, chemical, radionuclide inventory, safeguards and security, physical, and packaging, characteristics that their preclosure handling/behavior and postclosure behavior in a geologic disposal concept would be broadly similar (e.g., all commercial SNF packaged in purpose-built containers). *Waste Forms* with a highly distinct feature in any one of these traits may define a *Waste Group* unto themselves (e.g., directly disposed sodium-bonded fuel). Note that if a *Waste Type* has more than one *Waste Form*, it may appear in more than one *Waste Group*, depending on the properties of each alternative *Waste Form.*

The two terms related to disposal are disposal concept and disposal option. These are defined as follows.

- ***Disposal Concept***—the conjunction of a facility with a geologic environment (e.g., deep borehole in crystalline (granitic) rock; mined repository in salt)
- ***Disposal Option***—the consideration of a particular *Waste Group* disposed in a specific *Disposal Concept* (e.g., HLW glass in a repository in clay/shale). Note the total number of *Disposal Options* will be the product of the number of *Waste Groups* and the number of *Disposal Concepts*. If a *Waste Type* has more than one *Waste Form*, it may appear in more than one *Waste Group*, depending on the properties of each alternative *Waste Form*.

Waste Type is what exists today





E.g., existing tank waste, existing HLW glass

Waste Form is what could go underground

E.g., Canisters of HLW glass from multiple sites and sources

Waste Group is an aggregation of *Waste Forms* with similar disposal characteristics

E.g., All HLW glass regardless of origin

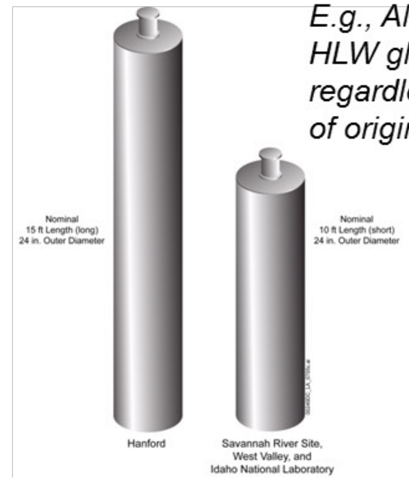


Figure 1-1. Illustration of waste type, waste form, and waste group

1.3 Organization of the Report

Chapter 1 of this report defines the purpose and scope of the study, identifies the wastes and representative disposal concepts under consideration, outlines the basic methodology, and defines the terminology used in the evaluation.

Chapter 2 provides a summary listing of all SNF and HLW currently existing or reasonably projected from current activities in the U.S. that is a candidate for deep geologic disposal. Waste is identified and characterized in terms of both present waste type and the waste form(s) in which the waste type may ultimately be disposed.

Chapter 3 describes the combination of waste forms into waste groups with similar characteristics. Waste groups are defined based on attributes relevant to disposal including radionuclide inventory, thermal output, chemical composition, physical properties, and packaging. Attributes relevant to safeguard and security topics are noted where relevant.

Chapter 4 defines the criteria and metrics by which disposal options are evaluated.

Chapter 5 presents the results of the qualitative evaluation for each metric for each disposal option, with primary pros and cons discussed for each waste group and for each disposal concept.

Chapter 6 summarizes the work of this evaluation, synthesizes the evaluation results by disposal concept, and provides answers to the study questions, including the primary conclusions of the study.

2 WASTE TYPES AND WASTE FORMS

This study is chartered to evaluate existing and "reasonably foreseeable" SNF and HLW inventories. The existing wastes are those that can be inventoried, and those scheduled to be generated by currently operating reactors (or reactors under construction) and HLW-generating activities. The boundary of "reasonably foreseeable" is selected to include wastes that can be forecast from current actions by industry or government, but is not intended to include potential waste streams from advanced fuel cycle technologies that may be—or may not be—deployed in the future. This enables the physical and radiological characteristics of both existing and reasonably foreseeable wastes to be sufficiently well defined for evaluation in disposal options.

Several examples help describe the boundary of "reasonably foreseeable" wastes.

1. Several prototype high-temperature gas-cooled reactors have been built and operated in the past in the U.S., and spent fuel from these reactors exists in current inventories. These fuels are clearly included in the scope of this study. The quantity and characteristics of these fuels are well known and included in the Section 2.1.2 inventories. In contrast, the DOE Office of Nuclear Energy (DOE-NE) program to develop and deploy a 'Next-Generation Nuclear Plant' seeks to build new high-temperature gas-cooled reactors in the future. Several types of fuel have been proposed for this reactor, but the final characteristics of discharged fuel are not well established. Furthermore, at this time there is not a construction commitment for this reactor, and there is no firm schedule for deployment. Spent fuel from these potential reactors is considered beyond the "reasonably foreseeable" boundary, and thus is not considered in the scope for this study, but may be addressed in the future.

2. Several prototype sodium-cooled fast-spectrum reactors have been built and operated in the past in the U.S., and spent fuel from these reactors exists in current inventories, are well characterized and included in Section 2.1.2 inventories—and are thus included within the scope of this study. Driver and blanket fuels from Experimental Breeder Reactor II (EBR-II) are currently being processed at DOE-NE facilities at INL, so the HLW generated from this processing is included in the scope of this study as both existing and reasonably foreseeable. The recent DOE-NE program to develop a fast-spectrum burner reactor proposed building new fast spectrum reactors in the future. However, this program is not currently being actively pursued, so spent fuel from these potential reactors is considered beyond the scope for this study.

3. Although prototype HLW glass with higher radionuclide loading is being studied currently, such potential glass waste forms are considered beyond the scope of this study. Currently existing glass waste forms and glass waste forms that are planned to be generated, such as at the vitrification facility at Hanford, are included within the scope of this evaluation.

Those spent fuel and high-level wastes that exist, or are scheduled to exist by current activities, and those that fall within the "reasonably foreseeable" boundary are described in detail below and in Appendix A. This study also included waste that may not necessarily be considered high-level waste but nonetheless requires deep geologic disposal, as discussed in Appendix A, Section A-3. Although it is recognized that some portion of the excess plutonium inventory (e.g., material not suitable for mixed-oxide fuel (MOX)) will need to be dispositioned, due to the uncertainty in the amount/type of that inventory and the uncertainty in its disposal method/form, that material is not included in this report.

2.1 Spent Nuclear Fuel

SNF is generally categorized as either commercial or DOE-managed. The current inventory of commercial SNF is discussed in Section 2.1.1, DOE-managed SNF is discussed in Sections 2.1.2 and

2.1.3, while SNF to be generated in small modular reactors (SMRs) is discussed in Section 2.1.4. Further details regarding all these types of SNF can be found in Appendix A.

2.1.1 Commercial Spent Nuclear Fuel

The current inventory of domestic SNF is massive, diverse, dispersed, and increasing (e.g., Wagner et al. 2012). As of 2012, 69,463 metric tons of heavy metal (MTHM) of commercial SNF (NEI 2013), representing a total of ~23 billion curies of long-lived radioactivity (Carter et al. 2012), are currently stored at 75 sites in 33 states (Figure 2-1) (GAO 2012). The distribution of the 2011 SNF inventory from pressurized water reactors (PWRs) and boiling water reactors (BWRs) in wet (pool) and dry storage is shown in Figure 2-2 (from Wagner et al. 2012, Figure 3). The commercial SNF inventory is increasing annually by ~2,000 MTHM (GAO 2011) and will increase at a greater rate in the future if the number of operating nuclear reactors increases. Assuming that 2,000 MTHM of SNF is created each year from 2012 through 2048, the commercial SNF inventory in 2048 is estimated to be 142,000 MTHM.

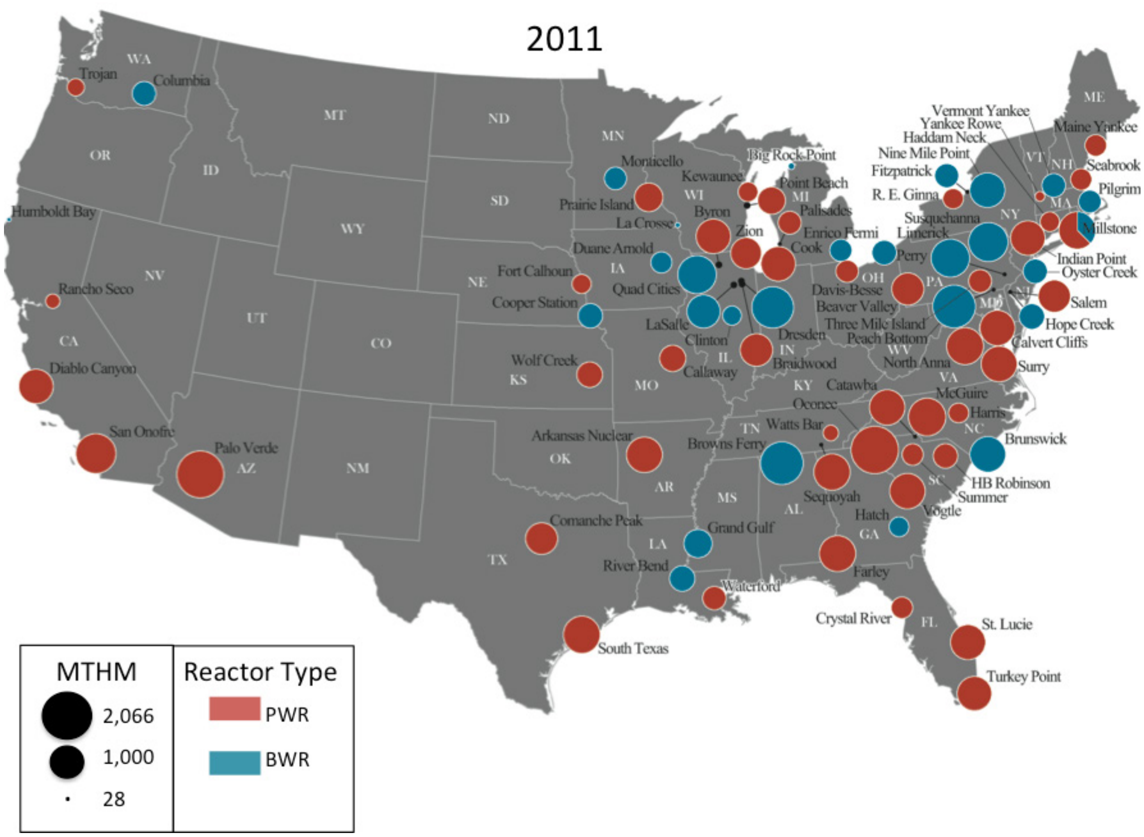
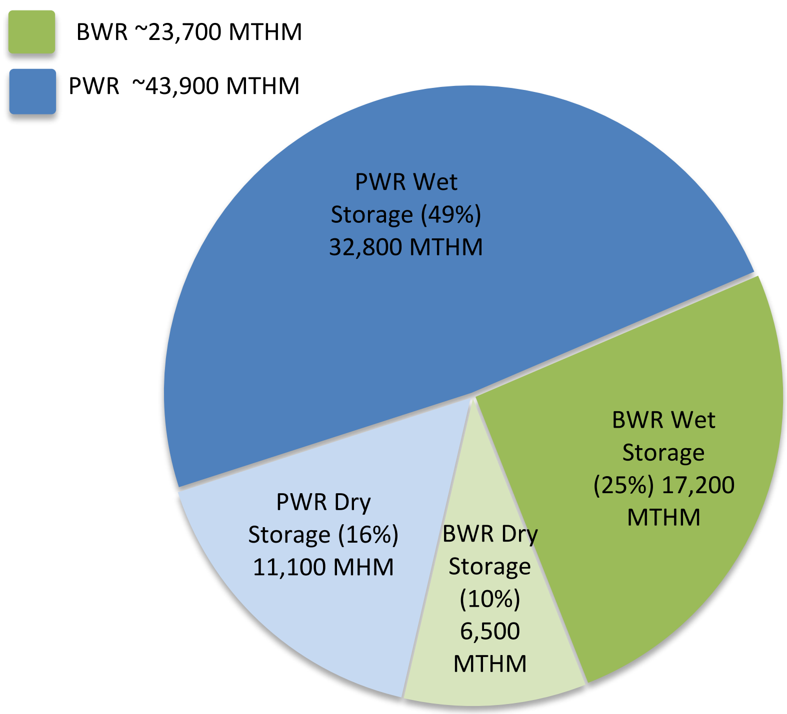


Figure 2-1. Location and quantity of discharged commercial SNF in the U.S. as of 2011



Source: Wagner et al. 2012, Figure 3.

Figure 2-2. Distribution of commercial SNF inventory in wet and dry storage (as of 2011)

Based on the characteristics described in detail in Appendix A, PWR and BWR assemblies are not sufficiently different to warrant classifying them as different waste types for the purposes of geologic disposal. Although specific exceptions may be identified in the future, there is currently no known compelling reason to process the bulk of the commercial SNF before disposal beyond the specifics of the packaging for disposal itself (Wagner et al. 2012). Although burnup is variable (Appendix A), it has not been used here to make distinctions among commercial SNF for purposes of deep geologic disposal. For commercial SNF, the benefits/constraints for geologic disposal considerations from the packaging of the SNF provide the largest distinctions for waste form categorization. Therefore, within this analysis, there is one waste type: commercial SNF (with included nonfuel assembly hardware); and two waste forms: purpose-built canisters (PBCs) and dual-purpose canisters (DPCs).[1]

For the purposes of this disposal option evaluation, PBCs are canisters that would be specifically designed (size, materials of fabrication, fabrication processes, etc.) and loaded for a particular repository concept. The design of PBC would take into account the specific geologic setting and how engineered and natural barriers are expected to evolve over time. PBCs could contain both commercial SNF and non-fuel assembly hardware.

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[1] Note that under the Standard Contract the DOE is only obligated to accept bare fuel for disposal and that contract holders who have packaged their spent nuclear fuel into DPC will have to sign a contract amendment with the DOE to have their DPC accepted.

In this work, commercial SNF is also considered as an alternative waste form disposed directly in DPCs that are designed and loaded to meet the current operational requirements of commercial nuclear power production facilities and the transportation and storage requirements of 10 CFR Part 71 and 10 CFR Part 72, respectively. The majority of SNF in existing dry storage is in DPCs and nearly all new dry storage transfers are in DPCs. These canisters typically hold as many as 32 PWRs assemblies (or 68 BWR assemblies) and recent designs hold even more. DPCs are not designed for a specific disposal concept and therefore the repository would need to account for and accommodate the existing DPCs sizes, materials of fabrication, fabrication processes and as-loaded and as-received content and conditions. DPCs contain commercial SNF and non-fuel assembly hardware.

The PBC design will depend significantly on the host geologic media and repository concept; therefore for this analysis there are four different representative designs/sizes of PBCs that were considered (corresponding to the four disposal concepts considered) and one representative design (with variable loading capacity) for DPCs. These different sizes can be seen in Table 2-1 and are covered in detail in Section A.1 of Appendix A. A summary of some of the physical characteristics of both PBCs and DPCs can be seen in Table 2-2. As expected, the number of disposal canisters needed varies inversely with the canister size and the range covers from about 11,400 DPC to about 470,000 PBC for disposal in boreholes, if this option was considered, of all commercial SNF generated through the year 2048.

Table 2-1. Potential purpose-built canister capacity for generic media

Canister type	Media/Design Concept	Representative Canister Capacity
PBC-Borehole	Deep Borehole	349 PWR fuel rods (consolidated) or 1 BWR Assembly[1]
PBC-Small	Clay/Shale or Crystalline: Enclosed	4 PWR or 9 BWR
PBC-Medium	Salt: Enclosed	12 PWR or 24 BWR
PBC-Large	Clay/Shale: Open	21 PWR or 44 BWR
DPC	N/A	32 PWR or 68 BWR

Notes: 1. Minimum values based on the smaller inner diameter (8.05 in. or 20.4 cm; Arnold et al. 2011) deep borehole PBC. Additional capacity for BWR fuel rods can be achieved with rod consolidation.
Sources: for deep borehole: Brady et al. 2009 and Arnold et al. 2011; and for rest Hardin et al. 2012b.

Table 2-2. Physical characteristics of both PBCs and DPCs

Canister type	Number of PWR canisters needed*	Number of BWR canisters needed*	Total number of canisters needed	Outer Diameter (m) **	Length (m) **
PBC-Borehole	173,163	296,900	470,063	0.27	4.6
PBC-Small	56,375	32,989	89,364	0.82	5
PBC-Medium	18,792	12,371	31,163	1.29	5.13
PBC-Large	10,738	6,185	16,924	1.6	5.13
DPC	7,047	4,366	11,413	2	5.13

* Data came from interpolating Scenario 2 of Carter et al. (2012) to the year 2048. Note that this data represents the number of canisters needed to disposal all of the commercial SNF generated up to the year 2048.
** Data came from Arnold et al. (2011) for deep borehole canisters and from Hardin et al. (2012b) for the remaining canister types.

An important concept in the safeguarding of commercial SNF is self-protection. Within the U.S. Nuclear Regulatory Commission (NRC) regulations in 10 CFR Part 73, self-protection is attributed to SNF "which is not readily separable from other radioactive material and which has a total external radiation dose rate in excess of 100 rem per hour at a distance of 3 feet from any accessible surface without intervening shielding." Previous studies have shown that the dose rate for typical discharged commercial SNF will fall below the current self-protection limit (100 rem/h at 3 ft) between 70 and 120 years after discharge (Durán et al. 2011). The BWR assembly will be closer to the 70-year time period, while the PWR assemblies will be closer to the 120-year time period. For the 2048 repository start-up target, many BWR fuel assemblies and some PWR assemblies discharged before 1978 (70 years old at 2048) may no longer be self-protecting.

Once the commercial SNF is no longer self-protecting, additional safeguards and security measures may be needed for storage and transportation purposes. This will be especially true with SNF assemblies that contain more than 2 kg of plutonium (classified as safeguards and security Category I—see Appendix D, Table D-1) and require the additional security and national safeguards (for commercial SNF see Table A-4).

2.1.2 DOE-Managed Spent Nuclear Fuel

The SNF under the purview of the DOE includes a broad range of fuels resulting from decades of nuclear research, development and testing, defense power, electric power production, experimental power production, and production of weapons and research materials. In addition to fuels from reactors operated by the DOE and U.S. Department of Defense, DOE fuels also include a number of university research reactors as well as foreign research reactor fuel returned to the U.S. as part of the Foreign Research Reactor Spent Nuclear Fuel Acceptance Program. DOE SNF has been regulated by agencies such as the Department of Defense, DOE, foreign research reactor entities, and the NRC. Some of the DOE fuels are packaged into multicanister overpacks (MCOs), however, other fuels remain in storage and will require packaging into either MCO or standardized canisters prior to movement to a repository.

DOE SNF is further described below to provide a general understanding of the variety of waste included. The DOE SNF has previously been delineated into 34 categories, or DOE fuel groupings (for details see Table A-5). The DOE groupings have proven convenient for safety analyses and are used in this study to define the specific waste types. These DOE groups are listed in Table 2-3 with their masses and the estimated numbers of disposal containers. Note that additional discussion is given in Section 2.1.3 of the DOE highly enriched uranium (HEU) fuel (naval SNF in DOE Group 32) in large packages. Also, because of the multiple possible disposal pathways, the DOE sodium-bonded fuel and its various waste forms (DOE Group 31) are discussed in Section 2.2.4.

2.1.2.1 DOE SNF General Description

DOE SNF includes a variety of geometries, fuel matrices, cladding types, fissile materials, enrichments and burnups. Both domestic and foreign suppliers supported the development of experimental fuels now managed by DOE (DOE 2007, Section 3.2). Some of these suppliers are no longer producing reactor fuels or have gone out of business.

DOE SNF and the associated reactors have used a variety of moderators such as beryllium, graphite, heavy water, light water, metal hydride, and organics. Additionally, DOE SNF has been used in many different reactors which used a variety of coolants such as air, helium, heavy water, light water, sodium-potassium alloy, nitrogen, organic, sodium, and even no coolant (DOE 2007, Section 3.1).

Fissile materials in DOE SNF include ^{233}U, ^{235}U, the various isotopes of plutonium, and other transuranics. The ^{235}U enrichment ranges from depleted uranium to over 93% (and up to 97% for naval fuel). The effective end-of-life enrichment values pertaining to DOE SNF are adjusted to account for the ingrowth of ^{233}U, plutonium, and other fissile radionuclides. DOE fuels can be grouped according to effective end-of-life enrichment as follows (DOE 2007, Section 3.2):

- "High"—those with enrichments greater than 20%, HEU.
- "Medium"—those greater than 5% but less than 20%, medium enriched uranium (MEU).
- "Low"—those with less than 5%, low enriched uranium (LEU).

The burnup of DOE SNF ranges from very slightly irradiated to over 500,000 MWd/MTU. For some DOE SNF, burnup is categorized in terms of ^{235}U burnup % and heavy metal burnup % consumed rather than MWD/MTU. The burnup for these fuels range from very slightly irradiated to over 80% of the initial ^{235}U to over 70% of the initial heavy metal. Thermal power for DOE fuel was estimated for 2010 and 2030. Figure A-14 shows a summary of the estimated decay heat for DOE SNF per transportation, aging, and disposal ("TAD") canister for 2030. Most (~98%) of the DOE SNF falls below an estimated 500 W per canister; significantly less than the anticipated 25-kW limits for commercial fuels packaged in transportation, aging, and disposal canisters (BSC 2007). There are a few fuels that are above 500 W per canister. However, the source term for these fuels was calculated based on very conservative assumptions because very little was known about these fuels. When these fuels are actually packaged, it is likely that the actual measured decay heat will be much less than these conservative estimates (DOE 2007, Section 3.2).

Compared to commercial SNF, DOE SNF is much more heterogeneous. The spent fuels exist in a variety of compounds, sizes, and configurations (e.g., rod, canister of scrap, plates), with various cladding materials, cladding conditions, and fuel matrices. A description of the properties of the fuels in the 34 DOE SNF categories is given in Table A-5.

2.1.2.2 Plans for Packaging and Disposal of DOE SNF

DOE plans to package most of its SNF (about 98% of the metric tons of heavy metal) into MCOs and standardized canisters suitable for storage, transport, and disposal without the need to be re-opened (DOE 2007, Section 3.2). A standardized disposal canister design was developed which includes canisters of 18- and 24-in. diameter and 10- and 15-ft length. Currently no SNF has been packaged into the standardized disposal canister design (see Table 2-3 for estimated numbers of these canisters). DOE SNF of commercial origin having handling features interchangeable with either BWR or PWR fuel assemblies and known to have no defects may be handled in the same manner as commercial SNF as specified in 10 CFR Part 961 (DOE 2007, Section 1.5.1.3), and thus will not be placed in standardized canisters.

The canister count estimates in Table 2-3 are estimates based on fuel dimensions and canister sizes. The actual canister count will change based on a variety of factors such as: fuel mixing, basket design and use, shield plug use, criticality considerations, and location of fuel and packaging facilities. See Appendix A-1.2.2 for further information.

Table 2-3. Mass and canister count estimates by fuel group for DOE-managed SNF

Fuel Group	MTHM[a]	MT[a]	Estimated Number of Canisters by Type[b]					Bare Fuel[c]	
			18x10	18x15	24x10	24x15	MCO[d]	PWR	BWR
01. U metal, zirc clad, LEU	2,096	3,130	—	2	—	—	388	—	—
02. U metal, non-zirc clad, LEU [e]	10	10	6	—	—	—	7	—	—
03. U-zirc	7	14	12	8	—	—	—	—	—
04. U-Mo	4	6	10	—	—	—	—	—	—
05. U oxide, zirc clad, intact, HEU	<1	23	3	55	—	—	—	—	—
06. U oxide, zirc clad, intact, MEU	2	4	8	—	—	—	—	—	—
07. U oxide, zirc clad, intact, LEU [e]	64	140	32	83	—	—	18	39	2
08. U oxide, stainless steel/hastelloy clad, intact, HEU	<1	2	13	—	—	—	—	—	—
09. U oxide, stainless steel clad, intact, MEU	<1	6	3	9	—	—	—	—	—
10. U oxide, stainless steel clad, intact, LEU	<1	3	1	3	—	—	—	1	—
11. U oxide, non-alum clad, non-intact or declad, HEU	<1	8	196	6	—	—	—	—	—
12. U oxide, non-alum clad, non-intact or declad, MEU	<1	4	112	1	—	—	—	—	—
13. U oxide, non-alum clad, non-intact or declad, LEU [e]	108	370	10	357	—	—	—	40	76
14. U oxide, alum clad, HEU	4	68	209	—	133	—	—	—	—
15. U oxide, alum clad, MEU and LEU	<1	2	9	—	—	—	—	—	—
16. U-ALx, HEU [e]	8	93	548	92	—	—	—	—	—
17. U-ALx, MEU	3	12	74	—	—	—	—	—	—
18. U_3Si_2 [e]	7	27	93	145	—	—	—	—	—
19. Th/U carbide, tristructural isotropic or buffered isotropic coated particles in graphite [e]	25	316	1	505	—	—	—	—	—
20. Th/U carbide, monopyrolytic carbon coated particles in graphite	2	33	—	63	—	—	—	—	—

Table 2-3. Mass and canister count estimates by fuel group for DOE-managed SNF (cont.)

Fuel Group	MTHM[a]	MT[a]	Estimated Number of Canisters by Type[b]					Bare Fuel[c]	
			18x10	18x15	24x10	24x15	MCO[d]	PWR	BWR
21. Pu/U carbide, non-graphite clad, not sodium bonded	<1	<1	3	3	—	—	—	—	—
22. MOX, zirc clad	3	4	6	—	—	—	—	—	5
23. MOX, stainless steel clad	11	53	13	127	—	—	—	—	—
24. MOX, non-stainless steel/non-zirc clad	<1	<1	2	1	—	—	—	—	—
25. Th/U oxide, zirc clad	43	85	9	12	—	27	—	—	—
26. Th/U Oxide, stainless steel clad [e]	8	12	11	1	—	—	—	—	—
27. U-zirc hydride, stainless steel/incoloy clad, HEU	<1	4	18	—	—	—	—	—	—
28. U-zirc hydride, stainless steel/incoloy clad, MEU	2	18	70	—	—	—	—	—	—
29. U-zirc hydride, alum clad, MEU	<1	6	18	—	—	—	—	—	—
30. U-zirc hydride, declad	<1	<1	7	—	—	—	—	—	—
31. Metallic sodium bonded	60	—	—	—	—	—	—	—	—
32. Naval	65 [f]	—	—	—	—	—	—	—	—
33. Canyon stabilization	N/A	—	—	—	—	—	—	—	—
34. Misc. (not previously listed) [e]	<1	<1	9	1	—	—	—	—	—
Totals	**2,532**	**4,453**	**1,506**	**1,474**	**133**	**27**	**413**	**80**	**83**

[a] MTHM and MT are rounded to next higher whole number or reported as <1, as applicable.
[b] Representation for standardized canister dimensions are "diameter in inches" x "length in feet", thus18x10 represents an 18 in. diameter by 10 ft long canister.
[c] Intact PWR and BWR will be shipped as bare fuel in a transport cask; it is assumed they will not need to be placed in a standardized canister.
[d] MCOs are 24 in. in diameter and 166 in. long
[e] Fuel group contains records with missing or incomplete MTHM and/or MT data.
[f] This is the expected generation of naval spent fuel mass through the year 2035 that was estimated for delivery to the Yucca Mountain site in its anticipated 25 years of operation (2010 through 2035).
Source: Query of Spent Fuel Database, Version 6.2.3 (DOE 2011).

2.1.3 Naval Spent Fuel

Naval propulsion spent fuel consists of HEU, which is defined as uranium that has been enriched to 20% or greater in the ^{235}U isotope and is considered to be weapons usable. HEU SNF, shown in Table 2-3 as DOE fuel group 32, comes in different forms, including metals, oxides, solutions, reactor fuel, and irradiated SNF (see also Section 2.1.2). It consists of solid metal and metallic components that are nonflammable and highly corrosion resistant, and neither pyrophoric, explosive, combustible, chemically reactive, nor subject to gas generation by chemical reaction or off-gassing. Naval SNF is from pressurized water reactors with the exception of one design operated in a sodium cooled reactor. A small amount of the naval SNF from the sodium-cooled reactor remains (approximately 0.0036% of naval SNF allocation). Residual sodium has been cleaned from this last type of naval SNF (DOE 2008, Section 1.5.1.4).

Naval SNF is being packaged in canisters pending shipment to a repository. Two canisters, one short and one long, were designed to accommodate different naval SNF fuel assembly designs. Both canisters were sized to fit within the proposed design for the Yucca Mountain repository waste package. The outer diameter of the canister is 66 in. nominal (66.5 in. maximum). The maximum external dimensions ensure naval SNF canisters fit into the waste packages. The short canister is 185.5 in. (nominal) in length (187 in. maximum), and the long canister is 210.5 in. (nominal) in length (212 in. maximum). With the exception of length, the other characteristics of the naval SNF canisters are identical. Approximately 400 canisters (310 long and 90 short) are planned to be packaged and temporarily stored pending shipment to a repository for disposal (DOE 2008, Section 1.5.1.4).

2.1.4 Small Modular Reactor Fuels

For most comparisons, an SMR fuel assembly (based on a light water reactor technology) can be generalized as an assembly from a large PWR with the only major difference being that the SMR assembly will be approximately half the length of a large PWR assembly. The SMR vendors have all stated that their assemblies will have enrichments below 5% and make use of burnable poisons/enrichment variations to control excess reactivity and power peaking in the core. The expected waste stream from these assemblies, when measured using a metric that includes burnup, will roughly be the same as that from the PWRs. This conclusion is based on minimal differences in the neutron spectrum, fuel burnup, and fresh fuel enrichment between an SMR and large PWR. The ratio of assembly support structure (end fittings) to active fuel will be higher for SMRs compared to that of the large PWR.

These SMR spent fuels will be similar enough to commercial SNF (though smaller in dimension and having more support structure per fuel mass) that they can be considered as a disposed waste form included in those considerations for commercial SNF described in detail in Section A-1.1. The absolute amount of these types of spent fuels will be small in comparison well into the foreseeable future of the fuel cycles.

2.2 High Level Wastes

HLW exists in several forms, which are discussed below. Tank waste from fuel reprocessing that has already been immobilized in borosilicate glasses is discussed in Section 2.2.1, while Section 2.2.2 presents the tank wastes from fuel reprocessing that is projected to be immobilized in borosilicate glass in the future. Calcine HLW at the Idaho site, for which several alternative treatment processes have been considered; sodium-bearing waste (SBW) at the Idaho site; and cesium/strontium capsules stored at the Hanford site are examined in Section 2.2.3. Finally, HLW types being generated at the INL from the electrochemical treatment of sodium-bonded fuels, including salt waste and metallic waste, are discussed in Section 2.2.4.

2.2.1 Existing Vitrified HLW

HLW tank waste from fuel reprocessing has been vitrified at WVDP in New York and SRS in South Carolina. The tank waste at WVDP is the result of commercial fuel reprocessing conducted in the between 1966 and 1972. The tank waste at SRS is the result of reprocessing fuels for nuclear weapons production and for recovery of special radioisotopes; fuel processing efforts at SRS began in the 1950s and continue today at H Canyon. In addition, the Hanford site produced some vitrified glass for the Federal Republic of Germany; this glass is currently stored at Hanford. These existing wastes are discussed below.

2.2.1.1 West Valley High-Level Radioactive Waste Glass

The plutonium uranium extraction (PUREX) process was used at WVDP in West Valley, New York, to recover plutonium and uranium from commercial SNF. The thorium extraction (THOREX) process was also used to process a batch of mixed uranium-thorium fuel. The liquid waste inventory at WVDP prior to vitrification was about 600,000 gallons with radioactivity content of 24 million curies. Vitrification operations were completed in 2002, resulting in over 1 million pounds of borosilicate glass in 275 canisters. Additional detailed discussion of this waste type can be found in Section A-2.1.1 of Appendix A.

The WVDP canisters were fabricated from austenitic stainless steel Type 304L. They comprised of four major components: canister shell, bottom head, top head and neck flange. The weld filler metal ASME SFA5.9 ER308L was used to assemble the canister. The 308L alloy was also used for the weld-beaded canister identification labels (Palmer and Barnes 2002). The physical characteristics of the HLW canisters are summarized as follows:

- Canister length: 300 cm
- Nominal outer diameter: 61 cm
- Thickness: 0.34 cm
- Empty canister weight: 181.4 kg
- Available volume: 0.83 m^3
- Material: Stainless Steel 304L
- Nominal fill height: 91%
- Nominal glass volume: 0.76 m^3
- Maximum allowable glass weight: 2,500 kg.

The heat generation rate for the HLW canister was calculated using the SCALE (Standard Computer Analyses for Licensing Evaluation) computer codes. The radionuclide concentrations provided the input for the SCALE computation. All predicted values are lower than the Waste Acceptance Product Specifications (DOE 2012a) limit of 1,500 W per canister. The total thermal output for WVDP HLW canisters in 2017 is 44,200 W (DOE 2008).

2.2.1.2 Savannah River Site High Level Radioactive Waste Glass

The SRS has been reprocessing spent fuel since 1954, and is in the midst of vitrifying the waste that resulted from that reprocessing. Through 2012, over 4 million gallons of waste has been treated at the SRS Defense Waste Processing Facility (DWPF) resulting in over 14 million pounds of borosilicate glass contained in 3,339 canisters. To facilitate operations, DWPF uses a macrobatch strategy where a large volume of sludge (approximately 2 million liters) is prepared and staged for transfer to DWPF for processing. In this manner, a macrobatch represents a consistent waste chemistry feed for glass production. The DWPF recently completed processing macrobatch 9. Data are available for glass produced through macrobatch 8 (early 2012) and is presented in this section. Table 2-4 provides data on the number of canisters and amount of glass produced in the first eight macrobatches at DWPF, representing operations from 1996 through early 2012, along with the calculated heat content of each canister. Canisters are 3 m long, 61 cm in diameter (outer), and 0.95 cm thick.

Table 2-4. Canisters produced, glass mass and canister heat content (at the time of production) in macrobatches 1 through 8

	Macrobatch							
	1	**2**	**3**	**4**	**5**	**6**	**7**	**8**
Number of canisters	495	726	363	727	314	323	194	197
Glass (kg x1,000)	890	1260	650	1,300	560	590	310	350
Heat Content (W/canister)	4.0	4.0	19.0	25.1	32.1	45.1	120.7	115.4

2.2.1.3 Radioactive Waste Glass at Hanford for Federal Republic of Germany

In 1986 and 1987, the Pacific Northwest Laboratory prepared isotopic heat and radiation sources to be used as part of the repository testing program by the Federal Republic of Germany (FRG) in the Asse Salt Mine (Kuhn and Rothfuchs 1989). Using the radioactive liquid-fed ceramic melter in the 324 Building, thirty stainless steel canisters were filled with borosilicate glass spiked with ^{137}Cs and ^{90}Sr to achieve the desired heat and dose targets (note: ^{135}Cs also is in the glass but contributes negligibly to the heat as it is a long-lived isotope). The ^{137}Cs (and ^{135}Cs) was obtained from cesium capsules (see Section 2.2.3.3) from the Hanford site, and the ^{90}Sr was obtained from strontium nitrate in B-Plant. In addition to the 30 sources, two production demonstration canisters and two instrumented canisters for heat transfer studies were also produced (for a total of 34 canisters). The FRG testing program was stopped before the canisters could be shipped and they have remained at the Hanford site. They are currently stored at the Central Waste Complex at the 200-West area on the central plateau of the Hanford site.

The 34 canisters were fabricated in Germany from stainless steel. They are 1.2 m long by 0.3 m in diameter. As part of the testing, the filled canisters passed through a 306-mm inside diameter tube to check their ovality.

The 30 isotopic source canisters were filled in three separate processing campaigns (RLFCM7, RLFCM-8, and RLFCM-9) with different source objectives for each set of 10 canisters (Brouns and Powell 1988). Table 2-5 summarizes the heat and dose characteristics of the 34 FRG glass canisters.

Appendix A provides additional information regarding the FRG glass. Table A-19 summarizes the glass compositions for the three campaigns. Table A-20 summarizes the weights of the canister components and the glass contained inside. Dimensions of the casks are shown in Table A-21 along with number of canisters in each cask.

Table 2-5. Average heat and dose characteristics for FRG glass canisters

Number of Canisters	^{137}Cs Content (kCi)			^{90}Sr Content (kCi)			Decay Heat (W/canister)	Surface Dose (R/hr)
	Average	Min	Max	Average	Min	Max	Average	Average
10	192	175	237	85	67	110	1490	272,000
10	78	17	187	143	90	159	1330	112,000
10	207	182	233	130	103	148	1860	310,000
2[a]	Unknown	—	—	Unknown	—	—	1020	190,000
2[b]	Unknown	—	—	Unknown	—	—	Unknown	Unknown

[a]Instrumented canisters for heat transfer studies
[b]Production demonstration canisters
Source: Brouns and Powell 1988, Holton et al. 1989.

2.2.2 Projected Vitrified HLW

Although progress has been made to vitrify HLW tank waste at SRS (see Section 2.2.1.2), the bulk of the waste (including most of the salt fraction) remains to be vitrified. Additionally, all tank waste at Hanford remains to be vitrified. This section includes descriptions of the forecasted waste treatment and glass products at Hanford and SRS.

2.2.2.1 Hanford High Level Radioactive Waste Glass

The Hanford Site, located in southeastern Washington State, has approximately 54.6 million gallons (~207 million liters) of radioactive and listed hazardous wastes stored in 177 underground tanks on Hanford's Central Plateau (Certa et al. 2011). The wastes were generated from plutonium production as part of the U.S. defense programs during World War II and the Cold War. Fuel from production reactors along the Columbia River (and some from other reactors) was reprocessed to extract plutonium and later uranium for recycling. Wastes from those reprocessing operations are stored in the underground tanks and in cesium and strontium capsules (see Section 2.2.3.3).

The construction of the Hanford Tank Waste Treatment and Immobilization Plant (WTP) HLW vitrification facility was scheduled to be completed by 2016 and waste treatment is projected to be concluded in 2043, with the production of between 9,000 and 15,000 (GAO 2009) canisters of HLW glass with a current nominal value of 10,586 canisters (Certa et al. 2011); however, at present, a later date for the completion of construction appears likely. The physical characteristics of the Hanford HLW glass canisters are summarized as follows (DOE 2008; DOE 2011):

- Canister length: 450 cm
- Nominal outer diameter: 61 cm
- Thickness: 0.95 cm
- Empty canister weight: 715 kg
- Available volume: 1,190 liters
- Material: Stainless Steel 304 L
- Nominal fill height: 95%
- Nominal glass volume: 1,135 liters
- Filled canister weight: 3,735 kg
- Glass weight: 3,020 kg
- Glass density: 2.66 g/cm^3

Details of the WTP process for vitrification and the resulting properties of the glass are provided in Section A-2.2.1 of Appendix A.

2.2.2.2 Savannah River Site Tank Waste

As discussed in Section 2.2.1.2, about 4 million gallons of liquid waste at SRS has already been incorporated into borosilicate glass, leaving about 33 million gallons still to be treated. The documented System Plan Revision 18 (Chew and Hamm 2013) projects that 7,824 canisters will be produced by the time the SRS HLW mission is completed. As reported in Section 2.2.1.2, 3,339 canisters were produced during macrobatch 1–8 campaigns. Therefore, it is projected that 4,485 additional canisters will be produced post macrobatch 8 (for a total of 7,824 canisters). System Plan Revision 18 projects that twelve additional macrobatches will be processed in DWPF post macrobatch 8. The waste form and the waste form canisters associated with future DWPF operations will be consistent with those for the existing waste described in Section 2.2.1.2. The projected heat output for the canisters (Chew and Hamm 2012) is shown in Table 2-6.

Table 2-6. Number of canisters in wattage ranges for projected DWPF canisters

Scenario	Number of DWPF Canisters in Wattage Range					
	<100 W	100–149 W	150–199 W	200–249 W	250–299 W	450–499 W
Nominal Salt Waste Processing Facility operations	459	897	1598	924	363	—
Full processing capacity	459	1491	1565	462	—	264

Note: Total decay heat for all 7,580 canisters during the DWPF mission is approximately 800,000 W. Numbers align with the number of canisters projected by System Plan Revision 17 (7,580; Chew and Hamm 2012).

2.2.3 Existing Wastes Other than Glass

Although the high-level radioactive tank waste at SRS and Hanford represent the most voluminous fractions of DOE HLW, there are other wastes that are classified as HLW or that may be disposed of in a HLW repository. These are discussed below.

2.2.3.1 Calcine Waste at Idaho National Laboratory

At the Idaho Nuclear Technology and Engineering Center (INTEC), previously known as the Idaho Chemical Processing Plant, SNF was reprocessed to recover enriched uranium and other nuclear-related products. The first-cycle raffinate from the uranium extraction was temporarily stored in underground tanks before being converted to a solid granular material called calcine. Fuel reprocessing began in 1953 and concluded in 1994, and the calcination operations ran from December 1963 to May 2000. Approximately 4,400 m^3 of calcine is currently stored in six Calcine Solids Storage Facility bin sets (Staiger and Swenson 2011).

The thermal output of the calcine 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). Over 99% of the radioactivity in calcine is due to $^{137}Cs/^{137m}Ba$ and $^{90}Sr/^{90}Y$.

Calcine Treatment—In 2010, DOE issued a Record of Decision (75 FR 137) documenting the selection of hot isostatic pressing (HIP) technology to treat the calcine and provide various possible waste forms suitable for disposal at a facility outside the State of Idaho. In the HIP process (CDP 2012; Russell and Taylor 1998), calcine retrieved from the Calcine Solids Storage Facility is heat-treated (in a fluidized bed) at temperatures up to 600°C to remove moisture and NOx. After heating, the calcine is mixed with amorphous silica, titanium (metal and oxides), 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 7,000 to 15,000 psi and is heated to 1,150°C. At these processing conditions, the calcine is converted to a glass ceramic.

The can itself shrinks around the glass ceramic as the interstitial voids in the calcine mix collapse. A volume reduction of approximately 30% is expected. The currently planned size of the HIP waste form can (before treatment) is 60-in. diameter by 30-in. tall with a volume of approximately 1.36 m^3. With a volume of reduction of 30%, the 4,400 m^3 of calcine would be reduced to approximately 3,080 m^3 and the pressed can would have a volume of approximately 0.95 m^3, yielding approximately 3,200 HIP processed cans at 100% fill. Heat load would be in the range of 4 to 54 W/can. The filled can would weigh approximately 4,500 pounds.

After the HIP process, the compressed cans will be placed in canisters 5.5-ft diameter by 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.

Processing the calcine with the silica and titanium additives is needed to provide a glass ceramic waste form that eliminates the Resource Conservation and Recovery Act (RCRA) hazardous waste characteristics (75 FR 137). 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).

The 2010 Record of Decision retains an option to HIP the calcine *without* the addition of the silica, titanium and calcium sulfate. The HIP process without the additives would generate an alternative waste form that is expected to provide *additional* volume reduction of up to approximately 50% (Hagers 2007). This alternative calcine waste form would include RCRA waste constituents however, and would be acceptable for disposal at a facility that accepts RCRA wastes without additional regulatory/institutional considerations for these constituents.

Calcine Vitrification—Additionally, vitrification has been considered as an option for treatment of the calcine for disposal (DOE 2002). The calcine would be mixed with a glass frit and would be converted to a calcine glass waste form in a Joule-heated melter. For 5,435 m^3 of calcine (assumes remaining SBW is calcine—i.e., includes the additional volume of SBW waste materials covered as direct disposed waste forms in Section 2.2.3.2), approximately 14,115 10-ft long by 2-ft diameter canisters of vitrified calcine would be produced (Lopez and Kimmitt 1998). This would correspond to 11,400 canisters for 4,400 m^3 of existing calcine. Heat load would be in the range of 1.2 to 15.4 W/canister. More recent work has considered the use of a cold-crucible induction melter to produce a glass or a glass ceramic at perhaps higher waste loading and fewer canisters for disposal (Maio 2011; King and Maio 2011). The specific volume reduction that could be achieved is not immediately available.

Direct Disposed Calcine—The last calcine waste form alternative would be to directly dispose of the calcine waste. A specific example of this option was evaluated. In this option, the calcine could be placed in a RH-72B canister. The RH-72B canister is 121 in. long and 26 in. diameter with a 0.25 in. diameter steel wall (Forrester et al. 2002). Internal volume is 0.9 m^3. Thus, the 4,400 m^3 of calcine would yield approximately 4,900 canisters at 100% fill or 5,400 canisters at 90% fill. Heat load would be in the range from a low of 2.4 W/canister (low heat calcine and 90% fill) to a high of 36 W/canister (high heat calcine and 100% fill). Additionally, direct disposal of calcine waste could utilize other packaging configurations, such as small diameter packages for deep borehole disposal.

2.2.3.2 Sodium-Bearing Waste at INL

SBW is defined as mixed hazardous, radioactive waste generated as a by-product of SNF reprocessing at the INTEC (Barnes et al. 2004). Approximately 850,000 gallons of SBW are stored in three underground tanks at the INTEC. The aqueous wastes are composed primarily of decontamination solutions used over the years in support of operations, but include small fractions of first (1%), second (2%), and third (4%) cycle extraction wastes from the fuel reprocessing (70 FR 44598). The acidic wastes are relatively high in sodium and potassium from the decontamination solutions, thus the name "sodium-bearing waste." SBW is high in transuranics, but has significantly less fission product activity than calcine (see Section 2.2.3.1) derived from first-cycle raffinate. The SBW programmatic baseline assumes that SBW is transuranic ("TRU") waste. However, a determination in accordance with DOE O 435.1 will be made to finalize the disposition path for SBW (70 FR 75165).

Fluidized-bed steam reforming has been selected as the treatment method for the SBW. The Integrated Waste Treatment Unit has been constructed east of the INTEC and will begin treating the SBW in 2014 and is planned to be completed by the end of 2014. In the fluidized bed steam reforming process, nitric acid, nitrates, and nitrites are converted to nitrogen gas; organic materials are converted to carbon dioxide and water; and the remaining inorganic species including the radionuclides are converted to a dry, granular/powder carbonate mineral product (ID-DEQ 2013).

The final product from the SBW fluidized bed treatment will be a combination of granular solids and fine powders, which will be placed in canisters for storage, shipment to, and disposal at, the ultimate disposal facility. The canisters are approximately 26 in. (0.66 m) in diameter by 10 ft (3 m) high. They have a capacity of approximately 34 ft^3 (0.96 m^3). The current estimate is that there will be 688 waste canisters from the steam reforming of the SBW. This includes the tank heels that will be flushed from the tanks and treated after the bulk of the solution is treated. The estimate varies depending on the assumptions made for bulk density, canister fill level, and carbon concentration. In addition, there are two smaller tanks containing wastes that will be treated after the SBW. Between 8 and 40 additional canisters will be produced from these wastes. Total heat load in the 688 SBW canisters (based on the radionuclide inventory in Appendix A, Table A-40) is 1,690 W for an average of 2.5 W/canister.

The fluidized-bed stream reforming process is expected to destroy hazardous organic constituents in the SBW. Testing with simulants spiked with chromium, lead, and mercury (Crawford and Jantzen 2007) has shown that the fluidized-bed stream reforming product meets Toxicity Characteristic Leaching Procedure Universal Treatment Standards in 40 CFR Part 268 for Land Disposal Restrictions. However, the fluidized-bed stream reforming product is not expected to meet Land Disposal Restrictions because some of the RCRA metals are not immobilized in the carbonate-based solid matrix (Olson 2006).

2.2.3.3 Cesium and Strontium Capsules at Hanford

There are 1,936 capsules stored on the Hanford Site (1,335 cesium capsules and 601 strontium capsules). The capsules on the Hanford Site contain strontium and cesium extracted from wastes generated from the chemical processing of defense fuel, a fraction of those elements in the form of the isotopes ^{90}Sr, ^{137}Cs and ^{135}Cs (other radioisotopes having decayed away). The cesium and strontium were separated from the wastes in B-Plant to reduce the heat load of the wastes stored in the underground tanks on the Hanford Site. The cesium and strontium capsules were fabricated at the Waste Encapsulation and Storage Facility in the 200 East Area beginning in 1974, with the final strontium capsule prepared in 1985. The cesium and strontium are stored underwater in double-walled capsules at the Waste Encapsulation and Storage Facility. The capsules contain approximately a third of the total radioactivity on the Hanford Site. Characteristics of the capsules are given in Table 2-7. Additional information on composition and thermal output of the capsules can be found in Section A-2.3.3 of Appendix A.

Several studies have identified and evaluated options for the treatment and disposal of the cesium and strontium capsules (Claghorn 1996; DOE 1996a; DOE 2012b). Common to the studies are two options for dispositioning the capsules. In one option, the capsules are left intact and are disposed in some sort of canister (direct disposal waste form). In the alternative, the cesium and strontium are extracted from the capsules and vitrified into a glass waste form at the Hanford WTP (HLW glass waste form). In the vitrification option, the cesium and strontium would be extracted from the capsules and the resulting slurry would be converted to a glass waste form in the HLW melters of the WTP. The Tank Closure and Waste Management Environmental Impact Statement (DOE 2012b) estimates that an additional 340 canisters of HLW glass would be produced solely from the inventory of cesium and strontium capsules. With a total heat content of 308 kW in the capsules, there would be an average of ~905 W per glass canister. Section A-2.2.3 of Appendix A provides additional information regarding secondary wastes generated for this scenario and possible blending options for incorporation of cesium and strontium capsule wastes into HLW glass.

Table 2-7. Characteristics of cesium and strontium capsules (Plys and Miller 2003)

Item	Containment Boundary	Material	Wall Thickness[a] (in.)	Outside Diameter (in.)	Total Length (in.)	Cap Thickness (in.)
CsCl Capsule	Inner	316L Stainless Steel	0.095 0.103 0.136	2.25	19.75	0.4
	Outer	316L Stainless Steel	0.109 0.119 0.136	2.625	20.775	0.4
CsCl Type W Overpack	Single	316L Stainless Steel	0.125	3.25	21.825	0.4
SrF_2 Capsule	Inner	Hastelloy C-276	0.12	2.25	19.75	0.4
	Outer	316L Stainless Steel or Hastelloy C-276	0.12	2.625	20.1	0.4

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

2.2.4 Sodium-Bonded Spent Nuclear Fuel and Associated Waste Forms

The DOE inventory of sodium-bonded SNF includes about 3.4 metric tons of heavy metal (MTHM) driver fuel and 57 MTHM blanket fuel. These fuels, which were generated during the operation of experimental fast-neutron breeder reactors, consist of HEU or depleted uranium alloy fuel surrounded by a layer of sodium metal (for heat transfer) within an alloy cladding. The driver fuel consists of HEU metal (approximately 65% ^{235}U upon discharge) alloyed with 10 wt.% zirconium. The fuel is completely surrounded by a layer of metallic sodium, which is contained within steel cladding. The blanket fuel (i.e., the fertile rod assemblies that surround the fissile core for the purposes of breeding) consists of depleted uranium in steel cladding. About 3.1 MTHM driver fuel and 22.4 MTHM blanket fuel are at the EBR-II facility in Idaho, ~0.3 MTHM driver fuel is from the Hanford Fast Flux Test Facility (FFTF), and about 34 MTHM blanket fuel is from the Detroit Edison Fermi Nuclear Power Plant facility.

These types of fuels represent a significant technical challenge for direct disposal due to the potentially energetic reaction of sodium metal with water to produce hydrogen gas and sodium hydroxide. Distillation of sodium from driver fuel is not effective because sodium becomes incorporated within the pore structure of the fuel. In addition, the fuel interacts with the cladding such that mechanical stripping is not effective. Several options for treating the fuel were considered, including: (1) electrometallurgical treatment (EMT); (2) melt and dilute; (3) distillation of blanket fuel (i.e., melt, drain, evaporate, carbonate—MEDEC process); (4) aqueous processing; and (5) the use of high integrity cans. The MEDEC process would produce a metallic spent fuel waste that could be considered for disposal without further treatment.

After development and demonstration of the EMT procedure (Benedict et al. 1999, which treats both the fuel and cladding), DOE made the decision to treat all sodium-bonded fuel except Fermi-1 blanket fuel using this process (65 FR 56565). The waste forms developed to dispose EMT wastes are to be qualified for disposal as HLW rather than as spent fuel (65 FR 56565; note that to this point only about ~0.8 MTHM of EBR-II driver fuel and ~3.2 MTHM of EBR-II blanket fuel have been processed with EMT). Because of its different physical characteristics, DOE decided to store the Fermi-1 fuel while alternative treatment processes are evaluated, although EMT remains a treatment option for this fuel.

The separation and refining of uranium using the EMT process will generate about 9,900 and 22,450 kg of LEU from treatment of driver and blanket spent fuels, respectively, plus two separate waste streams—high-level radioactive salt waste and metallic waste—that would be immobilized into waste forms for disposal. The recovered LEU will be stored until DOE decides on its future use, and the two waste types will be immobilized in suitable waste forms and disposed of in an HLW repository.

Salt wastes from EMT of sodium-bonded fuels are treated by first occluding the waste within a zeolite matrix and then microencapsulating the zeolite in a borosilicate glass (Goff et al. 1996). The resulting waste form is a glass-bonded sodalite material referred to as the ceramic waste form. An estimated 50,950 kg of ceramic waste will be produced from the combined EBR-II, and Hanford FFTF fuels. The ceramic waste form is being formed as a right cylinder up to 1 m tall with an outer diameter of about 0.5 m. Each 1-m cylinder will weigh about 400 kg (~128 cylinders total) and occupy a volume of about 0.2 m^3. Additional details of the process for producing the ceramic waste form are given in Section A-2.4.1 of Appendix A.

The EMT metallic waste stream will be immobilized by melting it in an induction furnace at about 1,600°C with added zirconium and depleted uranium to produce an alloyed metallic waste form. The metallic waste form products are being cast as ingots sized to fit in the 3 m long HLW canisters that are also to be used to store/dispose the ceramic waste form products. The disk-shaped ingots will be about 14 to 16 in. in diameter and up to 5 in. thick, and will weigh about 12 kg. The first metallic waste form ingot was produced in 2012 (Westphal et al. 2013). It is currently estimated that 5,850 kg of metallic waste form will result from EMT treatment of sodium-bonded spent fuel, yielding approximately 488 12-kg disks. Additional details regarding the process for producing the metallic waste form are given in Section A-2.4.2 of Appendix A.

The salt wastes from EMT of sodium-bonded fuels could also be disposed of directly without further treatment. For the analysis of performance in a salt repository, idealized waste containers and waste canisters were described by Lee et al. (2013). A thin walled stainless steel container with diameter of 25 cm and length of 50.5 cm was used to hold 40 kg of the EMT salt waste form. A larger canister was listed as holding three of these canisters (120 kg EMT salt waste form; 27 cm outer diameter; 155 cm length) and is to be inserted into a cylindrical thicker-walled overpack (with welded lid) to complete the waste package. Lee et al. (2013) estimated EMT salt waste totals of 1,017 kg for the 3.1 MTHM of EBR-II driver fuel, and 699 kg for the 22.4 MTHM of EBR-II blanket fuel, with a total of 15 waste packages to be disposed. Assuming that treatment of the ~0.3 MTHM of FFTF driver fuel and the 34 MTHM of Fermi blanket fuel yields about 100 kg and 1,050 kg of salt, respectively, an additional ten waste packages would be needed to dispose of the full treated sodium-bonded inventory discussed above, for a total of about 25 waste packages for direct disposal of all the EMT salt waste form. Additional details regarding disposal of the salt wastes are given in Section A-2.4.3 of Appendix A.

3 DELINEATING DISPOSAL CONCEPTS AND WASTE GROUPS

For the purposes of this evaluation, an option is defined as disposal of a particular waste group by means of a particular disposal concept. That is, the option includes both the waste group and how it is disposed of. The four disposal concepts considered in this study are discussed below, followed by a description of how the waste types and their associated waste forms described in the previous section are combined into waste groups.

3.1 Disposal Concepts Considered

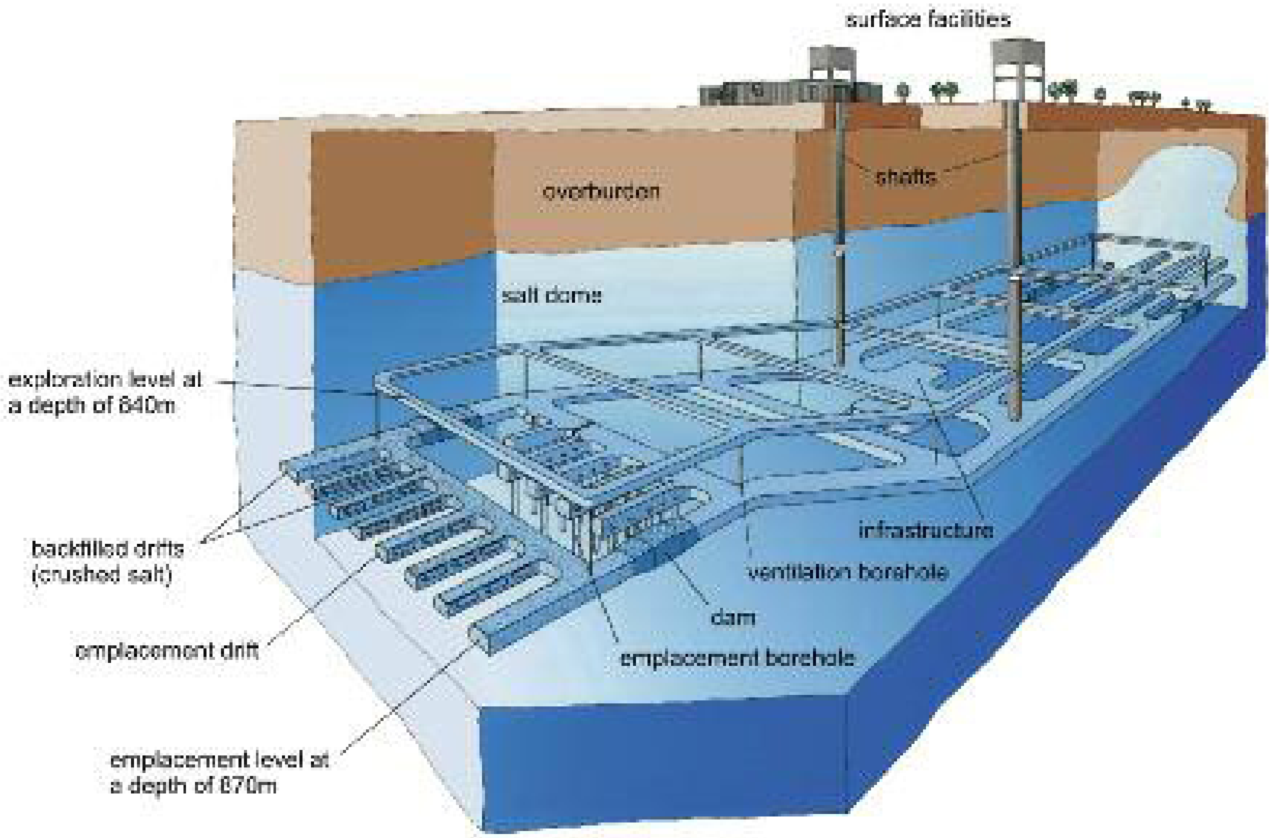
This report considers the four representative disposal concepts selected for further research and development activities by the DOE Office of Nuclear Energy’s Spent fuel Disposition Campaign (Rechard et al. 2011). These four concepts are mined repositories in three geologic media—salt, clay/shale rocks,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

Variants of the four primary concepts are also considered where appropriate. For example, as described by Hardin et al. (2012a), some mined repository concepts can, in principle, be implemented in both open modes (i.e., with active ventilation during the operational period) and closed modes (i.e., with early emplacement of backfill), depending on thermal load management needs. Relevant details of each disposal concept and its variants are discussed in Chapter 5 of this report in the context of the evaluations of specific disposal options.

Other geologic disposal concepts have been proposed and are potentially viable. For example, Canada is currently evaluating a mined repository for intermediate-level radioactive waste in carbonate rocks (NWMO 2011) and the U.S. has evaluated a potential mined repository concept in volcanic tuff (DOE 2008). Although these concepts have unique features that distinguish them from the four selected for consideration in this report, attributes of the four concepts discussed here are representative of a broad range of other disposal concepts.

3.1.1 Mined Repositories in Salt

The primary references for mined repositories in salt come from the U.S. WIPP program (DOE 1996b; DOE 2009) which is an operating repository disposing of defense-related transuranic waste, and the proposed German repository at Gorleben (e.g., BMWi 2008). Figure 3-1 shows a representative design for a salt repository. Emplacement of waste would occur in horizontal tunnels (referred to as “drifts” in mining terminology) or in boreholes drilled from drifts at depths between 500 and 1000 meters below the land surface. As proposed, access to the emplacement areas would be by hoists in vertical shafts. Primary isolation would be provided by the essentially impermeable nature of intact salt. Other attributes of salt relevant to repository design and waste disposal include a relatively high thermal conductivity that allows conductive transfer of heat away from the waste, and the plastic creep behavior of salt under pressure that causes it to flow, closing fractures and allowing for seal systems in access shafts that will compact under lithostatic loads to achieve extremely low permeabilities. Bedded salt, which occurs in horizontal layers of nearly pure sodium chloride originally deposited from shallow salt-saturated sea water, can contain both small quantities of trapped brine and interbedded layers of clays and other evaporite minerals such as anhydrite (calcium sulfate). Domal salt, which has moved from its original bedded form into dome-shaped structures due to plastic flow over geologic time, tends to have less water and fewer impurities and interbeds, but also occurs in more restricted geographic settings.



Source: BMWi 2008, Figure 15.

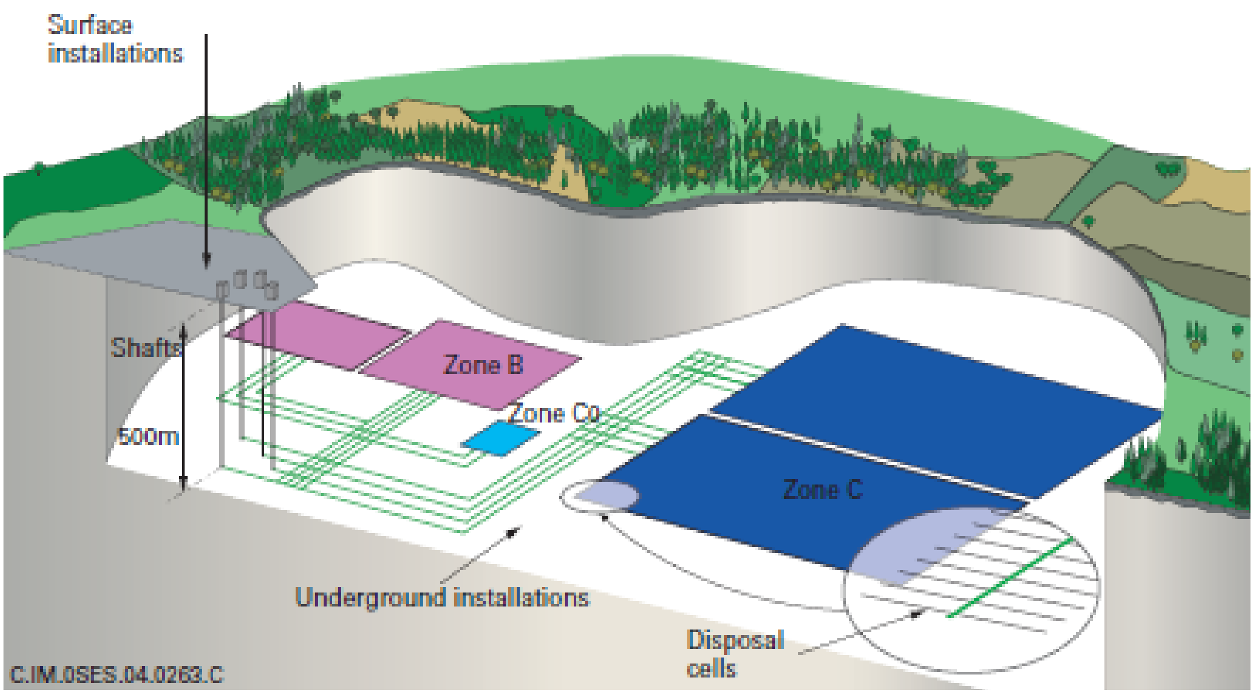
Figure 3-1. Schematic representation of a mined repository in salt

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

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

3.1.2 Mined Repositories in Clay and Shale Rocks

The primary references for mined repositories in clay and shale rocks come from the French, Swiss, and Belgian national programs, each of which is evaluating disposal in argillaceous host rocks (ANDRA 2005a, 2005b; NAGRA 2002; ONDRAF/NIRAS 2011). Figure 3-2 shows a representative design for a mined repository in clay or shale. Emplacement of waste would occur in horizontal holes bored laterally from access drifts at a nominal depth of 500 m below the land surface. As proposed, access to the underground emplacement region would be by hoists in vertical shafts. Isolation would be provided by long-lived waste packages, waste forms that are long-lived in the chemically reducing environment, and by the extremely slow rate of diffusion through the low-permeability host rock. Sorption of radionuclides on clay minerals within backfill and the host rock would effectively prevent long-term releases of all but the most mobile radionuclides, such as ^{129}I and ^{36}Cl, and long-term releases of these species would remain very low.

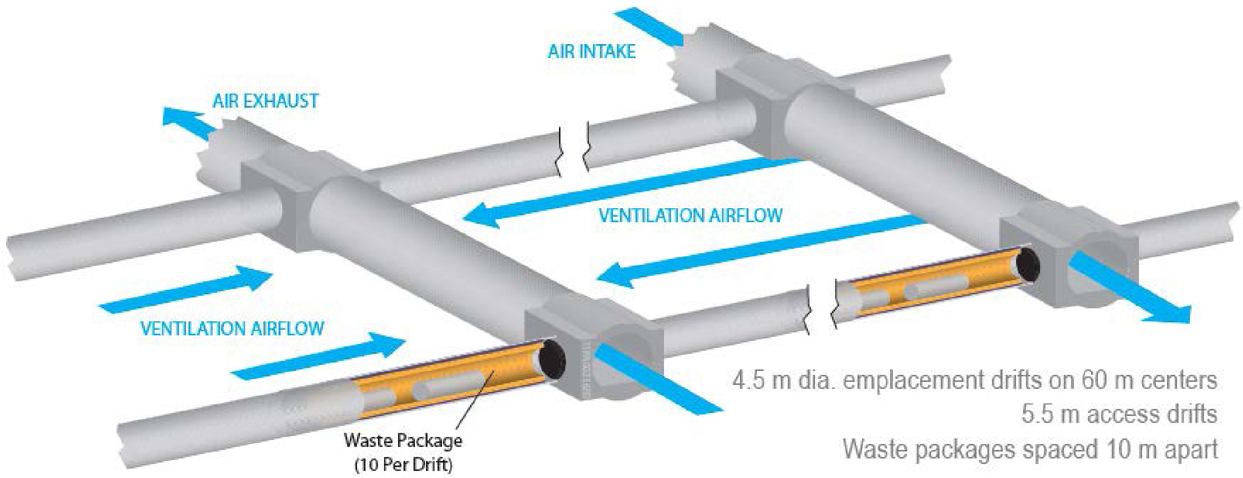


Source: ANDRA 2005b.

Figure 3-2. Schematic representation of a mined repository in argillaceous rock

Argillaceous rocks display a broad range of physical properties from weakly indurated clays capable of plastic flow (e.g., the formation being evaluated for disposal in Belgium), to laminated shales common in many sedimentary basins including in the U.S., to strongly indurated and massive argillites such as that being evaluated for disposal in France. All are characterized by extremely low permeability that will lead to diffusion-dominated release pathways and by an abundance of clay minerals that contribute to radionuclide sorption. All also have lower thermal conductivity than salt, and mined repository concepts in clay and shale rocks must be designed accordingly to accommodate thermal loads. The most widely adopted approach to manage decay heat in clay/shale rocks is to use relatively small waste packages (up to 4 spent fuel assemblies per package) and to space the emplacement drifts relatively far apart. Hardin et al. (2012a) evaluated the potential for increasing the thermal loading capacity of a mined repository in shale by considering an "open-emplacement" design concept in which emplacement drifts remain unbackfilled and open to allow extended ventilation to remove decay heat, as illustrated in Figure 3-3.

Backfilling and sealing of access drifts occurs at the time of repository closure, with the option of leaving the emplacement drifts unbackfilled permanently if the operational constraints so dictate.

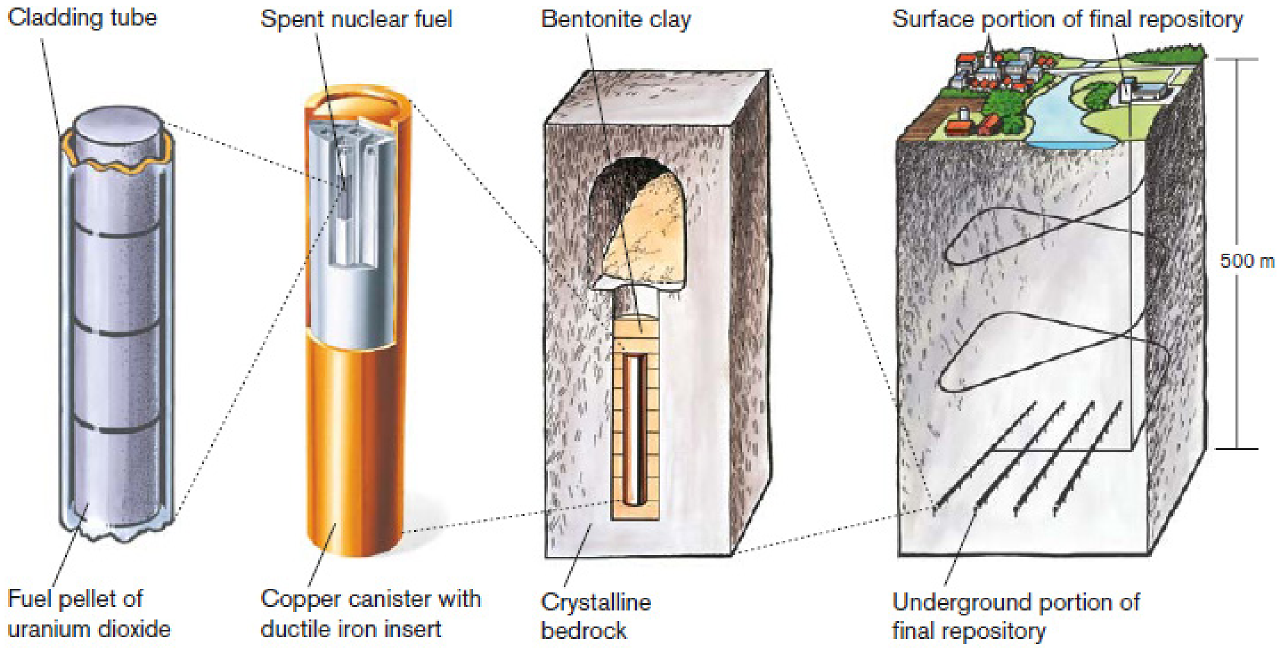


Source: Hardin et al. 2012a, Figure 1.5-3.

Figure 3-3. Schematic of shale unbackfilled open emplacement concept

3.1.3 Mined Repositories in Crystalline Rock

The primary references for mined repositories in crystalline rock come from the Swedish and Finnish programs (SKB 2011; Posiva Oy 2013), which are in the process of seeking licenses to construct and operate facilities for the permanent disposal of SNF. Multiple other nations are also conducting research on mined repositories in crystalline rock, including Canada, Japan, Korea, China, and the Czech Republic. Figure 3-4 shows a representative disposal concept developed for the Swedish program. Wastes (SNF in this example) are emplaced in vertical boreholes drilled in the floor of horizontal drifts at a nominal depth of 500 m below the land surface. Alternative design options call for emplacing waste in horizontal tunnels drilled into the sides of the access drifts. In either case, access to the waste disposal region is by an inclined ramp in this concept, rather than vertical shafts and hoists. Isolation is provided by long-lived corrosion-resistant waste packages (copper in this case, which is thermodynamically stable in the chemically reducing environment), by the durability of the uranium oxide waste form (also stable in reducing conditions) disposed of in the Swedish repository concept, and by the high sorption capability of the bentonite clay buffer that surrounds the waste packages. Other reduced waste forms (e.g., metallic fuels) would be closer to their equilibrium conditions and would corrode more slowly than in oxidizing environments. Still other waste forms (e.g., HLW glass) may not benefit from the reducing environment as much in terms of waste form lifetimes in such a disposal concept, but many radionuclide solubility limits would be very low and substantial performance would be expected based on the waste package lifetime and the bentonite backfill capabilities. Open and interconnected fractures, which can occur in crystalline rocks at these depths, have the potential to provide pathways for advective transport of radionuclides from the repository to the near-surface environment if the near-field barriers are breached, and design concepts therefore call for avoiding emplacement in regions intersected by fractures and for surrounding waste packages with a low-permeability bentonite clay buffer.



Source: SKB 2011, Figure S-1.

Figure 3-4. Schematic representation of a mined repository in crystalline rock

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

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

3.1.4 Deep Borehole Disposal in Crystalline Rock

Deep borehole repositories for permanent isolation of radioactive materials has been proposed and investigated intermittently for decades in the U.S. and other nations (e.g., O'Brien et al. 1979; Halsey et al. 1995; MIT 2003; Nirex 2004; Åhäll 2006; Brady et al. 2009). The earliest proposals for deep borehole disposal considered direct disposal of liquid HLW from reprocessing (National Academy of Sciences Committee on Waste Disposal 1957); subsequent analyses have considered disposal of solid wastes of various types, including glass HLW and surplus weapons-grade plutonium. Published analyses to date have concluded that the overall concept has the potential to offer excellent isolation, but deep borehole disposal of solid wastes has not been implemented in any nation, in part because of the availability of proven mining technologies at the time that national policy decisions were made, and in part because of concerns about the feasibility of retrieving waste from deep boreholes. Advances in drilling technologies over the last several decades (e.g., Beswick 2008) suggest that the construction of deep boreholes should no longer be viewed as a greater technical challenge than deep mines, and that retrieval, if required, should not be viewed *a priori* as unachievable. Retrieval of wastes is likely, however, to remain more difficult from deep boreholes than from most mined repository concepts, and if permanent disposal is not intended, deep boreholes should not be a preferred option.

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

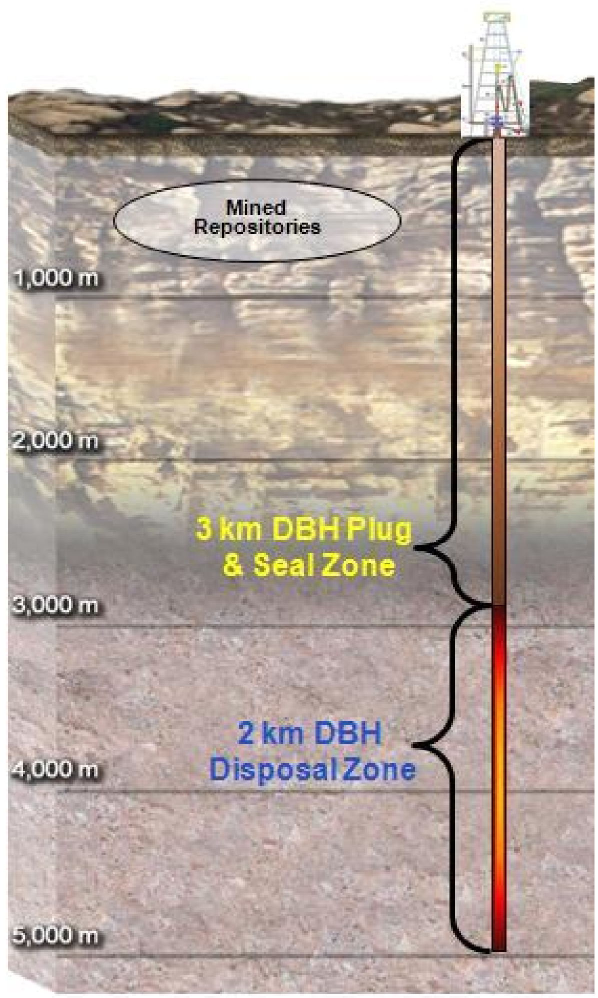


Figure 3-5. Schematic representation of a deep borehole repository

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

Isolation of the waste would be provided by the extremely low permeability of crystalline rocks at these depths (significantly deeper than the depths proposed for mined repositories), and by the long pathway for diffusive transport upward through the borehole seal system. Low permeability of the host rock and the absence of open fractures would need to be verified through borehole testing before waste was emplaced; testing would also confirm the absence of low-salinity or young groundwater. Because of the primary reliance on the geologic barriers and the long seal system, little long-term performance would be required from the waste packages, which could be constructed of standard drilling-industry steel pipe. The strongly reducing environment in the deep portion of the hole would stabilize reduced redox-sensitive species in the waste and would greatly limit the mobility of many radionuclides because of low radionuclide solubility limits under these geochemical conditions. Other reduced waste forms (e.g., metallic) would be closer to their equilibrium conditions and would corrode more slowly than in oxidizing environments. Still other waste forms (e.g., HLW glass) may not benefit from the reducing environment as much in terms of waste form lifetimes in such a disposal concept, but many radionuclide solubility limits would be very low and substantial performance would be expected the bentonite backfill capabilities.

Although recent work in the U.S. has focused on demonstrating that borehole disposal of SNF is feasible in boreholes (Arnold et al. 2011; 2012), the concept is more conducive to any waste that is physically small enough to be emplaced. Smaller waste forms could be emplaced in narrower and therefore less expensive boreholes that can be drilled using existing technology. The modularity of the concept (i.e., boreholes may be constructed one at a time) and the potentially broad geographic areas in which suitable geology may be found (large areas of the U.S. are underlain by stable crystalline rocks within 2 km of the land surface) may make the concept much more attractive for wastes forms other than SNF. Disposal of larger waste forms, including SNF and existing HLW glass canisters nominally 2 ft (0.61 m) in diameter, directly in boreholes drilled using current standard technology is not currently a viable option.

3.1.5 Considerations of General Aspects of Disposal Concept Performance

Three general aspects of disposal concept performance that may ultimately impact decisions about the choices of disposal options are largely independent of the specific details of the waste form, and are therefore discussed here at a generic level, rather than in Section 5 with the evaluations of individual disposal options. The first two topics, human intrusion and retrievability, are strongly dependent on future regulatory expectations and requirements, and are therefore not used as primary metrics for discrimination among disposal options. The third topic, system-level cost, is evaluated qualitatively as a metric in Section 5, but because available cost information for repository design concepts is largely independent of the details of the waste forms other than waste package size (DOE 2013), repository costs are summarized here.

3.1.5.1 Human Intrusion Scenarios

There is general agreement in the U.S. and internationally that the likelihood and consequences of inadvertent human intrusion, for example by drilling, should be considered in assessments of long-term repository performance. However, current U.S. regulations do not provide unambiguous guidance on how such scenarios should be constructed or evaluated. One standard (40 CFR Part 197, applied only to the proposed Yucca Mountain repository) requires consideration of a single drilling event regardless of probability and limits consequences to doses resulting from subsurface transport; the other standard (40 CFR Part 191, applied to the WIPP and potentially any repositories for SNF and HLW other than Yucca Mountain) requires estimates of the probability of future drilling events and includes releases at the land surface in the consequence analysis.

Setting aside uncertain regulatory requirements, three assumptions may help inform consideration of inadvertent human intrusion:

- The probability of future drilling events is independent of the waste form emplaced in the repository
- The probability of future drilling events is correlated with the potential for subsurface resources in the region
- Consequences of drilling events are strongly correlated with the waste form (i.e., some waste forms pose greater risks than others), but may not correlate strongly with the disposal concept (e.g., drilling penetration of a spent fuel assembly in crystalline or clay rocks may have consequences similar to the same event in salt, assuming that individual waste packages are sufficiently isolated from each other in both concepts)

At a generic level, conclusions that are consistent with these assumptions include:

- Repositories in sedimentary rocks may have a greater likelihood of future intrusion than other concepts because of the prevalence of hydrocarbon resources in sedimentary basins. Specifically, the probability of future inadvertent human intrusion may be greater for repositories in salt or clay/shale rocks than for mined repositories in crystalline rock or deep borehole disposal concepts. (Note that the EPA recognized this point in guidance accompanying 40 CFR Part 191 in 1985: "the Agency assumes that the likelihood of such inadvertent and intermittent drilling need not be taken to be greater than 30 boreholes per square kilometer of repository area per 10,000 years for geologic repositories in sedimentary rock formations, or more than 3 boreholes per square kilometer of repository area per 10,000 years for repositories in other geologic formations" (40 CFR Part 191, Appendix C).
- As long as disposal concepts call for isolating waste packages from each other, the consequences of inadvertent human intrusion events may provide little discrimination among disposal options.

Consistent with these assumptions, and in the absence of clear regulatory direction, this report does not use probability or consequence of future inadvertent human intrusion as a specific metric in evaluations of disposal option performance.

3.1.5.2 Retrievability

There is general agreement in the U.S. and internationally that the ability to retrieve wastes from a repository should be considered in the selection of disposal options. There is also general agreement that some disposal concepts are more conducive to future retrieval of the waste than others. For example, retrieval of wastes from deep boreholes after casing has been removed and seals have been emplaced is likely to pose significant technical challenges. Retrieval of wastes from repositories in salt and soft clays will be complicated, although not precluded, by rock creep and the collapse of the emplacement drifts. Retrieval of wastes from mined repositories in hard media such as crystalline rocks or strongly indurated argillites will be relatively simpler than in other concepts.

U.S. regulations do not require that retrieval be simple, straightforward, or inexpensive. In guidance accompanying the 1985 promulgation of the requirement at 40 CFR 191.14(f) that "disposal systems shall be selected so that removal of most of the wastes is not precluded for a reasonable period after disposal," the EPA stated that "the intent of this provision was not to make recovery of waste easy or cheap, but merely possible in case some future discovery or insight made it clear that the wastes need to be relocated" (50 FR 38082). Furthermore, relative ease of retrievability may be an undesirable attribute of a disposal concept if it is achieved at the expense of robust and permanent isolation. As the Nuclear Energy Agency of the international Organisation for Economic Co-operation and Development noted in a 2001 review of topics associated with retrievability of radioactive wastes, "[t]he introduction of provisions for retrievability must not be detrimental to long-term safety. Thus, for example, locating a repository at a depth that is less than optimum from a long- term safety perspective in order to facilitate retrieval is unlikely to be acceptable" (NEA 2001, Section 4.3.1).

Consistent with these observations, this report does not use retrievability as a primary metric in the evaluation of disposal options in Section 5.

3.1.5.3 Repository Design Concept Cost Estimates

The most recent cost estimates available for disposal concepts relevant to those considered in this report come from analyses conducted to support the DOE's January 2013 Nuclear Waste Fund Fee Adequacy Report (DOE 2013, Appendix B). These estimates were developed for specific scenarios assuming single repositories accepting 140,000 metric tons of commercial SNF, and they are not directly applicable to the range of disposal options considered in this report. They also do not address deep borehole disposal concepts. Conclusions from this analysis are qualitatively relevant to this study, however, and provide a foundation for the observations presented relevant to total system cost in Section 5.

Specific to repository costs alone, DOE estimated a total disposal cost for 140,000 MT of commercial SNF in salt ranging from \$24B to \$39B, in clay/shale from \$25B to \$93B, and in crystalline rock from \$61B to \$81B (DOE 2013, Appendix B, Table 2-4). For comparison, repository costs alone for the proposed Yucca Mountain repository were estimated to be \$51B (DOE 2013, Appendix B, Section 4). As noted in the DOE analysis (DOE 2013, Section 2.4), the range in cost estimates "reflects the different strategies for relying on engineered and natural barriers (i.e., natural barriers cost less)." Thus, mined repositories in salt, which call for the least reliance on engineered materials, present the lowest total costs for disposal, and crystalline and clay/shale repositories, which call for a large number of relatively small but robust waste packages, correspond to the highest cost estimates. Cost estimates presented in the 2013 Fee Adequacy Report are based on an assumption that SNF arrives at the repository in sealed packages of the correct size for the facility design, and they do not include costs associated with repackaging and handling prior to transportation to the facility.

Costs for the management of spent fuel prior to its arrival at the repository may be on the same order of magnitude as repository costs, and will be largest for those options that call for repackaging SNF into newly-constructed disposal packages. Nutt et al. (2012) analyzed scenarios that consider a range of options for canister sizes, locations for repackaging facilities, start dates for repository and centralized storage facilities, and throughput rates, and concluded that the total cost of storage, transportation, and repackaging operations for 140,000 metric tons of commercial SNF could be in the range of \$51B to \$135B (Nutt et al. 2012, Table 6-12). Cost estimates for SNF management are affected by multiple factors, with large contributions from storage costs (both at reactor sites and at a centralized storage facility) and repackaging costs, which are larger for smaller package sizes. Limiting the cost considerations to only repackaging required to dispose of commercial SNF in different mined repository options and limiting the throughput considerations to 3000 MTHM/year, which was the rate assumed in the 2013 Fee Adequacy Report, results in a smaller range driven by package size. Estimated repackaging-only costs were \$14.4B for the most expensive option, which was relatively small packages (4 PWR assemblies) in crystalline or clay/shale concepts; \$8.4B for medium-sized packages (12 PWR assemblies) in salt, and \$6.6B for large packages (21 PWR assemblies) in a ventilated and unbackfilled repository concept (Nutt et al. 2012, Table 6-12). The Nutt et al. (2012) study did not evaluate costs associated with SNF management for direct disposal of existing DPCs.

Arnold et al. (2011) estimated a total cost of approximately $27 million for drilling a 5 km borehole with a bottom-hole diameter of 0.43 m, casing it with a steel liner, and preparing it for disposal. Narrower diameter holes would be less expensive to drill. Preliminary estimates suggest that a single borehole with a bottom-hole diameter of 0.216 m would be sufficient to accommodate all of the cesium and strontium capsules currently stored at the Hanford Site, and could cost on the order of $17 million to drill and prepare for disposal. These estimates do not include the costs of packaging the waste, emplacing it in the borehole, and sealing the hole after emplacement operations are complete, and are therefore not directly comparable to the mined repository costs. Arnold et al. (2011) estimated that for a deep borehole disposing of SNF, these costs could raise the total cost per borehole to approximately $40 million. Cost estimates for packaging and emplacing smaller waste forms such as the cesium and strontium capsules are not available.

3.1.6 Considerations for Transportation and Storage

In Section 2 above, generation and packaging of the waste forms are discussed in terms of the characteristics that may affect disposal strategy. In this section, brief overviews are given covering considerations for transportation and for storage of waste forms that may also have some impact on strategies for disposal.

3.1.6.1 Transportation Considerations

While current storage and transportation conditions are safe and secure, much of the infrastructure used to store the Nation's SNF and HLW inventory was not designed for the long storage times that are now anticipated before future consolidated and permanent storage facilities are opened. In addition, the current infrastructure must not only provide safe and secure storage of our waste as it is stored, but it must maintain it in a condition so that the waste can withstand the stresses and strains of repackaging and transportation to a consolidated facility and permanent repository. Lastly, in order to run a successful transportation campaign, there must be a concerted effort to establish and maintain trust with the affected communities. This includes risk communication and emergency response planning and training.

Although the transportation of commercial SNF to consolidated or permanent storage has not yet started, there is experience transporting high-level waste across the country on truck and rail. Since 1957, the Naval Nuclear Propulsion Program has made over 800 rail shipments of SNF to INL with no release of radioactivity and no injury to workers or the public (BRC 2012). WIPP, in Carlsbad, New Mexico, has received over 11,000 truck shipments since opening in 1999 (WIPP n.d.). These programs show the ability to transport different types of waste safely over road and rail.

3.1.6.2 Storage Considerations

For much of the inventory of waste types and waste forms described in Section 2, experience with existing storage processes has provided information that is directly relevant to storage considerations associated with geologic disposal approaches. The DOE is responsible for storing waste generated including DOE SNF, naval SNF, and DOE HLW resulting from the reprocessing of DOE and naval spent fuel, as well as commercial SNF potentially generated by uranium fuel light water reactors (Carter et al. 2012). Additionally, the DOE operated or sponsored a variety of research, test, training, and other experimental reactors both domestically and overseas (Carter et al. 2012). Storage of radioactive waste is discussed in further detail in Section F-1 of Appendix F.

3.2 Characteristics of Waste Forms for Delineating Waste Groups

Given the above waste types and associated waste forms, this section of the report discusses the various characteristics considered in creating waste groups. These characteristics include radionuclide inventory, thermal output, physical dimensions, chemical reactivity, packaging of the waste form, and safeguards and security needed for handling, transporting, and disposing of the waste form. Considering these characteristics allows delineating similarities and differences among waste forms to provide high-level identification of any waste forms to be considered within distinct groups due to outstanding qualities in one or more of these characteristics. In general, any waste forms that are largely similar in these characteristics would be lumped together into a single waste group. The consideration of these characteristics for grouping waste forms is done in the context of the disposal concepts being considered for the disposal options.

3.2.1 Radionuclide Inventory

The radionuclide inventory of each waste form is the essential hazard that is being made safe via deep geologic disposal and, as such defines the primary nature of the hazard of any one particular waste form. Any additional known hazards for the waste forms are considered and discussed within the context of evaluating the particular waste form (see aspects discussed in Chemical Characteristics below). There is a range of variation regarding fission product content. For example, some spent fuels that have burnups higher than 45 MWd/kg uranium have higher fission product contents than spent fuel with lower burnup, as well as having concurrent changes to microstructure. In addition, some waste forms are either highly enriched in fissionable radionuclides to start, or may consist primarily of short-lived, high activity radionuclides. Each of these aspects may provide a basis for defining a unique waste group because of the degree of variation and has been considered in the waste groups listed below.

3.2.2 Thermal Characteristics

The thermal output of the waste form is related to the radionuclide inventory discussed above, but presents an additional consideration in terms of both handling and managing heat within a disposal concept. For waste forms that have a high thermal unit output (e.g., cesium/strontium capsules; FRG glass waste forms) or that have a high total thermal load (e.g., commercial spent fuel), these considerations may be distinct enough to delineate a waste group, or could be convolved with related aspects to define a waste group. Each waste form is considered for its thermal output per waste form unit (e.g., package, capsule, or container) in terms of handling aspects, but is also considered for the total amount of thermal load represented by the total quantity of the waste form. This integrated thermal mass is more relevant to considerations of thermal management impacts to both/either storage and/or disposal of the waste form. It needs to be recognized that such considerations become less relevant with time as the decay of the major heat producing radionuclides (e.g., ^{137}Cs, ^{90}Sr) reduces the thermal output over hundreds of years. As such, these characteristics focus on thermal limits for repository environments and thermal management strategies for storage and disposal systems.

3.2.3 Chemical Characteristics

The bulk chemistry of a waste form is considered from a number of standpoints for delineating waste groups. First, in terms of waste form lifetime for a disposal concept, the bulk chemistry defines the reactivity of the waste form under differing environmental conditions. Reduced oxide waste forms disposed of in a reducing repository concept may have extremely long lifetimes if there is no ready source of oxidants (e.g., spent fuels that have radiolytic oxidants scavenged by hydrogen formed from metal corrosion), whereas some waste forms (e.g., salt waste electro-refined from sodium-bonded fuel) dissolve readily and have very short lifetimes once exposed to water. In the example of spent fuel, waste form lifetime would potentially be the limiting factor for the radionuclide source-term for a reducing geologic repository, whereas in the case of salt waste form, the waste package longevity would be the limiting factor for release of radionuclides. Many waste forms would perhaps have lifetimes intermediate to these two cases, but those with lifetimes less than some hundreds of thousands of years would generally fall into the second group.

A second set of chemical characteristics also related to the bulk composition of the waste form is reactivity and ability to affect the bulk chemistry of solutions in the regions of the source term. For example, borosilicate glass waste imparts a relatively alkaline pH to water that reacts with it, so if it is in a condition where very slow flow exists, fluids may become alkaline in the region around the reacting waste form. Such bulk chemical effects are considered relative to interactions with engineered materials lifetimes in that region, as well as for potential to affect the lifetime of any other proximal waste forms. An extreme example, which relates both to this and the first aspect above, is the pyrophoric nature of the sodium-bonded metallic fuel (in consideration for direct disposal of that waste). Additionally, there are some trace constituents that may be detrimental to engineered barriers or other waste forms that are nearby and their potential effects are considered. Examples include fluoride that may be more abundant and labile in some waste forms compared to the composition of the hosting geologic formation, and as such is accounted for in terms of its relevance to corrosion processes. Such chemical aspects are considered on a case by case basis. Lastly, considerations of hazardous substances within some wastes that are covered by other regulations (e.g., RCRA) are discussed in relevant cases as additional regulatory considerations.

3.2.4 Physical Characteristics

The physical aspects of the waste form, including the overall dimensions and mass of the before-packaging waste form (length, radius, weight, volume) are relevant for both disposal and handling considerations. For existing calcine waste type, the physical traits of the potential waste forms (calcine directly disposed, glass-ceramic from HIP, or glass waste) vary largely and result in a range of possible waste form volumes for disposal. Also in the case of the condition of the waste form (e.g., glass log, intact cladding, fine-grained broken pieces), such aspects are relevant to waste form lifetime once the package was breached. As discussed below, the dimension/scale of the packaged waste form is a consideration for handling during transportation and storage, as well as during placement into a disposal concept.

3.2.5 Packaging

Depending on waste form, the packaging aspects for each may or may not distinguish that waste form as needing consideration as a specific waste group. That is, some packaging may require substantially unique handling considerations, like the DPCs because of size and weight constraints, and possibly thermal constraints also. For example, DPCs are large and heavy and may present particular challenges to hoist technology to lower them to a disposal level of a repository. In fact, such attributes would preclude placing DPCs in a disposal concept such as deep borehole. In addition, for some waste forms (e.g., direct disposed salt wastes from electro-refined sodium-bonded fuel), the packaging may be the primary (longest lived on average) aspect of the waste form that controls the rate of release of the radionuclides. Packaging is considered as a basis for defining a waste group in those cases where at least one aspect of the disposal process (including transportation and storage) is significantly affected.

3.2.6 Safeguards and Security

Within Section 2, some of the particular aspects of safeguards and security are discussed in the context of the waste types and waste forms descriptions. This section provides an overview of considerations for the relative difficulty in implementing safeguards and security for candidate disposal concepts, waste types, and waste forms. In addition to assessing safeguards and security for the candidate disposal concepts, transportation from the originating waste site to the disposal facility, and packaging for transportation and then again for disposal, are also considered.

More specifically, this report summarizes a preliminary study (see Appendix D) to (1) evaluate potential safeguards and security requirements of the regulating agencies of interest (NRC, DOE, and International Atomic Energy Agency); (2) discuss the safeguards and security characteristics of candidate waste types and forms; and (3) discuss possible safeguards and security metrics for the waste groups (see Section 4.6). Additionally, relevant information to assess the safeguards and security implications for two different disposal paths for the cesium and strontium capsules is discussed as an example in Appendix D. A preliminary conclusion following review of regulatory requirements for safeguards and security indicates waste containing special nuclear material such as spent fuel, will require safeguards through the operating life of a repository and may even require minimal safeguards following closure. However, HLW without significant special nuclear material would likely not require safeguards during operation or following closure.

An important concept in the safeguarding of commercial SNF is self-protection. Within the NRC regulations in 10 CFR Part 73, self-protection is attributed to SNF "which is not readily separable from other radioactive material and which has a total external radiation dose rate in excess of 100 rem per hour at a distance of 3 ft from any accessible surface without intervening shielding." Previous studies have shown that the dose rate for typical discharged commercial SNF will fall below the current self-protection limit (100 rem/h at 3 ft) between 70 and 120 years after discharge (Durán et al. 2011). The BWR assembly will be closer to the 70-year time period, while the PWR assemblies will be closer to the 120-year time period. For the 2048 repository start-up target, many BWR fuel assemblies and some PWR assemblies discharged before 1978 (70 years old at 2048) may no longer be self-protecting.

Once the commercial SNF is no longer self-protecting, additional safeguards and security measures may be needed for storage and transportation purposes. This will be especially true with SNF assemblies that contain more than 2 kg of plutonium (classified as safeguard and security Category I—see Appendix D, Table D-1) and require the additional security and national safeguards (for commercial SNF see Table A-4).

Two additional characteristics of a repository that could influence security measures are (1) ease of adversary access to the mined repository or borehole repository disposal concept and (2) length of time that an adversary would have to access a repository. These two characteristics are based upon known construction and operating differences. More specifically, a mined repository in soft media such as salt or clay may require vertical shaft access (in contrast to vehicle access in crystalline media) to avoid tunnel collapse since the operating duration may be many decades. Additionally, access is significantly different between an underground mined repository and borehole repository. Similarly, it is likely a borehole will be sealed relatively shortly after disposal of the waste form, whereas access to tunnels/shafts of an underground repository could be maintained for long periods of time.

3.3 Delineation of Waste Groups

Based on considerations of the aspects described above, each waste group (WG) has been defined to capture salient common characteristics of the waste forms collected within it. The set of waste groups encompasses the entire list of waste forms considered. Some of these waste groups may rely on one or more aspects that are particularly distinct (e.g., WG6 for the sodium-bonded metallic fuel considered for direct disposal) regarding the waste form(s) included in each group. This approach facilitates covering the full range of potential variation in waste forms for evaluation in this study. Table 3-1 lists the waste groups that were created from the 43 waste types and corresponding 50 waste forms. Each waste form in each waste group has a unique identifier, shown in the second column of the table. A letter in the waste form identifier (e.g., 1A, 1B) indicates alternative forms of treatment for the same waste type (e.g., commercial SNF in PBCs and commercial SNF in DPCs). Waste types that have multiple possible waste forms considered for disposal (e.g., commercial SNF, calcine waste) appear multiple times in the table. If a waste form is mutually exclusive from another waste form (e.g., cesium/strontium capsules disposed of as-is and cesium/strontium capsules that have been treated by vitrification for disposal), the waste form(s) with which it is mutually exclusive is identified in the third column. Detailed tabulation of the input information on waste types and waste forms considered in this report can be found in Appendix C.

3.3.1 Waste Group 1—All Commercial SNF Packaged in Purpose-Built Disposal Containers

Waste group 1 (WG1) includes all commercial SNF and considers the spent fuel assemblies (or rods in some variants) as being packaged into purpose-built containers. The purpose-built containers are considered in a general manner as being designed specifically for each disposal concept. This WG1 would entail repackaging of commercial SNF already packaged in existing DPCs.

3.3.2 Waste Group 2—All Commercial SNF Packaged in Dual-Purpose Canisters of Existing Design

Waste group 2 (WG2) includes all commercial SNF and considers the spent fuel assemblies as being packaged in DPCs of existing design (e.g., 32-37 PWR assemblies). Note that WG2 is mutually exclusive to WG1. No repackaging of existing DPCs is included as part of the waste form generation included in this WG.

3.3.3 Waste Group 3—All Vitrified HLW (all types of HLW Glass, existing and projected, canistered)

Waste group 3 (WG3) encompasses all types of HLW glass waste forms described in Section 2. WG3 includes waste form(s):

- SRS HLW glass
- WVDP HLW glass
- Hanford FRG glass
- Hanford projected HLW glass (from tank waste)
- SRS projected HLW glass
- Vitrified calcine (calcine HLW glass)
- Vitrified cesium/strontium capsules (cesium/strontium HLW glass)

For the purposes of this work, the general attributes of these glass waste forms are viewed as being sufficiently similar and distinct from the alternative waste forms that are considered for a number of these waste types that WG3 captures the salient considerations for evaluating their range of potential disposal options.

3.3.4 Waste Group 4—Other Engineered HLW Forms

Waste group 4 (WG4) consists of waste forms that are currently planned to be generated using existing treatment processes in place for a number of various waste types. Because these treatment processes are developed and have been designed to produce waste forms for disposal that are sufficiently distinct from the HLW glass in WG3, these are designated as other engineered HLW forms. WG4 includes waste form(s):

- Two waste forms produced after EMT of sodium-bonded fuel (produces salt and metallic waste streams)
  - glass-bonded sodalite ceramic waste form from salt waste stream
  - alloyed metallic waste form from metallic waste stream
- Calcine HIP-A (with silica, titanium and calcium sulfate additives) glass ceramic
- Calcine HIP-B (without additives) ceramic.

Table 3-1. Waste groups

Waste Group Identifier	Waste Type	Projected Quantity of Waste Type in 2048	Physical Description of Waste Form	Waste Form Identifier	Overlaps With
WG1—Commercial SNF purpose-built containers	Commercial SNF, currently existing and projected through 2048	142,000 MTHM	Purpose-built disposal canister (PBC)[a]	1A	1B
WG2—Commercial SNF in DPCs	Commercial SNF, currently existing and projected through 2048	142,000 MTHM	Dual-purpose canisters (DPCs)[a]	1B	1A
WG3—HLW glass	Savannah River HLW tank waste	4 million gallons of reprocessing waste in tanks	Existing Savannah River HLW Glass	36	
	West Valley HLW tank waste	600,000 gallons of reprocessing waste in tanks	Existing West Valley HLW Glass	37	
	Federal Republic of Germany glass at Hanford	34 canisters	Glass logs containing strontium and cesium[b]	38	
	Hanford tank waste	~54.6 million gallons of reprocessing waste in tanks	Projected glass waste from Hanford	39	
	Savannah River tank waste	28 million gallons of reprocessing HLW in tanks	Projected glass waste from Savannah River	40	
	Calcine waste	4,400 m^3	Calcine waste that has been vitrified[c]	41C	41A, 41B, 41D
	Cesium/strontium capsules at Hanford	1,335 cesium capsules, 601 strontium capsules	Vitrified cesium and strontium from capsules[d]	43B	43A
WG4—Other engineered waste forms	Metallic sodium bonded (EBR-II, INTEC, and FFTF) (DOE group 31)	26 MTHM	Glass-bonded sodalite waste form from EMT INL metal waste form resulting from EMT	32B	32A, 32C
	Metallic sodium bonded fuel (EBR-II, INTEC, and FFTF) (DOE group 31)	26 MTHM	INL metal waste form resulting from EMT	32C	32A, 32B
	Metallic sodium bonded (Fermi-1) (DOE group 31)	34 MTHM	Glass-bonded sodalite waste form from EMT INL metal waste form resulting from EMT	33B	33A, 33C

Table 3-1. Waste groups (cont.)

Waste Group Identifier	Waste Type	Projected Quantity of Waste Type in 2048	Physical Description of Waste Form	Waste Form Identifier	Overlaps With
WG4—Other engineered waste forms (cont.)	Metallic sodium bonded (Fermi-1) (DOE group 31)	34 MTHM	INL metal waste form resulting from EMT	33C	33A, 33B
	Calcine waste	4,400 m^3	Calcine waste treated by HIP, including silica, titanium and calcium sulfate	41A	41B, 41C, 41D
	Calcine waste	4,400 m^3	Calcine waste treated by HIP without silica, titanium and calcium sulfate[c]	41B	41A, 41C, 41D
WG5—Metallic spent fuels	U metal, zirc clad, LEU (DOE group 1, mostly N Reactor)	2,096 MTHM	Tubes	2	
	U metal, nonzirc clad, LEU (DOE group 2)	10 MTHM	Canister of scrap Tube Unknown	3	
	U-zirc (DOE group 3)	7 MTHM	Tube Cylinders Plates Assembly r	4	
	U-Mo (DOE group 4)	4 MTHM	Rod Tube Plates in can	5	
	U-ALx, HEU (DOE group 16)	8 MTHM	Rods array, Tubes, Plates, Pin cluster, Assembly, Element, Canister of scrap, Cylindrical sections, Stacked disks, Multi-pin cluster	17	
	U-Alx, MEU (DOE group 17)	3 MTHM	Assembly Element Plates	18	
	U_3Si_2 (DOE group 18)	7 MTHM	Tubes Multi-pin cluster Assembly Canister of scrap Plates	19	

Table 3-1. Waste groups (cont.)

Waste Group Identifier	Waste Type	Projected Quantity of Waste Type in 2048	Physical Description of Waste Form	Waste Form Identifier	Overlaps With
WG5—Metallic spent fuels (cont.)	Pu/U carbide, non-graphite clad, not sodium bonded (DOE group 21)	<1 MTHM	Canister of scrap Rod Rod hex array	22	
	U-zirc hydride, stainless steel/incoloy clad, HEU (DOE group 27)	<1 MTHM	Rod Element Rod array	28	
	U-zirc hydride, stainless steel/incoloy clad, MEU (DOE group 28)	2 MTHM	Element Canister of scrap Rod	29	
	U-zirc hydride, alum clad, MEU (DOE group 29)	<1 MTHM	Element Rod	30	
	U-zirc hydride, declad (DOE group 30)	<1 MTHM	Declad rod	31	
	Misc. DOE SNF (not previously listed) (DOE group 34)	<1 MTHM	Canister of scrap Tube Rod Unknown Plates	35	
WG6—Sodium-bonded fuels	Metallic sodium-bonded (Fermi-1) (DOE group 31)	34 MTHM	Untreated metallic sodium bonded (Fermi-1)[c]	33A	33B, 33C
	Metallic sodium bonded (EBR-II, INTEC, and FFTF) (DOE group 31)	22 MTHM[e]	Untreated metallic sodium bonded (EBR-II, INTEC, and FFTF)[c]	32A	32B, 32C
WG7—DOE oxide fuels	U oxide, zirc clad, intact, HEU (DOE group 5)	<1 MTHM	Rod Rod array Assembly Plates	6	
	U oxide, zirc clad, intact, MEU (DOE group 6)	2 MTHM	Rod Element Rod array	7	
	U oxide, zirc clad, intact, LEU (DOE group 7)	64 MTHM	Tube Rod Rod array Plates Assembly Unknown	8	

Table 3-1. Waste groups (cont.)

Waste Group Identifier	Waste Type	Projected Quantity of Waste Type in 2048	Physical Description of Waste Form	Waste Form Identifier	Overlaps With
WG7—DOE oxide fuels (cont.)	U oxide, stainless steel/hastelloy clad, intact, HEU (DOE group 8)	<1 MTHM	Tubes Canister of scrap Rod Plates Assembly	9	
	U oxide, stainless steel clad,intact, MEU (DOE group 9)	<1 MTHM	Rod Element	10	
	U oxide, stainless steel clad, intact, LEU (DOE group 10)	<1 MTHM	Tube Rod Rod array Rod hex array	11	
	U oxide, nonalum clad, nonintact or declad, HEU (DOE group 11)	<1 MTHM	Canister of scrap Assembly Tubes Filters Particulate Plate	12	
	U oxide, nonalum clad, nonintact or declad, MEU (DOE group 12)	<1 MTHM	Experiment capsule Canister of scrap Melted fuel	13	
	U oxide, nonalum clad, nonintact or declad, LEU (DOE group 13)	108 MTHM	Canister of scrap Scrap Rod Rod array Debris	14	
	U oxide, alum clad, HEU (DOE group 14)	4 MTHM	Plates Assembly	15	
	U oxide, alum clad, MEU and LEU (DOE group 15)	<1 MTHM	Plates Assembly Tubes	16	
	MOX, zirc clad (DOE group 22)	3 MTHM	Rod Canister of scrap Rod array Element	23	

Table 3-1. Waste groups (cont.)

Waste Group Identifier	Waste Type	Projected Quantity of Waste Type in 2048	Physical Description of Waste Form	Waste Form Identifier	Overlaps With
WG7—DOE oxide fuels (cont.)	MOX, stainless steel clad (DOE group 23)	11 MTHM	Rod Plates Element Canister of scrap Scrap Rod hex array Melted fuel	24	
	MOX, non-stainless steel/nonzirc clad (DOE group 24)	<1 MTHM	Scrap Canister of scrap Unknown	25	
	Th/U oxide, zirc clad (DOE group 25)	43 MTHM	Rod array Rod hex array Canister of scrap	26	
	Th/U oxide, stainless steel clad (DOE group 26)	8 MTHM	Canister of scrap Rod	27	
WG8—salt, granular solids, powder	Metallic sodium bonded (EBR-II, INTEC, and FFTF) (group 31)	26 MTHM	Salt waste from EMT[c]	32C	32A, 32B
	Metallic sodium bonded (Fermi-1) (DOE group 31)	34 MTHM	Salt waste from EMT[c]	33C	33A, 33B
	Calcine waste	4,400 m^3	Calcine waste that is disposed of without further treatment[c]	41D	41A, 41B, 41C
	Sodium-bearing waste at INL	810,000 gallons	Sodium-bearing waste treated by fluidized bed steam reforming (FBSR)	42	
	Cesium/strontium capsules at Hanford	1,335 cesium capsules, 601 strontium capsules	Untreated overpacked cesium/strontium capsules from Hanford[d]	43A	43B
WG9—coated particle spent fuels	Th/U carbide, tristructural isotropic or buffered isotropic coated particles in graphite (DOE group 19)	25 MTHM	Tubes Canister of scrap Carbon coated part	20	
	Th/U carbide, monopyrolytic carbon coated particles in graphite (DOE group 20)	2 MTHM	Element	21	

Table 3-1. Waste groups (cont.)

Waste Group Identifier	Waste Type	Projected Quantity of Waste Type in 2048	Physical Description of Waste Form	Waste Form Identifier	Overlaps With
WG10—naval fuel	Naval (DOE group 32)	65 MTHM[f]	Naval fuel in naval canister	34	

[a] Projected as of 2048.
[b] Existing or projected.
[a] For commercial SNF, both of these waste form disposal pathways are alternative pathways as neither has been finalized, but these represent two endmember pathways to evaluate the technical range of possibilities. Note that under the Standard Contract the DOE is only obligated to accept bare fuel for disposal and that contract holders who have packaged their SNF into DPC will have to sign a contract amendment with the DOE to have their DPC accepted by the DOE.
[b] Contains known amounts of ^{137}Cs and ^{90}Sr; contains an unknown amount of ^{135}Cs.
[c] This waste form disposal pathway is an alternative to the planned disposal pathway for this waste type shown in the table.
[d] For Cs/Sr capsules, both of these waste form disposal pathways are alternative pathways as neither has been finalized.
[e] For this alternative waste form disposal pathway, there is only about 22 MTHM of untreated waste left as about 4 MTHM have already been processed via EMT.
[f] This waste mass represents the expected generation rate that would have been deliverable to Yucca Mountain in its projected 25 years of operation through 2035.

Note that within WG4 there are two waste form variations considered for calcine waste (with additives and without additives). There are a number of differences between these waste forms, for example the volume reduction for calcine that has undergone HIP treatment is greater for the process without additives, that affect the details considered for the disposal option evaluations. However the general aspects for evaluating these two waste forms are sufficiently similar such that this waste group can be used to represent either of the two end-member cases or any intermediate case for these two waste forms of calcine.

3.3.5 Waste Group 5—Metallic and Non-oxide Spent Fuels

Waste group 5 (WG5) consists of a number of DOE spent fuels that (listed below by the DOE spent fuel group number—DGRP), in many cases, have not yet been packaged for disposal. In general the state of the wastes is highly variable and in some cases consists of scrap, tubes, rods, cylinders, and or plates. The range of waste forms included here covers those that contain HEU and MEU (with possible natural uranium or depleted uranium in some of these). These waste forms are not expected to have particularly robust lifetimes in postclosure environments either because of their physical state (small broken pieces) or their reactivity (e.g., metallic waste forms), though the expected lifetimes would be longer in reduced environments versus oxidizing conditions. WG5 includes waste form(s):

- DGRP 1—(N Reactor packaged in MCO, inerted)
- DGRP 2—(10 MTHM U metal, nonzirc clad LEU)
- DGRP 3—(7 MTHM, U-Zirc fuel, HEU)
- DGRP 4—(4 MTHM, U-Mo fuel, HEU)
- DGRP 16—(8 MTHM, U-Alx; HFIR, HEU)
- DGRP 17—(3 MTHM, U-ALx, MEU)
- DGRP 18—(7 MTHM, U_3Si_2, HEU)
- DGRP 21—(<1 MTHM, Pu/U carbide, non-graphite clad, not sodium-bonded, HEU)
- DGRP 27—(<1 MTHM, U-zirc hydride, stainless steel/Incoloy clad, HEU)
- DGRP 28—(2 MTHM, U-zirc hydride, stainless steel/incoloy clad, MEU)
- DGRP 29—(<1 MTHM U-zirc hydride, alum clad, MEU)
- DGRP 30—(<1 MTHM, U-zirc hydride, declad, HEU)
- DGRP 34—(<1 MTHM, miscellaneous DOE SNF not previously listed, HEU)

3.3.6 Waste Group 6—Sodium-Bonded Spent Fuels

Waste group 6 (WG6) is a single DOE spent fuel waste form: metallic sodium-bonded fuel. This includes the spent fuel from the FERMI-1 reactor that is HEU (34 MTHM, Fermi-1, metallic sodium-bonded) as well as other spent fuel alloy compositions. Conditions of these fuels are variable. Direct disposal of these fuels has not been considered in safety assessments for deep geologic disposal previously. Because of the reactive nature of these spent fuel waste forms and the challenges that they present, they have been separated from WG5. WG6 includes waste form(s):

- DGRP 31—(60 MTHM, Metallic sodium bonded)

3.3.7 Waste Group 7—DOE Oxide Spent Fuels

Waste group 7 (WG7) consists of a number of DOE spent fuels that (listed below by the DOE spent fuel group number—DGRP), in many cases, have not yet been packaged for disposal. These waste forms are expected to have comparable to shorter lifetimes relative to commercial SNF in postclosure environments because of their physical state (small broken pieces) including the variety of cladding materials and enrichments. WG7 includes waste form(s) (Note: some HEU in this group, with possible natural uranium or depleted uranium in some of these):

- DGRP 5—(<1 MTHM, U oxide, zirc clad, intact, HEU)
- DGRP 6—(2 MTHM, U oxide, zirc clad, intact, MEU)
- DGRP 7—(64 MTHM, U oxide, zirc clad, intact, LEU)
- DGRP 8—(<1 MTHM, U oxide, stainless steel/hastelloy clad, intact, HEU)
- DGRP 9—(<1 MTHM, U oxide, stainless steel clad, intact, MEU)
- DGRP 10—(<1 MTHM, U oxide, stainless steel clad, intact, LEU)
- DGRP 11—(<1 MTHM, U oxide, non-alum clad, non-intact or declad, HEU)
- DGRP 12—(<1 MTHM, U oxide, non-alum clad, non-intact or declad, MEU)
- DGRP 13—(108 MTHM, U oxide, non-alum clad, non-intact or declad, LEU)
- DGRP 14—(4 MTHM, U oxide, alum clad, HEU)
- DGRP 15—(<1 MTHM, U oxide, alum clad, MEU and LEU)
- DGRP 22—(3 MTHM, MOX, zirc clad)
- DGRP 23—(11 MTHM, MOX, stainless steel clad)
- DGRP 24—(<1 MTHM, MOX, non-stainless steel/nonzirc clad)
- DGRP 25—(43 MTHM, Th/U oxide, zirc clad)
- DGRP 26—(8 MTHM, Th/U oxide, stainless steel clad)

3.3.8 Waste Group 8—Salt, Granular Solids, and Powders

Waste group 8 (WG8) is the set of waste forms that are salt waste forms, granular solids or powdered waste form materials and are expected to have either little waste form lifetimes once exposed to the postclosure environment, or only moderate lifetimes. Many of the waste forms included here are alternative untreated versions of other processed waste forms. WG8 includes waste forms:

- Salt waste form from treated sodium-bonded fuel
- Untreated Calcine waste
- Treated INL sodium-bearing waste
- Untreated cesium/strontium capsules

It should be noted that among these, the cesium/strontium capsules are very hot in short time frames (~300 years), but also contain ^{135}Cs that is long lived. Because they are relatively small in size and total mass, if needed, their thermal mass can be spread out within the disposal system. In addition, these waste forms are characterized by constituents (halides in the salt waste form and cesium/strontium capsules, and small amounts of halogenated organics in the calcine) that would be potentially corrosive to other metallic barriers or other waste forms in a disposal system. This corrosive chemistry may result in a requirement to separate these from other waste packages sufficiently (via backfill or location or both). Additionally because of the generally short-lived nature of these waste forms, colloidal plutonium may be another consideration (e.g., salt waste form).

3.3.9 Waste Group 9—Coated-Particle Spent Fuel

Waste group 9 (WG9) is the set of DOE spent fuels (listed below by the DOE spent fuel group number—DGRP) that are particle fuels that are carbide-based fuel particles with graphite/carbon coatings. For Fort St. Vrain fuel the particles are contained in hexagonal graphite blocks. In some cases, these have not yet been packaged for disposal and are small enough for deep borehole disposal (e.g., Peach Bottom fuel particles). WG9 includes waste form(s):

- DGRP 19—25 MTHM, Th/U carbide, tristructural isotropic or buffered isotropic coated particles in graphite)
- DGRP 20—2 MTHM, Th/U carbide, monopyrolytic carbon coated particles in graphite

This covers the waste forms from used coated particle fuels originating at Fort St. Vrain and Peach Bottom. The Fort St. Vrain fuel with the tristructural isotropic coating is expected to be a robust waste form with long waste form lifetimes and these materials include HEU.

3.3.10 Waste Group 10—Naval Spent Fuel

Waste group 10 (WG10) is the set of naval spent fuel that occurs in large waste packages and is HEU. WG10 includes waste form(s):

- DGRP 32—65 MTHM, HEU

4 CRITERIA AND METRICS DEVELOPMENT

A disposal option consists of a waste group paired with one of the four geologic disposal concepts. Each disposal option is evaluated against several different criteria and metrics that consider factors such as long-term safety of the disposal option, the robustness of (or confidence in) available information, operational issues, technical readiness, system-level cost, secondary waste production, and safeguards and security. The evaluation of each disposal option against the metrics considers the specific interactions between the particular waste group and the particular disposal concept. Metrics for the study are necessarily qualitative, because of the complexity of the problem and the difficulty of quantifying estimates of the behavior of specific waste forms in generic disposal environments. In lieu of quantitative information about specific disposal sites, design concepts, and waste form behavior in such environments, insights are developed based on the full range of information available to the group, including detailed assessments done by previous repository programs in the U.S. and elsewhere in the world.

Evaluations are performed by a subgroup that represents expertise across the ranges of waste types, waste forms, handling, transportation, storage, safeguards and security, and disposal concepts considered in this work. The analyses subgroup performs the evaluation of the disposal options using the criteria and metrics developed below and all the available information described in the report. The evaluation of each disposal option against the metrics results in ratings for each waste group within each disposal concept that reflects the goodness of fit of each pairing.

The analyses subgroup provided an integrated assessment (scoring) for the criteria and metrics for each disposal option considered. This provided a consistent set of understanding by the group for the basis of the evaluation. There were no cases where there was a substantial disagreement within the analyses subgroup on how to score any particular metric for a disposal option, although the option of providing specific discussion of any such disagreement was communicated to the group.

In general, there are three levels of metrics results for each criterion described below, and these reflect high-level assessments of strong, moderate, or weak results for the particular criterion and metric rated. For most criteria and their metrics, the evaluation results are color coded as (with corresponding symbol indicators):

- Green = strong or positive result (✓)
- Yellow = moderate result (O)
- Purple = weak/uncertain result (✸)

Note that there is a fourth category that is unique in that it represents a "no go," or not feasible, result:

- Red = not feasible (✕)

This is meant to be used in very prescriptive cases and was used only for clearly defined incompatibilities that make a particular disposal option untenable. For example, disposal of DPCs in a deep borehole concept is not operationally feasible because DPCs do not fit down such a borehole, and therefore would be rated overall as a "no go" result because of the physical incompatibility of the waste form with the disposal concept. Note also that evaluation of DPCs within a disposal concept that requires hoist emplacement/retrieval entails some large challenges to emplacement, but that situation is not a clear cut case of incompatibility, and as such, would not be classified as a "no go" result. Lastly, in cases where a disposal option was not analyzed, the criterion and metric boxes are grey and marked for indication that such a disposal option was not analyzed (NA).

The criteria and their associated metrics used in the evaluation are summarized in 1 and described in more detail in Appendix F. Note that in practice not all criteria are equally relevant to all components of a disposal option. For example, secondary waste generation and safeguards and security concerns are primarily relevant to the characteristics of the waste group, whereas confidence in expected performance bases and operational feasibility are more strongly influenced by the disposal concept.

Table 4-1. Evaluation criteria and associated metrics

Evaluation Criteria	Metrics	Possible Evaluation Results
Disposal Option Performance	Likely to comply with long-term standards?	Yes No
	Identification of key attributes of disposal system	N/A
Confidence in Expected Performance Bases	Additional engineered barrier system components needed above baseline for each design concept	Few Moderate Many
	Robustness of information bases; simplicity vs. complexity; knowledge gaps	Strong Moderate Weak
Operational Feasibility	Ease in ensuring worker health and safety at all stages	Standard Moderate Difficult
	Special physical considerations at any stages based on physical characteristics	Standard Moderate Difficult
Secondary Waste Generation	Amount of low-level waste (LLW) generated during handling and treatment	Minimal Moderate Large
	Amount of mixed waste generated	Minimal Moderate Large
Technical Readiness	Status of waste form technologies	Ready In-process Conceptualized
	Status of transportation and handling systems	Ready In-process Conceptualized
	Status of disposal technologies	Ready In-process Conceptualized
Safeguards and Security	National security implementation difficulty	Minimal Moderate High
	Radiological dispersion device prevention implementation difficulty	Minimal Moderate High

Additional discussions regarding differential-to-baseline system level costs will be provided for the disposal options to identify the first order aspects that would contribute to cost decreases or increases relative to the baseline. Consistent with the goal of the study to provide technical input to strategic decisions and policy decisions regarding disposal options, this study acknowledges programmatic constraints, including legal, regulatory, and contractual requirements, where applicable, but does not use those to constrain the technical evaluations. For example, the identification of waste types requiring deep geologic isolation is based on consideration of overall risk, rather than on specific U.S. legal and regulatory requirements. Any such programmatic constraints would need to be addressed explicitly prior to implementing any strategy or policy that would be based on technical recommendations that are currently subject to additional legal, regulatory, or contractual considerations.

4.1 Disposal Option Performance

This is the fundamental assessment of the expected behavior of a disposal option in postclosure. If there is an indication that a particular waste group disposed of in a specific disposal concept would not be able to meet expected health and safety requirements then the overall disposal option would have been listed as a red "no go" result. In addition to the overall assessment of safety, consideration in the metrics is also given to the attributes of the particular disposal option that provide the expected postclosure performance and evolution of the system. Key attributes were identified but were not "scored."

4.2 Confidence in Expected Performance Bases

Within this metric, notation of any additional considerations for the engineered barrier system of a repository concept beyond those already accounted for in the basic conceptualization (e.g., specific locations for specific waste forms, additional engineered barriers for specific additional isolation needs) would indicate that a particular disposal option is not directly and simply covered by the baseline concept. This metric also considered the simplicity vs. the complexity of the safety bases, the level of difficulty in generating confidence of the conclusions regarding safety, whether the analyses available are from an existing site vs. a generic study vs. a qualitative assessment, and if there appears to be any clear knowledge gaps for this disposal option.

4.3 Operational Feasibility

The criterion for operational feasibility is composed of two metrics that focus on (1) the health and safety of workers starting with the generation of the waste form from the waste type all the way through the disposal of the waste form and (2) the physical considerations involved with handling, transporting, storing, emplacing and ultimately disposing of the waste forms.

The consideration of worker health and safety is affected by factors such as the radionuclide content of the waste, the amount and type of handling of the waste form, and packaging of the waste form. Options that make it more difficult to meet health and safety standards do not score as well on this metric.

The physical considerations metric is used to assess whether additional challenges exist for a particular disposal option versus others. Options that introduce additional operational challenges compared to other options do not score as well on this metric.

4.4 Secondary Waste Production

For some waste types, different treatment options are available. For example, calcine waste may be treated by HIP into a monolithic ceramic waste form, or it may be vitrified into a glass waste form, or it may be directly disposed of as calcine waste form. In each of these cases, differing amounts of processing are involved with a variety of additionally generated wastes that then need to be dispositioned themselves. The metrics associated with this criterion consider the additional waste generated in a particular disposal option; one metric examines additional LLW generation while the other examines additional mixed waste generation. It should be noted that the metrics focus on the amounts generated outside of the currently planned waste generation that is in place today (e.g., glass waste at Hanford or SRS) and for which standard handling approaches are already developed.

4.5 Technical Readiness (e.g., TRL)

This criterion considers the current state of the technology needed to implement a disposal option from the stage of waste form generation through the closure of a disposal concept. There are three levels identified (which relate roughly to the Technical Readiness Levels assigned to engineered systems), and these levels indicate systems that are ready to be implemented, are in the process of reaching that implementation stage, or are still being developed (i.e., only conceptualized). These readiness levels are assessed as metrics for (1) waste form generation, (2) transportation and handling systems, and (3) the disposal concept technologies.

4.6 Safeguards and Security

The criterion of safeguards and security is based on the relative difficulty in implementing safeguards and security for disposal options. One metric is used to assess the need for additional domestic material control and accounting (MC&A) and international safeguards measures to ensure that there is minimal likelihood of material theft/diversion. This metric is a measure reflecting the fissile content and the related MC&A/safeguards implementation difficulty. The other metric is used to assess the need for additional security measures to ensure that there is minimal likelihood that materials could be sabotaged in-place or be taken (e.g., theft) for use in a radiological dispersive device. This metric is a measure reflecting the dose and dispersal risks, and the related security implementation difficulty.

5 DISPOSAL OPTIONS EVALUATION

The evaluation of the forty disposal options (defined as a single waste group (WG) paired with a single disposal concept) was performed by considering how the particular waste form(s) in the WG and the particular disposal concept performed against the metrics outlined in Section 4. Following the discussion for each metric, a group consensus was reached with regard to the "color" of the rating for that option for a given metric: green, yellow, purple, or red (ratings for each metric are discussed in Section 4 and defined in Appendix F). In every case, the group was able to reach a consensus regarding the rating for each metric (note that there was the option to include any major dissenting opinions where they were encountered). To record a summary of what the group considered and what the final rating was, a spreadsheet was created and used for each WG with disposal concepts as rows and metrics as columns. In general, the evaluation progressed across the spreadsheet (discussing all the metrics for the WG paired with a disposal concept), evaluating all four disposal concepts. The spreadsheets used during the evaluation process include notations regarding the bases for the ratings assigned to the metrics and are in Appendix E. The detailed results for each WG are given in Tables E-1 through E-10, while the detailed results for each disposal concept are given in Tables E-11 through E-14.

A summary of the evaluation results is discussed in this section. This summary was produced by synthesizing the result to the criterion level using the rating of the lowest rated metric within a given criterion to rate the entire criterion. Thus, if an option was rated "green" for a particular metric and "purple" for another metric associated with the same criterion, then the option evaluation summary indicates a "purple" for that criterion.

With respect to total system level cost, factors that would introduce cost differentials for each option were discussed, but no rating was assigned because the evaluation group concluded that there was not enough information to assign a meaningful rating.

In addition, with regard to programmatic and regulatory considerations, factors that might need to be considered in a programmatic or regulatory context were discussed, but no rating was assigned. One of the consent-based siting considerations that was mentioned for all WGs and all disposal concepts is that there are many sites around the country associated with each of the four disposal concepts. Another consideration is that there is a significant amount of operational experience in salt mining, and a significant amount of international repository experience in salt, crystalline rock, and in clay/shale, which might make it easier for a community to have confidence in entering into a consent-based licensing agreement. In contrast, there is no operational experience with respect to deep boreholes, which might present challenges in finding a consenting host community in the absence of further R&D activities.

For the deep borehole disposal concept, it was noted that current disposal regulations did not contemplate deep borehole disposal, and that underground injection control issues may be associated with deep borehole disposal.

It was also noted that for some wastes, there are open regulatory issues with respect to postclosure criticality screening. In addition, some wastes are likely to be subject to RCRA requirements. These concerns are discussed in conjunction with the WGs for which criticality and/or RCRA requirements are an issue.

The results of the evaluation organized by WG are in Section 5.1, while the results organized by disposal concepts are in Section 5.2. Additional cross-cutting characteristics for evaluated waste groups and information concerning waste volumes and masses are discussed in Appendix F.

5.1 Option Evaluations Organized by Waste Group

The following sections present a summary of the evaluation results by WG. That is, each section presents the results of the evaluation for that WG and each of the four disposal concepts.

5.1.1 WG1 Evaluation Results

WG1 includes all commercial SNF that has been placed into containers that have been designed specifically for disposal in the particular disposal concept being considered. For example, when evaluating the disposal of WG1 in deep boreholes, it was assumed that the spent fuel rods were placed in containers that were designed specifically to be disposed of in deep boreholes. Although the group initially agreed to assume that repackaging would occur after spent fuel had been removed from the reactor site, the impacts of this assumption were subsequently noted to be largely limited to institutional and cost issues, and therefore outside the scope of the technical criteria.

5.1.1.1 Evaluation Summary and Highlights for WG1

The results of evaluating WG1 and each of the four disposal concepts are summarized in Table 5-1 (see Table E-1 for details). Some of the pros and cons of disposing of this WG in these four disposal concepts are:

- Pros:
  - The disposal container is designed for the repository environment, which is not the case for WG2. This helps manage the thermal and chemical interactions between the waste form and the disposal concept.
  - The salt disposal concept offers the highest operational flexibility for thermal management because of the favorable thermal properties of salt.
  - For disposal in salt, crystalline rock, and clay/shale, confidence in disposal performance is high because of the amount of world-wide experience in the respective geologic media.
- Cons:
  - Existing DPCs containing commercial SNF will have to be repackaged for all of the disposal concepts, thus generating a significant amount of LLW and introducing additional challenges to ensuring worker health and safety.
  - For disposal in deep boreholes, confidence in disposal performance is not as high because there has been no demonstration of the concept as of yet.
  - For disposal in deep boreholes, only small packages can be disposed of, thus requiring not only repackaging but also consolidation for at least some of the commercial SNF. The necessary rod consolidation technology has not yet been demonstrated at the scale needed for borehole disposal of large quantities of spent fuel. In addition, the large number of small packages and the presence of bare fuel rods (during the repackaging and consolidation process) introduce additional challenges in ensuring worker health and safety, and in mitigating safeguards and security concerns.

Table 5-1. Summary of evaluation results for WG1

Disposal Concept	Disposal Option Performance	Confidence in Expected Performance Bases	Operational Feasibility	Secondary Waste Generation	Technical Readiness	Safeguards and Security
Salt	✓	✓	○	✹	✓	✓
Crystalline	✓	✓	○	✹	✓	✓
Clay/Shale	✓	✓	○	✹	✓	✓
Deep Borehole	✓	○	✹	✹	○	○

Legend:

✓	○	✹	✕
Strong	Moderate	Weak/Uncertain	Not Feasible

5.1.1.2 Cost Differential Considerations for WG1

Regardless of disposal concept, costs are shifted from the subsurface to the surface if spent fuel is repackaged before disposal. That is, costs associated with repackaging the waste (and consolidating as necessary for deep borehole disposal) on the ground surface would be offset by lower costs associated with repository operations in the subsurface because of the benefit of having uniform waste packages. There would also be a cost differential if repackaging were to occur at the reactor.

A salt repository would probably require the fewest packages because of the higher thermal conductivity of salt. A repository in crystalline rock or in clay/shale would require more packages because of the lower thermal conductivities of their geologic media (compared to salt), while deep borehole disposal would require many more packages because of restrictions on the diameter of the package. On the other hand, the packages to be disposed of in deep boreholes might be less expensive than those to be disposed of in the other three disposal concepts because of the lack of reliance on waste package performance and the less stringent design requirements likely to be associated with the smaller packages disposed of in a deep borehole.

5.1.1.3 Institutional/Regulatory Considerations for WG1

It was also noted that disposing of WG1 in deep boreholes would require a significant quantity of land.

5.1.2 WG2 Evaluation Results

WG2 includes waste from commercial SNF in DPCs that are currently in use. The waste would not be repackaged, but the canisters would be fitted with a purpose-built overpack before being emplaced in the repository. The DPCs are very large, massive, and can present challenges with regard to postclosure criticality control because of the amount of fissile material in a single canister.

5.1.2.1 Evaluation Summary and Highlights for WG2

The results of evaluating WG2 and each of the four disposal concepts are summarized in Table 5-2 (see Table E-2 for details). As indicated in Table 5-2, it is not physically possible to dispose of WG2 in deep boreholes because the DPCs are too large to fit in a deep borehole. Some of the pros and cons of disposing of this WG in the remaining three disposal concepts are:

- Pros:
  - The overpacks used would be engineered for the specific disposal environment
  - Repository performance in a salt environment is not dependent on the lifetime of the waste package, so the fact that DPCs were not designed for disposal is not as important in salt as it is in the other environments.
  - In contrast to WG1, the commercial SNF would not be repackaged prior to disposal, thus reducing the amount of LLW or mixed waste produced and reducing the effort needed to ensure worker health and safety.
  - The very large size of the DPC and the lack of repackaging minimize safeguards and security concerns.
- Cons:
  - There is a dearth of modeling experience and analysis for disposal of commercial SNF in DPCs in a salt repository, a crystalline repository, and in a clay/shale repository.
  - For a crystalline repository, the lifetime of the overpack would likely be important to repository performance, but such performance is yet unproven.
  - For all repository environments, components may need to be added to the engineered barrier system to address criticality control.
  - For both a salt and a clay/shale repository, it could be a challenge to keep the large shafts or ramps open during the operational period, and to seal them upon repository closure. A clay/shale repository might need to remain open longer than a salt repository to allow for ventilation.
  - For a salt repository, knowledge gaps exist regarding the behavior of salt under the high thermal loads associated with DPCs.
  - For a crystalline or clay/shale repository, the high thermal load could complicate reliance on bentonite as a buffer.
  - For both the salt and clay/shale repositories, there could be challenges associated with ensuring waste could be retrieved during the preclosure period.
  - For all repository environments, conveyance methods for maneuvering the large DPCs (e.g., hoists) in the repository have not yet been developed. Development of such methods may pose a significant challenge because of the mass and size of the DPCs.

Table 5-2. Summary of evaluation results for WG2

Disposal Concept	Disposal Option Performance	Confidence in Expected Performance Bases	Operational Feasibility	Secondary Waste Generation	Technical Readiness	Safeguards and Security
Salt	✓	○	✸	✓	○	✓
Crystalline	✓	✸	○	✓	○	✓
Clay/Shale	✓	○	✸	✓	○	✓
Deep Borehole	NA	NA	✕	NA	NA	NA

Legend:					
	✓ Strong	○ Moderate	✸ Weak/Uncertain	✕ Not Feasible	NA Not analyzed

5.1.2.2 Cost Differential Considerations for WG2

Costs are shifted from the surface to the subsurface (compared to WG1). That is, costs are reduced (compared to WG1) on the ground surface because the commercial SNF is not repackaged, but this reduction in cost is coupled with increased costs (compared to WG1) in subsurface operations associated with the equipment needed to handle very large and massive packages, and additional measures that would need to be taken for thermal management and perhaps criticality control.

5.1.2.3 Institutional/Regulatory Considerations for WG2

Disposal of DPCs may reduce the number of potentially suitable sites if the option eliminates some geologic media and/or places greater demands on the performance of the geologic barrier system. In addition, there may be regulatory issues associated with criticality screening for postclosure performance assessment for this waste form.

5.1.3 WG3 Evaluation Results

WG3 includes HLW glass, both existing and projected. That is, it includes the following waste forms:

- SRS HLW glass
- West Valley HLW glass
- FRG Hanford glass
- Hanford projected HLW glass from the tanks
- SRS projected HLW glass
- Calcine waste that has been vitrified
- Cesium/strontium waste currently in capsules that has been vitrified

Both the calcine waste and the cesium/strontium capsules have alternative waste forms that were evaluated as a part of other WGs. The calcine waste could be disposed of as-is (WG8) or it could be treated by HIP prior to disposal (WG4). The cesium/strontium capsules could also be disposed of as-is (WG8).

5.1.3.1 Evaluation Summary and Highlights for WG3

The results of evaluating WG3 and each of the four disposal concepts are summarized in Table 5-3 (see Table E-3 for details). Note that for deep borehole disposal (shown in the last row of the table), the split cells indicate that only some of the HLW glass in WG3 can be disposed of in boreholes drilled using current standard technology. It was assumed for the purposes of this evaluation that existing HLW glass (SRS, WVDP, FRG) could not be disposed of in deep boreholes. These existing waste forms are too large to fit in a deep borehole and repackaging these wastes is not feasible. Thus, the evaluation focused on those HLW glass waste forms that have not yet been produced (projected from Hanford tank waste, projected from SRS, calcine waste that could be vitrified, and cesium/strontium capsule waste that could be vitrified). Some of the pros and cons of disposing of this WG in these four disposal concepts are:

- Pros:
  - Thermal management presents fewer challenges than does disposal of commercial SNF because the waste is not as hot as commercial SNF.
  - For disposal in a salt environment, a crystalline environment, and a clay/shale environment, confidence in the information bases was thought to be good because of the quantity of international experience in these disposal concepts.
  - For disposal in a salt environment, a crystalline environment, and a clay/shale environment, no issues related to operational feasibility were identified.
  - Regardless of the disposal concept, the LLW, mixed waste, and waste incidental to reprocessing that would be produced has already been accounted for; that is, no additional LLW, mixed waste, or waste incidental to reprocessing would be produced.
  - Transportation casks have not been developed for these wastes, but no significant technical challenges are expected.
  - The size and form of the waste reduces the challenges associated with safeguards and security.
- Cons:
  - For disposal in deep boreholes, confidence was not as high as the other three disposal concepts because of the lack of a demonstration of the concept.
  - For disposal in deep boreholes, about four times as many canisters would have to be created and handled, compared to the other three disposal concepts, because of the smaller-diameter canisters that would be necessary.
  - For disposal in deep boreholes, treatment plants that are already being built would have to be redesigned to accommodate smaller-diameter glass logs.
  - Regardless of the disposal concept, technology development is required for the WTP at the Hanford Site.

Table 5-3. Summary of evaluation results for WG3

Disposal Concept	Disposal Option Performance	Confidence in Expected Performance Bases	Operational Feasibility	Secondary Waste Generation	Technical Readiness	Safeguards and Security
Salt	✓	✓	✓	✓	○	✓
Crystalline	✓	✓	✓	✓	○	✓
Clay/Shale	✓	✓	✓	✓	○	✓
Deep Borehole	✓ / NA	○ / NA	○ / ✕	✓ / NA	✸ / NA	✓ / NA

Note: Split scores indicate that size constraints preclude disposal of some, but not all, waste forms in this group.

Legend:	✓ Strong	○ Moderate	✸ Weak/Uncertain	✕ Not Feasible	NA Not analyzed

5.1.3.2 Cost Differential Considerations for WG3

As discussed in Section 2 and Appendix C, some fraction of Hanford tank waste may be managed as transuranic waste. Managing these wastes as transuranic, rather than HLW, could result in lower costs. In addition, vitrifying the calcine waste and/or the cesium/strontium capsules could also result in higher costs compared to directly disposing of these waste forms (see WG8).

Disposal of future vitrified waste forms in deep boreholes would be associated with higher costs also because of the need to re-design vitrification facilities to produce smaller-diameter glass logs.

5.1.3.3 Institutional/Regulatory Considerations for WG3

Some of the wastes in WG3 may be subject to RCRA.

5.1.4 WG4 Evaluation Results

WG4 consists of engineered waste forms, other than glass. Specifically, it includes:

- Glass-bonded sodalite of the salt waste from electrometallurgical treatment (EMT) of sodium-bonded fuel (projected)
- Metal waste form resulting from EMT of sodium-bonded fuel (both existing and projected)
- Calcine waste that has been treated by HIP, with additives (projected)
- Calcine waste that has been treated by HIP, without additives (projected)

A small quantity of the metal waste form resulting from EMT of sodium-bonded fuel already exists. Additional glass bonded sodalite and metal waste is projected from treatment of total of 26 MTHM sodium-bonded fuels from EBR-II, INTEC, and FFTF, and more is projected (if the 34 MTHM of sodium-bonded fuel from Fermi-1were to be subject to EMT). Disposal of sodium-bonded fuel as-is is covered in Section 5.1.6 for WG6. In addition, in this WG the two waste forms of calcine waste are alternatives that refer to the same physical waste; the two waste forms represent two different treatment options for this waste type. Direct disposal of calcine waste (i.e., as-is) is addressed in Section 5.1.8 covering WG8, while disposal of calcine waste that has been vitrified is addressed in Section 5.1.3 covering WG3.

5.1.4.1 Evaluation Summary and Highlights for WG4

The results of evaluating WG4 and the four disposal concepts are summarized in Table 5-4 (see Table E-4 for details). Some of the pros and cons of disposing of this WG in these four disposal concepts are:

- Pros:
  - The waste is not as hot as commercial SNF, so the wastes can be packed more densely and there are fewer issues with thermal management.
  - The size of the waste forms can be optimized for deep borehole disposal because they have not yet been produced (except for one metal ingot)
  - No additional EBS considerations were identified.
  - For disposal in salt, clay/shale, or in deep boreholes, because there is less reliance on waste form performance, it is easier to have confidence in these options.
  - The part of the treatment process for sodium-bonded fuel that produces the metal waste form has already been demonstrated.
  - Transportation casks have not been developed, but no significant technical challenges are expected.
  - Fissile content of the waste should be no greater than SNF.
  - Dose and dispersion concerns are no greater than those associated with SNF.
- Cons:
  - Galvanic coupling between the metal waste form and its packaging needs to be considered in disposal concepts that rely on waste package performance.
  - The deep borehole disposal concept has not yet been designed or demonstrated, making it a challenge to build confidence in the disposal option without further R&D.
  - Treatment of the sodium-bonded fuel and the calcine waste will generate more LLW and mixed waste than would be generated without treatment prior to disposal.
  - The part of the treatment process for sodium-bonded fuel that produces glass-bonded sodalite has not been demonstrated at scale.
  - The HIP technology is in development and has not been fully demonstrated.

5.1.4.2 Cost Differential Considerations for WG4

There is a cost differential between the treatment of sodium-bonded fuel and the treatment of calcine waste compared to direct disposal of these wastes (as-is, if possible) or other forms of treatment.

5.1.4.3 Institutional/Regulatory Considerations for WG4

Some of the wastes in WG4 may be subject to RCRA. Specifically, the waste form processed by HIP without additives will still be characteristically hazardous, while the waste form processed by HIP with additives may not be characteristically hazardous.

Table 5-4. Summary of evaluation results for WG4

Disposal Concept	Disposal Option Performance	Confidence in Expected Performance Bases	Operational Feasibility	Secondary Waste Generation	Technical Readiness	Safeguards and Security
Salt	✓	✓	✓	○	○	✓
Crystalline	✓	○	✓	○	○	✓
Clay/Shale	✓	✓	✓	○	○	✓
Deep Borehole	✓	○	✓	○	○	✓

Legend:	✓ Strong	○ Moderate	✹ Weak/Uncertain	✕ Not Feasible

5.1.5 WG5 Evaluation Results

WG5 consists of metal and non-oxide spent fuels as shown in Table 3-1. As shown in Tables C-3 and C-4, the enrichment, cladding, cladding condition, fuel compound, fuel matrix, configuration, and size of these fuels vary significantly.

5.1.5.1 Evaluation Summary and Highlights for WG5

The results of evaluating WG5 and the four disposal concepts are summarized in Table 5-5 (see Table E-5 for details). The split cells in the last row of Table 5-5 indicate that only some of the waste in WG5 can be disposed of in deep boreholes. Specifically, N Reactor fuel in MCOs (DOE Group 1) and some of the spent fuel in DOE Groups 2 and 16 is already packaged in containers too large to fit in a borehole drilled using current standard technology. It was assumed that the N Reactor fuel would not be repackaged into a waste package small enough to fit in a deep borehole. Some of the pros and cons of disposing of WG5 in these four disposal concepts are:

- Pros:
  - Disposal in a salt environment reduces criticality concerns (compared to other disposal concepts) because of the relative lack of water and the high cross-section of chlorine for capture of thermal neutrons.
  - Some smaller waste types would be candidates for disposal in deep boreholes.
  - For disposal in salt, clay/shale, or deep boreholes, it is easier to have confidence in these options because there is less reliance on waste form performance.
  - No additional concerns were identified with respect to ensuring worker health and safety.
  - For the waste in WG5 that can be disposed of in deep boreholes, transportation of waste would likely not present challenges. This is because the waste in WG5 that cannot be disposed of in deep boreholes is the waste that could present challenges with respect to transportation because of the characteristics of the waste itself.
  - Little LLW or mixed waste would be generated by packaging the unpackaged fuel.
  - Metal fuels are less dispersible than oxide fuels, reducing some safeguards and security concerns.

- Cons:
  - Galvanic coupling between the metal fuel waste form and its packaging needs to be considered in disposal concepts that rely on waste package performance.
  - The deep borehole disposal concept has not yet been designed or demonstrated, making it a challenge to build confidence in the disposal option without further R&D.
  - For disposal in salt, crystalline rock, or clay/shale, transportation of waste could present challenges because of some characteristics of some of the spent fuels. The waste in WG5 that cannot be disposed of in deep boreholes is the waste that could present challenges with respect to transportation because of the characteristics of the waste itself.
  - Neutron absorbers in the waste form need to be evaluated for disposal in salt, crystalline rock, or clay/shale.
  - The fissile content of some of the wastes in WG5 is high, requiring increased MC&A safeguards.

Table 5-5. Summary of evaluation results for WG5

Disposal Concept	Disposal Option Performance	Confidence in Expected Performance Bases	Operational Feasibility	Secondary Waste Generation	Technical Readiness	Safeguards and Security
Salt	✓	✓	○	✓	○	✓
Crystalline	✓	○	○	✓	○	✓
Clay/Shale	✓	✓	○	✓	○	✓
Deep Borehole	✓ / NA	○ / NA	✓ / ✗	✓ / NA	○ / NA	✓ / NA

Note: Split scores indicate that size constraints preclude disposal of some, but not all, waste forms in this group.

Legend:	✓ Strong	○ Moderate	✹ Weak/Uncertain	✗ Not Feasible	NA Not analyzed

5.1.5.2 Cost Differential Considerations for WG5

If the sodium-bonded fuel (WG6) were to be treated such that the sodium was removed but the fuel was left largely intact, the fuel would then be in this group. Treatment in such a manner would introduce additional costs.

5.1.5.3 Institutional/Regulatory Considerations for WG5

Criticality analysis has not yet been performed for all the wastes in this WG.

5.1.6 WG6 Evaluation Results

WG6 consists of untreated sodium-bonded metal fuels (driver and blanket). This waste could also be treated by EMT, creating either a salt waste (WG8) and a metal waste (WG4) or a glass-bonded sodalite waste (WG4) and a metal waste (WG4). Those waste forms all have viable disposal options.

Sodium-bonded fuel reacts vigorously with water and is variably enriched (0.1% to 93% enrichment). This waste is the only waste type considered in the present evaluation that was not included in the License Application for the proposed Yucca Mountain repository (DOE 2008). Because direct disposal of this waste form has not yet been studied, the evaluation group decided that there was not enough information to determine whether it could be disposed of safely or to perform a meaningful evaluation of this waste form. Therefore, it was not evaluated explicitly.

However, in the course of evaluating other WGs, some observations that are also relevant to WG6 were made. First, wastes with a high fissile content will require increased MC&A safeguards. Second, metal fuels are less dispersible than oxide fuels. Finally, if this waste were to be treated such that the fuel was left intact but the sodium removed (e.g., using MEDEC), then this waste would be considered to be a metal waste form included in WG5.

5.1.7 WG7 Evaluation Results

WG7 includes DOE oxide fuels as shown in Table 3-1. As shown in this Tables C-3 and C-4, the enrichment, cladding, cladding condition, fuel compound, fuel matrix, configuration, and size of these fuels varies significantly.

5.1.7.1 Evaluation Summary and Highlights for WG7

The results of evaluating WG7 and the four disposal concepts are summarized in Table 5-6 (see Table E-7 for details). The split cells in the last row of Table 5-6 indicate that only some of the fuels in this WG are potential candidates to be disposed of in deep boreholes because some of the fuels in this WG are too large to fit in a borehole drilled using current standard technology. Some of the pros and cons of disposing of WG7 in the four disposal concepts are:

- Pros:
  - WG7 has properties similar to those of WG1, but the packages would be smaller and cooler, and thus easier to manage for disposal.
  - As most of the wastes in the WG7 have not yet been packaged for disposal, the disposal containers could be designed for the repository environment.
  - For disposal in salt, crystalline rock, and clay/shale, confidence in disposal performance is high because of the amount of world-wide experience in the respective geologic media.
  - In a crystalline or a clay/shale repository, retrieval of waste could be relatively easy.
- Cons:
  - For deep borehole disposal, only small packages could be disposed of.
  - For deep borehole disposal, some wastes might have to be repackaged or consolidated, increasing the amount of LLW produced, presenting challenges in ensuring worker health and safety, and possibly requiring technologies not yet developed.
  - The deep borehole disposal concept has not yet been designed or demonstrated, making it a challenge to build confidence in the disposal option without further R&D.
  - For some wastes, the fuel has high fissile content, requiring increased MC&A safeguards.
  - Oxide fuel is more dispersible than metal fuel or glass.
  - For deep borehole disposal, the repackaging of waste and the small waste packages increase safeguards and security concerns.

Table 5-6. Summary of evaluation results for WG7

Disposal Concept	Disposal Option Performance	Confidence in Expected Performance Bases	Operational Feasibility	Secondary Waste Generation	Technical Readiness	Safeguards and Security
Salt	✓	✓	✓	✓	✓	✓
Crystalline	✓	✓	✓	✓	✓	✓
Clay/Shale	✓	✓	✓	✓	✓	✓
Deep Borehole	✓ / NA	○ / NA	✸ / ✕	○ / NA	○ / NA	○ / NA

Note: Split scores indicate that size constraints preclude disposal of some, but not all, waste forms in this group.

Legend:					
	✓ Strong	○ Moderate	✸ Weak/Uncertain	✕ Not Feasible	NA Not analyzed

5.1.7.2 Cost Differential Considerations for WG7

No cost differential considerations unique to WG7 were identified.

5.1.7.3 Institutional/Regulatory Considerations for WG7

No institutional or regulatory considerations unique to WG7 were identified.

5.1.8 WG8 Evaluation Results

WG8 includes waste forms that are salt, granular solids, and powders. This WG includes the following waste forms:

- Salt waste from EMT of sodium-bonded fuels
- Calcine waste that is disposed of without further treatment
- Sodium-bearing waste treated by fluidized bed steam reforming
- Cesium/strontium capsules disposed of without further treatment

Some of the salt waste from EMT of sodium-bonded fuels already exists, and some is projected to be produced. This salt waste would be produced if the Fermi-1 sodium-bonded fuel (WG6) is treated by EMT and the resulting salt waste is not turned into a glass-bonded sodalite waste form (WG4). Calcine waste can also be treated by HIP (WG4) or by vitrification (WG3). The cesium/strontium capsules can also be treated by vitrification (WG3).

The wastes in WG8 are characterized by having short waste form lifetimes once contacted by water. That is, after disposal, once the waste package has been breached and water has entered the package, the salt or granular solid or powder will dissolve readily in the water, compared to other waste forms. This means that isolation of the waste from humans and the environment will, in general, be a function of the properties of the disposal medium and the waste packages, rather than the properties of the waste form. In addition, these wastes are granular, contains respirable fines, and easily dispersible.

Some of the wastes inWG8 are also characterized by being corrosive (e.g., high halide content). Because of the presence of corrosive chemical components, it might be necessary in some disposal concepts to separate these wastes from wastes in other WGs. The separation distance need not be great, just sufficient to keep the corrosive chemicals from negatively affecting the performance of other waste forms/packages disposed of in the same repository.

As shown in Table C-2, the cesium/strontium capsules have the highest estimated thermal output (in 2048) on a volume basis of any of the waste forms considered in this study, assuming SNF is allowed to cool for at least five years prior to disposal.

5.1.8.1 Evaluation Summary and Highlights for WG8

The results of evaluating WG8 and the four disposal concepts are summarized in Table 5-7 (see Table E-8 for details). Some of the pros and cons of disposing of this WG in these four disposal concepts are:

- Pros:
  - Both a salt repository and a clay/shale repository have limited far-field transport in the respective geologic media, which is of particular importance for the short-lived wastes in this WG.
  - These wastes would fit easily in deep boreholes.
  - Deep boreholes would make it easy to segregate halide-bearing wastes from other wastes, or preclude the need for segregation.
  - Would be relatively easy to spread out the cesium/strontium capsules to avoid thermal issues, assuming they were packaged accordingly.
  - For disposal in salt, clay/shale, and deep boreholes, because there is less reliance on waste form performance, it is easier to have confidence in the option.
  - By disposing of the salt, calcine, and cesium/strontium capsules directly (i.e., without further treatment), potential worker doses are avoided and no additional LLW or mixed waste is generated.
  - There is little to no fissile content in much of this waste, minimizing criticality concerns.
- Cons:
  - The salt waste from EMT of sodium-bonded fuel contains plutonium, which could be transported in colloidal form through the fracture network in a crystalline repository.
  - Because of the corrosive chemical components in some of these wastes, these wastes might have to be isolated from waste packages relied on to provide isolation for a long period of time. The distances may not need to be as great in clay/shale as in a crystalline repository because of the limited far-field transport in a clay/shale repository.
  - More information is needed with regard to transport in a crystalline media because of the possibility of colloidal transport and the short waste form lifetime.
  - Some of these wastes contain respirable material, introducing challenges during transport and disposal (in the event of a low-probability accident).
  - For deep borehole disposal, surface storage might be needed, introducing security concerns because of the small packages containing respirable material.
  - This waste presents security challenges because the waste consists of highly dispersible powders, salts, and granular solids, potentially in small packages.

Table 5-7. Summary of evaluation results for WG8

Disposal Concept	Disposal Option Performance	Confidence in Expected Performance Bases	Operational Feasibility	Secondary Waste Generation	Technical Readiness	Safeguards and Security
Salt	✓	✓	○	✓	✓	○
Crystalline	✓	○	○	✓	✓	○
Clay/Shale	✓	✓	○	✓	✓	○
Deep Borehole	✓	○	○	✓	○	○

Legend:				
	✓ Strong	○ Moderate	✸ Weak/Uncertain	✕ Not Feasible

5.1.8.2 Cost Differential Considerations for WG8

Not treating the salt, the calcine, and the cesium/strontium capsules lowers costs, but these lower costs are coupled with additional costs to transport and manage packages of respirable and easily dispersible material. For deep borehole disposal, it is likely that there would have to be on-site storage of waste or a just-in-time type of management, introducing additional costs to secure small packages of respirable and easily dispersible material.

5.1.8.3 Institutional/Regulatory Considerations for WG8

No institutional considerations unique to WG8 were identified. Regulatory considerations for all disposal concepts include: (1) untreated calcine is both a listed waste and a characteristically hazardous waste, and (2) there may be regulatory issues with transporting respirable material.

5.1.9 WG9 Evaluation Results

WG9 includes coated-particle spent fuels, namely tristructural isotropic or buffered isotropic coated and monopyrolytic carbon fuels from Fort St. Vrain and Peach Bottom (DOE Fuel Groups 19 and 20).

5.1.9.1 Evaluation Summary and Highlights for WG9

The results of evaluating WG9 and the four disposal concepts are summarized in Table 5-8 (see Table E-9 for details). The split cells in the last row of Table 5-8 indicate that not all of the waste forms in WG9 can fit in a deep borehole. Specifically, the waste from Fort St. Vrain cannot be disposed of in deep boreholes because it is too large. When considering disposal in deep boreholes, only the waste in this WG that can be disposed of in deep boreholes was evaluated. Some of the pros and cons of disposing of WG9 in the four disposal concepts are:

- Pros:
  - Regardless of the disposal concept, disposal containers can be designed for the repository environment.
  - For a clay/shale repository, may not need a separate backfill.
  - For disposal in salt, crystalline rock, and clay/shale, confidence in disposal performance is high because of the amount of world-wide experience in the respective geologic media.

  - The waste would not need to be repackaged, thus making it easier to ensure worker health and safety, and generating no additional LLW or mixed waste.
  - Retrieval of the waste from a crystalline repository or from a clay/shale repository could be relatively easy.
- Cons:
  - The lack of a design and demonstration of deep borehole disposal makes it a challenge to have confidence in the disposal concept without further R&D and lowers the degree of technical readiness of the disposal concept.
  - The fuel has a high fissile content, requiring increased MC&A safeguards.
  - The particles pose a somewhat higher security challenge than spent fuel rods because they are dispersible

Table 5-8. Summary of evaluation results for WG9

Disposal Concept	Disposal Option Performance	Confidence in Expected Performance Bases	Operational Feasibility	Secondary Waste Generation	Technical Readiness	Safeguards and Security
Salt	✓	✓	✓	✓	✓	✓
Crystalline	✓	✓	✓	✓	✓	✓
Clay/Shale	✓	✓	✓	✓	✓	✓
Deep Borehole	✓ / NA	○ / NA	○ / ✕	✓ / NA	○ / NA	✓ / NA

Note: Split scores indicate that size constraints preclude disposal of some, but not all, waste forms in this group.

Legend:	✓ Strong	○ Moderate	✹ Weak/Uncertain	✕ Not Feasible	NA Not analyzed

5.1.9.2 Cost Differential Considerations for WG9

If the spent fuel from Fort St. Vrain were to be cored to remove the particles, all the waste in this waste group could be disposed of in deep boreholes. This would introduce additional costs.

5.1.9.3 Institutional/Regulatory Considerations for WG9

No institutional or regulatory considerations unique to WG9 were identified.

5.1.10 WG10 Evaluation Results

WG10 consists of naval spent fuel, which is DOE Spent fuel Group 32. There are projected to be 65 MTHM of this highly enriched spent fuel, and it will not be repackaged.

5.1.10.1 Evaluation Summary and Highlights for WG10

The results of evaluating WG10 and the four disposal concepts are given in Table 5-9 (see Table E-10 for details). As indicated in this table, the waste in WG10 cannot be disposed of in a deep borehole because the packages are too large. Some of the pros and cons of disposing of this waste in the remaining three disposal concepts are:

- Pros:
  - An overpack specifically designed for the repository environment can be used, making it easier to manage the thermal and chemical interactions between the waste form and the disposal concept. The waste canister was designed to be directly disposed of in a geologic repository and criticality concerns have been addressed.
  - Repackaging is not required, reducing challenges to worker health and safety, and minimizing the amount of LLW and mixed waste generated.
- Cons:
  - Higher thermal load in a crystalline repository complicate reliance on a bentonite buffer.
  - There are knowledge gaps regarding the behavior of salt under high thermal loads.
  - For disposal in a crystalline environment, overpack performance is unproven.
  - For disposal both in a salt environment and in a clay/shale environment, it could be a challenge to keep the large shafts or ramps open during the operational period, and to seal them upon repository closure. A clay/shale repository might need to remain open longer than a salt repository to allow for ventilation.
  - For all disposal concepts, conveyance options for the large packages have not yet been developed.
  - The fissile content of the fuel increases the need for MC&A safeguards, which may be partially mitigated by the large size and mass of the waste packages.

Table 5-9. Summary of evaluation results for WG10

Disposal Concept	Disposal Option Performance	Confidence in Expected Performance Bases	Operational Feasibility	Secondary Waste Generation	Technical Readiness	Safeguards and Security
Salt	✓	○	✸	✓	○	✓
Crystalline	✓	✸	○	✓	○	✓
Clay/Shale	✓	○	✸	✓	○	✓
Deep Borehole	NA	NA	✕	NA	NA	NA

Legend:					
	✓ Strong	○ Moderate	✸ Weak/Uncertain	✕ Not Feasible	NA Not analyzed

5.1.10.2 Cost Differential Considerations for WG10

Additional costs could be incurred for repository engineering for large package sizes.

5.1.10.3 Institutional/Regulatory Considerations for WG10

Disposal of the large naval SNF packages may reduce the number of potentially suitable sites if the option eliminates some geologic media and/or places greater demands on the performance of the geologic barrier system.

5.2 Disposal Option Evaluations Discussed by Disposal Concept

Section 5.1 presented the results of the evaluation by WG. This section presents the same results but organized by disposal concept. Each section below presents the results of the evaluation for a single disposal concept and all WGs.

5.2.1 Salt Disposal Concept

It was agreed that all WGs could be disposed of in a salt repository, with the exception of sodium-bonded fuels for which no assessment was made because of a lack of information.

5.2.1.1 Evaluation Summary and Highlights of the Salt Disposal Concept

The results of evaluating all ten WGs in a salt repository are shown in Table 5-10 (see Table E-11 for details). Some of the pros and cons of using this disposal concept are as follows:

- Pros:
  - For wastes that have a high thermal load, the high thermal conductivity and high temperature limit of salt allows packages to be spaced more closely than in either a crystalline repository or a clay/shale repository.
  - For wastes for which criticality is a concern, the relative lack of water and the high cross-section of chlorine for capture of thermal neutrons makes it easier to address criticality concerns.
  - The limited far-field radionuclide transport in salt reduces the importance of the waste form and waste package lifetime in evaluating the safety of the disposal option and increases confidence in the information bases. This is particularly important for wastes in WG8 (salts, granular solids, and powders).
  - The low permeability and reducing environment makes it easier to keep particular waste packages isolated from each other, should that be necessary.
  - Some untreated waste types may be appropriate for direct disposal in salt, potentially reducing costs and risks associated with waste treatment.
  - The experience at WIPP provides additional operational confidence.
- Cons:
  - For very large waste packages, keeping the large shafts and ramps open during the operational period will present a challenge, as will sealing these large shafts and ramps upon closure.
  - For very large waste packages, technologies for moving and emplacing the waste packages in salt have yet to be developed.
  - There are gaps in our knowledge concerning the response of salt to high thermal loads.
  - There may be a greater need for site-specific information regarding this type of disposal media because of the high reliance on the integrity of the host rock.

5.2.1.2 Cost Differential Considerations of the Salt Disposal Concept

Based on previous experience, baseline cost estimates for mining and operating a salt repository are anticipated to be lower than that associated with other mined repositories (see Section 3.1.5.3). Additional costs for a salt repository would be incurred for repository engineering for large package sizes.

Table 5-10. Summary of evaluation results for the salt disposal concept

Waste Group	Disposal Option Performance	Confidence in Expected Performance Bases	Operational Feasibility	Secondary Waste Generation	Technical Readiness	Safeguards and Security
WG1—Commercial SNF PBCs	✓	✓	○	✸	✓	✓
WG2—Commercial SNF DPCs	✓	○	✸	✓	○	✓
WG3—HLW glass	✓	✓	✓	✓	○	✓
WG4—Other engineered waste forms	✓	✓	✓	○	○	✓
WG5—Metallic and non-oxide spent fuels	✓	✓	○	✓	○	✓
WG6—Sodium-bonded fuel	Unknown	NA	NA	NA	NA	NA
WG7—DOE oxide fuels	✓	✓	✓	✓	✓	✓
WG8—Salt, granular solids, powders	✓	✓	○	✓	✓	○
WG9—Coated-particle spent fuel	✓	✓	✓	✓	✓	✓
WG10—Naval fuel	✓	○	✸	✓	○	✓

Legend:

✓	○	✸	✕	NA
Strong	Moderate	Weak/Uncertain	Not Feasible	Not analyzed

5.2.1.3 Institutional/Regulatory Considerations of the Salt Disposal Concept

There is a significant amount of mining and operational experience in salt, perhaps making it easier for a community to enter into a consent-based agreement. There are also many salt sites across the country that might be suitable for disposal.

5.2.2 Crystalline Disposal Concept

It was agreed that all WGs could be disposed of in a crystalline repository, with the exception of sodium-bonded fuels for which no assessment was made because of a lack of information.

5.2.2.1 Evaluation Summary and Highlights of the Crystalline Disposal Concept

The results of evaluating all ten WGs in a crystalline disposal concept are shown in Table 5-11 (see Table E-12 for details). Some of the pros and cons of using this disposal concept are:

- Pros:
  - There is a significant amount of world-wide experience with this medium.
  - If waste package retrieval is necessary, it is easiest in this disposal concept.
  - Keeping large shafts and ramps open during the operational period should not present a challenge, making this disposal concept more amenable to disposal of large packages.
  - Would be relatively easy to achieve separation distances between wastes, if needed.
- Cons:
  - The lower thermal conductivity of crystalline rock complicates the reliance on bentonite as a buffer for those wastes that generate a significant amount of heat.
  - Management of high heat loads could require emplacing backfill after a long period of ventilation.
  - Higher reliance on the waste package lifetime to ensure safety makes it more of a challenge to have confidence in disposing of those waste forms with short lifetimes (e.g., WG8).
  - May need to consider adding EBS component to address criticality concerns for some wastes.
  - Overpacks would be needed for some of the wastes, particularly the DPCs.
  - Some wastes would need to be segregated from other wastes because of possible corrosive chemical reactions.
  - Some wastes might contain or generate colloids, which could be transported readily in the fracture networks present in crystalline rocks.
  - For very large waste packages, technologies for moving and emplacing the waste packages in crystalline rock have yet to be developed.

5.2.2.2 Cost Differential Considerations of a Crystalline Disposal Concept

As discussed in Section 3.1.5.3, baseline cost estimates for a crystalline mined repository are higher than those for a salt repository. Additional costs would be incurred for large overpacks for some waste packages (e.g., DPCs).

5.2.2.3 Institutional/Regulatory Considerations of a Crystalline Disposal Concept

There is a significant amount of mining experience in crystalline rock, perhaps making it easier for a community to enter into a consent-based agreement. There are also many crystalline rock sites across the country that might be suitable for disposal.

Table 5-11. Summary of evaluation results for the crystalline disposal concept

Waste group	Disposal Option Performance	Confidence in Expected Performance Bases	Operational Feasibility	Secondary Waste Generation	Technical Readiness	Safeguards and Security
WG1—Commercial SNF PBCs	✓	✓	○	✸	✓	✓
WG2—Commercial SNF DPCs	✓	✸	○	✓	○	✓
WG3—HLW glass	✓	✓	✓	✓	○	✓
WG4—Other engineered waste forms	✓	○	✓	○	○	✓
WG5—Metallic and non-oxide spent fuels	✓	○	○	✓	○	✓
WG6—Sodium-bonded fuel	Unknown	NA	NA	NA	NA	NA
WG7—DOE oxide fuels	✓	✓	✓	✓	✓	✓
WG8—Salt, granular solids, powders	✓	○	○	✓	✓	○
WG9—Coated-particle spent fuel	✓	✓	✓	✓	✓	✓
WG10—Naval fuel	✓	✸	○	✓	○	✓

Legend:

✓	○	✸	✕	NA
Strong	Moderate	Weak/Uncertain	Not Feasible	Not analyzed

5.2.3 Clay/Shale Disposal Concept

It was agreed that all WGs could be disposed of in a clay/shale repository, with the exception of sodium-bonded fuels for which no assessment was made because of a lack of information.

5.2.3.1 Evaluation Summary and Highlights of a Clay/Shale Disposal Concept

The results of evaluating all ten WGs in a clay/shale disposal concept are shown in Table 5-12 (see Table E-13 for details). Some of the pros and cons of using this disposal concept are:

- Pros:
  - There is a significant amount of world-wide experience with this disposal medium.
  - Would be relatively easy to achieve necessary separation distances between wastes, if needed.
  - The low permeability and high sorptive capacity for clay/shale reduces the reliance on waste form and waste package lifetimes for generating confidence in disposal option performance. This is particularly important for wastes in WG8 (salts, granular solids, and powders).
- Cons:
  - For very large waste packages, keeping the large shafts and ramps open during the ventilation period will present a challenge, as will retrieval during preclosure.
  - May need to add EBS components to address criticality control.
  - Some wastes would need to be segregated from other wastes because of possible corrosive chemical reactions.
  - For very large waste packages, technologies for moving and emplacing the waste packages in clay/shale have yet to be developed.

5.2.3.2 Cost Differential Considerations of a Clay/Shale Disposal Concept

As discussed in Section 3.1.5.3, baseline cost estimates for a clay/shale mined repository show a broad range, but are generally higher than those for a salt repository. Additional costs for a clay/shale mined repository would be incurred for repository engineering for large package sizes.

5.2.3.3 Institutional/Regulatory Considerations of a Clay/Shale Disposal Concept

There is a significant amount of international experience in clay/shale repository concepts, perhaps making it easier for a community to enter into a consent-based agreement. There are also many clay/shale sites across the country that might be suitable for disposal.

Table 5-12. Summary of evaluation results for the clay/shale repository

Waste group	Disposal Option Performance	Confidence in Expected Performance Bases	Operational Feasibility	Secondary Waste Generation	Technical Readiness	Safeguards and Security
WG1—Commercial SNF PBCs	✓	✓	○	✹	✓	✓
WG2—Commercial SNF DPCs	✓	✹	○	✓	○	✓
WG3—HLW glass	✓	✓	✓	✓	○	✓
WG4—Other engineered waste forms	✓	○	✓	○	○	✓
WG5—Metallic and non-oxide spent fuels	✓	○	○	✓	○	✓
WG6—Sodium-bonded fuel	Unknown	NA	NA	NA	NA	NA
WG7—DOE oxide fuels	✓	✓	✓	✓	✓	✓
WG8—Salt, granular solids, powders	✓	○	○	✓	○	○
WG9—Coated-particle spent fuel	✓	✓	✓	✓	✓	✓
WG10—Naval fuel	✓	✹	○	✓	○	✓

Legend:

✓	○	✹	✕	NA
Strong	Moderate	Weak/Uncertain	Not Feasible	Not analyzed

5.2.4 Deep Borehole Disposal Concept

It was found that some of the wastes cannot be disposed of in deep boreholes. Specifically, all the wastes in WG2 (commercial SNF disposed of in DPCs) and in WG10 (naval fuel) are too big to fit in a deep borehole. Furthermore, some waste forms in three other WGs are too big to fit in a deep borehole drilled with current standard technology: existing HLW glass canisters (WG3), already-packaged DOE metallic and non-oxide spent fuels (WG5), already-packaged DOE oxide fuels (WG7), and the Fort St. Vrain fuel (WG9). As for the other three disposal concepts, no assessment was made for sodium-bonded fuels (WG6) due to lack of information.

5.2.4.1 Evaluation Summary and Highlights of a Deep Borehole Disposal Concept

The results of evaluating all ten WGs in a deep borehole concept are shown in Table 5-13 (see Table E-14 for details). Some of the pros and cons of using this disposal concept are:

- Pros:
  - Because of the depth of the disposal concept and the small size of the waste packages, the thermal buoyancy effect within the geologic medium is not an issue.
  - Because there is less reliance on waste form and waste package performance with regard to demonstrating safety, it is easier to have confidence in the performance bases.
  - Smaller waste types are candidates for this disposal concept, allowing for efficient disposal of such waste forms.
  - Some untreated waste types may be candidates for direct disposal in boreholes, potentially reducing costs and risks associated with waste treatment.
- Cons:
  - Limited to disposal of very small packages (around 1 ft (30 cm) diameter or less).
  - There has been no detailed design or demonstration of this disposal concept, limiting the confidence in performance bases.
  - Disposing of spent fuel would require repackaging and, in some cases, consolidating the spent fuel rods. This would introduce additional costs, create additional LLW, and introduce additional health and safety concerns.
  - Retrieving waste disposed of in deep boreholes could be difficult.
  - The transportation capacity and logistics for small volumes of waste likely would limit disposal operations, so surface handling and storage concepts would need further consideration. Surface storage would likely be needed, introducing security issues, particularly for small packages containing easily dispersible wastes.
  - Disposing of only very small-diameter packages results in handling more waste packages compared to the other three disposal concepts.

5.2.4.2 Cost Differential Considerations of a Deep Borehole Disposal Concept

As discussed in Section 3.1.5.3, costs for deep borehole disposal may be comparable to those associated with the other three disposal concepts. This disposal concept would require more waste packages than the other three disposal concepts. However, the waste packages and the waste package material would likely be less expensive because of the small package size and less stringent design requirements. If projected HLW glass were to be disposed of in deep boreholes, additional costs would be incurred because of the need to re-design vitrification facilities and handling of a larger number of canisters. Repackaging spent fuel rods that are already in canisters and rod consolidation would also incur additional costs. The additional surface handling and storage facilities that might be required at the deep borehole disposal site would also introduce additional costs.

5.2.4.3 Institutional/Regulatory Considerations of a Deep Borehole Disposal Concept

There are many possible geologic sites around the country, perhaps making it easier to find a community willing to enter into a consent-based agreement. On the other hand, there is no operational experience with deep boreholes, which might make communities less willing to enter into a consent-based siting agreement absent further R&D activities. Current regulations did not contemplate deep borehole disposal, and there are possible issues with Underground Injection Control requirements.

Table 5-13. Summary of evaluation results for the deep borehole concept

Waste group	Disposal Option Performance	Confidence in Expected Performance Bases	Operational Feasibility	Secondary Waste Generation	Technical Readiness	Safeguards and Security
WG1—Commercial SNF PBCs	✓	○	✸	✸	○	○
WG2—Commercial SNF DPCs	NA	NA	✕	NA	NA	NA
WG3—HLW glass	✓ / NA	○ / NA	○ / ✕	✓ / NA	✸ / NA	✓ / NA
WG4—Other engineered waste forms	✓	○	✓	○	○	✓
WG5—Metallic and non-oxide spent fuels	✓ / NA	○ / NA	✓ / ✕	✓ / NA	✓ / NA	✓ / NA
WG6—Sodium-bonded fuel	Unknown	NA	NA	NA	NA	NA
WG7—DOE oxide fuels	✓ / NA	○ / NA	✸ / ✕	○ / NA	○ / NA	○ / NA
WG8—Salt, granular solids, powders	✓	○	○	✓	○	○
WG9—Coated-particle spent fuel	✓ / NA	○ / NA	○ / ✕	✓ / NA	○ / NA	✓ / NA
WG10—Naval fuel	NA	NA	✕	NA	NA	NA

Note: Split scores indicate that size constraints preclude disposal of some, but not all, waste forms in this group.

Legend:	✓ Strong	○ Moderate	✸ Weak/Uncertain	✕ Not Feasible	NA Not analyzed

6 SUMMARY AND CONCLUSIONS

6.1 Scope

The study team was tasked with (1) identifying the U.S. inventories of commercial SNF and DOE-managed SNF and HLW (and other wastes) that require deep geologic disposal, and (2) evaluating technical aspects of potential disposal of the identified waste forms in various geologic disposal concepts to identify a range of disposal options for the waste forms. Specific questions posed to the team included:

- Is a "one-size-fits–all" repository a good strategic option for disposal?
- Do different waste types and forms perform differently enough in different disposal concepts that they warrant different treatment?
- Do some disposal concepts perform significantly better with or without specific waste types or forms?

6.1.1 Waste Inventory

The study included all existing SNF from commercial, defense, and research reactors, and SNF from reasonably foreseeable operations of existing reactors. The study also included existing HLW (e.g., vitrified HLW at SRS and WVDP) and waste forms projected to be generated in the future from existing process waste (e.g., vitrified HLW from Hanford and SRS). In addition, the study includes consideration of both direct disposal of waste forms that are not currently planned for disposal without further treatment (e.g., calcine waste at INL) and alternatives to planned treatments. The study acknowledges existing plans, commitments, and requirements where applicable, but the study evaluates options for disposal based primarily on technical, rather than programmatic, constraints.

6.1.2 Disposal Concepts

The four representative disposal concepts included in this study are mined repositories in three geologic media—salt, clay/shale rocks, and crystalline (e.g., granitic) rocks—and deep borehole disposal in crystalline rocks.

Other geologic disposal concepts have been proposed and are potentially viable. For example, Canada is currently evaluating a mined repository for intermediate-level radioactive waste in carbonate rocks (NWMO 2011) and the U.S. has evaluated a potential mined repository concept in volcanic tuff (DOE 2008). Although these concepts have unique features that distinguish them from the four selected for consideration in this report, attributes of the four concepts discussed here are representative of a broad range of other disposal concepts.

6.2 Method

Waste types were mapped to corresponding waste forms in which the waste could be disposed. The 43 waste types were mapped to 50 waste forms, which were then grouped into ten waste groups based on similarities of characteristics relevant to disposal performance. The ten waste groups are shown in Table 6-1.

Each waste group was evaluated against six primary criteria, summarized in Table 6-2, for potential disposal in each of the four disposal concepts. Scoring was done qualitatively, using the informed and consensus judgment of a subset of the multidisciplinary team contributing to the report.

Table 6-1. Waste group descriptions

Waste group	Description
WG1	All commercial SNF packaged in purpose-built disposal containers
WG2	All commercial SNF packaged in DPCs of existing design
WG3	All vitrified HLW (all types of HLW glass, existing and projected, canistered)
WG4	Other engineered waste forms
WG5	Metallic and non-oxide DOE spent fuels
WG6	Sodium-bonded fuels (driver and blanket), direct disposed[1]
WG7	DOE oxide fuels
WG8	Salt, granular solids, and powders
WG9	Coated-particle spent fuel
WG10	Naval fuel

Table 6-2. Evaluation criteria and associated metrics

Evaluation Criteria	Metrics
Disposal Option Performance	Likely to comply with long-term standards?
Confidence in Expected Performance Bases	Additional EBS components needed above baseline for each design concept Robustness of information bases; simplicity vs. complexity; knowledge gaps
Operational Feasibility	Ease in ensuring worker health and safety at all stages Special physical considerations at any stages based on physical characteristics
Secondary Waste Generation	Amount of LLW generated during handling and treatment Amount of mixed waste generated
Technical Readiness	Status of waste form technologies Status of transportation and handling systems Status of disposal technologies
Safeguards and Security	National security implementation difficulty Radiological dispersion device prevention implementation difficulty

6.3 Results from the Disposal Option Evaluation

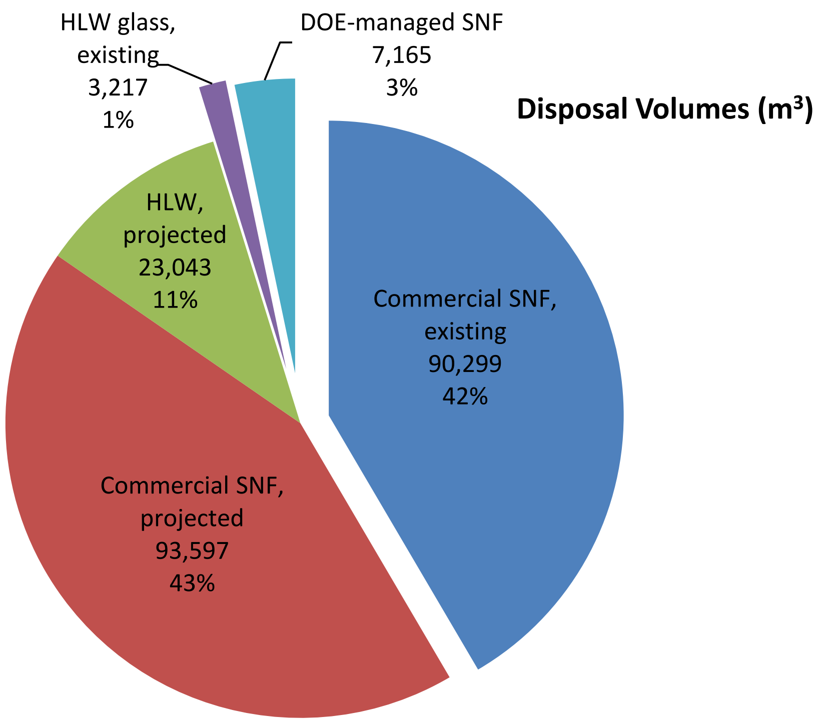
In the course of conducting the study, inventory information for the various waste types was collected. Observations regarding this waste inventory are presented below, followed by a summary of the evaluation results.

6.3.1 Observations Specific to the Waste Inventory

Commercial SNF is the largest component of the waste inventory requiring geologic disposal today, and it will increase in quantity to comprise a vast majority of the inventory by 2048 (Figure 6-1). Assuming for the purposes of the analysis that commercial power generation continues unchanged from today's rate, and that all commercial SNF is eventually packaged in existing-design DPCs for storage, it is estimated that SNF will comprise 88% (by volume) of the total inventory of HLW and SNF in 2048.

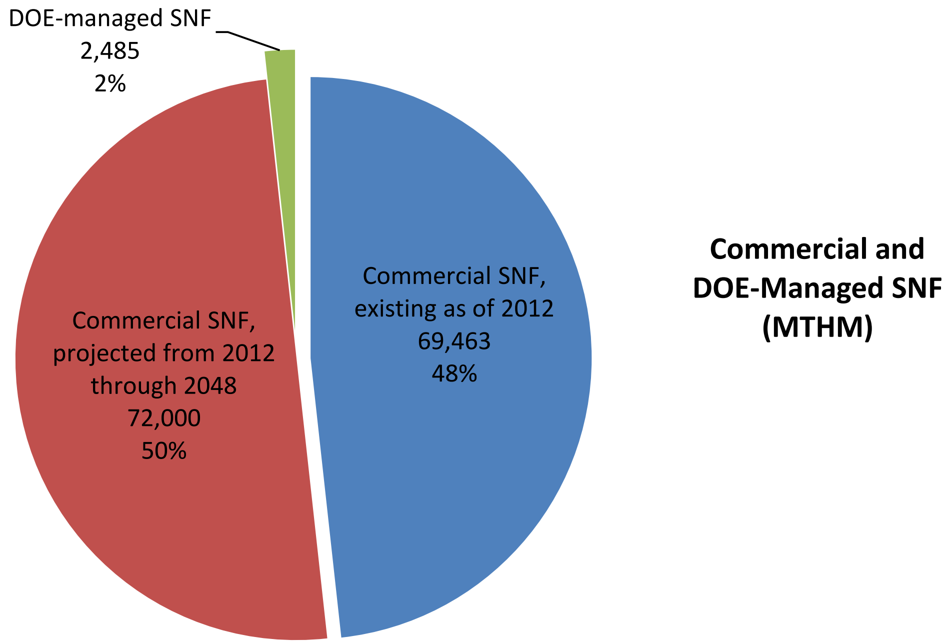
At this time, a total of 47% of all expected HLW and SNF wastes exist: commercial SNF is 42%, DOE SNF is 3% (all DSNF is assumed to exist for simplicity, though some naval SNF is yet to be generated), and HLW is 2%. It is estimated that the vast majority (i.e., 98% by mass) of the SNF inventory in 2048 will be commercial SNF (Figure 6-2), with the remaining 2% of the mass being DOE-managed SNF.

The large majority of the volume of DOE-managed SNF (which in itself will be ~3% by volume of the total estimated inventory of HLW and SNF in 2048—see Figure 6-1), will be naval fuel (64% by volume), which is packaged in large containers, followed by uranium metal fuels with Zircaloy cladding (7% by volume), primarily from the N Reactor at the Hanford Reservation. Other DOE-managed SNF, including a broad range of fuel types, comprise the remaining 29% by volume of the total inventory.



NOTE: Volume estimates assume (1) constant nuclear power generation in commercial reactors and disposal of all commercial SNF in dual-purpose canisters, and (2) relative volumes of various HLW based on calcine processed by hot isostatic pressing with additives, (3) sodium-bearing waste treated by fluidized bed steam reforming, (4) sodium-bonded fuels undergo electrometallurgical treatment, and (5) all other waste forms are vitrified. For simplicity, all DOE-managed SNF is shown as "existing"; approximately 3,500 m^3 of naval SNF remains to be generated.

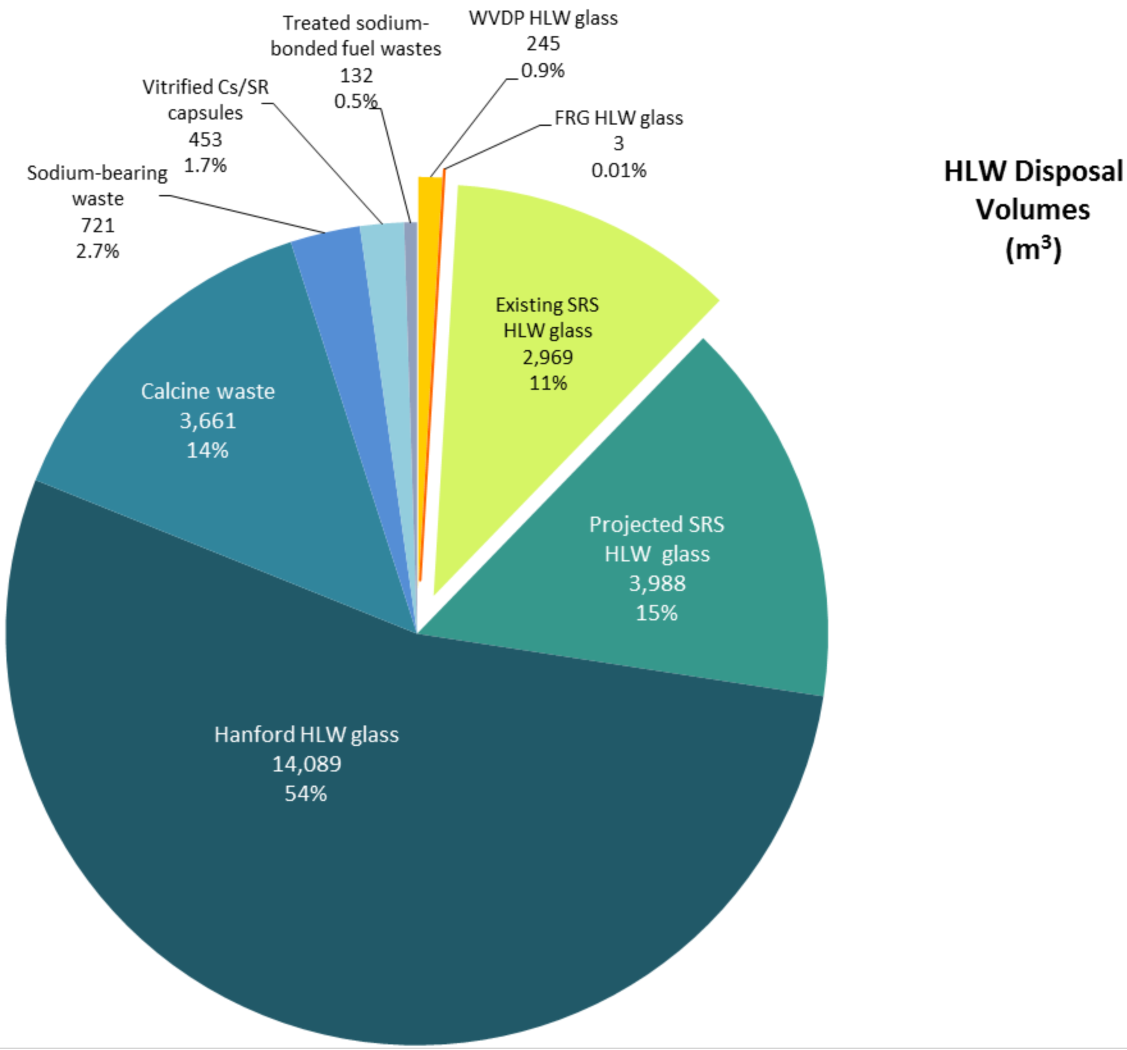
Figure 6-1. Disposal volumes of U.S. SNF and HLW, existing and projected in 2048



NOTE: For simplicity, all DOE-managed SNF is shown as "existing"; approximately 3,500 m^3 of naval SNF remains to be generated.

Figure 6-2. Masses of commercial and DOE-managed SNF, existing and projected in 2048, assuming constant rate of nuclear power generation in commercial reactors

Figure 6-3 shows relative volumes of HLW projected to 2048 and represents the breakdown of the two HLW sections (existing and projected HLW) shown in Figure 6-1. Of the HLW projected to be available for geologic disposal in 2048, the largest fraction (54%) presently exists as tank waste at the Hanford Reservation. Slightly less than 13% of the total projected volume of HLW in 2048 exists today in a vitrified form, with nearly all of that at the SRS and the balance at the WVDP (Figure 6-3).



NOTE: Volume estimates assume calcine processed by hot isostatic pressing with additives, sodium-bearing waste treated by fluidized bed steam reforming, sodium-bonded fuels undergo electrometallurgical treatment, and all other waste forms are vitrified.

Figure 6-3. Volumes of HLW and other waste, existing and projected in 2048

Table 6-3 gives disposal volumes of the various wastes shown in Figure 6-1, Figure 6-2, and Figure 6-3, both currently existing and projected to exist in 2048. As this table shows, commercial SNF comprises the largest fraction of the volume of waste to be disposed of.

Table 6-3. Disposal volumes of U.S. SNF and HLW, existing and projected in 2048

Waste	Present Volume (m^3)	Additional Projected Volume in 2048 (m^3)	Total Volume (m^3)
Commercial SNF disposed of in DPCs	90,299	93,597	183, 896
DOE-managed SNF	7,165	0	7,165
Savannah River Site vitrified HLW	2,969 (through macrobatch 8)	3,988	6,957
Hanford Site vitrified HLW	0	14,089	14,089
Calcine waste after treatment by HIP	0	3,661	3,661
SBW after treatment by fluidized bed steam reforming	0	721	721
Vitrified cesium/strontium capsules	0	453	453
West Valley Demonstration Project vitrified HLW	245	0	245
Treated sodium-bonded fuel (EMT)	0	132	132
Federal Republic of Germany HLW glass	3	0	3
Total	100,681	116,641	217,322

Note: Values estimated assuming constant nuclear power generation in commercial reactors. For simplicity, all DOE SNF is shown as "existing," although approximately 3,500 m^3 of naval SNF remains to be generated. In addition, all the waste from EMT of sodium-bonded fuel is shown as "projected" even though a small quantity was generated during demonstration of the treatment process.

6.3.2 Observations Specific to Disposal Options

6.3.2.1 Mined Repositories in Salt

Overall, mined repositories in salt showed strong results for most waste groups with respect to most metrics. At the level of resolution provided by this evaluation, scores for salt and clay/shale repositories are equivalent. The evaluation group noted the following pros and cons of the concept:

- Pros:
  - The high thermal conductivity and high temperature limit of salt provide greater flexibility (i.e., larger packages and closer spacing) for disposal of heat generating wastes than other geologic media.
  - The limited far-field radionuclide transport (low permeability) of salt reduces the reliance on the waste form and waste package lifetimes, providing greater confidence in estimates of long-term performance.
  - The low permeability and reducing (oxygen-poor) environment makes it easier to keep specific waste packages isolated from each other, should that be necessary.
  - Some untreated waste types may be appropriate for direct disposal in salt, potentially reducing costs and risks associated with waste treatment.
  - The operational experience at WIPP provides additional confidence.

  - The relative lack of water and the high cross-section of chlorine for capture of thermal neutrons make it easier to address criticality concerns (for waste forms for which criticality is relevant).
- Cons:
  - For very large waste packages (e.g., DPCs), keeping the large shafts and ramps open during the operational period will present a challenge, as will sealing these large shafts and ramps upon closure.
  - For very large waste packages, technologies for moving and emplacing the waste packages in salt have yet to be developed.
  - Knowledge gaps exist concerning the response of salt to high thermal loads.
  - There may be a greater need for site-specific information regarding this type of disposal media because of the high reliance on the integrity of the host rock.

6.3.2.2 Mined Repositories in Crystalline Rock

Overall, mined repositories in crystalline rock showed strong results for most waste groups with respect to most metrics; however, scores in several areas were lower than for salt or clay/shale disposal concepts suggesting that R&D needs could be greater for a crystalline repository. The evaluation group noted the following pros and cons of the concept:

- Pros:
  - There is significant world-wide experience with this medium.
  - If waste package retrieval is necessary, it is easiest in this disposal concept.
  - Stable rock properties enhance operational feasibility for very large packages (e.g., DPCs).
  - Would be relatively easy to achieve separation distances between waste forms, if needed.
- Cons:
  - The lower thermal conductivity of crystalline rock complicates the reliance on bentonite as a buffer for those wastes that generate a significant amount of heat.
  - Management of high heat loads could delay emplacing backfill until after a long period of ventilation.
  - Strong reliance on waste package lifetime results in lower confidence for high-heat waste forms and readily mobilized waste forms
  - Some wastes may need to be segregated from other wastes because of possible corrosive chemical reactions.
  - Some of the waste forms, particularly DPCs, would need robust overpacks that may pose design challenges.
  - Colloids pose a potential for transport in fracture networks
  - For very large waste packages, technologies for moving and emplacing the waste packages in crystalline rock have yet to be developed.
  - May need to consider adding EBS component to address criticality concerns for some wastes.

6.3.2.3 Mined Repositories in Clay/Shale

Overall, mined repositories in clay/shale showed strong results for most waste groups with respect to most metrics. At the level of resolution provided by this evaluation, scores for clay/shale and salt repositories are equivalent. The evaluation group noted the following pros and cons of the concept:

- Pros:
  - There is a significant amount of world-wide experience with this disposal medium.
  - The limited far-field radionuclide transport in clay/shale (low permeability and high sorption) reduces the reliance on the waste form and waste package lifetimes
  - Would be relatively easy to achieve necessary separation distances between wastes, if needed.
- Cons:
  - For very large waste packages, keeping the large shafts and ramps open during the ventilation period will present a challenge, as will retrieval during preclosure operations.
  - For very large waste packages, technologies for moving and emplacing the waste packages in clay/shale have yet to be developed.
  - May need to add EBS components to address criticality control.
  - Some wastes may need to be segregated from other wastes because of possible corrosive chemical reactions.

6.3.2.4 Disposal in Deep Boreholes

Overall, deep borehole disposal options received mixed scores. For wastes that currently exist either as unpackaged materials (e.g., untreated calcine waste and some DOE-managed SNF) or in small packages (e.g., cesium/strontium capsules), deep borehole disposal may be a feasible and potentially attractive option. For example, untreated cesium/strontium capsules, which represent about 40% of the total radioactivity originally in HLW at the Hanford site, could be disposed of in a single borehole. However, f or larger waste forms (e.g., commercial SNF disposed of in DPCs and the existing canisters of HLW glass), the option is simply not feasible because the wastes forms are larger than current standard drilling technology could accommodate. For many waste forms, engineering challenges associated with preparing the waste form in a small enough package for borehole emplacement are sufficient to make it a less attractive option. For example, deep borehole disposal of PWR SNF could require removal of assembly hardware and consolidation of the rods. Deep borehole disposal of future HLW glass would require redesign of existing vitrification facilities to make smaller canister pours. The evaluation group noted the following pros and cons of the concept:

- Pros:
  - Thermal load management concerns are minimized by the depth of the disposal concept and the relatively small size of the waste packages.
  - Because there is less reliance on waste form and waste package performance, it is easier to have confidence in the performance bases.
  - Smaller waste types are good candidates for this disposal concept (e.g., cesium/strontium capsules), allowing for efficient disposal of such waste forms.
  - Some untreated waste types may be candidates for direct disposal in boreholes (e.g., untreated calcine waste), potentially reducing costs and risks associated with waste treatment

- Cons:
  - Currently limited to disposal of very small packages (around 1 ft (30 cm) diameter or less).
  - Lack of detailed design or demonstration of this disposal concept limits confidence.
  - If considered for disposing of commercial spent fuel, it would require repackaging and, in some cases, consolidating the spent fuel rods.
  - Retrieving waste disposed of in deep boreholes could be difficult.
  - The transportation capacity and logistics for small volumes of waste likely would limit disposal operations, so surface handling and storage concepts would need further consideration.
  - For a given volume of waste, disposing of only very small-diameter packages results in handling more waste packages compared to the other three disposal concepts.

6.3.2.5 Waste Form Observations Independent of Disposal Concept

Although some of the criteria provided relatively little discrimination among disposal concepts, the evaluation identified potentially useful insights relevant to waste form properties independent of disposal concept. Specifically:

- Enough information does not currently exist to evaluate the performance of direct disposal of sodium-bonded SNF in any geologic disposal concept, and this waste type may require treatment regardless of the disposal concept.
- None of the disposal concepts considered posed significantly different concerns related to safeguards and security, and only those containing salts, granular solids, and powders, raised moderate security concerns associated with the potential for diversion for radioactive dispersal devices.
- All waste form treatment options that involve handling or processing waste carry the potential for increased generation of secondary waste. This is most significant, perhaps, in comparing WG-1 (repackaging of all commercial SNF for disposal in PBCs) and WG-2 (disposal of all commercial SNF in existing DPCs without repackaging).
- In general, demonstration of the technical readiness of a specific waste form is independent of the disposal concept being considered.

6.4 Conclusions

Technical conclusions relevant to each of the three questions posed at the beginning of the study are presented here.

Is a "one-size-fits-all" repository a good strategic option?

The study concludes that from a technical perspective, any of the mined repository concepts could accommodate all of the waste forms with the exception of untreated sodium-bonded SNF, for which available information was insufficient to support an evaluation, similar to conclusions by Rechard (1993; 1995; 1998). The study concludes that the deep borehole disposal option is a good option for small waste packages and provides flexibility to a disposal strategy. The study also notes that disposal options that utilize multiple repositories are also technically viable. Evaluation of strategic decisions regarding the choice of disposal options is outside the scope of this report.

Do different waste forms perform differently enough in different disposal environments to warrant different approaches?

The study did not identify any waste forms that required a specific disposal environment/concept. Other relevant observations include:

- With the exception of the untreated sodium-bonded SNF discussed above, all waste forms could be accommodated in multiple disposal concepts, although with varying degrees of confidence.
- Some disposal concepts may require segregating some waste forms from each other within a single repository. Specifically, halide-bearing wastes (including salt waste forms and the cesium/strontium capsules) may be corrosive, and if they are disposed of without treatment they should be isolated from other wastes in disposal concepts that rely on long-lived waste packages.
- Small waste forms are potentially attractive candidates for deep borehole disposal. Those wastes forms include salt wastes from EMT of sodium-bonded SNF, untreated calcine waste, cesium/strontium capsules, and some DOE-managed SNF that has not yet been packaged.
- Salt allows for more flexibility in managing high-heat waste in mined repositories than other media.
- The study did not identify technical issues associated with disposing of mixed waste (i.e., waste containing both radioactive and RCRA-regulated constituents).
- The study concluded that direct disposal of commercial SNF in existing DPCs was potentially feasible but could pose significant challenges both in repository operations and demonstrating confidence in long-term performance.

Do some disposal concepts perform better with or without specific waste forms?

The study concludes that all of the disposal options evaluated have the potential to comply with applicable regulatory requirements that protect both worker and public health and safety, and protect the environment. There were a few disposal options considered that were not evaluated for the full range of criteria. These exceptions are those deep borehole options that are physically infeasible due to size constraints, and the disposal of untreated sodium-bonded SNF, for which information is insufficient to support an evaluation. All other disposal options identified in this study could be designed, constructed, and operated to provide safe and robust isolation of the waste forms. However, as shown in the evaluation results presented in detail in Section 5 and summarized above, implementation and demonstration of robust performance may be simpler for some disposal options than others.

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