Preliminary Report on Dual-Purpose Canister Disposal Alternatives (FY13)

Fuel Cycle Research & Development

Prepared for :
U.S. Department of Energy
Office of Used Nuclear Fuel Disposition

E.L. Hardin and D.J. Clayton, Sandia National Laboratories
R.L. Howard, J.M. Scaglione, E. Pierce and K. Banerjee,
Oak Ridge National Laboratory
M.D. Voegele, Complex Systems Group, LLC
H.R. Greenberg, J. Wen and T.A. Buscheck,
Lawrence Livermore National Laboratory
J.T. Carter and T. Severynse,
Savannah River National Laboratory
W.M. Nutt, Argonne National Laboratory

December, 2013
FCRD-UFD-2013-000171 Rev. 1

Revision History

Revision	Description
FCRD-UFD-2013-000171 Rev. 0	Initial issue, assigned Sandia number SAND2013-7251P.
FCRD-UFD-2013-000171 Rev. 1	Issued to clarify Section 5.3, correct Section 6.3.3 and Table 6-7, add disclaimer below, and fix typos and formatting; assigned Sandia number SAND2013-10068P.

CONTEXT FOR THIS STUDY

This is a technical presentation that does not take into account the contractual limitations under the Standard Contract. Under the provisions of the Standard Contract, DOE does not consider spent fuel in canisters to be an acceptable waste form, absent a mutually agreed to contract modification.

Sandia National Laboratories

Sandia National Laboratories is a multi-program laboratory managed and operated by Sandia Corporation, a wholly owned subsidiary of Lockheed Martin Corporation, for the U.S. Department of Energy's National Nuclear Security Administration under contract DE-AC04-94AL85000.

Sandia Review and Approval Number: SAND2013-10068P.

Approved for Unclassified Unlimited Release

Table of Contents

Tables

Figures

Acronyms

AAR	Association of American Railroads
Andra	National Agency for Radioactive Waste Management
BPRA	Burnable Poison Rod Assembly
BRC	Blue Ribbon Commission on America’s Nuclear Future
BSC	Bechtel-SAIC Co.
BWR	Boiling Water Reactor
CARE	Cavern-Retrievable disposal concept
CASTOR	Trade name for vertical dry storage systems (GNS mbH)
CE	Combustion Engineering Inc.
CFR	Code of Federal Regulations
CRIEPI	Central Research Institute of Electric Power Industry
DIREGT	Acronym (German) for direct disposal concept from DBE TEC GmbH
DOE	U.S. Department of Energy
DPC	Dual-Purpose Canister
EBS	Engineered Barrier System
EDZ	Excavation Damage Zone
EIS	Environmental Impact Statement
EPA	U.S. Environmental Protection Agency
EPRI	Electric Power Research Institute
ESR	Excavation Support Ratio
FEPs	Features, Events, and Processes
FEM	Finite Element Method
FY	Fiscal Year
GW	Gigawatt
GW-d	Gigawatt-day
Hi-Storm	Trade name for vertical dry storage systems (Holtec International)
HLW	High-Level Waste
IAEA	International Atomic Energy Agency
ISFSI	Independent Spent Fuel Storage Installation
ISF	Interim Storage Facility
ISG	Interim Staff Guidance
KBS-3	Swedish Reference Disposal Concept
KENO	Criticality modeling framework
k_{eff}	Effective reactivity coefficient
K_{th}	Thermal Conductivity

MMC	Metal Matrix Composite
MOX	Metal Oxide
MPC	Multi-Purpose Canister (also used for canisters from Holtec International)
MT	Metric Tons
MTHM	Metric Tons of Heavy Metal
MTU	Metric Tons Uranium
Nagra	National Cooperative for the Disposal of Radioactive Waste
NAS	National Academy of Sciences
NAS/NRC	National Academies/National Research Council
NEPA	National Environmental Policy Act
NRC	U.S. Nuclear Regulatory Commission
NUHOMS	Trade name for vault –type dry storage system (TransNuclear)
NWPA	Nuclear Waste Policy Act
OCRWM	Office of Civilian Radioactive Waste Management
OECD	Organisation for Economic Co-Operation and Development
OFA	Optimized Fuel Assembly
OFF	Oldest Fuel First
ONDRAF/ NIRAS	Belgian Agency for Radioactive Waste and Enriched Fissile Materials
ONWI	Office of Nuclear Waste Isolation
PA	Performance Assessment
PWR	Pressurized Water Reactor
R&D	Research and Development
RCRA	Resource Conservation and Recovery Act
RMEI	Reasonably Maximally Exposed Individual
RMR	Rock Mass Rating
RQD	Rock Quality Designation
SCC	Stress Corrosion Cracking
SKB	Swedish Nuclear Fuel and Waste Management Co.
SNF	Spent Nuclear Fuel
SNL	Sandia National Laboratories
SS	Stainless Steel
STD	Standard Fuel Assembly
TAD	Transportation-Aging-Disposal canister
TEV	Transport-Emplacement-Vehicle
TSC	Transportable Storage Container (NAC International)
TSPA	Total System Performance Assessment

UFD	Used Fuel Disposition
UNF	Used Nuclear Fuel
UNF-ST&DARDS	Library of DPC fuel characteristics with calculation tools
WIPP	Waste Isolation Pilot Plant
w/w	Weight fraction
YFF	Youngest Fuel First
yr	years

Preliminary Report on Dual-Purpose Canister Disposal Alternatives (FY13)

Deliverable: M2FT-13SN0816112
Work Package: FT-13SN081611

E.L. Hardin and D.J. Clayton, Sandia National Laboratories
R.L. Howard, J.M. Scaglione, E. Pierce and K. Banerjee, Oak Ridge National Laboratory
M.D. Voegele, Complex Systems Group, LLC
H.R. Greenberg, J. Wen and T.A. Buscheck, Lawrence Livermore National Laboratory
J.T. Carter and T. Severynse, Savannah River National Laboratory
W.M. Nutt, Argonne National Laboratory

Executive Summary

Disposition of commercial spent nuclear fuel (SNF) that is stored in thousands of dry casks at reactor sites, will become a major part of back-end fuel management over the next few decades. This report documents the first phase of a multi-year project to understand the technical feasibility and logistical implications of direct disposal of SNF in existing dual-purpose canisters (DPCs) and other types of storage casks. The first phase includes a set of preliminary disposal concepts and associated technical analyses, identification of additional R&D needs, and a recommendation to proceed with the next phase of the evaluation effort.

The preliminary analyses presented in this report indicate that DPC direct disposal could be technically feasible, at least for certain disposal concepts. Preliminary analysis also suggests that cost savings might be realized compared to re-packaging DPCs, although further analysis is needed to understand economic consequences associated with the many possible scenarios.

The topical areas addressed by this study are summarized as follows:

Disposal Concepts – A range of possible geologic settings for DPC direct disposal was identified, drawing on previous disposal concept development. High-level engineering concepts of operation were developed for DPC disposal, to constitute a set of alternative disposal concepts. These include the salt concept, and emplacement in hard rock (i.e., crystalline) or argillaceous sedimentary rock, with or without backfill. This set is not exhaustive but it covers a range of behaviors potentially important to DPC direct disposal including thermal response, postclosure nuclear criticality, and long-term opening stability. Other factors such as ground support, waste package transport and emplacement, and shaft vs. ramp access, are also important and may depend more on site-specific characteristics and in some instances, local experience and preference.

The salt concept would be backfilled immediately after emplacement, while openings in hard rock and sedimentary rock would be ventilated for decades (approximately 50 years or longer) to remove heat. Hard rock formations exist that could have excellent long-term stability, heat dissipation properties, and environmental conditions conducive to waste isolation. Argillaceous (clay-bearing) sedimentary media (e.g., clay or shale formations) could have very low permeability and chemically reducing conditions, but are likely to have more restrictive thermal constraints to limit alteration of the clay, relatively low thermal conductivity and more limited long-term stability. Backfill is an option for hard rock and sedimentary open-mode concepts, and

could provide an additional, redundant engineered barrier. However, the use of backfill would significantly elevate EBS temperatures.

The cavern-retrievable storage and disposal concept was first proposed about a decade ago and remains a potentially important alternative. Shielded dry-storage casks could be emplaced or installed underground, ventilated for decades to remove heat, and closed by installation of an encapsulating buffer. The use of existing surface storage casks (or cask designs) could lower costs but would require development and testing of a buffer system to assure waste isolation.

Safety – Important factors that help to ensure postclosure safety for DPC direct disposal include: 1) diffusion-controlled radionuclide transport in the EBS and NBS; 2) near-field transport properties that are relatively insensitive to temperature, or for which temperature effects can be modeled with confidence; 3) limited radionuclide transport in backfill (if present) and the host rock (particularly the far field); and 4) attributes that limit potential postclosure criticality. These characteristics could actually benefit any geologic repository. When prospective repository sites are identified, site-specific data will support more resolution of differences in postclosure safety associated with DPC direct disposal.

Thermal Management – The salt concept and the unbackfilled hard rock concepts could accept SNF in 32-PWR size packages, with SNF burnup to 60 GW-d/MT, and approximately 50 to 100 years decay storage depending on burnup. These concepts could close within the 150-year timeframe adopted for this study while meeting target values for peak host rock temperature (200°C in both types of media). Required repository layout area (drift and package spacing) to meet the peak temperature targets ranges from approximately 60 to 100 m^2/MTHM.

In sedimentary media, which have lower thermal conductivity and a lower target value for peak rock temperature (100°C) lower burnup PWR SNF (and BWR SNF with similar heat output) could be accommodated within the 150-year timeframe, with repository package and drift spacings similar to the hard rock concept. For higher burnup SNF (e.g., greater than 40 GW-d/MT) a modified concept would be needed that uses some combination of: 1) longer decay storage plus ventilation; 2) much larger spacing (roughly doubling the repository plan area); and/or 3) peak host rock temperature target greater than 100°C.

When backfill is added to the hard rock or sedimentary concepts, and installed at repository closure, the waste package temperature increases significantly. The temperature rise within the backfill (i.e., at the waste package surface) is much greater than the differences in temperatures between the hard rock and sedimentary concepts. Clay-based low-permeability backfill materials could be sensitive to temperature, and better understanding of clay behavior, or alternative materials, is needed to facilitate backfill options for DPC direct disposal. For a similar reason the enclosed emplacement modes such as those being pursued in Sweden or France for crystalline and sedimentary media, respectively, when applied to DPC direct disposal, cannot meet the peak buffer temperature target without decay storage much longer the assumed 150-year timeframe. It is therefore important to continue R&D that could support relaxation of thermal constraints on argillaceous host media and backfill/buffer materials. Such research could benefit any disposal concept, even those involving re-packaging.

Engineering Feasibility – Handling and packaging of large DPCs in surface facilities at the repository or at upstream installations, are within the state of available technology and current practice. Handling and packaging would be similar for any DPC direct disposal concept, no matter where the repository is located or in what geologic host medium. Thus, although engineering details need to be worked out, there appear to be no significant technical feasibility questions associated with repository operations until the waste is transported underground.

Several options exist for surface-to-underground waste package transport in shafts or ramps, including shaft hoists, funiculars, and rubber-tire or rail-mounted ramp transporters. These waste transport options are technically feasible although some systems, if implemented for DPCs, would be the largest of their kind. The choice is likely to depend on site-specific geology and local experience. Additional engineering is needed to develop systems for transport within the underground facility and for emplacement.

Criticality – Understanding the likelihood and consequences of in-package nuclear criticality for at least 10,000 years after disposal is a key issue for evaluating DPC direct disposal. Site characteristics and engineered system attributes that prevent or limit the probability of groundwater intrusion into failed waste packages are beneficial. Intrusion of brine (a possibility for the salt concept) could be inconsequential because natural ^{35}Cl is a neutron absorber. Preliminary analysis indicates that many, although not all existing DPCs could be sub-critical even if chemically and mechanically degraded in the disposal environment. Additional reactivity margin is available by using as-loaded assembly information, updated burnup credit, and taking into account groundwater salinity. With further analysis, existing DPCs can be categorized according to the potential for criticality in different disposal environments (i.e., different groundwater compositions). The consequences of criticality, conditioned on the probability of its occurrence, should also be evaluated as necessary and appropriate, as part of a complete postclosure safety analysis.

Waste Management Operational and Logistical Considerations – A waste management approach that uses DPC direct disposal to dispose of all SNF from existing or decommissioned nuclear plants in the U.S., could take longer to implement than a re-packaging approach that proceeds at a higher rate of throughput (e.g., 3,000 MTHM/yr). This is because of the decay storage time needed for DPC-based packages to cool sufficiently for disposal (e.g., cool to approximately 10 kW for emplacement in a repository in salt or hard-rock). One advantage of extended operations is that smaller capacity operating facilities could be deployed.

Re-packaging could use smaller canisters containing less SNF, to reduce the cooling time, expediting disposal. When a repository is sited the emplacement thermal power constraints will be better known by 2042 and probably sooner. At that time the potential for long decay storage times could be mitigated and throughput increased by loading bare fuel at the power plants into smaller, purpose-built canisters that can be disposed of sooner. More detailed evaluation of scenarios that compare direct disposal of existing DPCs with future re-packaging options is included in the list of R&D needs (Section 10).

The incremental cost of extended interim dry storage (for cooling DPCs) could be less than the life-cycle cost of building and operating a re-packaging facility. Depending on the repository emplacement thermal power limit, and the size of new canisters used in re-packaging, system cost savings on the order of $10B or more could be realized by not re-packaging. Note however that these statements are based on a preliminary analysis that may not account for all of the significant cost items in both scenarios being compared. For example, comparisons of incremental storage cost for DPCs vs. re-packaging cost do not address disposal costs, which go up significantly with smaller waste packages, and could potentially add tens of billions of dollars to the overall scenario compared with DPC direct disposal.

Acceptance – Once technical feasibility, safety and cost have been evaluated, it is important to communicate analysis findings, collaborate with industry, discuss safety with regulatory bodies, and promote reviews by external stakeholders. The current, ongoing feasibility evaluation represents the beginning of that process.

Preliminary Feasibility Statement – There appear to be no significant technical feasibility questions associated with repository operations (handling DPCs) at the surface, by analogy to current practices at power plants and storage sites. For transport underground, as concluded in a review of underground transport technology (Fairhurst 2012): "the method of transfer of heavy [175 MT]…loads to the subsurface might not pose an insurmountable technical constraint on siting and design of a geological repository." A significant engineering effort would be needed to develop surface handling and packaging, and underground transport and emplacement capabilities for DPC-based waste packages.

The preliminary analyses summarized above indicate that DPC direct disposal could be technically feasible, at least for certain disposal concepts. Preliminary analysis also suggests that substantial cost savings might be realized compared to re-packaging DPCs, although further analysis is needed to understand the economic consequences associated with the many possible scenarios.

Further technical and logistical analyses are needed to support a more definitive evaluation of feasibility, and this report provides a survey of topics that should be considered (Section 10). All of the DPC disposal concepts proposed here are probably not equally feasible due to limitations imposed by the geologic setting or engineered materials. Recommendations include steps to narrow the range of alternative concepts to be carried forward in the evaluation.

1. Introduction

The U.S. nuclear power industry is accumulating spent nuclear fuel (SNF) in dry storage at the rate of approximately 2,000 MT per year, at 36 sites including operating and decommissioned power plants. There currently are about 1,700 casks in use containing more than 17,000 MT of SNF (as heavy metal; Wagner et al. 2013). Projections show that by the year 2025 there will be more than 3,000 such casks in use (Figure 1-1) and that sometime before 2040 more than half of the SNF in the U.S. will be in dry storage (Hardin et al. 2013). Disposition of this SNF will become a major part of back-end fuel management strategy.

For most dry storage systems SNF is loaded and sealed into welded, stainless steel canisters which are then transferred to stationary dry storage casks. Exceptions include a few self-shielded, transportable casks that contain bare fuel assemblies. Canisters that can also be loaded into licensed transportation casks are referred to as dual-purpose canisters (DPCs). The majority of SNF in existing dry storage in the U.S. is in DPCs, and nearly all new dry storage transfers are to DPCs. These canisters typically hold as many as 32 PWR assemblies (or equivalent BWR fuel) and recent designs hold even more.

The technical objectives for direct disposal of SNF in DPCs are the same as for any geologic repository: 1) safety of workers and the public and protection of the environment; 2) respecting thermal limits for the fuel and the repository; 3) preventing or limiting criticality after waste emplacement; and 4) demonstrating engineering feasibility of underground construction and operations.

The possibility for direct disposal of existing DPCs without cutting them open and re-packaging the SNF is attractive because it could be more cost effective, reduce the complexity of fuel management operations, and result in less cumulative worker dose during custody and handling prior to eventual disposal in a geologic repository. These benefits are possible, but not proven. This report gives a technical description of some promising direct disposal concepts, proposes safety strategies for each, provides analysis of post-emplacement temperatures and the potential for nuclear criticality, and presents preliminary simulations of the timing and cost for direct disposal of DPCs containing all the SNF from commercial power plants in the U.S. It then provides a list of R&D needs to support future feasibility evaluations, which arise from discussion of the previously described work. Finally, it generates a preliminary statement of feasibility considerations at this early stage of the evaluation.

The concept of using a common canister design for storage, transport and disposal of SNF originated in the 1990's as dry-storage systems were being deployed by the U.S. utility industry. The potential advantages of standardized canisters were recognized, giving rise to multi-purpose canister (MPC) concepts developed for the U.S. Department of Energy (DOE 1994). Another study specifically addressed disposal of existing DPCs in unsaturated tuff (BSC 2003). It identified post-emplacement criticality as the most important technical issue, and that fuel burnup data from reactor operations should be used to perform criticality analyses taking into account the as-loaded configuration of each canister. Direct disposal of DPCs (up to 32-PWR size) was also examined by the Electric Power Research Institute (EPRI 2008a; 2008b). They looked at thermal and criticality issues and found no technical impediment to direct disposal for the disposal conditions that would be present in a repository in volcanic tuff, proposed at that time. More recently, a German team has proposed direct disposal of the CASTOR-V storage/transportation cask containing approximately 10 MT of SNF, in a repository in salt (Graf et al. 2012).

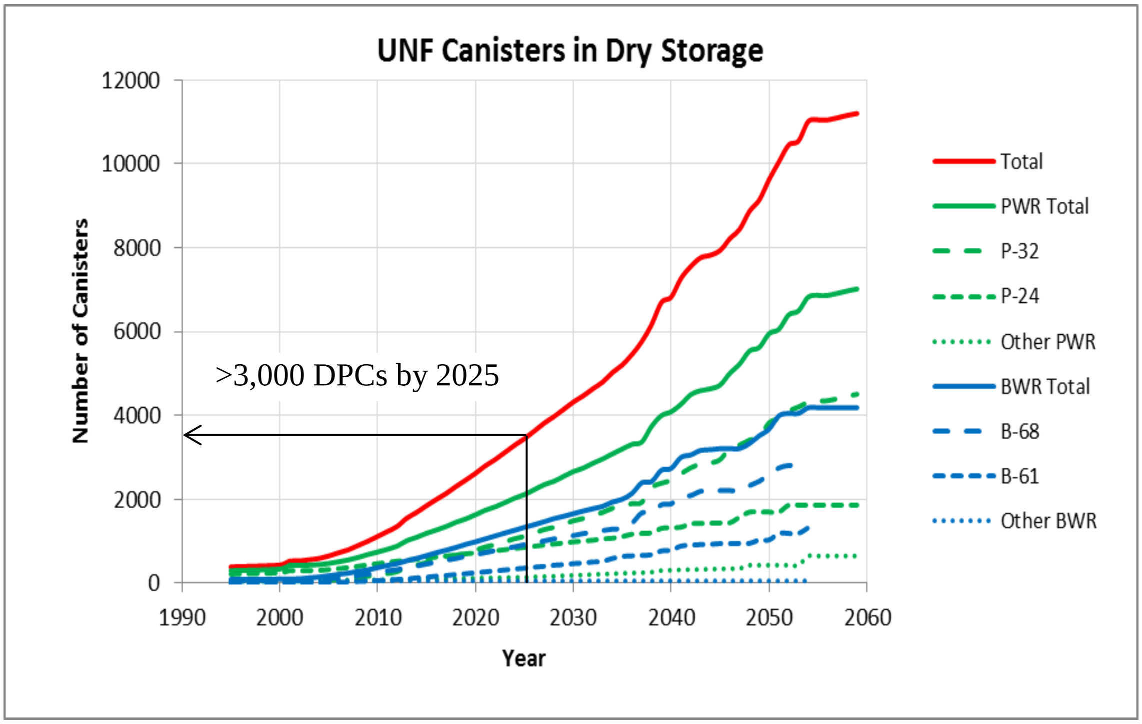


Figure 1-1. Dry storage canister projection for the U.S., using the TSL-CALVIN logistical simulator (Nutt et al. 2012) and assuming existing power reactors are operated with life-extension licenses.

The following sections describe evaluation steps undertaken in FY12 and FY13 as part of a multi-year effort (following Sections 3.1 and 3.2 of the workplan; Howard et al. 2012). The steps began with documenting a set of assumptions to focus the work (Section 2). Alternative DPC disposal concepts were then developed based on a survey of relevant technologies (Section 3) and a generic (non-site specific) analysis focused on possible engineering solutions and thermal management (Section 4). This was supplemented by additional thermal analysis (Section 5). The postclosure safety of alternative disposal concepts was approached by developing safety strategies for each (Section 6) and scoping out how generic, comparative probabilistic performance assessments could be done and what might be learned from them (Section 7). The potential for SNF criticality was analyzed for degraded DPC-based waste packages (Section 8). Finally, the potential duration and cost of DPC direct disposal were modeled (Section 9), and a set of potentially important R&D needs was identified (Section 10). A summary of these preliminary results is provided in Section 11.

References for Section 1

BSC (Bechtel-SAIC Co.) 2003. *The Potential of Using Commercial Dual-Purpose Canisters for Direct Disposal*. TDR-CRW-SE-000030 REV 00. Office of Civilian Radioactive Waste Management. Las Vegas, NV.

DOE (U.S. Department of Energy) 1994. *Multi-Purpose Canister System Evaluation*. DOE/RW-0445. Office of Civilian Radioactive Waste Management. Washington, D.C.

EPRI (Electric Power Research Institute) 2008a. *Feasibility of Direct Disposal of Dual-Purpose Canisters in a High-Level Waste Repository*. Palo Alto, CA. 1018051.

EPRI (Electric Power Research Institute) 2008b. *Feasibility of Direct Disposal of Dual-Purpose Canisters: Options for Assuring Criticality Control*. Palo Alto, CA. 1016629.

Graf, R., K.-J. Brammer and W. Filbert 2012 (in German). “Direkte Endlagerung von Transport- und Lagerbehältern - ein umsetzbares technisches Konzept.” *Jahrestagung Kerntechnik 2012*. Stuttgart. May, 2012.

Hardin, E., C. Stockman, E. Kalinina, and E.J. Bonano 2013a. “Integrating Long-Term Storage with Disposal.” *OECD/NEA International Workshop on Safety of Long Term Interim Storage Facilities*. Munich, Germany. May 21 – 23, 2013.

Howard R., J.M. Scaglione, J.C. Wagner, E. Hardin and W.M. Nutt. 2012. *Implementation Plan for the Development and Licensing of Standardized Transportation, Aging, and Disposal Canisters and the Feasibility of Direct disposal of Dual Purpose Canisters*. FCRD-UFD-2012-000106 Rev. 0. U.S. Department of Energy Fuel Cycle Technology Program, Used Fuel Disposition Campaign.

Nutt, M., E. Morris, F. Puig, E. Kalinina and S. Gillespie 2012. *Transportation Storage Logistics Model – CALVIN (TSL-CALVIN)*. FCRD-NFST-2012-000424. U.S. Department of Energy, Nuclear Fuel Storage and Transportation Planning Project. Washington, D.C. October, 2012.

Wagner, J.C., J.L. Peterson, D.E. Mueller, J.C. Gehin, A. Worrall, T. Taiwo, W.M. Nutt, M.A. Williamson, M. Todosow, R. Wigeland, W.G. Halsey, R.P. Omberg, P.N. Swift and J.T. Carter 2013. “Assessment of Used Nuclear Fuel Inventory Relative to Disposition Options.” *Proc. 14th Intl. High-Level Rad. Waste Mgmt. Conf.* Albuquerque, NM. April 28–May 2, 2013. American Nuclear Society.

THIS PAGE INTENTIONALL LEFT BLANK

2. Assumptions for Evaluating DPC Direct Disposal

Any finding of technical feasibility for DPC direct disposal will be based on targeted technical analyses, to be conducted over the course of the multi-year evaluation, based on results from R&D activities (Howard et al. 2012). Assumptions are needed to control this process because: 1) the analyses are generic (no site is specified); 2) it is recognized that statutory and regulatory changes or clarifications are required (BRC 2012); and 3) the timing of disposal is uncertain so that the future state of the overall fuel management system in the U.S. must be assumed. The basis for the following list of assumptions is a separate report (Miller et al. 2012). Note that one of the needs identified in Section 10 is to update that report as the evaluation activity proceeds.

The goal of these assumptions is to provide a common, underlying basis for targeted analyses, and not to specify how the analyses will be conducted. Assumptions categorized into three areas:

- Engineering and technology assumptions
- Statutory and regulatory framework for analysis of disposal
- Assumptions on storage and transportation that would support disposal

If and when further design or engineering activities are directed to proceed for DPC direct disposal, assumptions will be reevaluated, and any impacts on the analyses or conclusions of this report will be assessed.

2.1 Engineering and Technology Assumptions

2.1.1 DPC Characteristics

1) **DPCs contain commercial SNF.** Average burnup for existing spent nuclear fuel (SNF) in dry storage is nominally 40 GWd/MT, with a bounding value of 60 GW-d/MT for future DPCs. It is assumed that these values may be used in generalized analyses to evaluate DPC disposal (canister-specific or assembly-specific data may also used, for example in nuclear reactivity analysis).

   Basis: These burnup values are consistent with analysis and projections of Carter et al. (2012). Note that the enrichment and burnup of SNF in DPCs may be less than the bounding values reported in this source (e.g., PWR burnup may be less than 60 GW-d/MT).

2) **DPC capacity is 32 PWR assemblies or 68 BWR assemblies.**

   Basis: This is a typical value and includes most existing loaded systems although many are smaller, and larger systems have been loaded. The Magnastor® 37-PWR canister recently developed by NAC International has approximately the same dimensions and weight as typical 32-PWR systems. Heat output would be greater (for similar age and burnup) and could require longer decay storage before emplacement, depending on the selected disposal concept.

3) **Storage-only canisters can be included in the evaluations.**

   Basis: It is assumed that plant operators, vendors, or an implementing organization could develop approaches to allow these canisters to be transported to a centralized storage facility, and then a repository (or directly to a repository). Storage-only canisters currently exist at the Idaho National Laboratory, and Calvert Cliffs, Surry, Oconee, Arkansas Nuclear One, Palisades, Davis-Besse, Point Beach, Susquehanna, and H.B. Robinson nuclear power plants.

4) **DPCs designed for vertical storage can be approved for horizontal disposal.**

   Basis: It is assumed that with minor modifications, canisters designed for vertical storage (and horizontal transport) can be readily licensed and implemented to allow horizontal disposal, if this approach meets the requirements for disposal. The NUHOMS® canister systems are all designed for horizontal storage and transport (and are loaded and transferred vertically) and constitute a significant fraction of existing DPCs. Note that the 2003 BSC study (BSC 2003) raised the question of whether horizontal DPCs could be loaded vertically into disposal overpacks, meaning that additional engineering effort would be needed to establish how this would be done.

5) **Existing canisters can be analyzed for uniform average enrichment, average burnup, and average age for the assemblies contained.**

   Basis: This simplifying assumption avoids the complication of nonuniform thermal loading within canisters, for thermal analysis. With package surface temperatures of 200°C or lower, there is margin available to meet internal package temperature limits (subject to further analysis).

6) **Residual moisture in sealed DPCs can be estimated from the drying procedures specified in license documents.**

   Basis: Regulatory precedent. Drying standards and requirements provide a basis for residual water content.

2.1.2 Disposal Concepts

1) **Surface decay storage of DPCs and storage-only canisters is assumed for up to 100 years (out-of-reactor).**

   Basis: This assumption is used in thermal analysis. It implies that storage cask licenses can be extended to 100 yr, and that the DPCs can be transported after such storage. Extended storage (and transportation) to 100 years or longer may be possible, and this is the subject of ongoing R&D in the Used Fuel Disposition (UFD) campaign.

   The 100-year time frame of this assumption is generally consistent with the "No Action Alternative" considered in an Environmental Impact Statement (EIS) for a geologic repository previously analyzed in the U.S. (DOE 2002). The EIS assumed that storage facilities including the existing DPCs would be completely replaced in 100 years and possibly every 100 years afterward.

2) **Open emplacement modes (Hardin et al. 2012) are limited to 50 years of repository operation after emplacement of the last waste package.**

   Basis: The combined durations of surface storage and repository operation will not be evaluated beyond 150 yr, to limit any additional assumptions about long-term stability of institutions responsible for waste management.

3) **Near-field EBS and host rock temperatures will be used to evaluate thermal loading of the repository and repository performance, without specifying limits a priori.**

Basis: Near- and far-field temperature limits have been imposed previously (DOE 2008). Where such temperature limits are constraining, this study will treat them as targets, and evaluate what can be gained by relaxing them (e.g., allowing peak temperature greater than 100°C for clay-based materials). This approach implies that scientific understanding will increase and support such new thermal criteria.

4) **Underground handling and transport of DPCs (and all waste packages) will be shielded.**

   Basis: Shielded transporters and handling equipment substantially decrease the risk of accidental worker exposure, and are typical for disposal concepts being investigated world-wide.

5) **Disposal mode may be shielded (e.g., by borehole emplacement) or unshielded (e.g., in-drift emplacement).**

   Basis: Both shielded and unshielded modes continue to be investigated internationally, and have been investigated by previous studies in the U.S.

2.1.3 Criticality Analysis

1) **Low probability justifications for postclosure criticality may be used to exclude the criticality scenario class, by including burnup credit, and geologic media specific, fuel assembly specific, and cask specific characteristics in the analysis.**

   Basis: Past studies have identified situations where burnup credit and more detailed modeling (principal isotopes, BSC 2003; more complete isotopics, EPRI 2008) is needed in DPC disposal analysis. It is assumed that these refinements will allow use of low-probability screening arguments for some, if not all DPCs.

2) **Consequence analysis may also be used to show that criticality after disposal may be excluded on low consequence, i.e., that estimates of postclosure waste isolation performance are within acceptable limits.**

   Basis: Previous studies (e.g., Rechard et al. 1996; Mohanty et al. 2004) have shown that criticality events may not significantly change assessments of postclosure repository performance. The possibility of using criticality consequence analysis in addition to probability assessment, to evaluate risk, was included in a previously approved Topical Report (DOE 1998).

3) **Reactor operating records can be used for selecting more realistic modeling parameters to characterize the discharge isotopic composition and residual reactivity levels associated with SNF**

   Basis: Numerous studies (e.g., Wagner and Sanders 2003) have examined the impact of depletion and criticality analysis assumptions which suggest that a considerable amount of uncredited margin is incorporated into most cask loadings. Reducing uncertainty associated with parameter selection and calculating more realistic safety margins will enable a higher percentage of SNF to satisfy sub-criticality requirements.

2.1.4 Surface Facilities

1) **Canisters will be sealed at the reactors or at a centralized storage facility and will not be reopened at the repository as part of the disposal concept.**

Basis: Canister remediation options that involve re-opening the canister (even if fuel is not removed), such as addition of a filler material, may be feasible but at this time they are out of scope for this evaluation. Research on waste form conditioning internal to canisters may be conducted in the future (see R&D needs in Section 10). Note that all DPCs and storage-only canisters are not the same and only some may be disposable, such that the rest would need to be re-packaged for disposal.

2) **Surface facility throughput will be sufficient to dispose of all nominally storage-only canisters and DPCs at minimum age/burnup.**

   Basis: Surface facilities can be readily designed, constructed and operated to handle and package DPCs for disposal. Such facilities could be similar in scope, and with the same throughput rate, as facilities previously designed in the U.S. to package transportation-aging-disposal (TAD) canisters (DOE 2008).

3) **Any necessary DPC inspection can be done remotely in a hot cell without opening canisters or re-packaging.**

   Basis: Inspections may be required to confirm the condition of canisters prior to packaging and emplacement, to protect workers, and to conform to other related requirements as applicable. It is assumed that such inspections can be done externally without access to the canister interior or its contents.

2.2 Statutory and Regulatory Framework for Disposal Analysis

The generic health standard, 40CFR191, for mined geologic disposal first promulgated by the Environmental Protection Agency (EPA) in 1985 is still in force, and could, in concept be applied to future repositories. However, the evolution in the strategy adopted by the EPA and NRC in the site-specific regulations for a repository in tuff, 40CFR197 and 10CFR63, would likely be adopted with changes, for a future repository.

2.2.1 Statutory Framework

1) **Future repositories will be regulated by the NRC, implementing requirements of the National Environmental Policy Act (NEPA), and performance standards promulgated by the EPA.**

   Basis: These conditions are required by current statutes.

2.2.2 Regulatory Framework

In general, the regulatory framework controlled and implemented by EPA and NRC will be similar to existing site-specific regulations (40CFR197 and 10CFR63).

1) **Expected peak mean annual dose to a reasonably maximally exposed individual (RMEI) at the boundary of the accessible environment will be the primary measure of individual dose.**

   The peak mean annual dose measure would be calculated for two time periods: a limit of 0.15 mSv/yr before 10^4 yr, and 1 mSv/yr for the mean of simulations beyond 10^4 yr through the period of geologic stability, or approximately 10^6 yr.

2) **The accessible environment for performance assessment of DPC disposal will be at least 5 km away from the boundary of the repository**

Basis: 10CFR63.302.

3) **The NRC requirement for retrievability will remain similar:**

   "…the geologic repository operations area must be designed so that any or all of the emplaced waste could be retrieved on a reasonable schedule starting at any time up to 50 years after waste emplacement operations are initiated, unless a different time period is approved or specified by the Commission." (§63.111[e])

4) **In general, features, events, and processes (FEPs) and scenario classes will be retained or omitted based on their influence on performance in the first 10^4 yr (§63.114).**

   The criterion for screening FEPs and scenario classes based on probability will remain at 10^{-8} in any one year. Seismic and climate change effects will be projected beyond 10^4 years (§63.342).

5) **Lead, chromium or other materials used in fabrication of DPCs is part of waste packaging that will not be subject to regulation under the Resource Conservation and Recovery Act (RCRA).**

6) **NRC requirements for barriers of the disposal system will remain similar.**

   In particular: "Licensee must identify components of the disposal system that are important for isolation and demonstrate their performance. No subsystem containment requirements will be specified (§63.115)."

7) **Inadvertent human intrusion assessment will be included as an independent, conditional dose calculation.**

   A stylized calculation of individual dose to the RMEI would be assessed, conditioned on the human intrusion. The dose pathway will be limited to groundwater (or to airborne transport if significant). Dose to the drilling crew responsible for intruding will not be evaluated (§63.321).

8) **The circumstances of human intrusion will be similar.**

   A stylized calculation would be specified such that a human intrusion event would occur when a single well bypasses a portion of the natural barrier system vertically above and/or below the repository, but that the remainder of the natural barrier in the horizontal direction to the accessible environment would be retained (§63.321).

2.3 Assumptions for Storage and Transportation

The condition of DPCs or storage-only canisters during storage and transportation establish initial conditions for disposal. Other limits on storage and transportation such as permitted durations or age of SNF, also interface with disposal.

2.3.1 Storage

1) **Parts 71 and 72 will be substantially unchanged.**

   Basis: It is assumed that further licensing activities will proceed to allow transport of commercial SNF in DPCs (or existing storage-only canisters) for up to 100 years from reactor discharge, in accord with Assumption 2.1.2(1) above. The influence of shorter and long storage durations will be evaluated in sensitivity studies.

2.3.2 Transportation

1) **Transportation casks for all existing and future DPCs, and storage-only canisters, will be developed and licensed for use in transporting SNF to a centralized storage facility, and from there to the repository.**

   Basis: The availability of approved infrastructure for transporting DPCs to the repository is beyond the scope of this study.

2.3.3 Movement from Storage

1) **DPCs or storage-only canisters can be selected for transport to the repository using various strategies, including oldest fuel first (OFF) and youngest-fuel-first (YFF), and variations thereof.**

   Basis: Selection strategy could be important for a disposal system that can accommodate DPCs, particularly for thermal analysis. Also, once fuel is stored in a centralized facility, selection can be optimized for disposal and other fuel management priorities.

References for Section 2

BRC (Blue Ribbon Commission on America's Nuclear Future) 2012. *Report to the Secretary of Energy*. January, 2012. (www.brc.gov)

BSC (Bechtel-SAIC Company) 2003. *The Potential of using Commercial Dual Purpose Canisters for Direct Disposal*. TDR-CRW-SE-000030, Rev 0. Las Vegas, NV: Bechtel SAIC Company.

Carter, J., A. Luptak, J. Gastelum, C. Stockman and A. Miller 2012. *Fuel Cycle Potential Waste Inventory for Disposition.* FCR&D-USED-2010-000031 Rev. 5. July, 2012. U.S. Department of Energy, Used Fuel Disposition R&D Campaign.

DOE (U.S. Department of Energy) 1998. *Disposal Criticality Analysis Methodology Topical Report*. YMP/TR-004Q Rev. 0. Office of Civilian Radioactive Waste Management. November, 1998.

DOE (U.S. Department of Energy). 2002. *Final Environmental Impact Statement for a Geologic Repository for Disposal of Spent Nuclear Fuel and High-Level Radioactive Waste at Yucca Mountain, Nye County, Nevada*. DOE/EIS-0250F. Washington, DC: U.S. Department of Energy, Office of Civilian Radioactive Waste Management.

DOE (U.S. Department of Energy). 2008. *Yucca Mountain Repository License Application for Construction Authorization*. U.S. Department of Energy.

EPRI (Electric Power Research Institute) 2008. *Feasibility of Direct Disposal of Dual-Purpose Canisters: Options for Assuring Criticality Control*. Palo Alto, CA: Electric Power Research Institute. #1016629. December, 2008.

Hardin, E., J.A. Blink, H.R. Greenberg, M. Sutton, M. Fratoni, J.T. Carter, M. Dupont and R. Howard. 2012. *Disposal Concepts/Thermal Load Management (FY11/12 Summary Report)*. FCRD-UFD-2012-00219 Rev. 0. Idaho Falls, ID: U.S. Department of Energy Fuel Cycle Technology Program, Used Fuel Disposition Campaign.

Miller, A., R. Rechard, E. Hardin and R. Howard 2012. *Assumptions for Evaluating Feasibility of Direct Geologic Disposal of Existing Dual-Purpose Canisters*. FCRD-UFD-2012-000352 Rev. 0. September, 2012. U.S. Department of Energy, Used Fuel Disposition R&D Campaign.

Mohanty, S., R. Codell, J. Menchaca, R. Janetzke, M. Smith, P. LaPlante, M. Rahimi and A. Lozano 2004. *System-Level Performance Assessment of the Proposed Repository at Yucca Mountain Using the TPA Version 4.1 Code. Revision 2.* Center for Nuclear Waste Regulatory Analyses. San Antonio, TX. March, 2004.

NRC (U.S. Nuclear Regulatory Commission) 2001. 10CFRParts 2, 19, 20, 21, etc.: Disposal of High-Level Radioactive Wastes in a Proposed Geological Repository at Yucca Mountain, Nevada; Final Rule. Federal Register 2001;66(213):55732-816. U.S. Nuclear Regulatory Commission.

NWPA (Nuclear Waste Policy Act of 1982). 1983. Public Law 97-425. (96 Stat. 2201; 42 U.S.C. 10101 et. seq.).

Rechard, R.P., M.S. Tierney, L.C. Sanchez and M.-A. Martell. 1996. *Consideration of Criticality when Directly Disposing Highly Enriched Spent Nuclear Fuel in Unsaturated Tuff: Bounding Estimates*. SAND96-0866. Albuquerque, NM: Sandia National Laboratories.

Wagner, J.C., and C.E. Sanders. 2003. *Assessment of Reactivity Margins and Loading Curves for PWR Burnup Credit Cask Analyses*, NUREG/CR-6800, ORNL/TM-2002/6. Oak Ridge, TN: Oak Ridge National Laboratory.

THIS PAGE INTENTIONALLY LEFT BLANK

3. Technology Survey and Canister Configurations

This section outlines the technologies associated with existing dual-purpose canisters (DPCs) for storage and transportation of nuclear fuel, and with the geologic disposal of spent nuclear fuel (SNF) with possible application to direct disposal of DPCs. The DPC designation results when canisters can be both stored and transported, if placed in suitable storage or transportation overpacks.

3.1 Technologies Used in Existing DPCs

Standard materials of construction for DPC shells, closure lids and welds are the stainless steels SS-304L and SS-316. They are relatively inexpensive, have good fabrication and weldability, and are resistant to corrosion. Each DPC is an assembly typically consisting of a canister shell, baseplate, lids, fuel basket, and vent and drain assemblies. The following discussion focuses on DPCs, although there a few storage-only casks in use for dry storage at nuclear plants. It also focuses on canisters with welded closures, although there some bolted closure casks in use (including most of the storage-only designs).

3.1.1 DPC Construction

Dual-purpose canisters typically consist of a thin-walled (up to approximately 15 mm) stainless steel shell fabricated from cold-rolled sheet joined by welding, with an integral basket to hold SNF assemblies. Canisters are typically sealed, with a welded shell, welded to a bottom plate and two lids (see Figure 25 from Rigby 2010). Thick metal shield plugs are also incorporated as lids on the top, and also on the bottom for certain canisters (e.g., NUHOMS designs; see Figure 3 from Miller et al. 2012).

Material, Fabrication and Treatment – DPCs are generally fabricated using through-penetrating metal-inert-gas welds. Materials and fabrication methods are selected for reasonable cost and lifetimes (e.g., up to 100 years, more or less depending on environmental conditions). The plates comprising the canister shell are cold-rolled during fabrication and are not annealed, nor is the basket or the weld-affected zones throughout the shell and basket.

Carbon steel has also been used for DPC fabrication, and in some cases continues to be used, with coatings (e.g., zinc-based anodic coating) to prevent corrosion during loading in fuel pools and subsequent dewatering. Components such as baskets, inner lids, and shield plugs may use carbon steel. Only a minority of existing DPCs makes use of carbon steel in these ways, and the applications constitute only a small fraction of the material used (the shell is potentially most important to containment, and is usually made from stainless steel). In addition, aluminum-based neutron absorber materials are used as discussed below.

Stress corrosion cracking (SCC) in metals can occur from the combined effect of tensile stress and the presence of aqueous chloride. A bulk liquid phase need not be present because films of water are present due to sorption of moisture from the air. A series of natural exposure and accelerated corrosion tests of stainless steel fuel canisters was conducted by the Central Research Institute of Electric Power Industry (CRIEPI) in Japan. One set of experimental tests on types 304L and 316 stainless steel yielded SCC initiation times ranging from 1.6 to 3 years under extreme natural exposure conditions. Pitting or crevice corrosion was found to trigger SCC, which initiated at the same locations. The SCC penetration rate varied from 0.04 to 0.6 mm/year over a range of residual tensile stress, corresponding to penetration times of 25 to 375 years for a thickness of 15 mm, after canister temperature cools to the point where moisture films can occur (Rigby 2010).

Storage-only canisters or storage casks for bare fuel assemblies mostly have bolted closures (Rigby 2010). Metal seals with inner and outer rings are typically used, and are subject to leakage from misplacement or aging. Such systems comprise less than 10% of all canisters in dry storage, and will comprise a smaller percentage in the future as the number of welded canisters increases.

3.1.2 Internal Structure

Basket – The basket is a key DPC component that provides structural support, criticality control, and the primary heat transfer path for the fuel assemblies (Figure 3-1). The basket structure maintains spacing between assemblies, supports neutron absorber elements, prevents configurations that have not been analyzed for nuclear criticality, and transfers heat from the fuel assemblies to the canister shell. The basket as described by BSC (2003) may be designed using gridded longitudinal plates, or as an array of square tubes (one for each fuel assembly). For both configurations, spacer plates oriented perpendicular to the canister axis are typically used to hold the box in place. Some spacer plates may be thermal shunts, such as the aluminum plates in the TSC-24 design (Section 8).

Fuel basket designs utilize neutron absorber to maintain subcritical configuration. Fixed neutron absorbers are installed in most DPCs to reduce system nuclear reactivity and increase the flexibility of the system for storage and transportation of a wider array of fuels. Most fixed neutron absorbers used in domestic DPCs are composed of B_4C mixed with aluminum. Neutron absorber materials are discussed in more detail below.

Lid Assembly – The lid assembly typically consists of a primary lid or shield plug fixed by a closure ring, and one or two additional welded lids (Figure 3-1). The primary lid typically contains two stainless steel penetration ports open to the DPC interior. These penetrations are used for dewatering and gas purging as discussed below. They are sealed by welding, prior to installation of the next lid. Shield plug materials include stainless steel, coated carbon steel, and lead or depleted uranium encased in stainless steel.

Trunnions, Rings and Skirts – SNF canisters are often lifted and handled using fixtures attached to the outside, or recessed in the outer wall. These may take the form of lugs, trunnions, rings or skirts. Lifting lugs are attached to the canister lid or shell (Figure 3-2). Trunnions are cylindrical attachments used for lifting or pivoting. Lifting rings are affixed to the circumference of the shell. Canisters may be designed so that these fixtures are removable, or recessed so that the overall diameter of the canister is compatible with insertion in a cylindrical overpack. Skirts are created by recessing the top and bottom plates (lids), leaving a short length of the canister wall exposed at each end. All methods require specialized hoisting fixtures or mechanisms. Lugs and skirts typically protrude from the ends and are accommodated by additional overpack length. Trunnions and lifting rings that protrude from the canister wall could complicate overpack design (or require additional radial clearance).

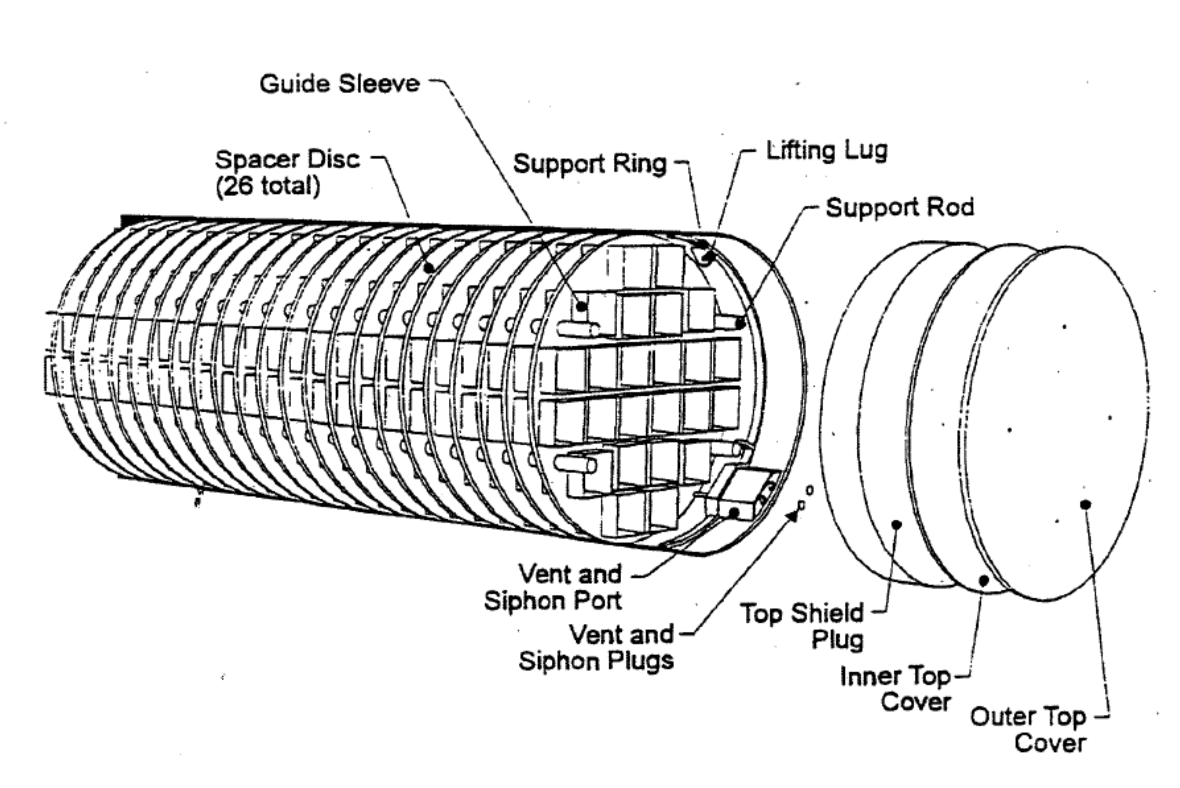
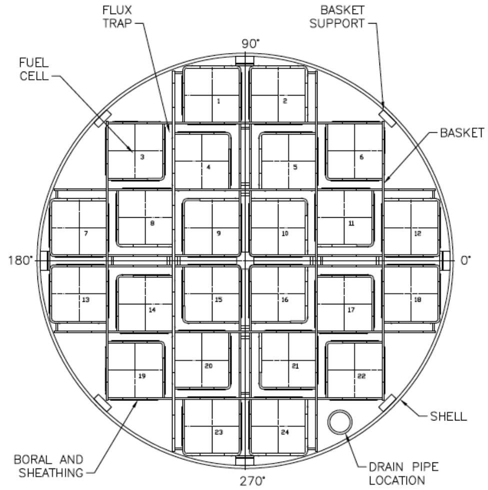
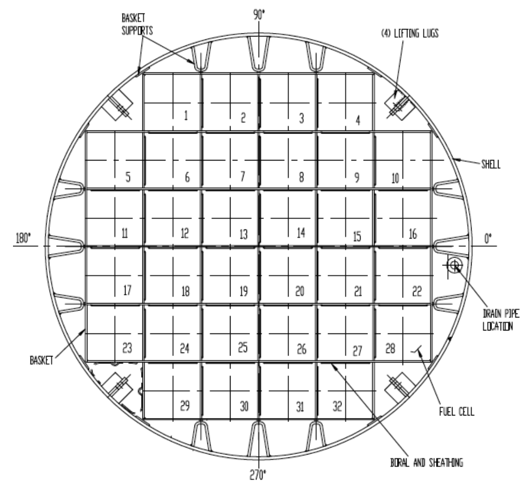


Figure 3-1. Dry shielded canister configuration (Transnuclear, Inc.) depicting the guide sleeve (fuel tube), spacer disc, top lid and vent and siphon (drain) port (from BSC 2003)

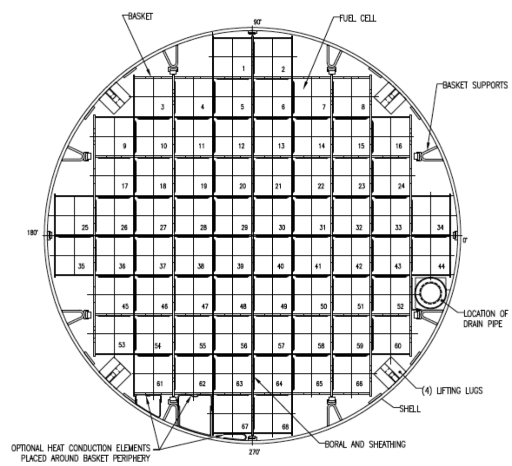
Many of the existing DPCs are NUHOMS® systems from Transnuclear Inc., which are loaded vertically with fuel assemblies, then transferred and stored in horizontal orientation. These canisters do not have external fittings for lifting, but are slid directly into transfer, storage and transportation casks that do have such fittings. For disposal of these DPCs, it is likely that disposal overpacks could be loaded horizontally (by analogy to loading transfer casks), then oriented vertically as needed for welding, then rotated back to horizontal for transport underground and emplacement in disposal drifts (see assumptions in Section 2).



MPC-24/24E/24EF



MPC-32



MPC-68

Figure 3-2. Cross sectional views of different DPC basket designs

3.1.3 Heat Dissipation

Dual-purpose canisters are designed to maintain fuel temperature at established limits during storage and transportation, and during dewatering operations (NRC 2003). Fuel cladding is sensitive to temperature exposure (peak and duration), and the UO_2 fuel itself is sensitive under certain environmental conditions (e.g., presence of air or moisture).

Heat transfer within DPCs includes radiation between fuel rods and the basket, conduction and convection in the fill gas, and conduction in the basket. DPCs are fabricated mostly from stainless steel (a relatively poor conductor of heat) which is suitable for immersion in fuel pools, exhibits little corrosion over the canister service life (preclosure), and is lower in cost than alternatives. Conductive dissipation can be improved by including thermal shunts in the basket design that increase heat transfer to the DPC shell. Heat transfer in the fill gas can be improved by using helium (the most conductive of readily available, chemically inert gases) and increasing the gas pressure (some canister designs call for several bars of gas pressure).

3.1.4 Criticality Control

Traditional methods of demonstrating criticality control for DPCs include the use of neutron absorbers that are incorporated into the basket, the use of geometry control (i.e. separation of fissile materials), and the use of moderator displacement and exclusion. Neutron absorber materials are incorporated as separate absorber plates affixed to the fuel basket, or they can be integrated into the metallic basket material. This section discusses technologies currently employed to reduce the nuclear reactivity of used fuel in DPCs.

Neutron Absorber Plates – Fixed neutron absorbers are installed in most DPCs to reduce system reactivity and increase the flexibility of the system for storage and transportation of fuel with a wider range of characteristics. Fixed neutron absorbers reduce the number of neutrons transmitted between basket cells. The majority of domestic DPC designs use sheets of a metal matrix material consisting of aluminum mixed with boron carbide (B_4C). Typical B_4C mixtures used in DPCs consist of either Boral® or a group of materials called metal matrix composites (MMCs). Boral has been the most common, having been used by vendors NAC International, Transnuclear, and Holtec International. It comprises B_4C particles and aluminum Alloy 1100 hot rolled together to form a neutron absorbing core. The core is then bonded to two outer skins of Alloy 1100 aluminum. Boral has a service history dating to the 1950s and has been used extensively for fuel storage in both fuel pools and DPCs. It has exhibited some blistering and bulging, primarily associated with earlier generations of the product and DPC drying operations.

Another category of metal matrix neutron absorber is metal matrix composites (MMCs), which are manufactured by extrusion and hot rolling of a fully dense cylindrical billet of aluminum and boron carbide produced using powder metallurgy techniques. Various alloys of aluminum can be used depending on whether the MMC absorber component has a structural function. Metamic® neutron absorber falls in the MMC category and has been used in spent fuel racks that contain degraded Boraflex®. Holtec International offers Metamic and Metamic-HT (High Temperature) neutron absorber in their spent fuel racks and dry storage casks. Some of the Holtec licensed fuel baskets are made entirely of Metamic. Additional details on neutron absorber materials such as material properties, characteristics, known degradation phenomena, and the current state of technology are available (EPRI 2009).

Flux Traps – The simplest and oldest method of controlling criticality in fuel storage systems is to physically separate the fuel assemblies from one another so that a smaller fraction of the neutrons being generated in one assembly is capable of arriving at an adjacent assembly. The principal method of ensuring separation of the fuel assemblies is to place them in storage cells which are separated from each other as is shown in Figure 3-1. The upper left portion of Figure 3-2 depicts a flux trap design with spaces between the neutron absorber panels separating each cell. In reactivity analyses for transportation accident conditions, these spaces are filled with fresh water. (Without flooding the reactivity is always subcritical.) The flux trap works to thermalize neutrons so that the absorber material is more effective at decreasing neutron communications between fuel assemblies. The upper right portion of the figure depicts a non-flux trap design as does the bottom image. In the non-flux trap design, a single neutron absorber panel is interstitial to adjacent basket cells.

Use of the flux trap designs was more common in early canister designs such as the TSC-24, the MPC-24, and the NUHOMS-24 series. The use of flux traps in conjunction with neutron absorber panels in those designs allowed the vendors to demonstrate subcriticality of fresh fuel, for higher enrichments (e.g., 4.2%). As burnup credit became an option for analysis supporting licensing of larger DPCs, non-flux trap designs such as the MPC-32 and the NUHOMS-32 series have become more common.

Control Rods – Regulatory guidance (NRC 2002) indicates that materials that are positioned or operated within the envelope of the fuel assembly during reactor operation may be approved for storage in the DPC. This includes items such as discharged control rods and burnable poison rods. These components can be inserted into the guide tubes of the fuel assemblies when they are placed into the DPCs. In some instances stainless steel rods have been inserted for moderator displacement.

Most control rods are removed from reactor service as they reach the end of their rated service life. Their life may be limited not by depletion, but by fretting on the upper portion of the control rod cladding, which does not impact structural integrity. Moreover, the absorber material is so effective that a small amount of depletion would not reduce their impact on reactivity. Hence, used control rods could be useful to control reactivity in certain fuel types and configurations, when inserted as DPCs are loaded.

All PWR reactor vendors have produced guide tube based control components. These are constructed of either B_4C or Ag-In-Cd and are used for shutdown of the core and for occasional power distribution control. In addition, some use axial power shaping rods (APSRs) composed of Inconel. Like discrete burnable absorbers, these components are often stored in the fuel assembly guide tubes when DPCs are loaded.

Thimble Plug Devices – These include guide tube plugs, orifice rod assemblies and similar devices with different names. Thimble plug devices (TPDs) are not utilized in all the assemblies in a reactor core, however, they are reused cycle to cycle. Except for the Westinghouse 14x14 water displacement guide tube plugs, TPDs extend only into the plenum region of the fuel assembly, so negative reactivity credit is not expected from these devices. Like burnable absorber rods, TPDs can be loaded with assemblies into any basket locations in DPCs.

Burnable Poison Rods – These are used in reactor operations to suppress power production in fresh fuel assemblies and reduce core wide reactivity. Burnable absorbers are generically classified into two categories, integral and discrete. Integral absorbers typically consist of either a rare-earth absorber such as Gd_2O_3 or Er_2O_3 which has been mixed into the fuel pellet, or a ZrB_2 coating which sprayed on the outside of the fuel pellet. Discrete absorbers are loaded into the guide tubes of the fuel assembly and typically consist of either B_2O_3 clad in stainless steel or a matrix of Al_2O_3-B_4C clad in zirconium alloy.

Characterization of the residual activity of integral or discrete burnable poisons is difficult and the credit gained for postclosure criticality analysis would generally be small due to depletion of the poison material. Neglecting neutron absorption leaves only the water displacement properties of control rods, which is the most important property in current nuclear reactivity analyses.

3.1.5 Dewatering Features

Most used fuel packaging is loaded under water (or borated water) in fuel pools. Drain ports are provided for removal of water after containers are closed. They are commonly located in the top lid and connected to an internal drain or "siphon" tube that extends to the bottom of the canister so that plugging and sealing operations can be conducted from above. A plug is provided to close each drain port, and for canisters closed by welding, such plugs are typically covered with a welded cover plate. The plugs are required to maintain containment isolation under normal and accident conditions.

In addition most canisters are equipped with a vent port. The port is used for transferring gas into or out of the canister during draining or filling with water, respectively. Purging with dry gas to remove residual moisture is done through this port. Some canisters may require the displacement of interior gas to remove or install the lid. Like drain ports, a plug is provided to close the vent, and may be sealed with a welded cover plate.

Each cell or tube within a canister basket butts tightly against the bottom plate, so cutouts ("mouse holes") are typically provided in the basket to provide pathways for water to drain.

3.2 DPC Operational Procedures

Although the DPCs have specific operational procedures depending on the DPC design, many of the fuel loading steps are similar. Following are the typical steps used to load DPCs.

Handling empty DPCs

- DPC is lifted by crane and placed into a transfer cask cavity using the DPC's lifting lugs. The annulus between the DPC outer shell and the transfer cask is then filled with demineralized water. Typically an inflatable annulus seal is installed in the upper end of the annular region between the DPC and transfer cask to prevent spent fuel pool water from contaminating the exterior of the DPC. DPC interior is also filled with either demineralized water or spent fuel pool water in accordance with the predefined loading procedures.
- The transfer cask containing an empty DPC is lifted by crane using the lift yoke and lowered into the spent nuclear fuel pool in the cask-loading pit.

Fuel loading operation

- Pre-selected assemblies are loaded in accordance with the DPC-specific Certificate of Compliance following a visual inspection to verify the assembly serial number or other identifications.

Removal of loaded DPC from the spent fuel pool

- The loaded transfer cask is raised to the pool surface after installing the DPC lid inside the spent fuel pool. The exterior surface of the transfer cask is sprayed with demineralized water while raising the cask from the pool.
- The transfer cask is placed in the designated cask preparation area.
- Decontamination of the transfer cask is then performed.
- The inflatable annulus seal is removed and if required an annulus shield is installed. The annulus shield provides additional radiation protection to the workers during the welding and drying operations. Other supplemental shielding may be installed as required by the procedure.

Dewatering

- At first the drain line is connected to the transfer cask to remove water from the annular region until the water is approximately 10 inches below the top of the DPC.
- The welding system is installed. A water pump is connected to the DPC drain line and DPC water level is slightly lowered to aid the welding operations.
- DPC closure lid is then welded to the DPC cylindrical shell in accordance with the procedures. Specific DPC design may require purging the space below the DPC lid with inert gas prior to and during welding of the DPC lid.
- Visual and dye-penetrant examinations are performed on the root and final passes of the welding following the procedures.
- Water is drained from the DPC cavity. After removing the bulk water, the remaining moisture from the DPC cavity is removed using either a vacuum drying system or a forced helium drying system in accordance with the procedures.
- DPC cavity is then backfilled with helium to provide inert atmosphere and enhance heat transfer.
- Cover plates are then installed and welded over the DPC vent and drain ports and dye penetrant examinations are performed followed by the leak testing of the cover plate welds.
- The DPC closure ring/top cover (also called structural lid) is then installed and welded.
- The transfer cask lid is installed after removing the remaining water from the transfer cask.

The DPC is then transferred from the transfer cask to the storage or transportation overpack using a transfer adapter.

3.3 Licensed DPC Systems

Comprehensive lists have been prepared of NRC licensed canisters that have been and continued to be loaded (Miller et al. 2012; JAI Corp. 2005; Greene et al. 2013). Table 3-1 presents the approved canister lists (mainly from JAI Corporation). Note that Table 3-1 may not present all the NRC-licensed DPCs to-date.

Table 3-1. List of approved canisters

Canister Class	Canister Type	Vendor	Capacity	Licensing Status
W21M-LD	S&T	Energy Solutions	21-PWR	Licensed for S&T
W21LM-LS	S&T	Energy Solutions	21-PWR	Licensed for S&T
W21M-SD	S&T	Energy Solutions	21-PWR	Licensed for S&T
W21M-SS	S&T	Energy Solutions	21-PWR	Licensed for S&T
W21T-LL	S&T	Energy Solutions	21-PWR	Licensed for S&T
W21LT-LS	S&T	Energy Solutions	21-PWR	Licensed for S&T
W21T-SL	S&T	Energy Solutions	21-PWR	Licensed for S&T
W21LT-SS	S&T	Energy Solutions	21-PWR	Licensed for S&T
W74M	S&T	Energy Solutions	64-BWR	Licensed for S&T
W74T	S&T	Energy Solutions	64-BWR	Licensed for S&T
MSB	Storage only	Energy Solutions	24-PWR	Licensed for storage
MPC-24	S&T	Holtec International	24-PWR	Licensed for S&T
MPC-24E/24EF	S&T	Holtec International	24-PWR	Licensed for S&T
MPC-32	S&T	Holtec International	32-PWR	Licensed for S&T
MPC-32F	S&T	Holtec International	32-PWR	Licensed for storage
MPC-68/68F	S&T	Holtec International	68-BWR	Licensed for S&T
MPC-68FF/68M	S&T	Holtec International	68-BWR	Licensed for storage
MPC-37	S&T	Holtec International	37-PWR	Licensed for storage
MPC-89	S&T	Holtec International	89-BWR	Licensed for storage
Yankee-MPC	S&T	NAC International	36-PWR	Licensed for S&T
CY-MPC	S&T	NAC International	24-PWR	Licensed for S&T
CY-MPC	S&T	NAC International	26-PWR	Licensed for S&T
UMS-Class 1	S&T	NAC International	24-PWR	Licensed for S&T
UMS-Class 2	S&T	NAC International	24-PWR	Licensed for S&T
UMS-Class 3	S&T	NAC International	24-PWR	Licensed for S&T
UMS-Class 4	S&T	NAC International	56-BWR	Licensed for S&T
UMS-Class 5	S&T	NAC International	56-BWR	Licensed for S&T
Magnastor PWR TSC	S&T	NAC International	37-PWR	Licensed for S&T
Magnastor BWR TSC	S&T	NAC International	87-PWR	Licensed for S&T
NUHOMS-24PS	Storage only	Transnuclear	24-PWR	Licensed for storage
NUHOMS-24PL	Storage only	Transnuclear	24-PWR	Licensed for storage
NUHOMS-24PHBS	Storage only	Transnuclear	24-PWR	Licensed for storage
NUHOMS-24PHBL	Storage only	Transnuclear	24-PWR	Licensed for storage
NUHOMS-24PTH-S	S&T	Transnuclear	24-PWR	
NUHOMS-24PTH-L	S&T	Transnuclear	24-PWR	
NUHOMS-24PTH-LC	S&T	Transnuclear	24-PWR	
NUHOMS-24PT-2S	S&T	Transnuclear	24-PWR	Licensed for storage
NUHOMS-24PT-2L	S&T	Transnuclear	24-PWR	Licensed for storage
NUHOMS-32PT-S100	S&T	Transnuclear	32-PWR	Licensed for storage
NUHOMS-32PT-S125	S&T	Transnuclear	32-PWR	Licensed for storage
NUHOMS-32PT-L100	S&T	Transnuclear	32-PWR	Licensed for storage
NUHOMS-32PT-L125	S&T	Transnuclear	32-PWR	Licensed for storage
NUHOMS-32PTH	S&T	Transnuclear	32-PWR	Licensed for storage
NUHOMS-FO	Transport only	Transnuclear	24-PWR	Licensed for transport
NUHOMS-FC	Transport only	Transnuclear	24-PWR	Licensed for transport
NUHOMS-FF	Transport only	Transnuclear	13-PWR	Licensed for transport
NUHOMS-24PT1	S&T	Transnuclear	24-PWR	Licensed for S&T
NUHOMS-52B	Storage only	Transnuclear	52-BWR	Licensed for storage
NUHOMS-61BT	S&T	Transnuclear	61-BWR	Licensed for S&T
NUHOMS-12T	S&T	Transnuclear	12 cans debris	Licensed for storage
NUHOMS-07P	Storage only	Transnuclear	7-PWR	Licensed for storage

3.4 Technologies External to DPCs Available for Handling and Disposal

3.4.1 Disposal Overpack

Disposal overpacks have been proposed for repository projects in the U.S. and internationally and can be single- or multi-layered and fabricated from a variety of materials. One corrosion resistant overpack consisted of a structural layer of stainless steel enclosed by an outer corrosion-resistant layer of nickel alloy (DOE 2008). Disposal overpacks generally provide structural support, and they provide corrosion allowance or corrosion resistant containment performance depending on the safety strategy (Section 6).

Disposal overpacks provide economical means to meet different requirements such as heat dissipation, impact damage limits, and corrosion lifetime.

Disposal overpacks of carbon steel and stainless steel have been proposed for previous repository projects in the U.S. (e.g., ONWI 1987a,b). Overpack materials that have been selected by repository programs internationally include carbon steel, type 316 stainless steel, copper, and titanium. All of these were considered in disposal concept development for the Used Fuel Disposition R&D program (Hardin et al. 2012). Carbon steel corrosion occurs by relatively well understood, general corrosion mechanisms making it suitable as a "corrosion allowance" material in applications where waste containment is required for only a few hundred to a few thousand years (DOE 1998). Thick-walled carbon steel overpacks could facilitate waste handling and ensure package integrity during repository operations. For longer term corrosion resistance, materials such as copper and passive alloys of nickel and titanium have been selected previously (DOE 2008). Passive materials (e.g., stainless steel and nickel alloys) may have low rates of general corrosion in the disposal environment, but are subject to localized corrosion at greater penetration rates.

Corrosion Resistant and Corrosion Allowance Materials – Corrosion resistant materials such as titanium, nickel-chromium alloys, etc., can provide long containment lifetimes for waste packages, on the order of 10^4 to 10^6 years (where containment is defined as no breach of any kind). Corrosion resistant materials generally are passive and subject to various modes of localized corrosion (pitting, stress-corrosion cracking, crevice corrosion, etc.). Localized corrosion produces only small penetrations, but penetration rates are far faster than general corrosion in these materials.

By contrast, corrosion allowance materials such as copper and low-alloy steel are not subject to localized corrosion in repository applications. However, general corrosion is faster than for corrosion resistant materials, and waste package penetration may occur in 10^3 to 10^5 years, especially for oxidizing conditions. Corrosion allowance materials perform better in reducing chemical environments, but are associated with minimum corrosion rates in the presence of water. Corrosion allowance materials can be used to protect the package contents during the period of elevated temperature, gamma radiolysis, etc. (Section 6).

Resistant Coatings – Amorphous metal and ceramic thermal spray coatings have been developed with excellent corrosion resistance and neutron absorption. These coatings, with further development, could be cost-effective options to enhance the corrosion resistance of waste packages and other engineered barrier systems (EBS) components, and to limit nuclear criticality in canisters for transportation, aging, and disposal of SNF. Iron-based amorphous metal formulations with chromium, molybdenum, and tungsten have shown corrosion resistant properties. Rare-earth additions enable very low critical cooling rates to be achieved. Ceramic coatings may provide even greater corrosion resistance for EBS applications (Plinski 1999) although the boron-containing amorphous metals are still favored for criticality control. These amorphous metal and ceramic materials have been produced as gas-atomized powders and applied as nonporous coatings with nearly full density, using the high-velocity oxy-fuel process. Blink et al. (2009) summarized the performance of coatings as corrosion-resistant barriers and as neutron absorbers, and also presented a simple cost model to quantify the economic benefits possible with these new materials.

3.4.2 Shielding

Typical DPCs have thin-walled cylindrical shells for which the primary function is to contain radioactive materials. The canister provides minimal shielding against the gamma and neutron radiation emitted by irradiated nuclear fuel.

Significant decrease in DPC external radiation levels occurs with decay storage time, particularly because gamma emission during storage, transportation and disposal operations comes from short-lived fission products (e.g., Sr-90 and Cs-137 with ~30-yr half-lives). Nevertheless, DPCs generate high radiation fields and require gamma and neutron shielding to reduce radiation levels to those required by operational radiation protection programs. Shielding of DPCs would be provided for all operations leading to final disposal, but disposal overpacks would not necessarily be shielded. Instead, transporters used for waste packages in the repository and surface facilities would be shielded.

The shielding function of the storage and transportation overpacks consists of gamma and neutron shielding. Gamma shielding materials include thick-walled steel, iron, other high-Z materials such as lead and depleted uranium, and concrete. Among the most effective neutron shielding materials are compounds with high concentrations of hydrogen such as polyethylene, polypropylene, and borated polymers.

Shielding is also provided within DPCs by internal shield plugs, which are thick plates of steel or other material that typically form the first lid. They are put into place in the fuel pool after fuel loading, before welding, to shield workers during welding and associated activities. Horizontally emplaced storage canisters such as the NUHOMS® series may have another shield plug in the bottom, integrated during canister fabrication, for additional shielding as canisters are handled in storage. Some canister types may also include a layer of neutron shielding material (see above discussion of overpacks). Internal shielding components are designed to work with fixed shielding (e.g., external shielding at welding stations) or shielded casks for transfer, storage or transportation.

DPC designs also permit regionalized loading, whereby hotter fuel assemblies are loaded in central assembly locations within the basket, than would otherwise be authorized under thermal and radiation limits under uniform loading conditions. This practice can reduce external dose substantially.

On-site transfer casks (e.g., NUHOMS OS200 and HI-TRAC developed by TransNuclear and Holtec International, respectively) are used at nuclear power plants to transfer loaded DPCs from reactor pools to storage or transportation overpacks. These casks incorporate both gamma and neutron shielding materials. An outer water jacket may be used for neutron shielding. Transfer casks feature a retractable bottom lid that allows a loaded DPC to be transferred into the storage or transportation overpack. Examples of transfer casks used with horizontal and vertical dry storage systems are shown in Figures 3-3 and 3-4, respectively.

Transfer casks for handling DPCs at the repository will be similar to those used at power plants, but different because they will: 1) handle the additional size and weight of the disposal overpack; 2) limit total weight as needed to meet transporter payload limits; 3) have shielding designed to take credit for the disposal overpack; and 4) have features designed to address conditions specific to repository application.

3.4.3 Surface-to-Underground Transport

DPC-based waste packages including disposal overpacks are expected to weigh on the order of 80 MT (Section 4). Additional shielding could weight 60 MT, by analogy to the total weight of the largest transportation casks (e.g., Magnastor® casks weigh approximately 50 MT and their transportation casks add 90 MT). In addition this 140 MT combination will require a vehicle or carriage that could weigh 20 to 35 MT, giving a total transported weight of up to 175 MT. There are several ways that such loads might be safely transported from the surface to emplacement areas underground in a repository. For this report they are divided into two categories: shafts and ramps. Note that access and transport are also needed for hauling men, materials, and waste rock, and for ventilation. A typical repository for U.S. spent fuel could have as many as a dozen access openings (Hardin et al. 2012) of which only one could be configured for waste transport. Selection of shafts or ramps for these other applications would depend on different criteria than selection for waste transport. Shafts tend to be favored, except for transport of heavy waste packages which is the focus of the following discussion.

Shafts and Heavy Hoists – Vertical shaft construction methods and hoisting systems have advantages and disadvantages to be considered when selecting the appropriate form of conveyance for waste transport (Fairhurst 2012). Hoist-actuated systems allow the power source to be located at the surface, and separated from the transported payload, which greatly reduces the risk from a potential fire. However, there are risks from hoists over-winding, or from component failure leading to free fall.

Hoists being considered for repository applications have safety features that can prevent or mitigate potential failures. Friction hoists, unlike drum hoists, use redundant cables to support the payload (as many as 8 have been proposed; see Graf et al. 2012). The resulting capacity could be more than 6 times the payload (safety factor). Other safety features include interlocks to control loading and unloading, and a hoist car with a floating floor that is latched to solid supports during loading and unloading, to stabilize the payload and balance cable loads.

Another safety feature that mitigates potential impact from over-winding accidents is the SELDA arrestor (Englemann et al. 1993). The SELDA arrestor is a type of shock absorber that can be installed at the top and/or bottom of a shaft. In the event of over-winding (which may occur in either direction, at the end of travel) the arrestor could bring the system to a stop over a distance of several meters, limiting dynamic loads. It works by the controlled deformation of a set of steel bars that run through rollers (Englemann et al. 1993), which converts kinetic energy into plastic strain that generates heat.

Source: http://www.nwtrb.gov/meetings/2012/jan/williams.pdf

Figure 3-3. NUHOMS® transfer cask being transported to a storage module

Source: LeDuc (2012).

Figure 3-4. HI-TRAC 100 transfer cask

Hoist systems require a head frame structure above the shaft collar. They are typically constructed from steel and reinforced concrete, and they can be readily designed to support loads developed by the hoist payload, plus a counterweight, plus the weight of cables and the hoist car. Other considerations for selecting shaft or ramp access are discussed for alternative disposal concepts in Section 4. Some of these considerations are specific to the geologic setting, and the ultimate choice may be influence by local experience and preference.

Ramps – Any entry into a geologic repository that is not a vertical shaft is considered a ramp in this discussion. This includes straight ramps, spiral declines, and adits (a horizontal or low-angle entry in mountainous terrain). Each system has been considered for a geologic repository somewhere in the world. The following discussion describes these alternatives and the conveyance systems that could be used for waste transport.

A spiral decline is a descending tunnel that typically circles up to 2 times down to the repository level, at a grade of up to 10%. The spiral configuration keeps the ramp portal near other repository facilities. Because of the spiral geometry, transporters must be motorized, and powered, either by diesel or electricity (battery, pantograph). Rubber-tire vehicles are preferred because the grade exceeds traction limits for rail (on the order of 2.5% or less; AAR 2008). Friction of rubber on concrete is sufficient to assure traction at 10% grades. Rubber-tire conveyances for heavy haul in industrial applications are readily available, and one was recently tested at the Äspö underground laboratory in Sweden, with a 90-MT payload (Fairhurst 2012). Conveyances of this general type can be configured with sufficient payload capacity for shielded transport of DPC-based packages. Modularity and intrinsic safety features (e.g., associated with independently controlled hydraulic drives) reduce risks from equipment failure.

A straight incline can also be used, although with a 10% grade the ramp portal could be more than 5 km from the repository (for depth of 500 m or greater). A straight incline is being considered for waste transport at the French repository site at Bure. It could have a grade of 15° (26%) and use a funicular conveyance (Fairhurst 2012). The funicular is a counterweighted rail car moved by cables connected to a hoist at the surface. Much of the load is carried by the rails, so the friction hoist capacity is much less than required for vertical lifting. Multiple brakes and other safety features would be provided.

An adit could be used in mountainous terrain allowing horizontal or gradually descending entry to the repository (e.g., DOE 2008). The principal advantage is that conventional railroad equipment could be used (grades up to 2.5%), although shielded waste transporters would be heavy and highly specialized. Runaway, fire and breakdown are still safety considerations.

As noted above other considerations for selecting shaft or ramp access are discussed in Section 4, for alternative disposal concepts in various geologic settings.

3.4.4 Underground Transport and Emplacement

Once a waste package has been transported underground to the repository, it must be transported to an emplacement drift and placed in its final disposal position. Some of the technologies that can be used to accomplish this are discussed below.

Underground Transporters – The same transporter can be used for transport from the surface, and for transport underground. Alternatively, a transfer station can be constructed underground to change conveyances. Whereas direct transport from the surface to disposal locations is efficient, a possible reason for using a transfer station is to decrease payload weight (e.g., for a shaft hoist or funicular). Another reason is to use a specialized emplacement machine as discussed below.

Repository layouts will be mostly horizontal, possibly with different levels connected by low-angle ramps. Accordingly, either rubber-tire or rail conveyance could be used. Rail systems offer more precision in locating packages for disposal, but have the disadvantages of rail installation cost, the potential for derailing, and sensitivity to rockfall. Rubber-tire systems may require installation of ballast or concrete floors (e.g., in sensitive media, see Section 4). They also are capable of colliding with walls or other equipment, and may be more complex mechanisms.

Previous design experience has shown that large amounts of steel would be needed to construct extensive repository rail networks (DOE 2008). For the large repository layouts discussed in Section 4, rubber-tire conveyance could be more cost effective.

Emplacement Machines – As noted above the same transporter could be used to haul waste packages down a ramp, then place them in disposal drifts (in-drift emplacement). An example is the transport-emplacement-vehicle (TEV) designed for a repository in unsaturated tuff (DOE 2008). For other modes of emplacement such as in vertical or horizontal boreholes or vaults (Section 4) a separate emplacement machine may be used. Such machines could incorporate gantries or jacks to hoist packages into position, or they could use sliding tracks to insert packages into boreholes. An example of the latter configuration is the deposition machine test performed at the Äspö underground laboratory in Sweden (Mützel et al. 2001).

Pallets – These are fixtures that support waste packages in the final disposal position, for in-drift emplacement modes. They facilitate unloading by providing a precise position for emplacement. For certain unbackfilled, in-drift disposal concepts they may also facilitate cooling, and elevate packages above rockfall debris for a period of time before eventual drift collapse. Pallets may be independent components or they may be permanently affixed to waste packages. They may also serve as fixtures for holding additional engineered barrier components that are installed after waste emplacement.

3.4.5 Emplacement Modes

Emplacement modes influence repository layout, construction, waste package size, and waste package handling. Emplacement modes may also influence worker safety, facility inspection and monitoring, performance confirmation, and retrieval. An important distinction among emplacement modes is whether waste packages are in direct contact with any surrounding engineered or geologic medium, particularly buffer material, backfill, or the host rock. These enclosed modes are contrasted with open modes in which packages are surrounded by connected air space that can be ventilated to remove heat. Uses of open and enclosed emplacement modes are discussed in Section 4.

Vertical and Horizontal Borehole Modes – For these enclosed modes waste packages would be emplaced in vertical boreholes drilled into the floor or the walls of access drifts. The depth of each borehole is sufficient to accommodate one or more waste packages, sealing or buffer materials, and possibly a shield plug. A liner may be installed in each borehole after characterization, except where not needed for borehole stability and/or waste package alignment. If a buffer around the waste package is part of the disposal concept, the borehole is sized accordingly. Advantages of borehole emplacement include heat transfer in highly conductive host media such as salt, and shielding to facilitate worker access after emplacement.

Disadvantages include heat transfer in low conductivity media, the cost of drilling, and the complexity of handling and potentially rotating heavy waste packages during emplacement.

In-drift Emplacement Modes – The in-drift emplacement, open mode concept consists of waste packages placed horizontally on the emplacement drift floor, either parallel or transverse to the axis of the drift. The drift remains open for cooling and inspection, and for drift maintenance, during a period of repository operations. After up to 100 years the drift may be backfilled prior to permanent closure. Advantages include relative simplicity of design and of the emplacement of large, heavy waste packages. Disadvantages include potential effects from seismic ground motion and rockfall during repository operations and after closure (for unbackfilled concepts), no shielding provided during operations by either the host rock or backfill, and the challenge of emplacing backfill in an elevated temperature radiation environment (for backfilled open concepts).

3.4.6 Buffer and Backfill Materials

Naturally derived bentonite has been a common choice for buffer material used in R&D reported in the international literature. Bentonite is actually a naturally occurring mixture of Na-montmorillonite (a smectite clay) with other clay minerals (e.g., illite), minerals such as silica and plagioclase, and organic matter. It is prepared by drying to a low moisture content (around 3% w/w), crushing, and compaction into pellets, bricks or other shapes. The material swells when rehydrated, and if compacted to sufficient dry density (1.9 to 2.0 g/cc) and confined during rehydration, it can readily attain swelling pressures of 6 MPa or greater (Pusch 1992). Under such conditions the permeability decreases to approximately 10^{-20} m^2 (hydraulic conductivity 10^{-13} m/sec) or less. Maximum water content (unconfined) is on the order of 50% depending on composition and processing.

Bentonite is used extensively for drilling mud and borehole sealing applications. Swelling behavior makes dehydrated bentonite especially useful for sealing because it expands to fill any voids. Swelling pressure is also useful to constrain heavy packages from sinking when surrounded by hydrated buffer material, and it may inhibit microbial mobility (SKB 2006, 2010, 2011). Mixtures of clay materials such as bentonite, with more indurated granular materials (e.g., crushed rock or sand) have been studied as possible backfill materials that have greater shear strength and lower cost, although with less swelling pressure and greater permeability (e.g., Pakbaz and Khayat 2004). Addition of graphite as a minor constituent to increase thermal conductivity has also been investigated (Jobmann and Buntebarth 2009).

There is no intrinsic requirement to use bentonite of the type known and sold in the U.S. Other clay-based materials could also suffice in the same applications, for example, local clay materials have been evaluated by the Swedish program (Dixon et al. 2011). Also, low permeability can be achieved with materials that potentially have better resistance to temperature than clays, but without swelling behavior (Hardin and Voegele 2013, Appendix B).

3.4.7 Water Diversion

For saturated disposal environments water is diverted from waste packages using low permeability materials as discussed above. For unsaturated conditions in free-draining host formations, water moves downward through the EBS, and additional engineered barriers can be used to divert water from waste. Drip shields made of corrosion resistant material have been proposed (DOE 2008) as have capillary barriers (Richards barrier; CRWMS M&O 1999a).

3.4.8 Pressure Management

Iron corrosion in anoxic environments produces hydrogen gas that could achieve elevated pressure limited only by the overburden stress, in low-permeability host media (e.g., salt or clay). The corrosion reactions consume water and are tied to the transport of moisture to the waste package. For host media with higher permeability the hydrogen could readily dissipate. The mechanical effects from pore pressure on the waste package might be insignificant, but the effects of gas generation on aqueous radionuclide transport are considered to be more complex, and are subject of recent investigations (described by Norris 2009).

3.4.9 Moisture Getters

Moisture is important to post-closure performance. Liquid groundwater is the principal vector for transport of released radionuclides. Moisture is a component in most corrosion reactions that degrade engineered barriers (e.g., metals, ceramics) and the spent fuel waste form. In addition, water has an important role in nuclear reactivity as a neutron moderator. In geologic host media that are fractured, or are otherwise sufficiently permeable to admit groundwater, the potential accumulation of moisture is likely to exceed the capacity of any getter. Similar findings were reached with respect to radionuclide getters, which could be overwhelmed by natural background chemistry in flowing systems (CRWMS M&O 1999b).

However, for certain host media such as salt or shale in which the availability of moisture in the disposal environment is more limited, the addition of moisture getters to the EBS could make it easier to demonstrate waste isolation and/or exclusion of FEPs such as postclosure criticality. In salt, moisture influx occurs in response to gradients of stress or temperature. Both are transient responses because brine movement is likely to slow as the repository cools and the host salt reconsolidates. Also, if brine influx occurs then fluid inclusions will become depleted in the near field. The total accumulation of moisture (as brine) at the waste package would depend on the initial moisture content of the salt and other factors. This limited moisture could partially mobilize radionuclides inside corroded packages, and could impact nuclear reactivity, unless it is incorporated into corrosion products or another getter phase outside of the waste package.

Disposal overpacks made of corrosion allowance materials such as low-alloy steel could serve as moisture getters in such low permeability, low moisture host media. The moisture capture rate could be limited by the material degradation rate (see BSC 2004). Special-purpose getter materials such as granulated metal and dehydrated clay might react at higher rates, possibly forming corrosion products with low permeability or capacity to sorb radionuclides.

3.4.10 External Neutron Poisons

Neutron poisons external to the DPC in the disposal environment may affect external reactivity (of fissile materials released from waste packages and selectively redeposited) but do not provide a means to control internal reactivity. Any external material is too many neutron mean-free-path lengths away from the fuel to be effective as either a moderator or absorber, and will only interact with neutrons that were already leaving the system.

References for Section 3

AAR (Association of American Railroads) 2008. *Performance Specification for Trains Used to Carry High-Level Radioactive Material*. Standard S-2043.

Blink, J., J. Farmer, J. Choi, and C. Saw 2009. "Applications in the nuclear industry for thermal spray amorphous metal and ceramic coatings." *Metallurgical and Materials Transactions A.* June, 2009. 40A, pp. 1344-1354.

BSC (Bechtel-SAIC Co.) 2003. *The Potential of Using Commercial Dual-Purpose Canisters for Direct Disposal*. TDR-CRW-SE-000030 REV 00. Office of Civilian Radioactive Waste Management. Las Vegas, NV.

BSC (Bechtel-SAIC Co.) 2004. *Aqueous Corrosion Rates for Waste Package Materials*. ANL-DSD-MD-000001 Rev. 1. Prepared for the U.S. Dept. of Energy, Office of Civilian Radioactive Waste Management, Las Vegas, Nevada.

CRWMS M&O (Civilian Radioactive Waste Management System Management & Operations contractor) 1999a. *License Application Design Selection Report*. B00000000-01717-4600-00123 Rev. 01 ICN 01. Las Vegas, Nevada: CRWMS M&O.

CRWMS M&O (Civilian Radioactive Waste Management System Management & Operations contractor) 1999b. *Diffusive Barrier and Getter Under Waste Packages VA Reference Design Feature Evaluations*. B00000000-01717-2200-00213 Rev. 00. Las Vegas, Nevada: CRWMS M&O.

Dixon, D., T. Sandén, E. Jonsson and J. Hansen 2011. *Backfilling of deposition tunnels: Use of bentonite pellets*. P-11-44. Swedish Nuclear Fuel and Waste Management Co. February, 2011.

DOE (U.S. Department of Energy) 1992. *Characteristics of Potential Repository Wastes, Volume 1*. DOE/RW-0184-R1. Prepared for the U.S. Department of Energy, Office of Civilian Radioactive Waste Management, Washington, D.*C.*

EPRI (Electric Power Research Institute) 2009. Handbook of Neutron Absorber Materials for Spent Nuclear Fuel Transportation and Storage Applications – 2009 Edition. 1019110. Palo Alto, CA. November, 2009.

Graf, R., K.-J. Brammer and W. Filbert 2012 (in German). "Direkte Endlagerung von Transport- und Lagerbehältern - ein umsetzbares technisches Konzept." *Jahrestagung Kerntechnik 2012*, Stuttgart, May, 2012.

Greene, S.R., J.S. Medford and S.A. Macy 2013. *Storage and Transport Cask Data for Used Commercial Nuclear Fuel – 2013 U.S. Edition.* ATI-TR-13047. Energx, Oak Ridge, TN, and Advanced Technology Insights, LLC, Knoxville, TN.

Hardin, E. and M. Voegele 2013. *Alternative Concepts for Direct Disposal of Dual Purpose Canisters*. FCRD-UFD-2013-000102, Rev.0. U.S. Department of Energy, Used Fuel Disposition R&D Campaign.

JAI Corp. 2005. Shipping and Storage Cask Data for Commercial Spent Nuclear Fuel. Fairfax, VA. March, 2005.

Jobmann, M. and G. Buntebarth 2009. “Influence of Graphite and Quartz Addition on the Thermo-Physical Properties of Bentonite for Sealing Heat-Generating Radioactive Waste.” *Applied Clay Science*. 44 (2009) pp. 206-210.

LeDuc, D.R. 2012. *Dry Storage of Used Fuel Transition to Transport*. FCRD-UFD-2012-000253. August, 2012. U.S. Department of Energy, Used Fuel Disposition R&D Campaign.

McConnell, J.W., Jr., A.L. Ayers and Jr., M.J. Tyacke 1996. *Classification of Transportation Packaging and Dry Spent Fuel Storage System Components According to Importance to Safety*. NUREG/CR-6407. U.S. Nuclear Regulatory Commission. Washington, D.C. February, 1996.

Miller, A., R. Rechard, E. Hardin and R. Howard 2012. *Assumptions for Evaluating Feasibility of Direct Geologic Disposal of Existing Dual-Purpose Canisters*. FCRD-UFD-2012-000352 Rev. 0. September, 2012. U.S. Department of Energy, Used Fuel Disposition R&D Campaign.

Mützel, W., E. Huth, R. Brausam, J. Keller, S. Pettersson, R. Bäck and H. Jendenius 2001. *Demonstration deposition machine for canisters: Description and experience from design and test operation*. IPR-01-38. Äspö Hard Rock Laboratory. Swedish Nuclear Fuel and Waste Management Co. July, 2001.

Norris, S. (ed). 2009. *Summary of Gas Generation and Migration: Current State of the Art*. Report D1.2R. Fate of Repository Gases (FORGE) Project. European Commission Work Package FP7. 176 pp.

NRC (U.S. Nuclear Regulatory Commission) 2002. *Interim Staff Guidance - 9, Revision 1, Storage of Components Associated with Fuel Assemblies.* Spent Fuel Project Office. April, 2002.

NRC (U.S. Nuclear Regulatory Commission) 2003. *Interim Staff Guidance - 11, Revision 3. Cladding Considerations for the Transportation and Storage of Spent Fuel.* ISG-11, Rev. 3. U.S. Nuclear Regulatory Commission. Washington, D.C.

NRC (U.S. Nuclear Regulatory Commission) 2011. *Standard Review Plan for Renewal of Spent Fuel Dry Cask Storage System Licenses and Certificates of Compliance – Final Report*. NUREG-1927. U.S. Nuclear Regulatory Commission, Washington, D.C.

NWTRB (Nuclear Waste Technical Review Board) 2010. *Evaluation of the Technical Basis for Extended Dry Storage and Transportation of Used Nuclear Fuel*. December, 2010. (www.nwtrb.gov)

ONWI (Office of Nuclear Waste Isolation) 1987a. *Waste package/repository impact study: Final report*. BMI/ONWI/C-312. (OSTI ID: 6718042). Columbus, OH: Office of Nuclear Waste Isolation, Battle Memorial Institute. June, 1987.

ONWI (Office of Nuclear Waste Isolation) 1987b. *Conceptual designs for waste packages for horizontal or vertical emplacement in a repository in salt for reference in the site characterization plan.* BMI/ONWI/C-145 (OSTI ID: 6915229). Columbus, OH: Office of Nuclear Waste Isolation, Battle Memorial Institute. June, 1987.

Pakbaz, M.C. and N. Khayat 2004. “The Effect of Sand on Strength of Mixtures of Bentonite-Sand.” *Engineering Geology for Infrastructure Planning in Europe: Lecture Notes in Earth Sciences*. V. 104. pp. 316-320.

Plinski, M.J. 1999. *LADS Feature report - Ceramic Coatings*. B00000000-01717-2200-00208 Rev 00. Las Vegas, Nevada: CRWMS M&O. March, 1999.

Rigby, D.B. 2010. *Evaluation of the Technical Basis for Extended Dry Storage and Transportation of Used Nuclear Fuel*. U.S. Nuclear Waste Technical Review Board. December, 2010. (www.nwtrb.gov)

SKB (Swedish Nuclear Fuel and Waste Management Co.) 2006. Long-term safety for KBS-3 repositories at Forsmark and Laxemar — A first evaluation. Technical Report TR-06-09.

SKB (Swedish Nuclear Fuel and Waste Management Co.) 2010. *Buffer, backfill and closure process report for the safety assessment SR-Site.* Technical Report TR-10-47.

SKB (Swedish Nuclear Fuel and Waste Management Co.) 2011. *Long-term safety for the final repository for spent nuclear fuel at Forsmark: Main report of the SR-Site project, Volume I*. Technical Report TR-11-01.

4. Alternative DPC Direct Disposal Concepts

This section develops a set of alternative concepts with reasonable prospects for providing safe and feasible direct disposal of spent nuclear fuel (SNF) in dual-purpose canisters (DPCs). The list of concepts presented here is not exhaustive, but it is sufficient for input to generic performance analyses such as mechanical damage to engineered barriers, postclosure criticality, and preclosure and postclosure safety analyses. The results draw heavily from previous concept development (Hardin et al. 2012), but with emphasis on DPC-size waste packages and thermal management. The concepts presented here are not exclusive to disposal of DPC-based waste packages, but may apply to other geologic disposal missions as well. The section also briefly discusses some concepts that were identified previously for smaller, cooler waste packages, but are not suitable for direct disposal of DPCs.

The alternative concepts are presented in numbered subsections. The numbering scheme is consistent with previous work (Hardin et al. 2012). The concepts are based on information accumulated from technical literature including repository conceptual design reports, and from a working group of technical staff from the Used Fuel Disposition (UFD) R&D campaign. The concepts may continue to evolve in distinction or detail, as future feasibility evaluation activities proceed.

Most international high-level waste (HLW) and SNF disposal programs are focused on enclosed modes in crystalline or clay-based host rock types, with inherent limits on heat generation and SNF capacity for waste packages. Assuming current understanding of thermal constraints on host rock media and engineered buffer and backfill materials, the salt concept and the open, unbackfilled emplacement modes are best suited for disposal of DPC-based packages with heat output on the order of 10 kW at emplacement. A disposal solution using larger packages is attractive for the U.S., which currently faces the disposition of more than twice as much SNF as any other nation without the means of reprocessing it. These ideas are explored further in this section, including description of enclosed and open emplacement modes, and thermal analysis that compares enclosed vs. open concepts, backfilled and unbackfilled concepts, and different host media.

Repository design attributes for discussion in describing alternative concepts include:

- **Host medium characteristics –** Geomechanical, geochemical and hydrologic characteristics that are important for defining the disposal concept.
- **Excavation and ground support –** Open-mode concepts could involve approximately 50 to 100 years of repository operations during which the drifts would be kept open for ventilation. Rock characteristics that determine long-term opening stability may also affect groundwater movement and other aspects of the disposal environment.
- **Waste package and EBS functions and materials –** Disposal concepts may include corrosion resistant waste packaging intended to ensure long-term containment (e.g., up to 10^6 years) or they may use simpler packaging limited to near-term functions.
- **EBS dimensions –** The size and weight of DPC-based waste packages are important for disposal operations and other aspects of repository performance.
- **Waste handling shaft or ramp access –** Surface-to-underground movement of large, heavy waste packages in shielded transporters could in principle be done using either

shafts or ramps. The rationale could depend on site-specific factors, and could also depend on local experience and preference. However, certain geologic settings are more amenable to one solution or the other, and these relationships are discussed.

- **Postclosure criticality control features –** DPCs would not be opened prior to disposal, so the criticality control features they contain cannot be changed (see Section 2). Some of these features will likely degrade after closure in the disposal environment. Note that long-term structural integrity of the spent fuel basket, on the order of 10^5 years, could decrease the probability of criticality as indicated in Section 8.
- **Thermal performance and schedule for operation and closure –** Thermally driven processes can degrade natural and engineered barriers, particularly clay-based engineered materials, and host media that exhibit temperature sensitivity. Heat output is a major difference between DPC direct disposal and concepts involving smaller packages.
- **Repository plan area –** Included because alternative disposal concepts and thermal management strategies may differ significantly with respect to how much area would be needed.

These attributes are discussed in the numbered subsections below. For additional introduction, the following paragraphs outline some basic concepts, including the approach to opening stability and ground support.

Enclosed vs. Open Concepts – Geologic disposal concepts are readily divided into "enclosed" and "open" modes of waste package emplacement (Hardin et al. 2012). The enclosed modes involve emplacing packages directly into contact with engineered materials or host rock that have temperature limits (called "targets" in this report). By contrast, the open modes maintain connected air space around each package that can be ventilated to remove heat prior to permanent closure of the repository. These spaces may remain open and continue to enhance heat dissipation after closure. Open emplacement concepts combine the functions of surface decay storage (i.e., in fuel pools or dry storage) with geologic disposal in the same underground facility. An open-concept repository can be operated much sooner than enclosed concepts that require surface decay storage of 100 years or longer (Hardin et al. 2012). Earlier emplacement of SNF waste could allow much of the disposal cost to be incurred sooner, potentially before currently operating nuclear power plants are shut down (and while Nuclear Waste Fund fees are still being collected).

Overpacks and Waste Package Dimensions – Disposal overpacks would be used with all disposal concepts. A range of overpacks could be designed to accommodate the range of DPCs, which differ with respect to dimensions and external handling features. Overpacks would be the interface between DPCs and other elements of the disposal system such as transporters, emplacement equipment, other engineered barriers, and the host rock. DPCs are typically constructed from relatively thin stainless steel plate, and overpacks could provide additional, robust mechanical strength for handling and transport, and for repository closure operations such as backfilling. The weight of fully loaded DPCs ranges up to approximately 50 MT depending on capacity, and a 5-cm thick steel overpack would add another 30 MT. The overall diameter of waste packages containing DPCs would be approximately 2 m or less, with length of 5 m or slightly more depending on fuel type.

Robust overpacks could also help to ensure containment for a range of potential accidents or disruptive events during repository operations. Overpacks made from low-alloy steel with wall thickness of a few centimeters have been proposed for various disposal concepts (Hansen et al. 2010; Sevougian et al. 2012). The materials used and the methods for fabrication and treatment would be selected for performance in the disposal environment, as has been demonstrated for a range of disposal concepts internationally (DOE 2008; SKB 2011; Andra 2005).

Thermal Analysis – Thermal analysis presented in this section were generated using an analytical superposition solution described by Greenberg et al. (2012) implemented in software as described by Sutton et al. (2011) and Hardin (2013). The same approach was used in previous studies of disposal concepts and thermal load management (Hardin et al. 2011; 2012). The one exception is thermal analysis of disposal concepts for salt (Section 4.2) which was done using the finite element method in order to accommodate emplacement geometry and changes in crushed salt backfill (Hardin et al. 2012). Further description of thermal analysis methods and results is provided in Section 5.

Opening Size, Excavation and Ground Support – Disposal concepts described here would require excavations that accommodate waste transport and handling, emplacement, monitoring, and repository closure. Some concepts would involve up to 100 years of ventilation, then backfilling, so consideration of long-term opening stability for such concepts is important.

Empirical ground support estimation methods that are widely accepted in the mining industry are used to understand the types of ground support that might be needed. Such methods provide little direct information on the time for which excavations could remain stable, but they do describe the stand-up time, defined as how long excavations remain stable if left unsupported. Current approaches to underground construction such as the New Austrian Tunneling Method, have built on the stand-up time concept. Modern underground support design typically considers that emplacement of ground support should occur as soon as possible after excavation to prevent rock movements that could result, in the long run, in greater support requirements.

The size of underground openings is key to understanding long-term stability and ground support requirements. For this report the sizes of underground openings are estimated based on handling of large DPC-size waste packages for the various emplacement modes. The discussion is less important for the enclosed emplacement modes (Sections 4.1 through 4.3) because the openings are filled with backfill or buffer material soon after emplacement. DPC-based package dimensions are somewhat larger than waste packages considered in earlier design work (BSC 2008a). International programs have developed a range of concepts for waste package handling, and corresponding drift size requirements (e.g., Andra 2005).

For those emplacement modes that involve either vertical borehole emplacement, horizontal borehole emplacement, or in-drift transverse emplacement, allowance is made for maneuvering waste packages into place. Typically, this means that the critical emplacement dimension is the diagonal length of the package. Allowing for additional thickness for the waste package, the critical dimension is therefore on the order of 7 to 8 m (Table 4-1). The other dimension of the excavation would be approximately 5 m to accommodate the width or height of the emplacement system.

Circular openings could be excavated by a tunnel boring machine, while more rectangular openings may best be excavated by the drill-and-blast method using pre-splitting to limit damage to the rock mass. If the compressive strength of the rock is low enough, excavation could be accomplished by a road-header type excavator. The dimensions discussed in this section are summarized in Table 4-1. Note that dimensions given in Table 4-1 are for drifts requiring ground support, either access drifts (for emplacement in boreholes or lined borings), or emplacement drifts (for in-drift emplacement). For in-drift emplacement in crystalline rock, hard rock, or sedimentary rock, a circular cross section is indicated for long-term stability. For access drifts in any medium, which are readily maintained, circular or rectangular openings could be used. For in-drift disposal in less competent sedimentary media a smaller drift diameter could be selected to enhance stability. In salt, all drifts and alcoves could be rectangular to conform to host salt stratigraphy and promote reconsolidation. For access drifts or emplacement drifts in the cavern-retrievable concept, a larger opening could be selected to accommodate heavy hauling equipment used to transport storage or disposal casks.

For evaluating opening stability and ground support in this report, an excavation diameter (circular) or span (rectangular) of 7 m is used throughout. The Rock Mass Rating (RMR), Rock Quality Designation (RQD) and "Q" Rock Mass Classification methods chosen to estimate the rock mass stability and support requirements (Hardin and Voegele 2013, Appendix A) are not especially sensitive to slight changes in the excavation dimensions. All concepts are assumed to be constructed at a depth of approximately 500 m, although shallower emplacement could be effective in some geologic settings, and could be necessary for unsaturated settings (Hardin et al. 2012).

4.1 Crystalline Rock Enclosed Emplacement Concepts

Although not recommended for DPC direct disposal, this group of concepts is included here for completeness. They rely on waste package integrity and a swelling-clay based buffer installed at emplacement, which can be implemented in vertical or horizontal emplacement boreholes or as in-drift emplacement. Access drifts would be backfilled with low-permeability engineered material prior to closure. The engineered barriers would include the buffer and waste package, and the host rock could serve as an additional barrier especially if it has low permeability (e.g., $<10^{-16}$ m^2) and/or chemically reducing conditions. These concepts are important because the vertical borehole emplacement variant is under development in both Sweden and Finland. These concepts could be used in saturated or unsaturated geologic settings (see Hardin and Sassani 2010 for analysis of clay buffer performance in unsaturated settings). With multiple engineered barriers, for example a long-lived waste package made from copper or titanium, these concepts could also be used in more permeable host media (EPRI 2010, Appendix B). Enclosed concepts in crystalline rock (or hard rock, including igneous and metamorphic types) are not well suited to DPC direct disposal for thermal reasons as discussed below.

Constructability is generally good in crystalline (e.g., granite) or other competent hard rock media in which stable openings can be readily constructed with spans on the order of 10 m or larger (e.g., at tunnel intersections). These media will be fractured from the effects of crystallization (i.e., cooling), subsequent tectonic loading, and excavation. Borehole variations (Hardin and Voegele 2013) could be used to emplace and isolate waste packages beyond the disturbed rock zone (DRZ), to avoid induced, interconnected fractures that could form preferential pathways for transport of groundwater and released radionuclides.

A rock mass selected for geologic disposal application would likely be massive (uniform, relatively unfractured) with high compressive strength. The spacing of fractures or other discontinuities could be large, and rock quality could be high. Infrequent joints could be present, but could be very tight and rough, with no infilling or weathering. Groundwater inflow would be relatively low with occasional water bearing fractures, so that excavations would be nearly dry even in the saturated zone. Because of the small number of discontinuities, emplacement drifts could be oriented favorably with respect to the fracture system. As a result, the Rock Mass Rating could be in the very good range or perhaps the high end of the good range. For such a rock mass the stand-up time for 7-m excavations would be on the order of 10 years. Stability would require little support other than occasional rock bolting. If the rock mass were of somewhat lower quality (good rock category) additional support might consist of rock bolts and wire mesh for support.

Similarly, using the "Q" Rock Mass Classification System (Hardin and Voegele 2013, Appendix A) the RQD could be excellent and the joint set number could correlate to a massive body with few if any joints. The joint roughness number could be determined by discontinuous joints or irregular undulating jointing, and the joint alteration number could reflect fresh joint walls. The expected case of dry excavations or minor water inflow would be reflected in the joint water reduction factor, and the stress reduction factor could be appropriate for a medium stress, a favorable situation. Using an excavation support ratio number of 0.8 the estimated support category using the "Q" Rock Mass Classification System would be in the unsupported or no support category, with at most occasional rock bolts needed.

Other aspects of crystalline enclosed concepts such as thermal response, construction methods and materials, shaft or ramp access, waste packaging, prefabrication, and postclosure criticality are discussed elsewhere (Hardin et al. 2012; Hardin and Voegele 2013). Borehole emplacement (horizontal or vertical) and in-drift emplacement concepts have been described. However, these are poorly suited to DPC direct disposal because many hundreds of years of decay storage would be needed before emplacement. While host rock peak temperature could readily be limited to 200°C (a typical limit used to avoid extensive micro-cracking; Hardin et al. 2012), without protracted aging the buffer peak temperature could be much higher than 100°C (and greater than the 200°C peak rock temperature). Thermal performance of backfill is discussed further for backfilled concepts (Sections 4.3, 4.5.1 and 4.6.2).

Table 4-1. Dimensions of emplacement or access drifts requiring ground support, for disposal concepts presented in this report.

Concept (Section #)	Critical Waste Package Dimension	Long-Term Ventilation Required	Approximate Excavation Shape
Crystalline rock, enclosed, swelling clay-based buffer			
4.1.1 Crystalline enclosed (vertical borehole emplacement)	Diagonal	(enclosed, not possible)	Vertical rectangle (7 m high x 5 m wide)
4.1.2 Crystalline enclosed (horizontal borehole emplacement)			Horizontal rectangle (5 m high x 7 m wide)
4.1.3 Crystalline enclosed (in-drift emplacement)	Diameter		Circular (5.5-m diameter)
Salt concept			
4.2.1 Horizontal in-alcove or in-drift transverse emplacement	Diagonal or Diameter	(enclosed, not possible)	Horizontal rectangle (5 m high x 7 m wide)
4.2.2 Borehole vertical emplacement	Diagonal		Vertical rectangle (7 m high x 5 m wide)
Clay/shale, enclosed			
4.3 Clay/shale enclosed	Diameter	(enclosed, not possible)	Horizontal rectangle (5 m high x 7 m wide)
Sedimentary, unbackfilled open			
4.4.1 Sedimentary unbackfilled, low-temperature	Diameter	✓	Circular (4.5 m diameter)
4.4.2 Sedimentary unbackfilled, high-temperature	Diameter	✓	Circular (4.5 m diameter)
Sedimentary backfilled open			
4.5.1 Sedimentary backfilled open	Diameter	✓	Circular (4.5 m diameter)
Hard-rock, open emplacement			
4.6.1 Hard-rock, unsaturated, unbackfilled open	Diameter	✓	Circular (5.5 m diameter)
4.6.2 Hard-rock, backfilled open	Diameter	✓	Circular (5.5 m diameter)
Cavern-retrievable			
4.7.1 Surface storage systems (shielded) in underground galleries	Diagonal	✓	Vertical rectangle (8 m high x 6 m wide)
4.7.2 Purpose-built, shielded, ventilated storage/disposal casks (vaults)	Diagonal	✓	Vertical rectangle (8 m high x 6 m wide)

4.2 Salt Disposal Concepts

A salt disposal concept that was originally proposed for heat-generating HLW glass (Carter et al. 2011) and extended to SNF (Hardin et al. 2012) is incorporated as an alternative in this report. All drifts could be excavated using boom-type road headers. Floors could be bare rock as proven feasible at salt mines and the Waste Isolation Pilot Project (WIPP) for rubber-tire equipment. Ground support could consist only of rock bolts in traffic areas. In each variation of this concept packages would be enclosed by intact salt or crushed salt backfill at the time of emplacement. Access drifts and other service openings would be backfilled with crushed salt prior to closure. Because the salt disposal concept alternatives are enclosed modes, there could be little radiological risk to workers performing repository closure operations once the emplacement alcoves, drifts or boreholes were backfilled.

For DPC disposal in salt, the excavation and ground support methods would likely be very similar to those used for the WIPP. Excavations at the WIPP generally require only rock bolts for support, and may be as large as 8 m wide by 4 m high, with wider spans at intersections. Experience has shown that it is possible to maintain these large openings until waste has been emplaced, and the rooms are backfilled and closed.

Both bedded and domal salt could be suited for DPC disposal. Both have been proposed previously for SNF disposal (e.g., the former Salt Repository Project in the U.S., and the German facility at Gorleben). The water content of domal salt is much lower, typically 0.5% (w/w) or less, compared to 1 to 3% for bedded salt (Roedder and Chou 1982). This difference affects the magnitude of potential brine migration in response to mechanical loading and heating. However, both bedded and domal salt formations have limited quantity and mobility of brine, and water-borne radionuclide releases under nominal, undisturbed conditions would be insignificant. Accumulation of brine at waste packages might conceivably be important in analysis of postclosure criticality, but approximately 75% of natural chlorine is ^{35}Cl, a thermal neutron absorber (see Section 8).

Waste package overpacks could consist of low-alloy steel (or nodular cast iron, etc.) to maintain integrity throughout repository operations, and for a period of time after emplacement. The minimum time could be on the order of 50 years to facilitate retrieval as required by current regulation (§10CFR60.111(b)). The overpack could be made of corrosion allowance material, and robust to withstand mechanical loading by salt creep during this period. Because moisture is scarce in the salt disposal environment, corrosion of such an overpack may be limited so that containment integrity is maintained for hundreds or thousands of years. This performance question is a potential area for future investigation.

For DPC disposal in salt, repository layouts could be designed for simplicity and to spread out heat-generating packages. No ventilation of waste packages would be possible after emplacement and backfilling, other than to maintain the filled panels at lower ventilation pressure than other areas where repository construction and operations are underway. The benefit from ventilating nearby dedicated drifts for heat removal was analyzed and found to be limited, comparable to a few years additional decay storage prior to emplacement (Section 5.2).

Waste packages could be handled underground in the horizontal orientation to limit the necessary height of excavations (more important in bedded salt, which is likely to have a limited disposal interval thickness). To limit package handling operations underground, they could also be transported from the surface in a horizontal orientation. Transport of DPC-based waste packages would require a ramp, or shaft hoist such as those used and tested at Gorleben, scaled up to sufficient payload capacity (e.g., 175 MT). The cost and performance of such hoists is an area of ongoing engineering analysis. Shaft hoist designs have been developed and tested with payload capacity up to approximately 85 MT (a full-scale test of hoist equipment and safety features for 85 MT capacity was performed at Gorleben, Germany). For DPC disposal the payload capacity would be up to 175 MT, depending on DPC size and how much additional shielding is used around the loaded canister and disposal overpack. A similar capacity has been proposed by the German program for the DIREGT direct disposal concept (Graf et al. 2012).

Vertical shafts could generally be used to access a repository for DPC disposal in salt, although this depends on site-dependent stratigraphy, rock characteristics and hydrology. Shafts are favored over ramps because the geometry tends to minimize the excavation area exposed to any overlying, water bearing strata. This facilitates construction and eventual sealing of shaft openings when the repository is closed. In addition, strata in sedimentary basins where bedded or domal salt is found may be poorly indurated (i.e., not lithified and exhibiting low compressive and tensile strengths) complicating ground support during construction, maintenance and closure.

Shaft access would be used for initial construction, operation, and ventilation of any geologic disposal facility in salt, but is not required for waste transport. Consideration of ramps is motivated by uncertainty as to the feasibility of vertically lowering (or lifting if necessary) large heavy waste packages, compared with conveying them down ramps on wheeled transporters. Ramps are common within salt or potash mines to access different levels, so the stability of ramp openings within evaporite sequences is not a major concern. For domal salt settings a waste handling ramp could be constructed well outside the dome structure, with horizontal access to the host formation, possibly at different levels. Thus, the most important seals associated with a ramp might be located well away from waste emplacement areas, mitigating the postclosure risk associated with groundwater inflow.

A similar approach to ramp design, construction and sealing could be taken for a repository in bedded salt. Access to the stratigraphic interval for disposal could be provided by a ramp located at a significant distance from emplacement areas. Postclosure risk would be mitigated by sealing a long horizontal drift within the repository salt layer, as well as backfilling, plugging, and sealing the ramp itself. Heavy rubber-tire transporters can be safely used in ramps with up to 10% grade, as demonstrated by testing of a 24-tire Cometto® transporter with 90-MT payload, in the spiral decline at the Äspö underground laboratory (Fairhurst 2012). This transporter technology (including others such as Wheelift® systems) is modular so that nominal payloads are essentially unlimited. Another option is to build a steep funicular railway (e.g., linear, with a 15° or 27% slope) with sufficient hoist capacity and safety features to accommodate DPC packages, thereby shortening the ramp length by a factor of 2 or more. A recent summary of approaches for waste transport being considered by repository R&D programs worldwide (Fairhurst 2012) concluded that "…the method of transfer of 'heavy' [of the order of 135 to 200 MT]…loads to the subsurface might not pose an insurmountable technical constraint on siting and design of a geological repository."

Importantly, site-specific factors are likely to constrain or eliminate possibilities for shaft or ramp construction, for example, the existence of an aquifer in the geologic section above the host salt formation could favor vertical shafts for all mined accesses to the repository. Cost considerations could also be important (de la Vergne 2003).

Salt has unique thermal properties that facilitate disposal of larger, hotter waste packages. Thermal conductivity is high (5.2 W/m-K for WIPP salt at ambient temperature and 3.2 W/m-K at 200°C) and salt can tolerate peak temperatures of 200°C or greater (Hardin et al. 2012). Finite-element thermal analysis of the disposal of large packages (32-PWR) in salt was reported previously (Hardin et al. 2012; Jove-Colon et al. 2012). The results in Table 4-2 were obtained for a range of package sizes and decay periods, with packages spaced 20 m apart on a grid, using an updated model for crushed salt that incorporates thermally activated creep. The model does not include the effect of moisture in the backfill, which accelerates reconsolidation and further lowers peak temperatures (Jove-Colon et al. 2012). Thermal analyses for the alcove and in-drift emplacement variations, with 32-PWR size packages spaced 30 m apart and containing high-burnup SNF, are compared as temperature histories in Figure 4-1. Both variations meet the 200°C peak salt temperature target for salt, and both can be closed well within the time frame assumed in Section 2 (150 years out-of-reactor).

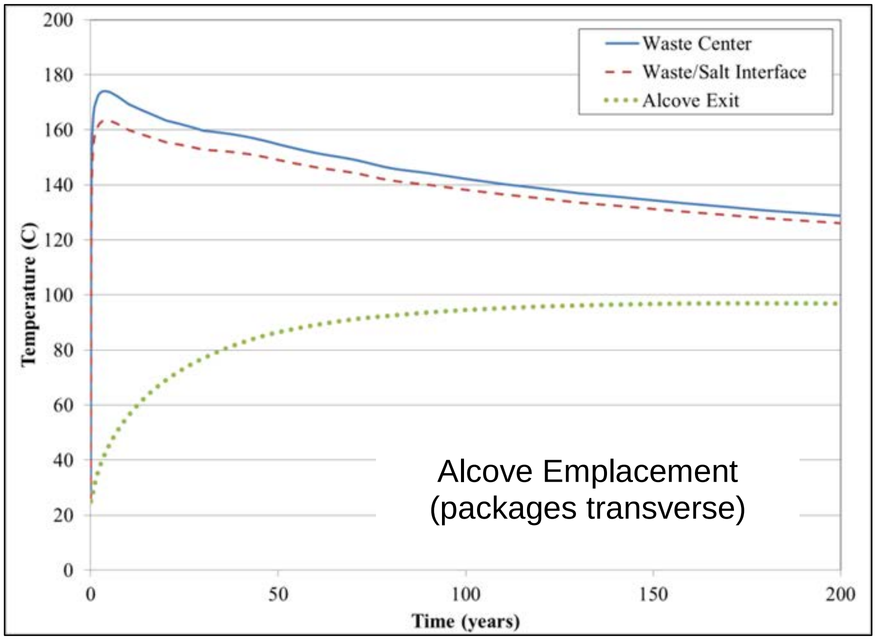
4.2.1 Horizontal In-Alcove or In-Drift Emplacement in Salt

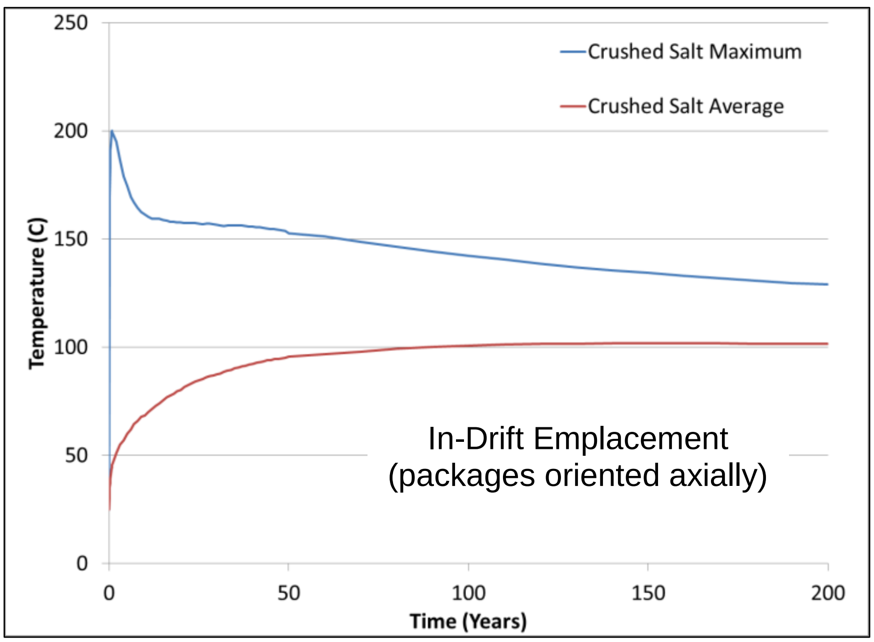
Emplacement in alcoves constructed from linear access drifts was originally proposed for HLW (Carter et al. 2011) to spread out heat-generating HLW packages on a grid. The access drifts could remain accessible after package emplacement, until eventually backfilled. Semi-cylindrical cavities (same radius as the packages) could be milled in the floor to accept the packages and improve heat transfer to intact salt. Placement in cavities lowers peak temperature on the order of 10 to 20 C° for the alcove mode (Figure 4-2).

Transverse orientation of packages (e.g., horizontal, perpendicular to the alcove axis) could facilitate moving smaller waste packages into alcoves. Once the alcoves are backfilled with crushed salt, the access drifts could remain accessible for maintenance or monitoring. However, handling large, heavy DPC-based packages in similar circumstances could be more difficult. In-drift emplacement with packages aligned parallel (Figure 4-3) could allow a simpler transporter design that distributes the package weight over more wheels. The transporter could use existing technology in the form of independent, hydraulic, kneeling wheel trucks such as those used in shipyards, and for transferring SNF in some dry storage facilities (Fairhurst 2012).

The in-drift variation (Robinson et al. 2012) has simpler, linear emplacement drift geometry that could be readily adapted to large SNF packages. This variation converts the access drifts for alcoves to emplacement, so there would be no access to the emplacement panel after emplacement (e.g., for monitoring).

The salt disposal concept is scalable and well suited for disposal of DPC-sized packages, mainly because of superior heat dissipation properties of salt. SNF with moderate burnup (40 GW-d/MT) can be disposed in 32-PWR sized packages after slightly more than 50 years of decay storage while meeting a 200°C target for peak salt temperature. Higher burnup fuel (60 GW-d/MT) can be disposed of after approximately 70 years decay storage (emplacing waste packages in cavities excavated in the floor to maximize heat transfer to intact salt).





Notes:

1. Waste-salt interface temperature is the maximum at the package surface, contacting crushed salt backfill.
2. SNF burnup 60 GW-d/MT, age at emplacement 70 years out-of-reactor, package spacing (x and y) 30 m.
3. Both cases include a semi-cylindrical cavity in the floor for maximizing package contact with intact salt.

Figure 4-1. Comparison of waste-salt interface temperature histories for disposal of 32-PWR size packages using the alcove (upper) and in-drift (lower) emplacement modes.

Table 4-2. Summary of peak waste package-salt interface temperatures, from FEM simulations of the salt concept, alcove emplacement mode, 20-meter package spacings.

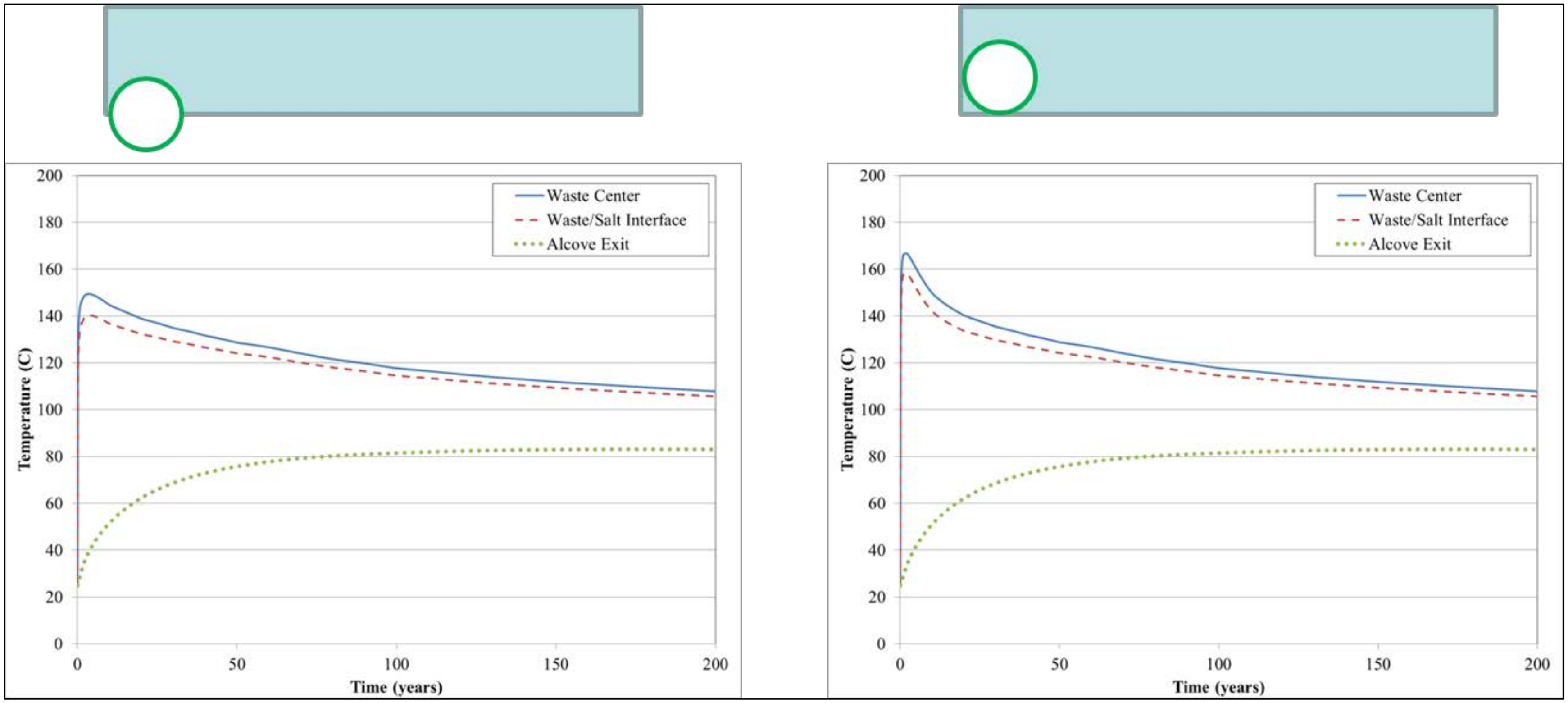
Package Type	Burnup (GW-d/MT)	MTHM	Age OoR (yr) [A]	Initial Heat Output (kW) [B]	Ventilation	Approx. Peak Salt Temperature (°C) [C]
4-PWR	40	1.88	10	2.7	No	75
4-PWR	60	1.88	10	4.5	No	110
4-PWR	40	1.88	50	1.3	No	50
4-PWR	60	1.88	50	2.0	No	65
12-PWR	40	5.64	10	8.0	No	160
12-PWR	60	5.64	10	13.5	No	275
12-PWR	40	5.64	50	3.8	No	90
12-PWR	60	5.64	50	5.9	No	130
21-PWR	40	9.87	10	14.1	No	270
21-PWR	40	9.87	50	6.7	No	145
21-PWR	60	9.87	50	10.4	No	220
21-PWR	60	9.87	60	8.9	No	190
32-PWR	40	15.04	50	10.2	No	210
32-PWR	60	15.04	50	15.8	No	330
32-PWR	40	15.04	60	8.8	No	190
32-PWR	60	15.04	100	8.2	No	210

NOTES:

[A] SNF age out-of-reactor (OoR) at emplacement, for the salt disposal concept.

[B] Package heat output at emplacement. SNF heat generation functions from Carter et al. (2012).

[C] Simulations represent packages emplaced in semi-cylindrical cavities in the floor, at the ends of alcoves, to maximize heat transfer to the intact salt. Crushed salt backfill consolidation is represented using a time-history developed from fully coupled thermal-mechanical simulation with the crushed salt creep model (Callahan 1999), which converges for zero porosity to the intact salt model (Munson et al. 1989). Waste packages are assigned the thermal and elastic properties of steel.



Note: SNF burnup 40 GW-d/MT, age at emplacement 50 years out-of-reactor, package spacing (x and y) 30 m.

Figure 4-2. Comparison of waste-salt interface temperature histories for emplacement of 32-PWR size waste packages directly on the alcove floor (right) vs. in a semi-cylindrical cavity (left) (burnup 40 GW-d/MT, age 50 years, package spacing 30 m).

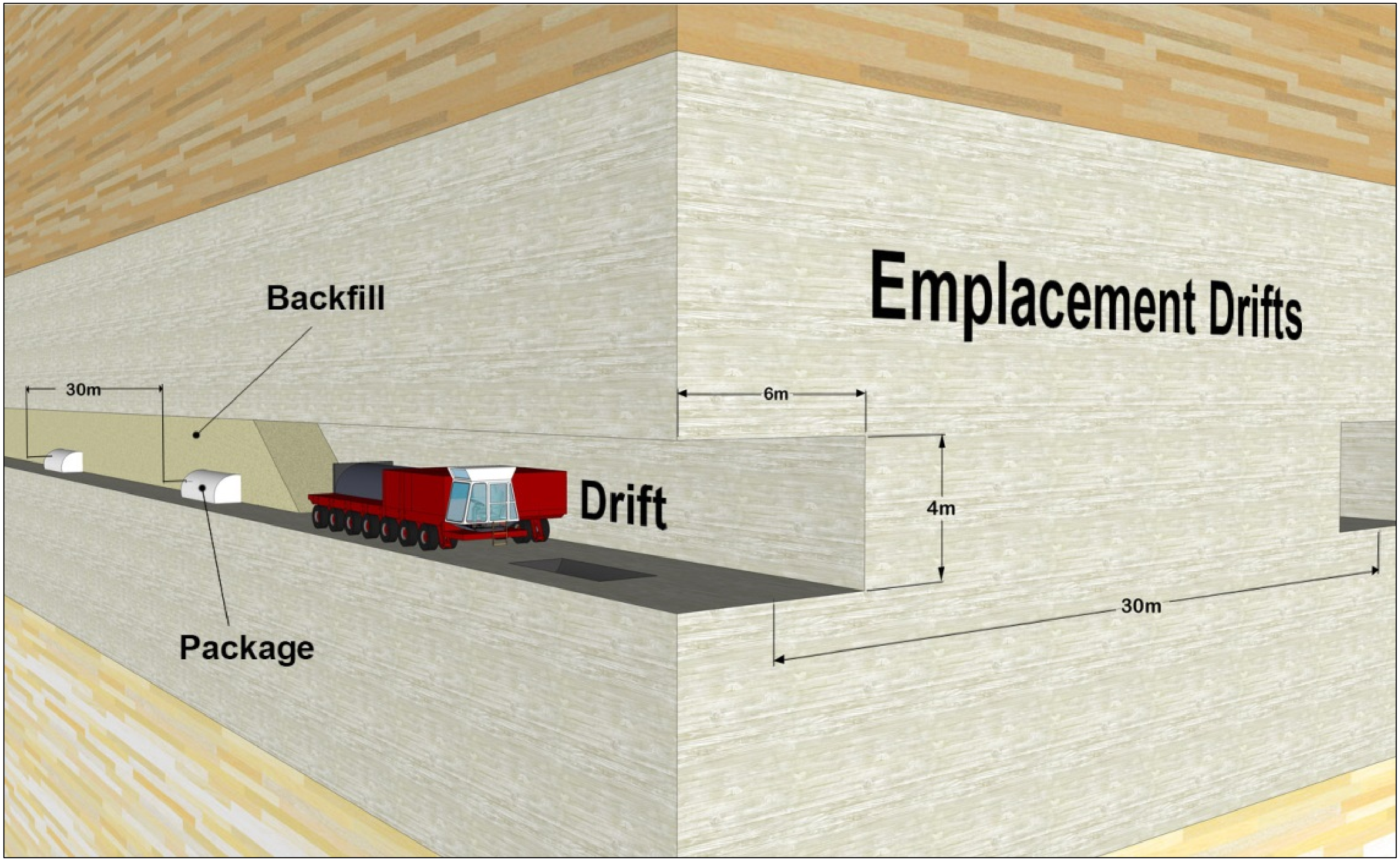


Figure 4-3. Conceptual drawing for the salt disposal concept with in-drift emplacement in long parallel drifts

4.2.1 Borehole Emplacement in Salt

Borehole emplacement openings could be drilled or bored, which is readily accomplished in salt although the boreholes can distort after construction, complicating emplacement. Borehole emplacement tends to optimize heat transfer compared to concepts that use crushed salt backfill, especially in the first few years after emplacement when peak temperatures occur (and crushed salt has not yet fully reconsolidated and has low thermal conductivity).

Horizontal borehole emplacement could more readily accommodate bedded stratigraphy and thinner disposal intervals, helping to ensure that all emplacement and access openings are constructed in an interval selected for low permeability and consolidation behavior. For horizontal emplacement a shielded handling machine could be used similar to that used for the NUHOMS® dry storage system or that proposed for the DIREGT concept (Graf et al. 2012). Skids, casing, and/or dedicated trolleys can be used to help slide each package into its borehole. A rigid, prefabricated borehole plug could be installed after emplacement for immediate shielding, and to stabilize the collar opening as the salt creeps. Such a plug could occupy less volume (requiring a shorter borehole) than full backfilling. After emplacement of all packages, the access drifts could be completely backfilled.

The slant borehole variation would involve construction of downward dipping emplacement boreholes, to facilitate emplacement by using gravity to provide some or all of the force needed to overcome sliding friction as packages are emplaced. Also, the downward orientation would tend to retain crushed salt backfill so that rigid shielding plugs would not be needed.

A more extreme concept for borehole emplacement in domal salt could involve stacking of waste packages in a deep boring tens to hundreds of meters in extent, excavated from an underground access drift (EPRI 2010, Appendix B). This is a reference concept for disposal of consolidated SNF assemblies in small canisters in salt domes (Bollingerfehr and Filbert 2010). It would not be well suited for DPC disposal because of the size and weight of the containers, the difficulty of retrieval, and the challenge of implementing borehole plugs to support large, heavy packages so that crushing loads do not develop.

4.3 Clay/Shale Enclosed, Borehole Emplacement Concept

Although not recommended for DPC direct disposal for the reasons discussed below, this concept is included for completeness. It is based loosely on the French concept for SNF (type C) waste in a clay or shale repository (Andra 2005; Hardin et al. 2012). In the original concept for 4-PWR size packages, horizontal emplacement borings 40-m long and spaced 30 m apart, would be excavated from parallel access drifts spaced approximately 100 m apart. For DPC disposal applications these spacings and the dimensions of the emplacement borings could be increased, with emplacement borings approximately 70 m apart, and up to 3 m in diameter (consistent with the buffer thickness needed for waste isolation). The horizontal length of each emplacement boring would be nominally 50 m to accommodate three packages spaced 20 m apart (center-center). The horizontal orientation is best for stratified sediments, but construction challenges could limit boring length and the number of packages (EPRI 2010, Appendix B).

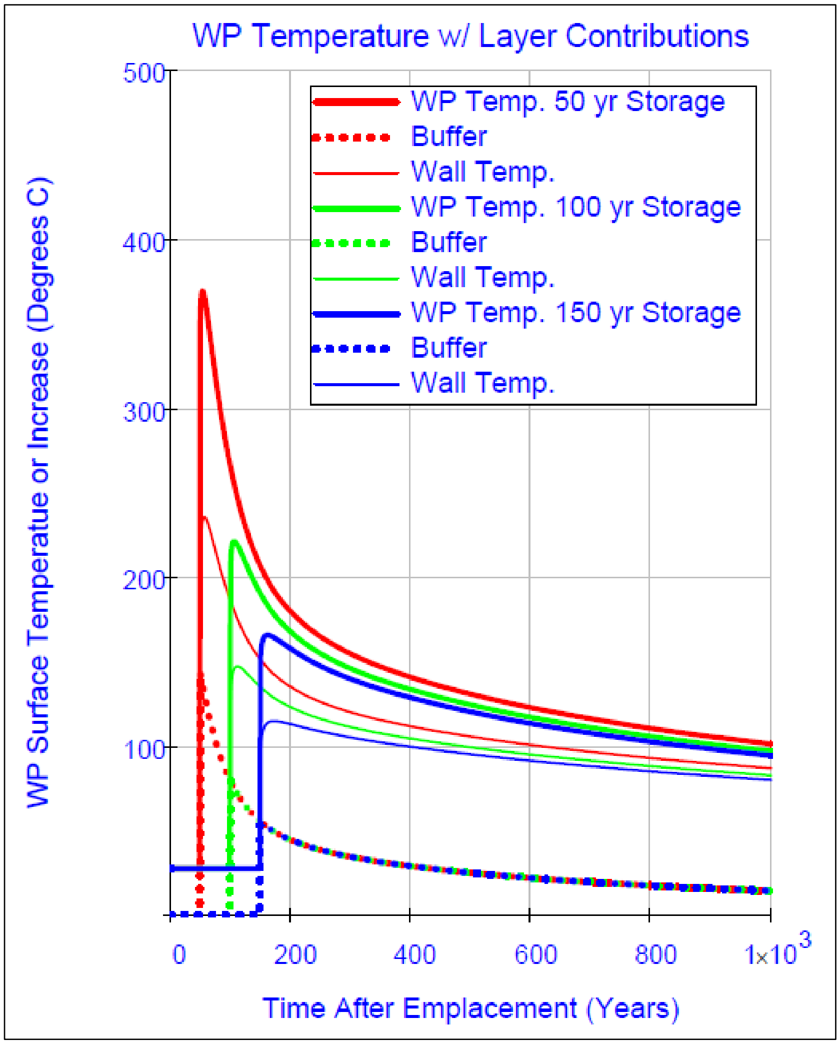
Emplacement borings could be lined with steel tubing or segmented plate (e.g., 2-cm wall thickness) to ensure opening stability throughout repository operations. An inner liner could be installed to accept packages, and the annulus filled with compacted, dehydrated, swelling-clay buffer material. Both the inner and outer liners would be subject to thermal expansion that could require accommodation in the design (EPRI 2010, Appendix B). Packages with low-alloy steel overpacks could be slid into place within the inner liner, alternating with plugs of buffer material, and finishing with a sealing/shielding plug at the collar.

There would be no long-term requirements on access drift stability, because this is an enclosed mode. Access drifts, service drifts, etc., could be constructed using, rock bolts, and other measures as needed to ensure service during repository operations. The emplacement mode is shielded so there could be no little or no radiological risk to workers performing repository closure operations.

The clay/shale enclosed geologic disposal concept would likely be implemented in mudstone or claystone media that behave more like cohesive soil than massive rock. Excavation could be accomplished using road-header type equipment or tunnel boring machines. Ground support in access and service drifts would provide full support to the circumference of the excavation, and consist of either: 1) shotcrete, steel sets and lagging; 2) pre-cast concrete liner segments; or 3) liner plate. With properly designed ground support there would be little uncertainty associated with maintaining stability for the time needed to emplace waste packages, buffer, and/or backfill.

To test thermal performance for an enclosed mode for large packages in clay/shale media, an "optimistic" thermal analysis was performed for 32-PWR packages (SNF burnup 40 GW-d/MT), expanded spacings (90-m drift spacing and 20-m package spacing), typical clay/shale thermal properties, an optimistic K_{th} of 0.9 W/m-K for dehydrated clay-based buffer material, and emplacement at 50, 100 and 150 years out-of-reactor. If thermal limits (e.g., peak temperature targets) are not met with such a case, then they are unlikely to be met with any similar, plausible case. The lowest peak buffer temperature (at the waste package surface) is 166°C (Figure 4-4) which is significantly greater than current maximum temperature targets for clay-based buffer material (approximately 100°C). The corresponding peak drift wall temperature is 115°C.

Use for DPCs would require long (150 years or longer) decay storage to limit peak buffer temperature to less than 200°C, and to limit peak rock wall temperature to 100°C. This would exceed the assumed disposal timeframe for this study (up to 100 years of surface decay storage, and up to 150 years of combined decay storage and repository ventilation; see Section 2).



Source file: *Open UOX40-32 (90 m) Clay WP20m Enclosed.xmcd*.

Figure 4-4. "Optimistic" thermal calculation for average SNF (40 GW-d/MT) in 32-PWR size packages, for an enclosed (clay buffer) emplacement mode in clay/shale media.

4.4 Sedimentary Unbackfilled, Open, In-Drift Emplacement Concepts for SNF

This concept is based on a previously defined shale unbackfilled open-mode reference case (Hardin et al. 2012) extended to a broader range of similar sedimentary rock types, with drift and package spacings expanded to at least 70 m and 20 m, respectively, for DPC direct disposal. The concept involves long, parallel emplacement drifts and in-drift emplacement of DPC-based waste packages. Larger spacings (e.g., drift and package spacings up to 100 m and 30 m, Figure 4-5) can be used for hotter packages (larger, younger fuel and/or higher burnup).

Waste package spacing is more effective than drift diameter at lowering peak EBS temperatures outside the waste package, and slightly more effective than drift spacing. A drift spacing of 70 m is selected because it incorporates much of the peak temperature reduction possible using drift spacing (Figure 4-6). Drift spacing has a greater effect on temperatures in later time (e.g., after 300 years) when the entire repository heats up and the relative contribution from adjacent drifts increases. The impact of these spacings on the overall repository layout area for several concepts was compared (Hardin and Voegele 2013); the needed area ranges from approximately 60 to 100 m^2 per MTHM, although very large spacings in sedimentary media could approach 200 m^2 per MTHM (Section 6.1).

The sedimentary unbackfilled concept would involve segmented emplacement drifts that are ventilated for up to 100 years (consistent with the timing assumptions (Section 2) then sealed in segments containing small numbers of packages. Thus, packages would not be sealed off from others within the same segment, but segments would be sealed off from other segments in the repository. The idea is that isolating every package with low-permeability backfill is not necessary in a massive, low-permeability formation that lacks through-going faults or other features that conduct groundwater flow. However, the uncertain possibility of such faults or other features intersecting emplacement drifts would make it prudent to isolate segments of the repository so that only a few waste packages could be affected. By isolating each segment from the remainder of the repository, the potential radionuclide migration from each segment would effectively resemble that from a single, very large waste package.

The unbackfilled approach would not require backfilling emplacement drifts at closure, and therefore would not need to meet backfill temperature targets. Backfill installed in non-emplacement areas would be situated far from waste packages, where temperatures are much lower. The approach would also leave the emplacement drifts open after closure for heat transfer to occur. The drifts would be expected to collapse eventually, but mostly after the time period when peak temperatures occur.

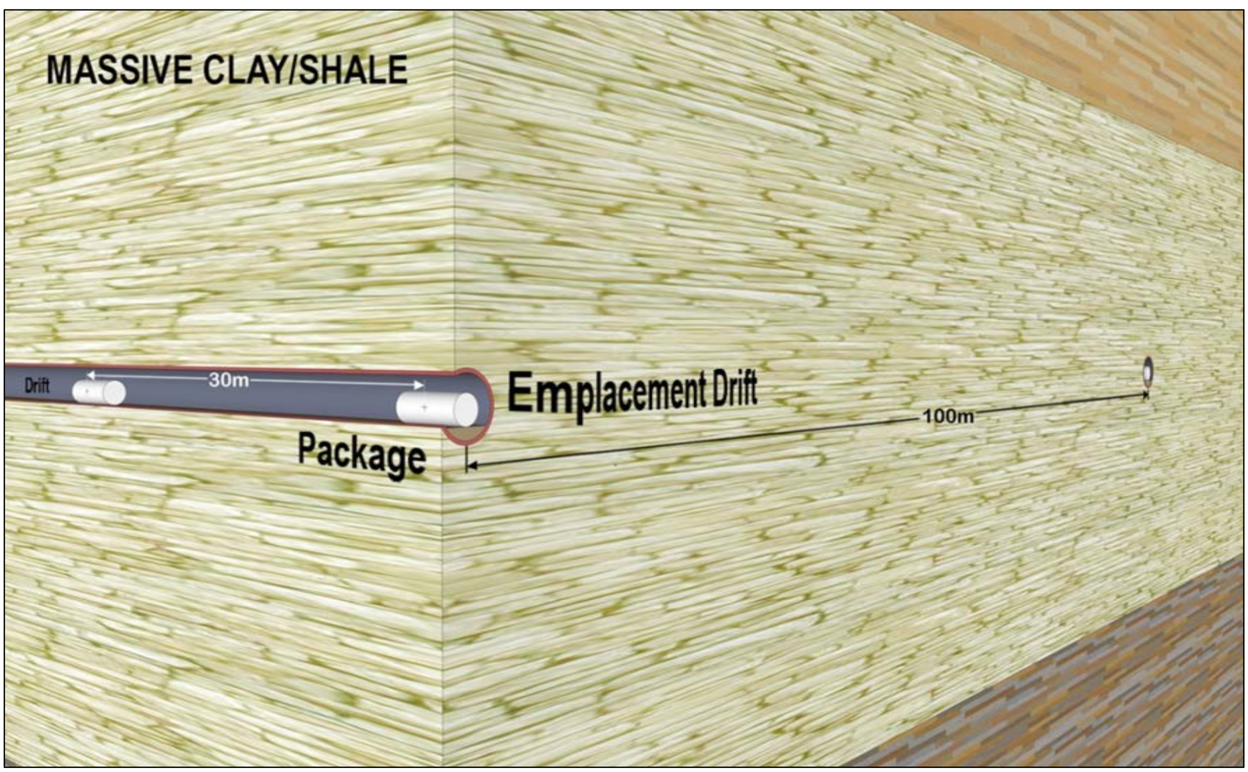
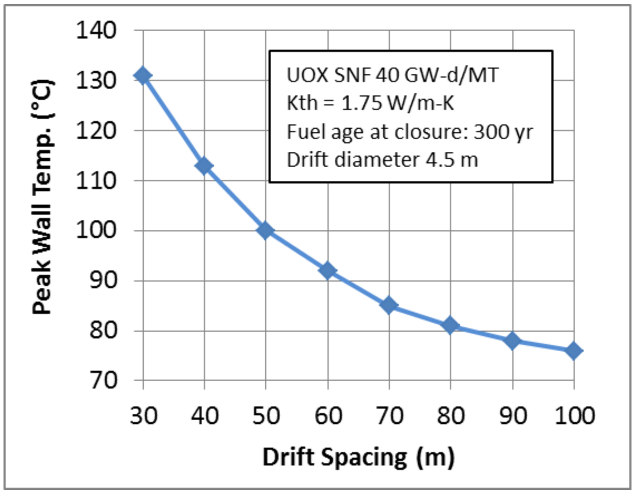


Figure 4-5. Schematic of sedimentary unbackfilled disposal concept with in-drift disposal, with expanded drift and package spacings.



Source files: *Open UOX40-21 (xx m) Clay.xmcd*, where *xx* is drift spacing in meters.

Figure 4-6. Effect of drift spacing on peak drift wall temperature, for typical SNF in 21-PWR size packages, in a sedimentary repository, unbackfilled, that closes at 300 years out-of-reactor.

Using empirical rock mass classification systems without site-specific information, it is difficult to differentiate between sedimentary rock and hard rock, with respect to opening stability. Perhaps the best way to look at this is in comparison to the Rock Mass Rating and "Q" Rock Mass Classifications presented earlier for crystalline rock (Section 4.1). Sedimentary rock could be well indurated; but if not, excavation and support methods would be similar to those described above for the clay/shale enclosed concept (Section 4.3). Sedimentary rock could be excavated using specialized tunnel boring machines. Compressive strength could be less than half that of granite, and there would likely be more frequent discontinuities and lower RQD. However, any rock mass considered would likely have tight unweathered joints with rough mating surfaces. As with any potential host medium, a rock mass with limited groundwater flow would be sought, and underground excavations would be oriented favorably with respect to discontinuities. Thus, the Rock Mass Rating could be in the good category or perhaps the very good category. The corresponding stand-up time (unsupported) could be from 10 years to perhaps as short as 1 year.

Using the "Q" Rock Mass Classification System for such a rock mass, RQD could be classified as good or excellent. One or more joint sets would likely be present. A rock mass with rough irregular undulating joint walls would be sought, which means the joint roughness and joint alteration characteristics would be favorable. The joint water reduction factor likely would reflect dry excavations or limited water inflow, while the stress reduction factor would reflect a well chosen rock mass with medium stress and favorable stress orientation. Using an excavation support ratio number of 0.8, the estimated support category using the "Q" Rock Mass Classification System would require rock bolting and perhaps shotcrete.

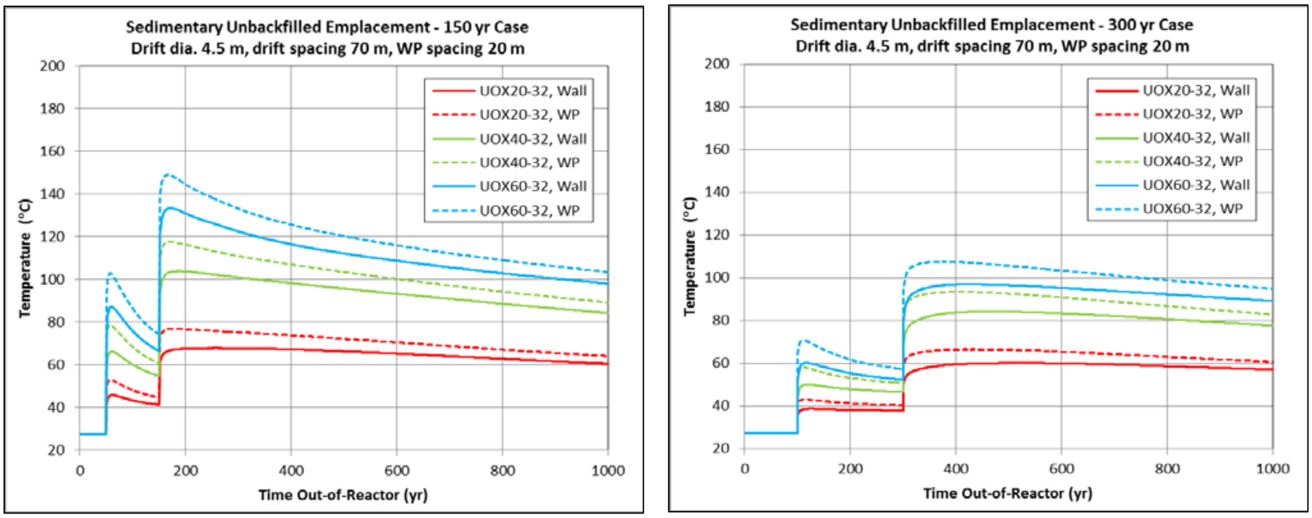
The foregoing discussion of rock mass stability and ground support applies to argillaceous media such as mudstone, claystone, shale, etc. (Hansen et al. 2011). There are other sedimentary media such as limestone, marl, etc., that could be indurated, with low permeability and other desired characteristics of repository host media. The suitability of more indurated sedimentary media for DPC direct disposal applications may be better represented by the discussion of hard rock

(Section 4.6). The sedimentary concepts described in this report include a range of argillaceous media, including some that may be also be calcic, and are not limited to plastic clay or soft shale.

Waste packages would have overpacks made from a corrosion resistant material such as a nickel-based alloy, thus providing a long-term containment barrier in addition to far-field plugs and seals, and the host rock (Section 6). Overpacks would be robust to facilitate handling and emplacement, and resist damage from rockfall. Corrosion-resistant overpacks would also limit radionuclide releases from human intrusion (Section 7) because: 1) corrosion resistance increases the likelihood that a future driller would recognize the potential for breaching a waste package before it occurs; and 2) interception of one package would not necessarily mobilize radionuclides from neighboring packages, for a very long time until there were many package containment breaches from corrosion. Human intrusion is more likely in sedimentary basins than hard rock (following 40CFR191 Appendix C), and for quiescent tectonic conditions it could be the most important scenario (Section 7).

Underground openings and ground support would be designed to allow only minor rockfall for up to 100 years to facilitate ventilation, inspection and possible retrieval. Accumulating rockfall after closure would increase temperature within the debris piles, and at the waste packages, but drift wall temperature targets could still be met. Package temperatures are likely to be lower than limits that could be imposed to preserve fuel cladding integrity (350°C). Low-temperature sensitization of stainless steel in DPCs could occur depending on thermal exposure of the canister, for example in weld affected zones after long-term exposure at 100 too 300°C (e.g., Fox and McCright 1983; Farmer et al. 1988). The disposal overpack, however, would be designed, fabricated and treated to limit such degradation. Accordingly, blanketing by rockfall debris would not produce excessive package temperatures, if it occurs some time after closure (see BSC 2008b for analysis showing 90 years of postclosure susceptibility to over-temperature conditions from rockfall). Eventually, rockfall would be limited by debris bulking that fills the opening.

Thermal calculations for this concept include gray-body radiative heat transfer from the waste package to the drift wall. They show (Figure 4-6) that closure at approximately 200 years out-of-reactor (storage plus ventilation time) limits the peak drift wall temperature to 100°C for SNF with 40 GW-d/MT burnup. Segments with lower burnup SNF (20 GW-d/MT) could be closed sooner, while those with higher burnup SNF (60 GW-d/MT) could take up to 300 years. For higher burnup, these durations exceed the disposal timeframe assumed for this study (up to 50 years decay storage and 100 years repository operations; see Section 2). More flexibility is available if the host rock peak temperature target is greater than 100°C (see Section 5.3 for calculations of required ventilation time and extent of heating the host rock). Thermal management options are discussed in the next sections.



Source files: *Open UOXxx-32 (70 m) Clay WP20m.xmcd*, where *xx* is burnup of 20, 40 or 60 GW-d/MT.

Figure 4-7. Temperature histories (drift wall and package surface) for 32-PWR size packages containing SNF with a range of burnup, in a sedimentary unbackfilled repository, for: (left) 50-year decay storage and 100-year ventilation, and (right) 100-year decay storage and 200-year ventilation.

4.4.1 Sedimentary Unbackfilled, Open, In-Drift, Low-Temperature Concept

As shown on the left-hand part of Figure 4-6, with these spacings only SNF with moderate burnup (e.g., less than 40 GW-d/MT) could be emplaced in sedimentary host rock in DPC-based packages, while maintaining drift wall temperature at 100°C or less without exceeding the assumed 150-year disposal timeframe (Section 2). Higher burnup SNF could be accommodated with greater spacings, or by relaxing the host rock peak temperature target (Section 5.3).

4.4.2 Sedimentary Unbackfilled, Open, In-Drift, High-Temperature Concept

This concept could accept the full range of SNF burnup in DPC-based packages (32-PWR size) if heating of the near-field host rock to temperatures greater than 100°C is allowable. This possibility was investigated previously as a reference case (Hardin et al. 2012) which would heat the drift wall to 130°C and push the 100°C isotherm into the host rock as much as 3 m beyond the drift wall. The 100°C isotherm would envelope all of the waste packages and much of the near-field host rock in an emplacement segment, but adjacent host rock where plugs and seals were installed would see much lower temperatures. The repository performance impact of heating the drift wall to temperatures greater than 100°C in argillaceous media is an area of ongoing research.

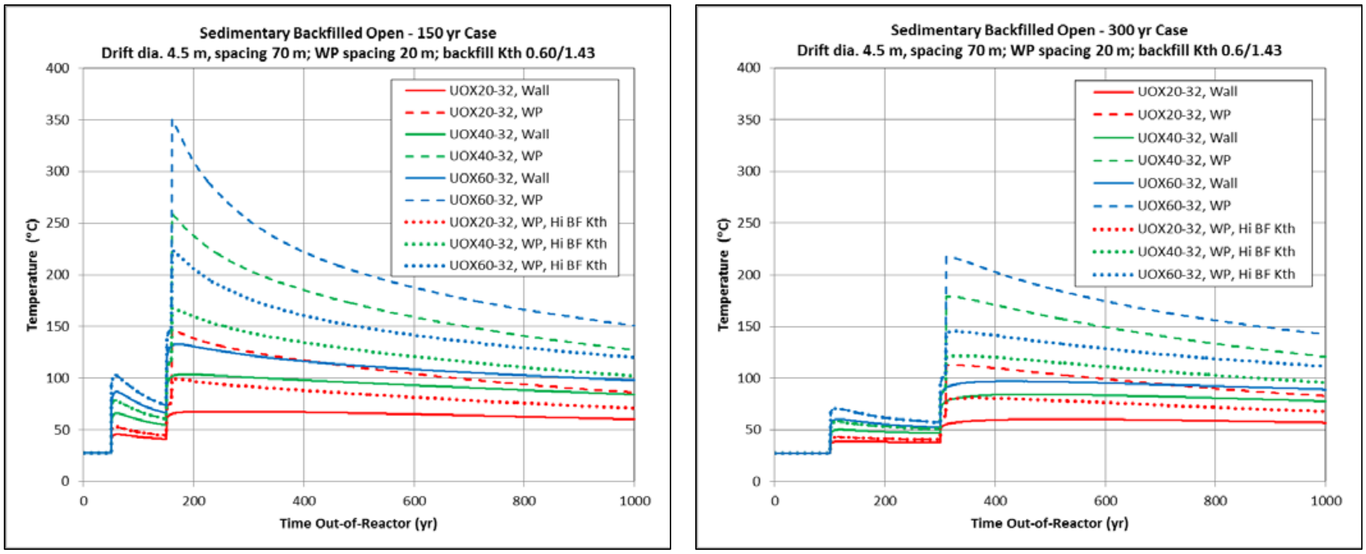
4.5 Sedimentary Backfilled, Open, In-Drift Emplacement Concept for SNF

For saturated host rock, repository openings need to be plugged and/or sealed to prevent groundwater moving preferentially throughout the repository. Low-permeability backfill of all repository openings is one approach. Backfill could also mitigate effects from roof collapse, seismic shaking, and other processes. The advantages of backfill, and the related thermal management issues, are also discussed in Section 6 for hard rock. Since these are open emplacement modes, the plugging, sealing and/or backfilling must be undertaken after ventilation, at elevated temperature, in a radiation environment.

In a low-permeability host formation the function of backfill may depend less on swelling properties than for higher permeability host rock settings (see Sections 1.3 and 6.2). Thus, a non-swelling low-permeability material could be selected, or a water-sensitive swelling material could be emplaced in a wet state. Some possible options for non-clay and/or non-swelling materials are discussed by Hardin and Voegele (2013, Appendix B). The function of backfill as an engineered barrier would trade with more corrosion resistant waste packaging. That is, a more thermal resistant but less expansive and more permeable backfill could be used, with a more resistant waste package. For this analysis, waste packaging is corrosion resistant as described above for all sedimentary open backfilled concepts. However, less resistant packaging could be used if complementary backfill function is demonstrated.

Excavation method and ground support requirements would be the same as those discussed for the sedimentary unbackfilled open concept (Section 4.4) except that selection of backfill could address the presence of more rock fracturing and/or potential groundwater flow. With greater fracturing or flow, more robust ground support measures could be used to keep emplacement drifts open for 100 years, such as shotcrete, with steel supports as needed.

Drift wall temperature would be the same as calculated for the unbackfilled case (Section 4.5, Figure 4-7). However, peak backfill temperature (at the waste package surface) would exceed 100°C for all burnups, for hundreds of years (Figure 4-8). For higher burnups (greater than 20 GW-d/MT) peak backfill temperature could be 200°C or greater. Like the corresponding hard rock concept (Section 6.2) increased drift spacing and waste package spacing would decrease the drift wall temperatures but would have less influence on backfill temperatures. The viability of this concept would depend on a backfill thermal strategy that could include: 1) reducing drift diameter and/or the effective backfill thickness; 2) increasing the minimum thermal conductivity for backfill material, for example using wet emplacement at closure; and/or 3) establishing a higher temperature tolerance (e.g., up to 200°C or higher) for backfill material, for example, by proving the performance of smectite clay-based materials or selecting a different material not subject to the same limitations. Backfill materials with greater thermal conductivity, or tolerance to temperatures of 150°C or higher, are the subject of ongoing R&D.



Source files: Open *UOXxx-32 (70 m) Clay Backfilled WP20m.xmcd*, where *xx* is burnup of 20, 40 or 60 GW-d/MT.

Notes: WP = waste package, and Hi BK Kth refers to hypothetical high backfill thermal conductivity of 1.43 W/m-K.

Figure 4-8. Temperature histories (drift wall and waste package surface) for various SNF burnup levels in 32-PWR sized packages, in a sedimentary backfilled repository with spacings shown, for: (left) 50-year decay storage and 100-year ventilation, and (right) 100-year decay storage and 200-year ventilation.

4.6 Hard-Rock Open, In-Drift Emplacement Concepts for SNF

This concept would use in-drift emplacement in long, parallel emplacement drifts, with drift and package spacings adjusted to limit postclosure temperature at the drift wall (Figure 4-9 shows drift and package spacings of 70 m and 20 m, respectively). The repository would be ventilated to remove heat for 50 to 100 years after emplacement, and it would be closed when the SNF age is 150 years or less out-of-reactor.

Hard rock has the potential to provide more reliable opening stability and allow less maintenance (for up to 100 years) depending on site-specific factors such as the hazard from seismic ground motion. In addition, hard rock (i.e., crystalline; and igneous or metamorphic) typically has greater thermal conductivity and is associated with higher temperature targets for repository applications (e.g., 200°C) than argillaceous sedimentary media containing significant total abundance of hydrous clay minerals (e.g., 100°C).

As discussed previously, using empirical rock mass classification systems without site-specific information, it is difficult to differentiate between sedimentary rock and hard rock. Given the strength, fracturing and groundwater characteristics of a well-chosen hard rock setting, the Rock Mass Rating could be in the good category or perhaps the very good category, and the corresponding stand-up time could be up to 10 years. The RQD could be classified as good or excellent. Joint characteristics, stress conditions, and groundwater flow conditions in a hard rock setting would be comparable to (or possibly better than) those for sedimentary clastic rock settings. Using an excavation support ratio number of 0.8, the estimated support category using the “Q” Rock Mass Classification System would require rock bolting and perhaps shotcrete.

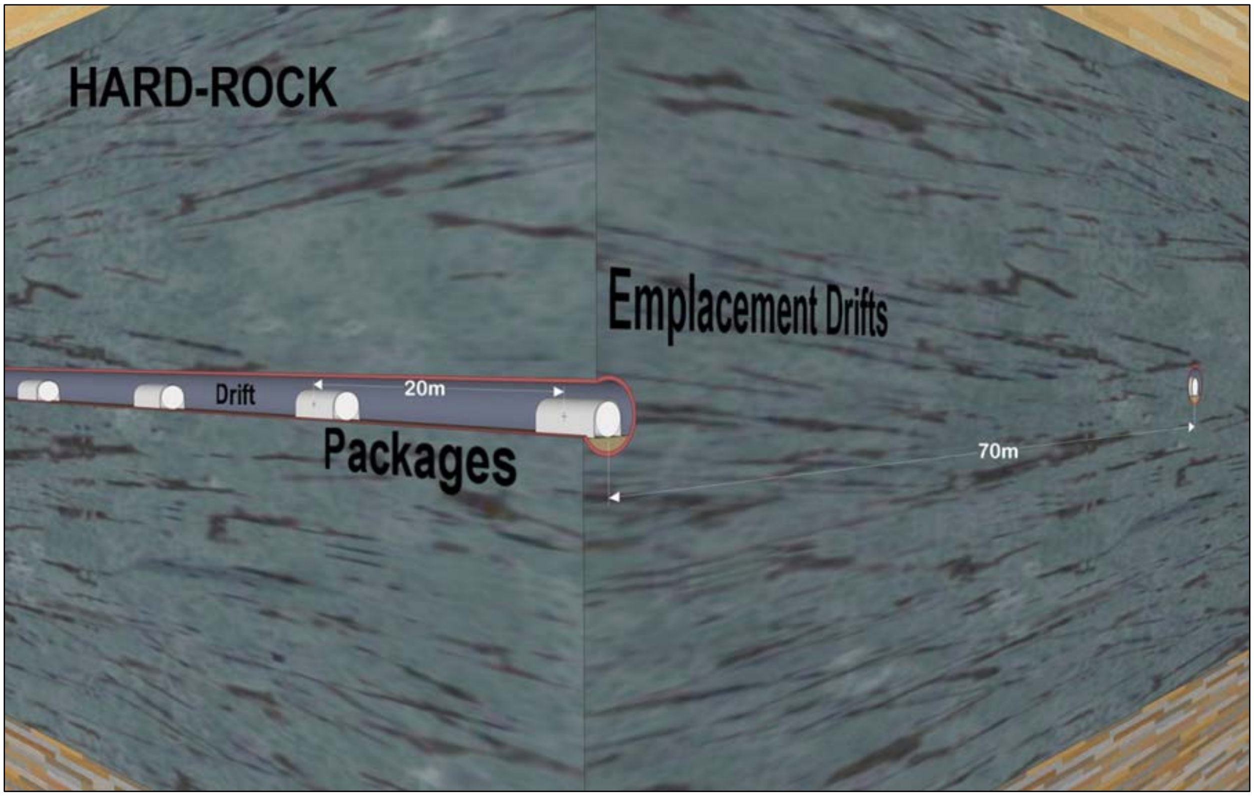


Figure 4-9. Schematic of hard rock unbackfilled disposal concept with in-drift disposal.

Indurated hard rock types may be subject to brittle behavior in response to geologic processes. The relative lack of plasticity or fracture healing ensures that fractures persist over geologic time periods. With a network of fractures, bulk permeability may be orders of magnitude greater than the small-scale, intact rock permeability. Virtually all hard rock units have some fracturing that needs to be addressed in disposal concept development and repository design.

If the host rock is unsaturated, bulk permeability will make it free-draining (DOE 2008). With free drainage throughout the host rock there is little possibility of focused groundwater flow along repository openings, so plugging and sealing of emplacement and access drifts are not needed (DOE 2008). For saturated host rock, such flow is possible and repository openings need to be plugged, sealed and/or backfilled with low permeability material. Since these are open emplacement modes, the plugging, sealing and/or backfilling must be undertaken at elevated temperature, in a radiation environment.

For all repositories in hard rock, either shaft or ramp access could likely be used for waste transport, because opening stability is advantageous for construction of liners, plugs and seals to control groundwater inflow.

For enclosed emplacement modes in which backfill/buffer is emplaced concurrently with waste packages, see the crystalline enclosed concepts (Section 4.1).

4.6.1 Hard-Rock Unbackfilled, Open, Unsaturated, In-Drift Concept

This concept would use a corrosion-resistant overpack and other engineered barriers as needed to enhance performance and facilitate licensing. Other barriers could include water diversion features (e.g., drip shields) or multiple corrosion-resistant materials used for waste packages (e.g., Ti and Hastelloy).

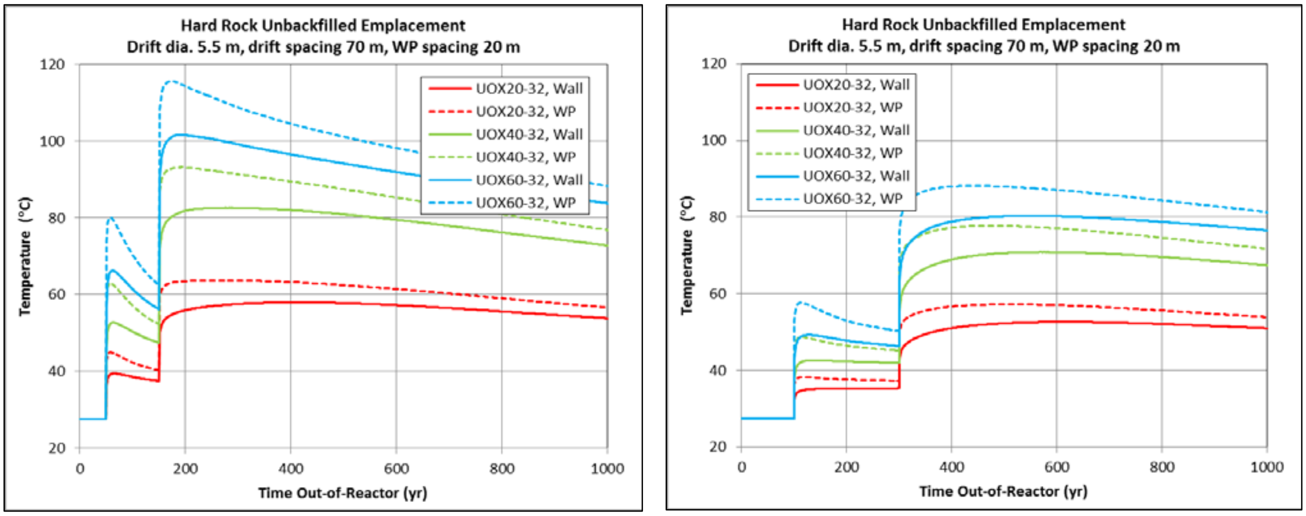
This concept is similar to previous work (DOE 2008) and to a previous proposal for direct disposal of DPCs (Kessler et al. 2008), so it could well be suited for disposal of DPC-sized packages. Hard rock offers long opening stand-up times and resistance to temperatures up to approximately 200°C. Thermal calculations (Figure 4-10) show that a 200°C wall temperature target could be met for SNF with high burnup, with fewer than 150 years decay storage plus ventilation (except possibly for minor fuel types such as irradiated Pu-MOX which could require longer storage/ventilation). The left-hand side of Figure 4-10 corresponds to the maximum storage and ventilation durations assumed for this study (Section 2), while the right-hand side extends closure to 300 years out-of-reactor. This calculation is based on host rock thermal conductivity of 2.5 W/m-K, which is greater than typical welded tuff but less than some granites. The results shown in Figure 4-10 are similar to low- and high-thermal operating mode calculations published previously (DOE 2002). They suggest that the hard rock repository layout could be optimized with smaller drift and waste package spacings, and shorter durations for decay storage and ventilation.

4.6.2 Hard-Rock Backfilled, Open, In-Drift Concept

This concept would use a corrosion-resistant overpack, and a low permeability backfill installed prior to closure since this would be a nominally saturated hydrologic setting. Postclosure performance would be similar to the crystalline enclosed concept with in-drift emplacement discussed above (Section 4.1.3). Possible strategies for closure operations involve backfilling using remotely operated equipment, and turnouts, plugs or labyrinths for worker shielding (Hardin et al. 2012). Low-moisture content granular backfill material could be emplaced remotely using conveyors, pneumatic delivery, or auger feeds.

As discussed for sedimentary host media (Section 4.5), the function of backfill as an engineered barrier would tend to trade against more corrosion resistant waste packaging. A more temperature resistant but less expansive and more permeable backfill could be used, with a more resistant waste package. Alternatively, less resistant packaging could be used if complementary backfill function is demonstrated.

The concept could also be implemented in unsaturated formations, where clay-based backfill/buffer materials might perform better than in saturated formations (Hardin and Sassani 2010). Compared with the unbackfilled concept (Section 4.6.1) backfill could mitigate effects from roof collapse, seismic shaking, and other processes (Section 7). Backfill might also produce long-lasting, reducing conditions at the waste package and waste form, which could be reflected in the EBS design (e.g., less reliance on corrosion resistant materials).



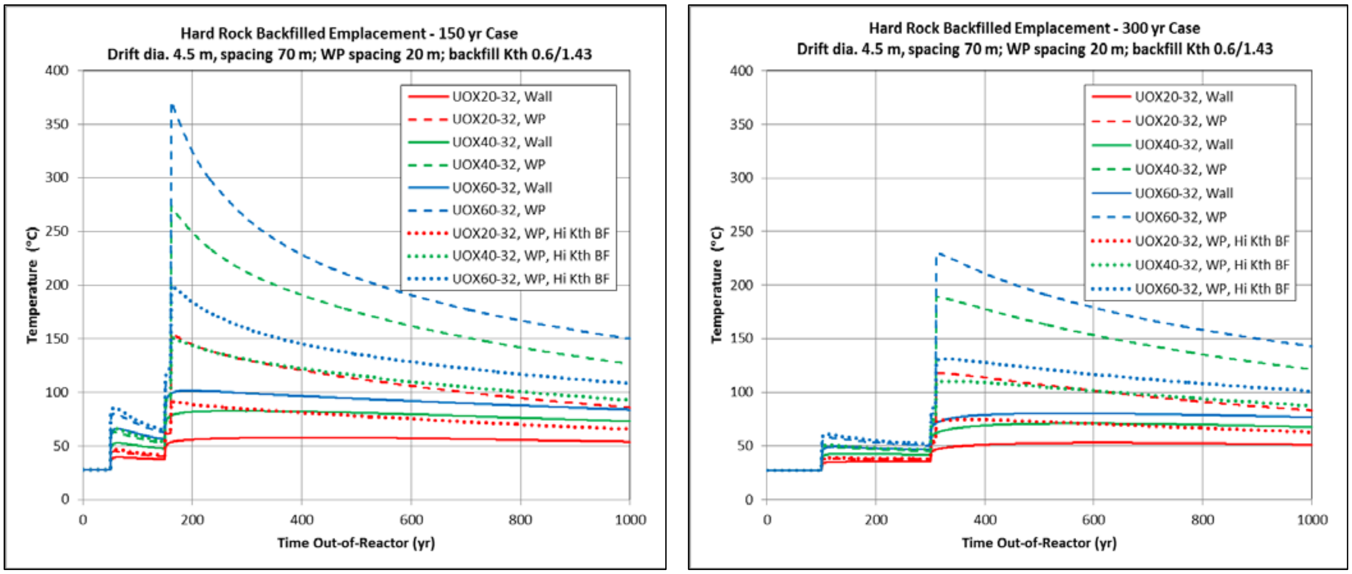
Source files: Open UOXxx-32 (70 m) Hard Rock WP20m.xmcd, where xx is burnup of 20, 40 or 60.

Figure 4-10. Temperature histories (drift wall and waste package surface) for various SNF burnup levels in 32-PWR sized packages, in a hard rock open (unbackfilled) repository with spacings shown, for: (left) 50-year decay storage and 100-year ventilation, and (right) 100-year decay storage and 200-year ventilation.

Thermal calculations show that limiting the peak postclosure temperature of backfill near the waste package surface would drive the timing of surface decay storage, emplacement, repository ventilation, and backfilling at closure (Figure 4-11). The left-hand side of Figure 4-11 corresponds to the maximum storage and ventilation durations assumed for this study (Section 2), while the right-hand side extends closure to 300 years out-of-reactor. Even closure at 300 years produces peak backfill temperatures of 120°C or higher depending on SNF burnup. These calculations were made using a drift diameter of 5.5 m (same as used for the unbackfilled cases) and a backfill thermal conductivity of 0.6 W/m-K representing dehydrated, compacted clay.

For backfilled concepts, larger drift diameter increases buffer thickness and therefore package temperature, whereas for unbackfilled concepts, larger drift diameter decreases package temperature. Increased drift spacing and waste package spacing would decrease the drift wall temperatures, but would not significantly lower peak backfill/buffer temperatures.

As discussed for sedimentary concepts (Section 4.5) the viability of this concept would depend on a strategy that could include: 1) reducing drift diameter and/or the effective backfill/buffer thickness; 2) increasing the minimum thermal conductivity for backfill material; and/or 3) establishing a higher temperature tolerance (e.g., up to 200°C or greater) for backfill material. Backfill materials with greater thermal conductivity or tolerance to temperatures of 150°C or higher, are the subject of ongoing R&D.



Source files: Open UOXxx-32 (70 m) Hard Rock Backfilled WP20m.xmcd, where xx is burnup of 20, 40 or 60.

Notes: WP = waste package, and Hi BK Kth refers to hypothetical high backfill thermal conductivity of 1.43 W/m-K.

Figure 4-11. Temperature histories (drift wall and waste package surface) for various SNF burnup levels in 32-PWR sized packages, in a hard rock open (backfilled) repository with spacings shown, for: (left) 50-year decay storage and 100-year ventilation, and (right) 100-year decay storage and 200-year ventilation.

4.7 Cavern-Retrievable Storage and Disposal Concept for SNF

The disposal concepts described in this section emphasize retrievability of the DPCs from the underground facility throughout the repository operations period (up to SNF age of 150 years according to the timing assumptions adopted for this study; see Section 2). Shielded storage systems similar to those used at nuclear power plants would be emplaced underground in large galleries. At repository closure casks containing DPCs would be encapsulated in low-permeability backfill/buffer material. The storage casks would be fully encapsulated on the top, sides and below. The cavern-retrievable (CARE) concept was proposed more than a decade ago and continues to be discussed in the literature (see McKinley et al. 2001, 2006 and 2008; EPRI 2010).

The galleries would have a minimum 100-year opening stability, although loaded storage casks and their contents could be readily moved for gallery maintenance. One motivation for cavern-retrievable storage is to maintain an option to extend storage well beyond 100 years (with extended ventilation) if desired in the future, and another is the small repository footprint that could be useful in limited settings (EPRI 2010, Appendix B).

The concepts could be implemented in saturated or unsaturated host media, although unsaturated settings might offer less potential for groundwater penetration of the engineered barriers. Also, clay-based backfill/buffer materials could perform well in unsaturated settings (Hardin and Sassani 2010).

The cavern retrievable concepts could potentially be developed in rock masses encompassed by the range of support estimates given for the crystalline, sedimentary, and hard rock as discussed above. Because the excavations would likely have smooth, curved roof spans their stability would be similar to that of circular openings. The excavation dimensions considered for the cavern retrievable concepts are only modestly larger than those considered for the in-drift emplacement concepts described previously. With high-quality rock (e.g., hard rock or indurated sedimentary rock as discussed above) ground support requirements could be minimal.

The problem faced when designing for a potential ventilation period of 100 years or longer is accommodating ground support maintenance. Because of the lengthy ventilation times, and limited access to emplacement drifts for ground support maintenance, previous design work selected stainless steel liner plate and stainless steel rock bolts that were point-anchored (DOE 2008). If the geochemistry of the disposal system is compatible, fully grouted rock bolts could provide one alternative for limiting maintenance and extending opening stability.

The cavern-retrievable emplacement concepts for DPC storage and disposal have an advantage in being shielded, ventilated systems in an underground environment. Other concepts such as the borehole variations for disposal in crystalline, salt or sedimentary media are shielded by rock, but not ventilated. This would allow access for preventive maintenance and would significantly enhance confidence in long-term opening stability. With shielded emplacement and long-term stability (and maintainability), the repository could then serve as the principal venue for decay storage.

Storage casks cool by natural convection that would be interrupted by installation of backfill at closure. Thermal performance would be bounded by the hard rock backfilled concept (Section 6.2), although this concept would likely be less hot because the large size of surface storage casks would spread the heat flux out and reduce thermal gradients in the surrounding backfill. Also, for the subterranean concept (Section 7.2), optimal buffer properties would be ensured during pre-construction, and the backfill emplaced at closure would contact only the top plug.

The most direct access for waste handling would be via shallow ramp (a few percent maximum grade, accessible to heavy-haul equipment similar to that used to transport storage casks at the surface) into the side of an escarpment in mountainous terrain. The facility could be built above or below the regional water table, which could impact the steps taken to seal the openings at closure.

These concepts would require a host medium suitable for construction and maintenance of long galleries (totaling tens of km) with spans of approximately 7 m, and height of at least 8 m (for vertical emplacement) to accommodate transport and emplacement equipment. To optimize constructability and geohydrologic performance the host medium would be compositionally uniform and relatively unfractured (e.g., relatively young granite, or thick volcanic tuffs, with bulk permeability of 10^{-15} m^2 or less).

4.7.1 Surface Storage Systems Emplaced Underground in Unsaturated-Zone Galleries

This concept is close to that proposed originally (McKinley et al. 2001, 2006 and 2008). It would use existing dry cask storage systems, which would be relocated in large galleries or caverns underground. Ramp access would be needed to move the heavy shielded casks.

The means to limit groundwater contact with the casks after repository closure would be installed around the casks, starting during initial construction, e.g., by emplacing storage casks on engineered pads of low-permeability material with sufficient mechanical strength for long-term stability. This could be similar to hydraulic containment liners for landfill disposal applications, which are typically constructed using mixtures of sand and clay (Kenney et al. 1992). Such construction might be especially effective in the unsaturated zone where pore pressures and groundwater flow velocities are minimal (Hardin and Sassani 2010).

Surface storage casks for DPCs cool by natural convection into the surrounding air. Emplacement galleries would be ventilated using a combination of forced and natural convection to remove this heat. Conditions would be dry during ventilation, especially for host rock of sufficiently low permeability, or in the unsaturated zone. Operations to close the facility would consist of removing services such as electrical conductors, and backfilling with granular material. Closure could proceed when heat output decays sufficiently to maintain backfill temperature at prescribed limits.

This concept would use the storage casks already deployed wherever DPCs exist, and it could also be used for self-shielded containers such as CASTOR casks. Use of existing hardware could limit disposal cost. The concept is similar in principle to in-drift disposal in crystalline rock (Section 1.3). It has not been evaluated in the technical literature and would require significant R&D to understand feasibility, for example: 1) investigate the performance of licensed storage casks in the underground environment, both preclosure and postclosure; 2) evaluate whether the concept would function differently in the unsaturated zone; 3) define the geometry, materials, and waste isolation performance of the backfill and other engineered barriers installed around the storage casks; and 4) simulate temperature histories for the casks and the EBS.

4.7.2 Transfer DPCs into Purpose-Built, Self-Ventilating, Large-Vertical-Borehole Casks

This concept would use specially built vaults in an underground facility to store, and eventually dispose of SNF in DPCs. Emplacement could be horizontal or vertical; in either case vaults would be constructed using low-permeability material, while also providing for cooling by natural convection. Vaults would be similar to surface storage concepts such as the NUHOMS® systems (horizontal) or the subterranean Hi-Storm 100 system (vertical), but with added features (e.g., low-permeability buffer) for waste isolation after closure. Vault design would incorporate features of an enclosed mode (Sections 4.1 through 4.3) with heat dissipation capability of an open mode during repository operation. This concept is similar to that originally proposed (Section 4.7.1) but with vaults built for both preclosure storage and postclosure waste isolation. Further R&D would be needed to pursue these concepts, for example, developing a configuration for the subterranean storage system that accepts a range of existing DPC types, facilitates heat removal, and ensures long-term waste isolation after closure.

4.8 Simplification of DPC Direct Disposal Concept Lists

For safety strategy discussion (Section 6) the enclosed modes that are not well suited for DPC direct disposal on thermal considerations (within the 150-year timeframe) are eliminated. Also, the low- and high-thermal cases of the sedimentary unbackfilled, open, in-drift concept can be combined because they are essentially the same with different host rock temperature distributions. Thus, the 13 items listed in Table 4-1 can be reduced to six general concepts (Table 4-3).

For performance assessment (Section 7), the backfilled crystalline-open and sedimentary-open (ventilated) concepts are simpler to analyze because rockfall, drift collapse and seismic ground motion hazards are potentially much less important. Hence, the backfilled concepts are recommended in Section 7 for initial analysis. Also, the cavern-retrievable concept is basically similar to the hard-rock backfilled case (with different geometry and packaging) for purposes of modeling potential waste isolation performance. Thus, the six concepts for which safety strategies are discussed (Section 6) can be reduced to three as a starting point for performance assessment (Table 4-3).

Table 4-3. Simplification schemes for DPC direct disposal concepts

Subsection (this section)	**Concept**	**Safety Case Discussion (Section 6)**	**Performance Assessment Discussion (Section 7)**
4.1.X and 4.3	Crystalline and Sedimentary (clastic) Enclosed Concepts	(not well suited for DPC direct disposal on thermal considerations)	
4.2	Salt Concept	**Salt** Concept	**Salt** Concept
4.4.1	Sedimentary Unbackfilled, Open, In-Drift Low-Temperature Case	**Sedimentary Unbackfilled, Open, In-Drift**	**Sedimentary Backfilled, Open, In-Drift (unbackfilled option)**
4.4.2	Sedimentary Unbackfilled, Open, In-Drift, High-Temperature Case		
4.5	Sedimentary Backfilled, Open, In-Drift	**Sedimentary Backfilled, Open, In-Drift**	
4.6.1	Hard-Rock Unbackfilled, Open, In-Drift	**Hard-Rock Unbackfilled, Open, In-Drift**	**Hard Rock Backfilled, Open, In-Drift (unbackfilled option)**
4.6.2	Hard-Rock Backfilled, Open, In-Drift	**Hard-Rock Backfilled, Open, In-Drift**	
4.7.1	Surface Storage Systems Emplaced Under- ground in Unsaturated-Zone Galleries	**Cavern-Retrievable**	
4.7.2	Transfer DPCs into Purpose-Built, Self-Ventilating, Borehole Casks		

4.9 Summary

A number of alternatives have been identified for direct disposal of DPCs (up to 32-PWR size, or BWR equivalent). Meeting maximum temperature targets for EBS materials and the near-field host rock is potentially a significant factor in developing concepts and determining feasibility. Control of postclosure criticality is also important, and the alternatives presented here offer features external to the DPC such as salt host rock, low-permeability host rock and/or engineered materials, and corrosion resistant packaging, that could help limit conditions leading to in-package postclosure criticality (Section 8). Other factors, such as opening stability and ground support, waste package transport and emplacement, and shaft vs. ramp access, are also important and depend on site-specific characteristics and possibly local experience and preference.

Thermal analysis indicates that the salt concept and the unbackfilled hard rock concepts could accept SNF in 32-PWR size packages, with SNF burnup to 60 GW-d/MT, and approximately 50 to 100 years decay storage depending on burnup. These repository concepts could close within the 150-year timeframe adopted for this study (Section 2), while meeting target values for peak host rock temperature (approximately 200°C in both types of media, based on rock characteristics specific to salt and hard rock).

In sedimentary media with lower thermal conductivity and a lower target value for peak rock temperature (100°C is used but higher values could be supportable) only lower burnup SNF (the 20 GW-d/MT case) could be accommodated given the 150-year timeframe. Disposal of higher burnup SNF might meet the same peak temperature targets, using longer decay storage and repository ventilation periods, or making an allowance for temperature exceedance within a limited region of the near-field host rock (Section 5.3).

For backfilled concepts in sedimentary or hard rock, with higher burnup SNF (40 GW-d/MT or greater) a modified concept would be needed that uses some combination of longer decay storage plus ventilation, and/or peak buffer/backfill temperature targets greater than 100°C. The enclosed emplacement modes identified previously (Hardin et al. 2012) for crystalline and sedimentary media would not likely meet a peak buffer temperature target without decay storage much longer than the assumed timeframe.

The cavern retrievable storage and disposal concept was first proposed about a decade ago and remains an important alternative that combines the heat removal performance of an open mode prior to closure, with an enclosed mode after closure. The storage casks would be shielded, facilitating facility management and closure operations underground. One variation would use existing storage casks, transported to the repository and emplaced underground, where they are eventually surrounded by low-permeability backfill for permanent disposal. Another concept would use standardized cask designs that accommodate a wide range of existing DPC types, and are constructed underground, where they are optimized for permanent SNF disposal (e.g., built-in buffers). The additional cost of the latter idea might be justified if underground storage/disposal was undertaken in lieu of centralized surface storage. The cavern-retrievable concepts are being investigated internationally but have not received much attention in the U.S.

Various R&D needs are identified in this section, associated with alternative DPC disposal concepts. This information anticipates later steps in the DPC disposal feasibility evaluation where key issues will be addressed (Howard et al. 2012). Some of the major needs identified in this section are included in the summary of key issues (Section 10). In addition, as pointed out in the introduction, the concepts described in this report will continue to be described and refined, including handling and transport of large, heavy packages, as needed to evaluate and communicate the technical issues related to feasibility of DPC direct disposal.

References for Section 4

Andra 2005. *Dossier 2005 argile – architecture and management of a geological disposal system.* December 2005. http://www.Andra.fr/international/download/Andra-international-en/document/ editions/268va.pdf.

Apted, M. 1998. "A Modest Proposal: A Robust, Cost-Effective Design for High-Level Waste Packages." *Mat. Res. Soc. Symp. Proc*. V. 506, pp. 589–596.

BSC (Bechtel-SAIC Co.) 2008a. *Transport and Emplacement Vehicle Envelope Calculation*. U.S. Department of Energy, Office of Civilian Radioactive Waste Management. 800-MQC-BEO0000100-000-00C.

BSC (Bechtel-SAIC Co.) 2008b. *Postclosure Analysis of the Range of Design Thermal Loadings*. U.S. Department of Energy, Office of Civilian Radioactive Waste Management. ANL-NBS-HS-000057 REV 00. January, 2008.

Callahan, G.D. 1999. *Crushed Salt Constitutive Model*. Sandia National Laboratories, Albuquerque, NM. SAND98-2680.

Carter, J.T., F. Hansen, R. Kehrman, and T. Hayes 2011. *A generic salt repository for disposal of waste from a spent nuclear fuel recycle facility*. SRNL-RP-2011-00149 Rev. 0. Aiken, SC: Savannah River National Laboratory.

de la Vergne, J. 2003. *Hard Rock Miner's Handbook*. Edition 3. McIntosh Engineering, Tempe, Arizona.

DOE (U.S. Department of Energy) 2008. *Yucca Mountain Repository License Application for Construction Authorization*. DOE/RW-0573. Washington, D.C.: U.S. Department of Energy.

EPRI (Electric Power Research Institute) 2010. *EPRI Review of Geologic Disposal for Used Fuel and High Level Radioactive Waste: Volume III – Review of National Repository Programs.* Final Report, December, 2010. Intera, Inc. for the Electric Power Research Institute. #1021614.

Fairhurst, C. 2012. *Current Approaches to Surface-Underground Transfer of High-Level Nuclear Waste*. Itasca Consulting Group, Minneapolis, MN.

Farmer, J.C., R.D. McCright and J.N. Kass 1988. *Survey of Degradation Modes of Candidate Materials for High-Level Radioactive-Waste Disposal Containers, Overview*. UCID-21362. Lawrence Livermore National Laboratory. Livermore, CA. June, 1988.

Fox, M.J. and R.D. McCright 1983. *An Overview of Low Temperature Sensitization*. UCRL-15619. Lawrence Livermore National Laboratory. Livermore, CA.

Graf, R., K.-J. Brammer and W. Filbert 2012 (in German). "Direkte Endlagerung von Transport- und Lagerbehältern - ein umsetzbares technisches Konzept." *Jahrestagung Kerntechnik 2012*, Stuttgart, May, 2012.

Greenberg, H.R., M. Sharma, M. Sutton and A.V. Barnwell 2012. *Repository Near-Field Thermal Modeling Update Including Analysis of Open Mode Design Concepts.* LLNL-TR-572252. U.S. Department of Energy, Used Fuel Disposition R&D Campaign. August, 2012.

Hansen, F.D., E.L. Hardin, R.P. Rechard, G.A. Freeze, D.C. Sassani, P.V. Brady, C.M. Stone, M.J. Martinez, J.F. Holland, T. Dewers, K.N. Gaither, S.R. Sobolik, and R.T. Cygan 2010. Shale

Disposal of U.S. High-Level Radioactive Waste. SAND2010-2843. Sandia National Laboratories. Albuquerque, NM. May, 2010.

Hardin, E. and D. Sassani 2011. "Application of the Prefabricated EBS Concept in Unsaturated, Oxidizing Host Media." Proceedings: 13th International High-Level Radioactive Waste Management Conference, Albuquerque, NM. April, 2011. American Nuclear Society. Paper #3380.

Hardin, E., J. Blink, H. Greenberg, M. Sutton, M. Fratoni, J. Carter, M. Dupont and R. Howard 2011. *Generic Repository Design Concepts and Thermal Analysis (FY11)*. FCRD-USED-2011-000143 Rev. 2. December, 2011. U.S. Department of Energy, Used Fuel Disposition R&D Campaign.

Hardin, E., T. Hadgu, D. Clayton, R. Howard, H. Greenberg, J. Blink, M. Sharma, M. Sutton, J. Carter, M. Dupont and P. Rodwell 2012. *Repository Reference Disposal Concepts and Thermal Management Analysis*. FCRD-USED-2012-000219 Rev. 2. November, 2012. U.S. Department of Energy, Used Fuel Disposition R&D Campaign.

Hardin, E. 2013. *Temperature-Package Power Correlations for Open-Mode Geologic Disposal Concepts*. SAND2013-1425. Sandia National Laboratories. Albuquerque, NM. February, 2013.

Hardin, E. and M. Voegele 2013. *Alternative Concepts for Direct Disposal of Dual Purpose Canisters*. FCRD-UFD-2013-000102, Rev.0. U.S. Department of Energy, Used Fuel Disposition R&D Campaign.

Jove-Colon, C. et al. 2012. *Evaluation of Generic EBS Design Concepts and Process Models: Implications to EBS Design Optimization*. U.S. Dept. of Energy, Office of Used Nuclear Fuel Disposition, FCRD-USED-2012-000140.

Kessler, J., A. Sowder and M. Kozak 2008. *Feasibility of Direct Disposal of Dual-Purpose Canisters in a High-Level Waste Repository*. Palo Alto, CA: Electric Power Research Institute. #1018051.

McKinley, I.G., H. Kawamura and H. Tsuchi 2001. "Moving HLW-EBS Concepts in to the 21st Century." *Mat. Res. Soc. Symp. Proc.* v. 663.

McKinley, I.G., F.B. Neall, H. Kawamura and H. Umeki 2006. "Geochemical optimization of a disposal system for high-level radioactive waste." *Journal of Geochem. Explor*., v. 90, pp. 1–8.

McKinley, I.G., M. Apted, H. Umeki and H. Kawamura 2008. "Cavern disposal concepts for HLW/SF: assuring operational practicality and safety with maximum programme flexibility." *International Technical Conference on the Practical Aspects of Geological Disposal of Radioactive Waste*. June 16-18, 2008. Prague, Czech Republic.

Munson, D.E., A.F. Fossum and P.E. Senseny 1989. *Advances in Resolution of Discrepancies between Predicted and Measured WIPP In-situ Room Closures*. Sandia National Laboratories, Albuquerque, NM. SAND88-2948.

Robinson, B.A., N.Z. Elkins and J.T. Carter 2012. "Development of a U.S. Nuclear Waste Repository Research Program in Salt." *Nuclear Technology*. v. 180, N. 1. October, 2012. pp. 122-138.

SKB (Swedish Nuclear Fuel and Waste Management Co.) 2006. *Long-term safety for KBS-3 repositories at Forsmark and Laxemar — A first evaluation.* Technical Report TR-06-09.

SKB (Swedish Nuclear Fuel and Waste Management Co.) 2010. *Buffer, backfill and closure process report for the safety assessment SR-Site.* Technical Report TR-10-47.

SKB (Swedish Nuclear Fuel and Waste Management Co.) 2011. *Long-term safety for the final repository for spent nuclear fuel at Forsmark: Main report of the SR-Site project.* (3 volumes). Technical Report TR-11-01.

Sutton, M., J.A. Blink, M. Fratoni, H.R. Greenberg and A.D. Ross 2011. *Investigations on Repository Near-Field Thermal Modeling – Repository Science/Thermal Load Management & Design Concepts (M41UF033302) Rev. 1.* Lawrence Livermore National Laboratory, LLNL-TR-491099-REV-1.

THIS PAGE INTENTIONALLY LEFT BLANK

5. Thermal Analysis Approaches

Calculations of peak temperature at the waste package surface and in the host rock are used in Section 4 to identify disposal concepts with thermal management aspects suitable for dual-purpose canister (DPC) direct disposal. In general, thermal analyses are used in this study to evaluate:

- Peak temperatures in the host rock and engineered barrier system (EBS)
- Timing of disposal (i.e., whether temperature targets can be met in accordance with timing assumptions in Section 2)
- Total repository area (i.e., waste package and drift spacings; see Table 6-1)
- R&D needs (Section 10).

With the exception of the salt concept, all thermal results presented in this report were calculated using the semi-analytical solution discussed below. For the salt concept a finite-element model was developed to understand the effects from thermally activated changes in geometry and backfill thermal conductivity. Both approaches are briefly described, citing references where more details can be found.

This section also includes a parameter study for peak temperature in sedimentary host media with open emplacement, supplementing thermal analysis results from Section 4. This attention to thermal management in sedimentary media (e.g., clay/shale) is appropriate because they have relatively low thermal conductivity and low tolerance to elevated temperature. As a result, peak temperature targets in sedimentary media (e.g., 100°C in the host rock) are most challenging to meet among the media types considered.

5.1 Hard Rock and Sedimentary Open-Mode Thermal Analysis

The semi-analytical method is based on an approach developed originally for enclosed emplacement modes (Sutton et al. 2011; Hardin et al. 2011) supplemented by new features to represent open modes and the associated effects from ventilation and backfill (Greenberg et al. 2012; Hardin et al. 2012; Hardin 2013). As will be discussed in Section 5.3, the same model has been applied in new schemes to find spacings and aging durations that achieve certain thermal goals.

The solution is used to calculate: 1) temperature history at the interface between the EBS and the host rock (i.e., drift wall or borehole wall); 2) temperature history a short distance outward from the EBS interface; and 3) temperature history at selected locations within the EBS, such as the waste package surface. For calculating temperatures at the EBS interface (and outward) the model assumes homogeneous, isotropic host rock with the EBS region replaced by host rock. Heat sources are superposed using analytical solutions for instantaneous heating by a finite line source (central waste package), point sources (nearby packages), and line sources (neighboring drifts). This superposition solution is convolved numerically with the thermal decay history for a single waste package to produce a transient solution for temperature at the EBS interface (and outward). Thermal decay history data for spent nuclear fuel (SNF) with 20, 40 and 60 GW-d/MT burnup are derived from Carter et al. (2012). The 60 GW-d/MT cases are considered bounding with respect to SNF burnup.

For selected locations within the EBS a steady-state calculation is performed for each point in time, representing the EBS as a set of concentric annular layers. For each point in time the thermal power is propagated inward through the annular layers starting at the EBS interface. The EBS interface temperature is used as the temperature boundary condition on the outermost layer.

This is an approximate method because: 1) the transient solution assumes that the heat capacity of the EBS is the same as the host rock; 2) the steady state solution neglects the heat capacity of the EBS; and 3) the steady state solution is 2-D, treating the waste package and EBS as an infinitely long concentric cylindrical arrangement and neglecting waste package end effects. The effect of these approximations is probably to slightly overestimate near-field temperatures. In addition, heat transfer is limited to conduction and does not include mechanisms such as phase change and convection that could also lower near-field temperatures. Validation of the method was undertaken by comparing to a lumped parameter finite-difference model with more dimensional detail (Huff and Bauer 2012; Greenberg et al. 2013).

The analysis approach was modified to represent open emplacement modes for SNF by defining a ventilation period during which a fraction of the heat output is removed, using a constant ventilation heat-removal efficiency factor. To calculate the waste package temperature during ventilation, heat transfer across the air gap is represented by gray-body thermal radiation (Hardin 2013). At repository closure ventilation stops so heat output is increased to the full output of the waste packages. If backfill is emplaced then the air gap is replaced by conductive backfill. Further description of the approach is provided by Greenberg et al. (2013).

The remainder of this section discusses results that pertain to DPC direct disposal, using figures presented in Section 4. The representative cases use 32-PWR waste packages, 2 m in diameter and 5 m long. Drift spacing is 70 m and waste package spacing is 20 m, as proposed in Section 4 (see Section 5.3 for analysis of other spacings). The discussion focuses on the analyzed cases with 50 years of decay storage and 100 years of ventilation, which comply with the timing assumptions from Section 2.

Results for Sedimentary Open Modes – For the unbackfilled sedimentary cases (assuming that emplacement openings remain mostly intact for hundreds of years after closure) the calculated peak host rock temperature is just above 100°C for all burnup values (Figure 4-7). For the 60 GW-d/MT case the peak is less than 140°C, but greater than 100°C suggesting a need to investigate the performance consequences from heating the near-field host rock to higher temperatures (Section 10). The waste package emplacement thermal power limit needed to ensure that host rock peak temperature would be less than 100°C is approximately 10 kW (Hardin 2013). Waste package surface temperatures calculated using thermal radiation across the air gap, are only slightly higher than drift wall temperatures (Figure 4-7). The drift and waste package spacings (70 m and 20 m, respectively) were selected as reference conditions (Section 4) but analysis in Section 5.3 shows that lower temperatures could be attained with larger spacings.

With the addition of backfill the calculated host rock temperatures are unchanged, but the waste package surface temperature (and the peak backfill temperature) is much higher (Figure 4-8). This is true even for a hypothetical backfill with the thermal conductivity of fully hydrated bentonite (1.43 W/m-K in Figure 4-8; from Hardin et al. 2012, Table D-2). Thus, without longer decay storage the backfill temperature will be much greater than 100°C. The need to investigate higher temperature tolerance and/or higher conductivity backfill is addressed in Section 10. An identical problem with peak temperature targets for backfill arises for the hard rock concepts discussed below.

Results for Hard Rock Open Modes – For the unbackfilled hard rock cases (also assuming that emplacement openings remain mostly intact for hundreds of years after closure) calculated peak host rock temperatures are well below the 200°C temperature target (Figure 4-10). The waste package emplacement thermal power limit needed to ensure that host rock peak temperature would be less than 200°C is approximately 12 to 15 kW (depending on burnup; Hardin 2013). The 60 GW-d/MT bounding case is cooler than the corresponding sedimentary case (Figure 4-7) because hard rock is assigned a higher thermal conductivity in the calculation. Accordingly, no thermal management issues are identified for open drifts in hard rock, given the timing assumptions used (Section 2). This result is consistent with previous analysis (DOE 2008). Note that rockfall and drift collapse debris could bury waste packages, especially with significant seismic ground motion, and that the debris has low thermal conductivity (BSC 2008). Even the 60 GW-d/MT case exhibits a peak temperature near 100°C with closure at 150-year SNF age, suggesting that sub-boiling rock temperatures might be possible for SNF of any burnup, in large packages in hard rock, by using small adjustments to spacings and decay storage time (and the approach presented in Section 5.3).

With the addition of backfill the host rock temperatures are unchanged, but the waste package surface temperature (and the peak backfill temperature) is much higher (Figure 4-11), as noted above for sedimentary rock. The temperature drop across the backfill is identical for sedimentary and hard rock concepts, and its magnitude is much greater than the difference in drift-wall temperature between sedimentary and hard rock concepts. The need to investigate higher temperature tolerance and/or higher conductivity backfill is addressed in Section 10.

5.2 Salt Concept Thermal Analysis Approach

The salt disposal concept was simulated using the FEM because crushed salt backfill initially has low thermal conductivity, and is subject to thermally-activated creep reconsolidation during the first few decades after emplacement. After substantial reconsolidation (e.g., to 95% fractional density) thermal conductivity approaches that of intact salt, which is the most conductive medium investigated. Also, the process of reconsolidation is nonuniform in response to thermal-mechanical conditions, and the geometry of salt disposal concepts (e.g., alcove emplacement) was considered to be more complex than simpler in-drift emplacement concepts. The FEM approach used is straightforward and could also be used to generate more accurate thermal analyses for other concepts.

Finite Element Solution – FEM models described here were run using the coupled Sierra codes, in a high-performance computing environment at Sandia. The FEM model used initially (see Hardin et al. 2012, Appendix C) was run using well-known creep models for intact and crushed salt. The initial crushed salt backfill porosity was set to 20% to accelerate numerical performance (preventing the backfill from consolidating under its own weight and pulling away from the roof, creating a free surface). Peak salt temperature was similar for fully coupled thermal-mechanical simulations, and thermal-only runs with the 20% backfill porosity. Model grid geometry, boundary conditions, waste package representation, and other aspects are discussed by Hardin et al. (2012). The model situated waste packages in semi-cylindrical cavities in the salt floor, enhancing heat transfer.

Alcove Emplacement Mode – After further constitutive model development (Jove-Colon et al. 2012) the crushed salt porosity could be initialized to 35%, and the mechanisms for creep consolidation were modified to include pressure solution (with fixed 1% volumetric moisture content). At the same time, heat generation functions for SNF were updated to be consistent with Carter et al. (2012). These developments introduced slightly higher peak salt temperatures, which occur early in the simulations, because backfill consolidation began at higher porosity (and lower thermal conductivity).

Peak salt temperature occurs at the waste package surface where it contacts the crushed salt backfill. With fewer than 100 years of decay storage, and alcove emplacement, peak salt temperatures are less than 200°C even for high burnup (Figure 5-1). For larger package spacings (e.g., 30 m horizontal grid) peak salt temperature occurs within just a few years after emplacement, whereas more closely spaced packages (e.g., 20-m grid) interact and produce peak temperatures several decades later (Figure 5-2). Note that these calculations represent a repository in a thick salt unit (e.g., a salt dome, or bedded salt on the order of 30 m thick). The waste package emplacement thermal power limit needed to ensure host rock peak temperature less than 200°C is approximately 10 kW (Hardin et al. 2012, Figure D-5).

Evaporite sequences are typically thick and may contain beds of anhydrite, clay, or other impurities. Where these beds are thin or represent a small portion of the overall thickness (mostly halite) the thermal effect is minor. If the host salt bed is enclosed by thick, lower conductivity strata, the calculated temperatures could be higher (requiring longer decay storage).

In-Drift Emplacement Mode – The original set of calculations (Hardin et al. 2012) represented alcove emplacement with packages emplaced transversely against the back wall of the alcove. Later calculations compared in-drift emplacement, with packages placed axially, along the centerline of a long drift. The in-drift arrangement facilitates emplacement of large, heavy packages using existing types of equipment (Figure 4-2). Temperature histories for fully coupled thermal-mechanical analysis of in-drift emplacement (without cavities in the floor) are shown in Figure 5-3, and the resulting history of crushed salt backfill porosity (averaged over the entire drift segment for one waste package) is shown in Figure 5-4.

To accelerate runs comparing effects on peak temperature from decay storage time, burnup, cavities in the floor, etc., a representative history of average backfill porosity from fully-coupled thermal-mechanical simulation (Figure 5-4) was used as input to thermal-only runs. The approach is approximate since backfill consolidation locally proceeds faster or slower than the average, giving rise to slight differences in the calculated temperature histories (Figure 5-3). For example, backfill consolidates faster directly above the waste package which acts as a rigid inclusion, increasing the local compressive strain in response to roof closure. Note, however that the temperature dependence of the crushed salt creep rate is uncertain, so the timing of backfill porosity changes is uncertain and an approximate approach is justified.

Temperatures for in-drift emplacement are higher because the packages are not situated close to intact salt such as the wall at the back of each alcove. For in-drift emplacement the walls of the drift are approximately 2 m from the package on each side (Figure 4-2). A comparison of alcove and in-drift emplacement, for 32-PWR size packages with 60 GW-d/MT burnup SNF aged 70 years at emplacement, and packages placed in cavities in the floor, is shown in Figure 4-1. When semi-cylindrical cavities or longer decay storage (100 vs. 70 years) are included in the model, peak salt temperature is significantly decreased (Figure 5-5).

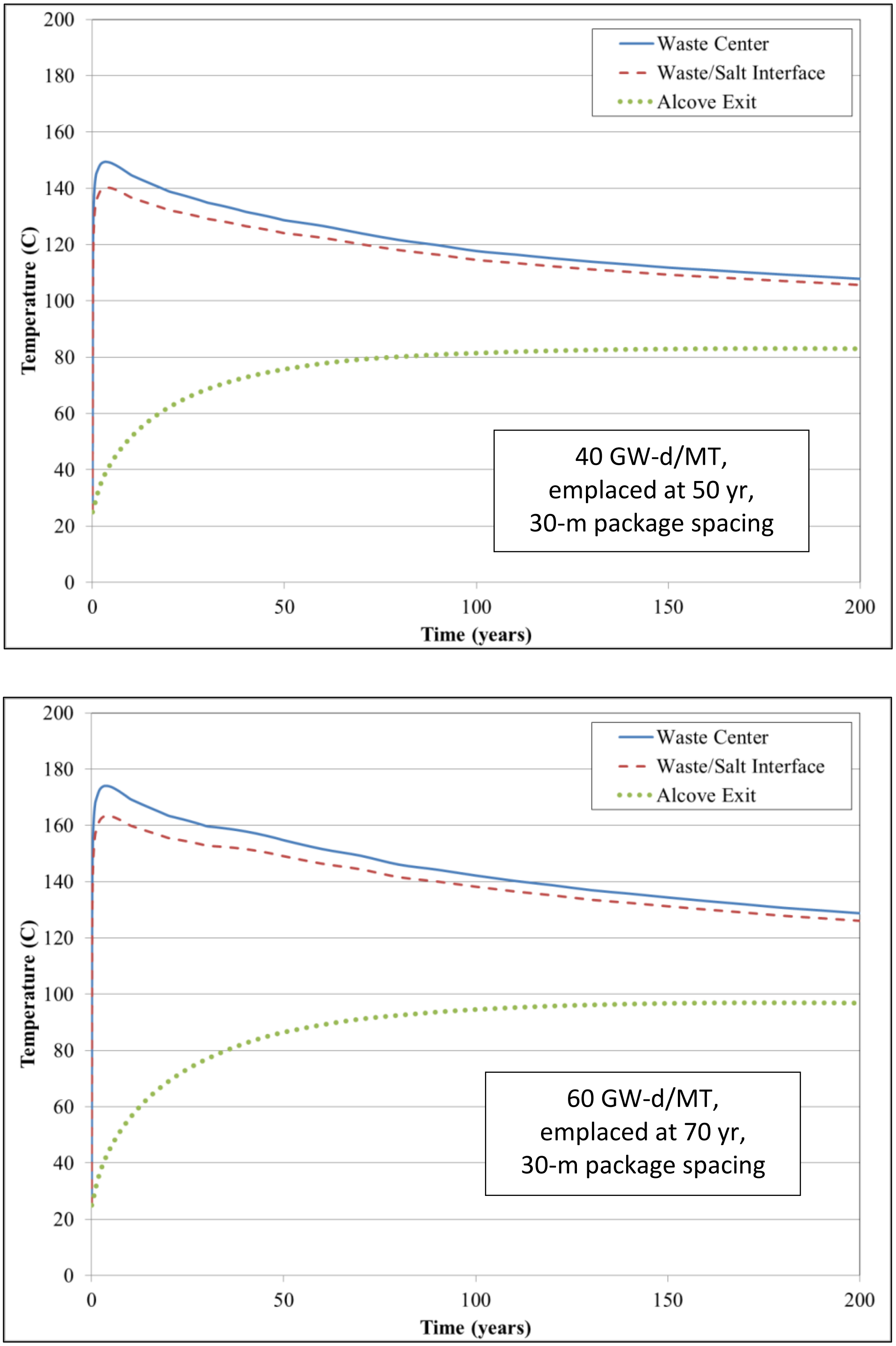


Figure 5-1. Temperature histories for 32-PWR size packages, for the salt concept, with alcove emplacement, and moderate (upper) and high burnup (lower).

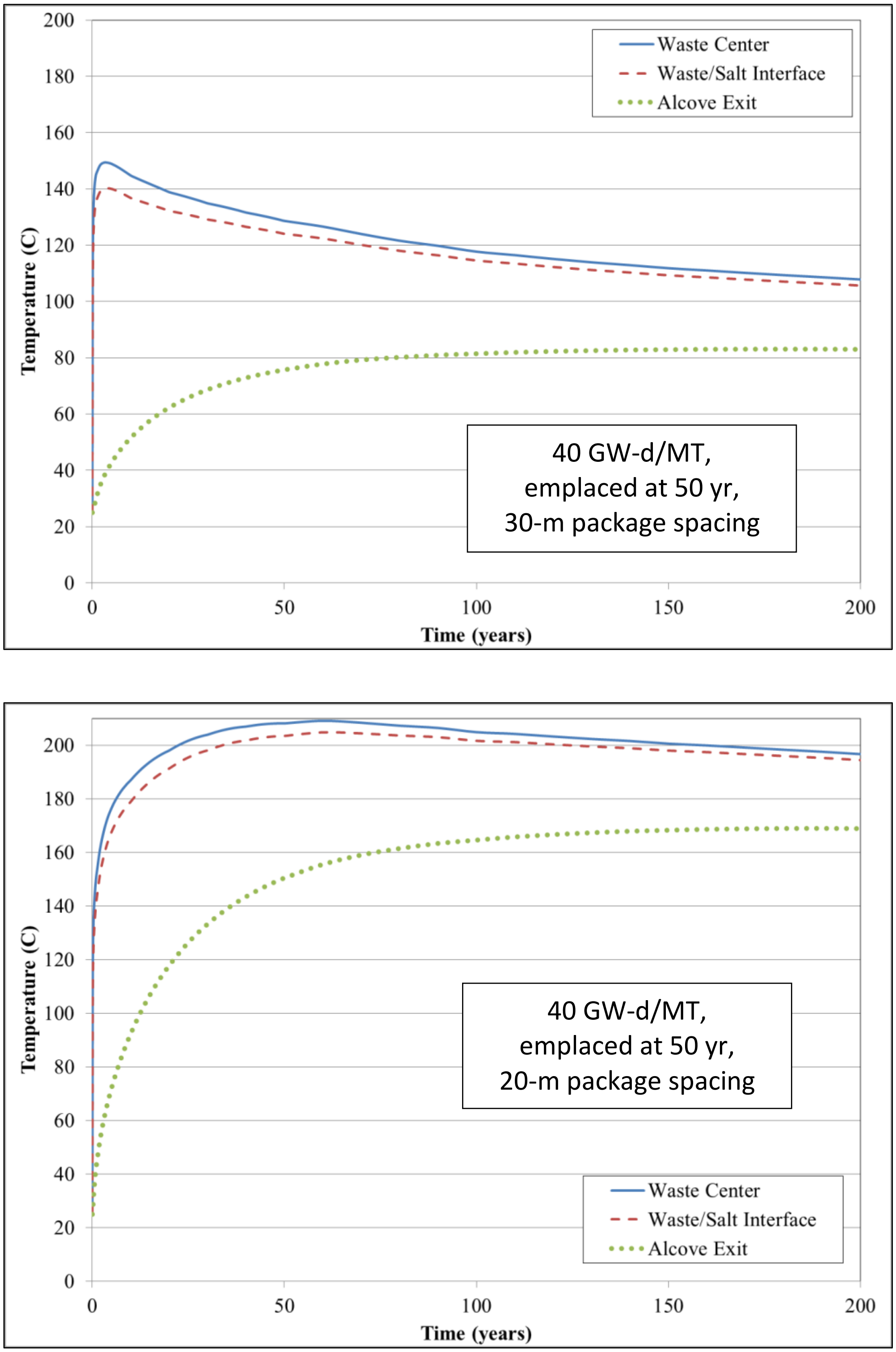
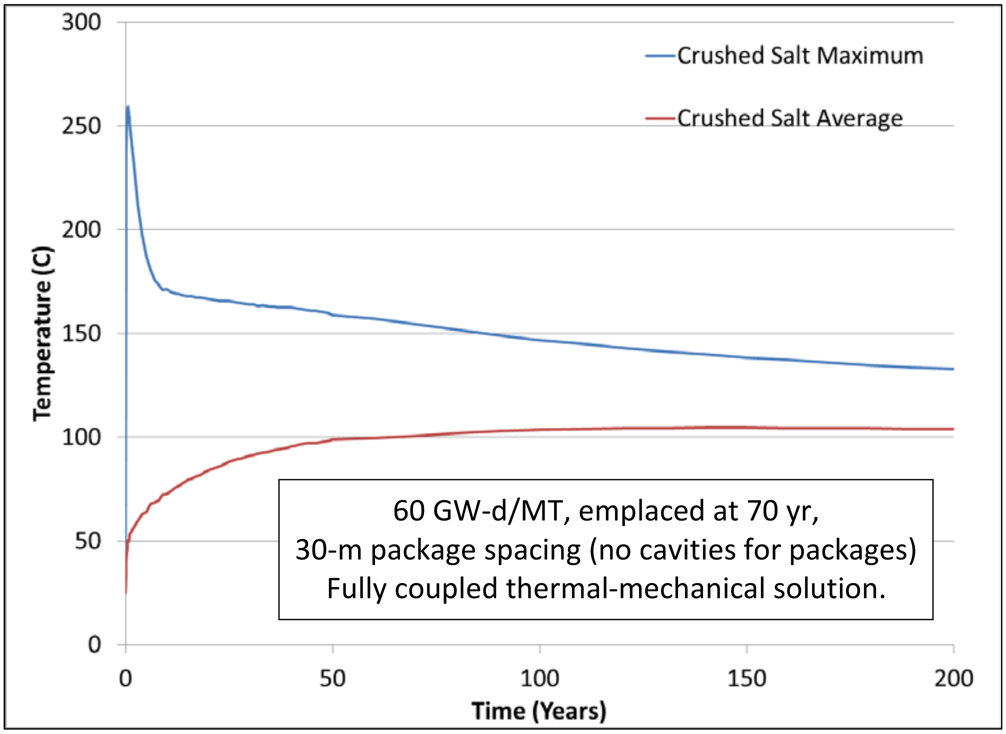


Figure 5-2. Temperature histories for 32-PWR size packages, for the salt concept, with alcove emplacement, and 30-m (upper) and 20-m package spacing (lower).



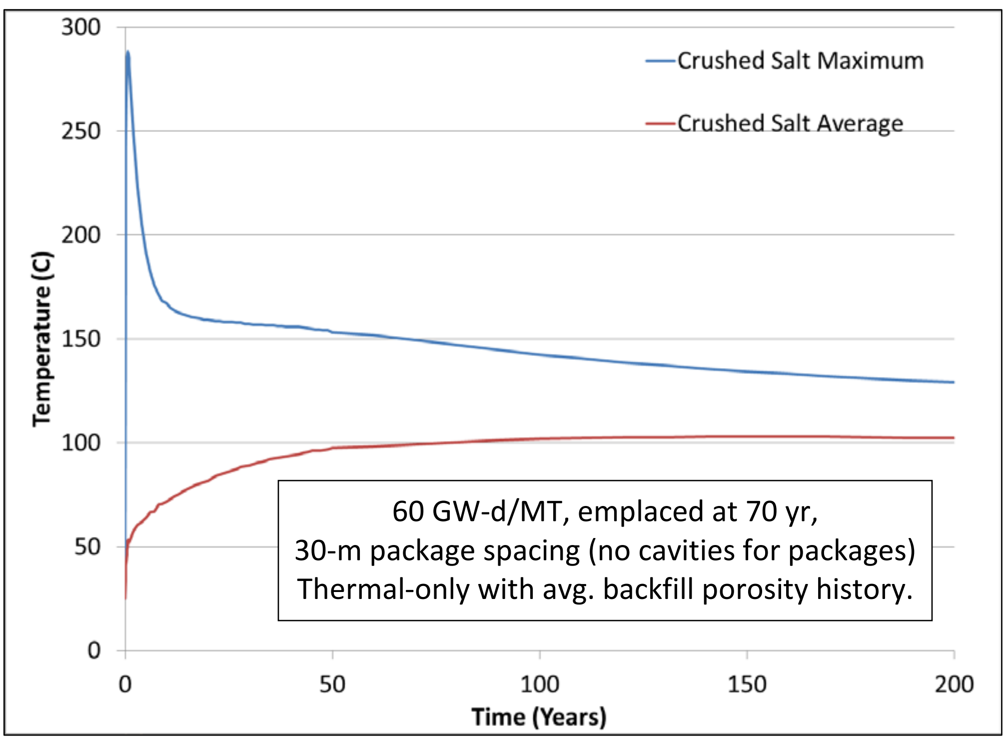
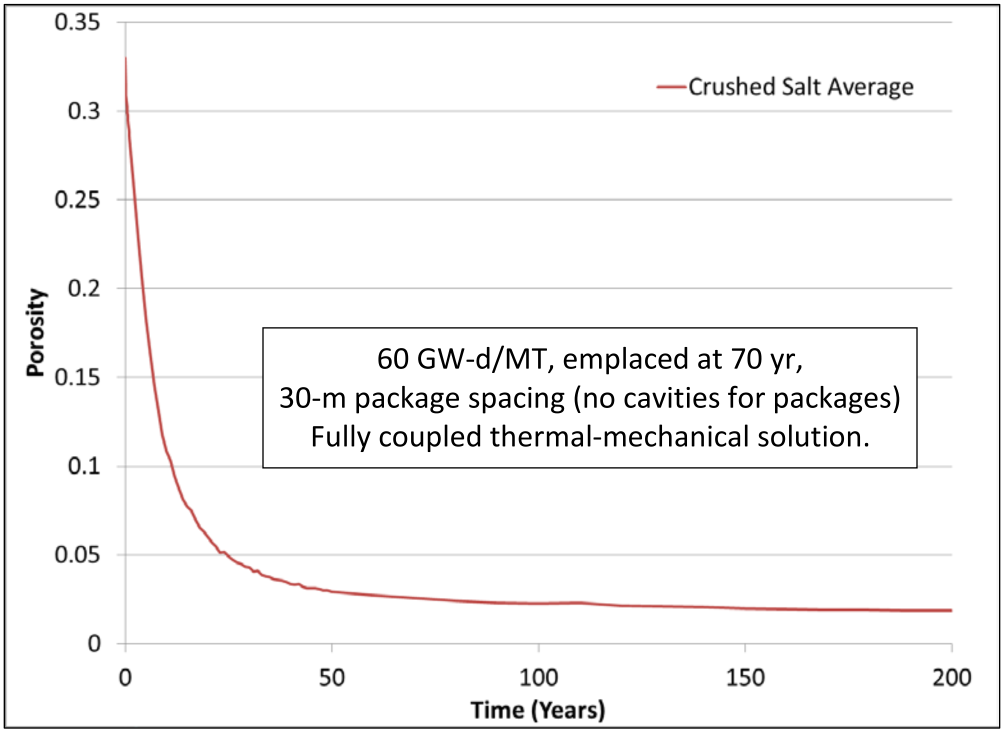


Figure 5-3. Temperature histories for 32-PWR size packages, for the salt concept, with in-drift emplacement and 30-m spacing, comparing fully coupled T-M (upper) with backfill porosity history approximation (lower).

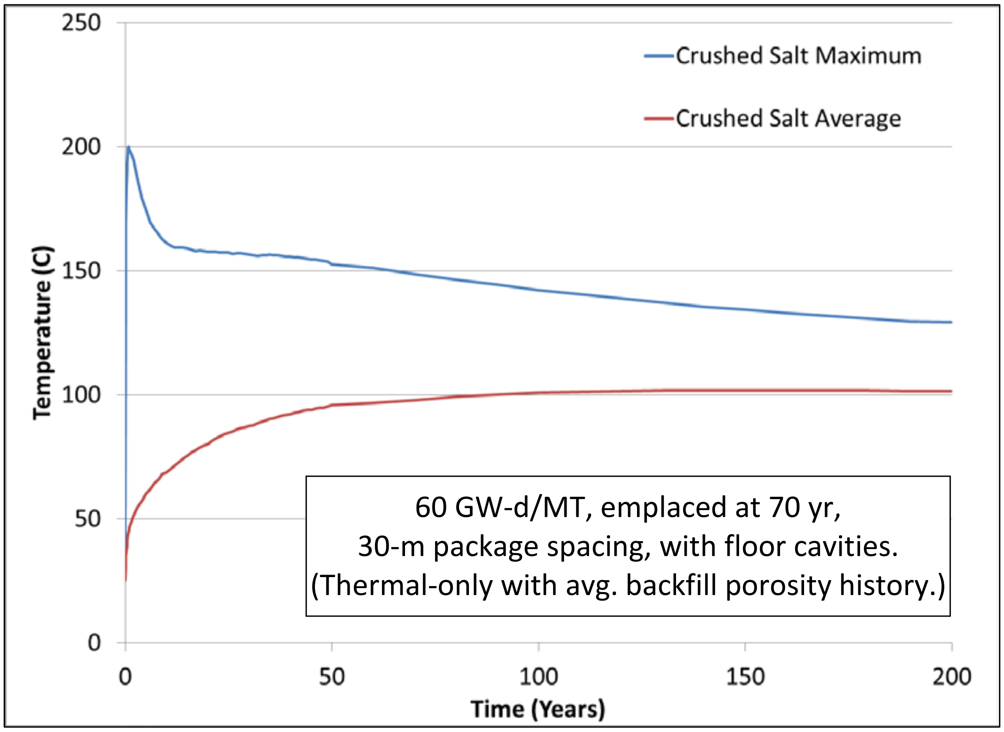


Note: Backfill porosity is averaged at each point in time, over the entire volume surrounding a waste package.

Figure 5-4. Average backfill porosity history for 32-PWR size packages, for the salt concept, with in-drift emplacement and 30-m spacing, and fully coupled thermal-mechanical solution.

Ventilated Salt Concept – A proposal for a ventilated repository for heat-producing waste in salt ("hybrid" approach) was analyzed previously (Hardin et al. 2012, Appendix C) and is relevant to thermal management for DPC direct disposal.

Salt is not suited to open emplacement modes because the closure of emplacement drifts will accelerate at elevated temperature. Accordingly, to stay open for heat removal the ventilated openings must be maintained at close to the ambient salt temperature, which means they must be vigorously ventilated and situated a few meters away from the waste packages. To test the idea, the finite element model was modified to include ventilation drifts between each set of alcoves (parallel to the access drifts) with circular cross-section to maximize stability. By comparison, the disposal alcoves and access drifts would be rectangular to promote closure. The duration of ventilation could be a few years to several decades, depending on cooling needs. The results (Hardin et al. 2012, Table C-4) show that ventilation could reduce peak temperatures by approximately 50 C° (for temperatures in the range 200 to 300°C) for a moderate ventilation rate. Temperature rise at the access drift wall is only a few degrees, for ventilation air introduced at the ambient temperature of 27°C. Only a fraction of the SNF heat generation rate was actually removed by ventilation.



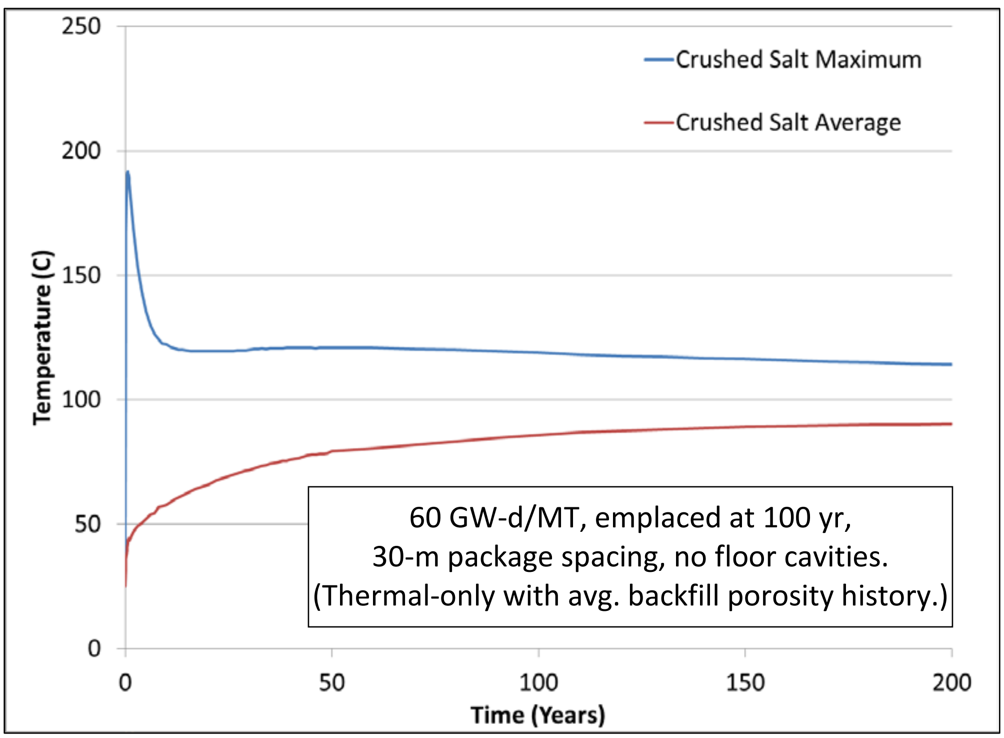


Figure 5-5. Temperature histories for high burnup SNF in 32-PWR size packages, for the salt concept, showing ways to lower temperature with in-drift emplacement: floor cavities (upper) and 100-year decay storage (lower).

To provide ventilation the circular drifts must remain open with reasonable maintenance. To evaluate long-term stability, the plane-strain mechanical response of a circular opening was simulated at various values of the (homogeneous) rock temperature. The closure rate increases with temperature (Hardin et al. 2012, Figure C-7). For the opening to close by 10% of its original diameter at 25°C would take about 33 years, while this would take 14 years at 40°C. Thus, ventilation could be effective for a few decades, effectively limiting peak salt temperatures in the first 10 to 20 years. This performance is equivalent to, and trades directly against a similar duration of extended surface decay storage. Finally, note that peak salt temperature is the most stringent test on the effectiveness of heat removal by ventilation, and that ventilation would be much more effective at limiting the average temperature rise across the repository footprint, if that proves to be a more important constraint.

Summary – FEM simulations of the salt disposal concept show that thermal-mechanical consolidation of the crushed salt backfill occurs in tens to hundreds of years, and acts to decrease near-field repository temperatures. Taking backfill consolidation in to account, peak salt temperatures (at the waste package surface) are calculated to be less than 200°C for 32-PWR size packages containing high-burnup SNF, with sufficient package spacing (30 m), surface decay storage (70 years), and measures to enhance heat transfer to the intact salt (see below).

Axial emplacement of waste packages on the floor in long, parallel drifts is a few tens of degrees hotter than transverse emplacement in separate alcoves (because of the proximity of the package to intact salt). Emplacement of waste packages in semi-cylindrical cavities in the floor improves this heat transfer by an amount that is similar in magnitude. A sensitivity case looked at ventilation of a dedicated (no waste), parallel drift for heat removal. The results showed that while a fraction of the generated heat could be removed, the effect is directly comparable to a few years to tens of years of additional surface decay storage.

5.3 Parametric Study of Spacings for Sedimentary Open Concepts

This parametric study of repository drift and waste package spacings was developed by Greenberg et al. (2013). The objectives are to explore repository layout alternatives (i.e., spacings) for DPC direct disposal, to determine: 1) what spacings could be used to meet temperature targets (100°C for argillaceous host media and backfill); and 2) how to minimize the extent of host rock around the emplacement drifts where temperature exceeds these targets. This analysis could be done for any type of host medium, but it is especially needed for sedimentary media that have both low thermal conductivity, and low tolerance for elevated temperature. A similar exercise was undertaken for the salt concept, resulting in the selection of 30-m package spacings. For media such as the hard rock category (Section 5.1) the main thermal management challenge is peak backfill temperature, which is controlled by near-field heat transfer and not significantly affected by drift and waste package spacings. Host rock peak temperature targets of 100°C and 120°C are used in this study.

Parametric analysis is done for waste package and drift spacings, assuming:

- 32-PWR size waste packages
- SNF burnup of 40 or 60 GW-d/MT
- Decay storage of 50 years before disposal

- Ventilation time of 25, 50, 75, and 100 years (after decay storage) for the 40 GW-d/MT analysis, and 150 years for the 60 GW-d/MT analysis
- Ventilation system heat removal efficiency of 75%
- Drift diameter of 4.5 m
- Drift spacing of 70 m for the 40 GW-d/MT analysis; 70 m and 90 m for the 60 GW-d/MT analysis
- Host rock thermal conductivity of 1.75 W/m-K

Note that some of the decay storage and ventilation duration values do not comply with the timing assumptions in Section 2, and are presented here only for comparison.

Parametric Spacing Study for 40 GW-d/MT Cases – The sensitivity of required ventilation duration to spacing options is presented in Figure 5-6 for cases meeting T ≤ 100°C at the drift wall, and in Figure 5-7 for T ≤ 120°C at the drift wall. For package spacing on the order of 20 m there are diminishing returns for increased drift spacing beyond approximately 70 m. Therefore, this analysis focuses on repository layout options for a range of package spacings near 20 m and drift spacing of 70 m. These values are the same spacings selected in Section 4 as reference values for disposal alternatives.

Figure 5-7 indicates that there is a range of spacings that could maintain the host rock temperature at or below 120°C, provided that the ventilation duration is at least 100 years. Figure 5-8 shows the relationship between waste package spacing and peak temperature, for several values of depth into the drift wall, and ventilation time of 100 years. The plot shows that package spacing of 23 m would ensure the peak drift wall temperature is at or below 100°C.

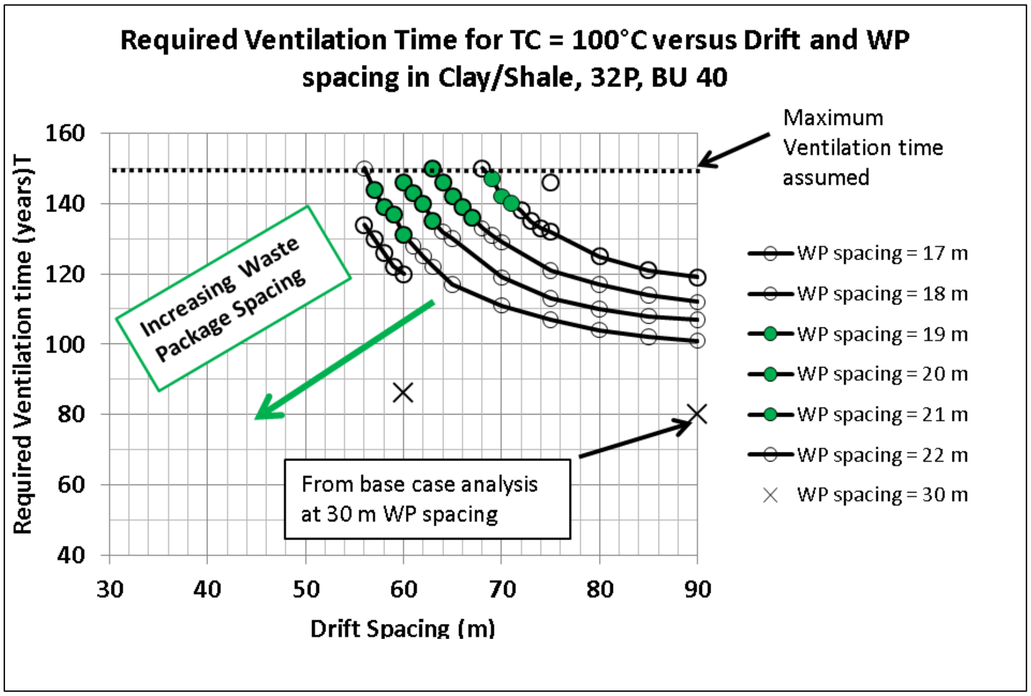


Figure 5-6. Ventilation time vs. drift spacing for drift wall T ≤ 100°C, for the sedimentary open concept, for values of package spacing (32-PWR size packages, 40 GW-d/MT burnup)

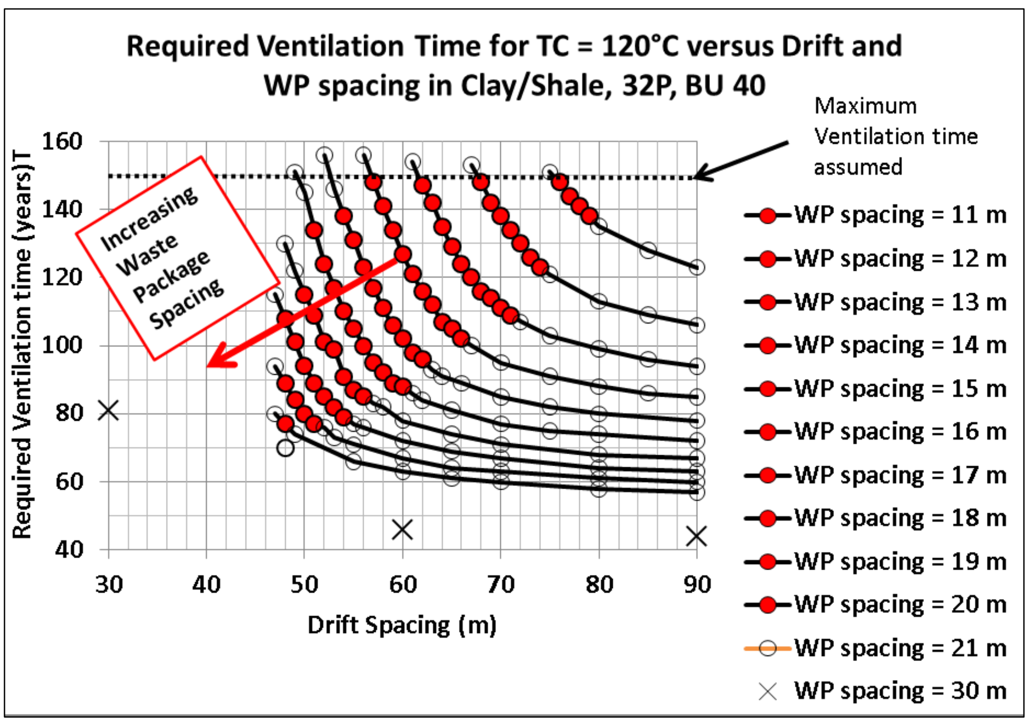


Figure 5-7. Ventilation time vs. drift spacing for drift wall T ≤ 120°C, for the sedimentary open concept, for values of package spacing (32-PWR size packages, 40 GW-d/MT burnup)

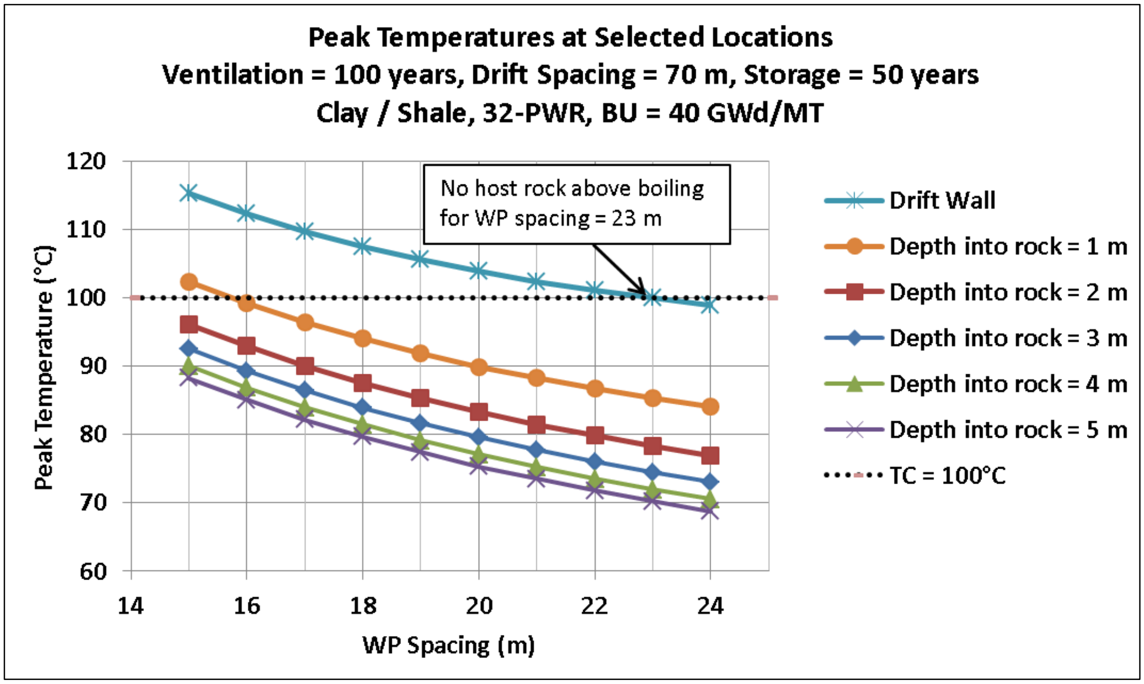


Figure 5-8. Peak temperature vs. package spacing for values of depth into the rock, for the sedimentary open concept (32-PWR size packages, 40 GW-d/MT burnup)

Another look at the same calculations is provided in Table 5-1, which presents the depth into the drift wall of the 100°C and 120°C peak temperature isotherms, as a function of ventilation time and waste package spacing. This is a measure of the extent of the host rock that would be affected by temperatures above the target. The table also indicates that package spacing of 23 m would ensure the peak drift wall temperature is at or below 100°C.

Table 5-1. Depth into the drift wall for which peak temperature equals the target value (40 GW-d/MT)

	Package Spacing (m) [B]				
	16	18	20	22	24
Ventilation Time (yr) [A]	Depth (into drift wall) to Meet T=100°C				
100	0.9	0.6	0.3	0.1	0.0
75	1.5	0.9	0.7	0.5	0.4
50	2.2	1.7	1.3	1.1	0.9
25	3.6	2.8	2.3	1.9	1.8
	Depth (into drift wall) to Meet T=120°C				
100	0.0	0.0	0.0	0.0	0.0
75	0.0	0.0	0.0	0.0	0.0
50	0.7	0.4	0.3	0.1	0.0
25	1.6	1.2	1.0	0.8	0.8
[A] After 50 years decay storage. [B] Drift spacing 70 m; SNF burnup 40 GW-d/MT.					

Figure 5-9 shows the temperature histories at the drift wall and at various depths into the host rock, for package spacing of 23 m (drift spacing 70 m). Figure 5-10 shows that the relative contributions to the peak drift wall temperature from heating by adjacent waste packages in the same drift, and by adjacent drifts, are relatively small compared to the contribution of the central waste package. This indicates that increasing the spacings would have only a small effect on the peak drift wall temperature.

Parametric Spacing Study for 60 GW-d/MT Cases – Because of the greater heat output, disposal of DPC-based, 32-PWR size waste packages with higher SNF burnup (60 GWd/MT) will need to use larger spacings and/or longer ventilation times than the cases discussed above. Extrapolating from the previous results, and the results presented in Section 4, a base case is selected here as a starting point for parametric study. The base case uses 150 years of ventilation (in addition to 50 years decay storage), 90 m drift spacing, and 30 m drift spacing, with the objective to maintain T ≤ 100°C at the drift wall. Note that this ventilation duration, and the total duration of 200 years until final closure, exceed the assumptions stated in Section 21.2. From Figure 5-11 (analogous to Figure 5-8) a package spacing of 34 m would meet this temperature target.

Based on the results obtained for the 40 GW-d/MT analysis, additional workable combinations of spacings likely exist that meet T ≤ 100°C at the drift wall. Figure 5-12 compares of 70 m and 90 m drift spacing, for package spacings from 16 m to 40 m. The results are only a few degrees hotter for the 70 m drift spacing layout, and an additional calculation (Figure 5-13) shows that a waste package spacing of 38 m, with drift spacing of 70 m, can meet drift wall T ≤ 100°C. The layout area represented by the 70 x 38 m spacings is 13% smaller than the 90 x 34 m spacings, but the length of drift required for the same waste inventory is approximately 12% greater.

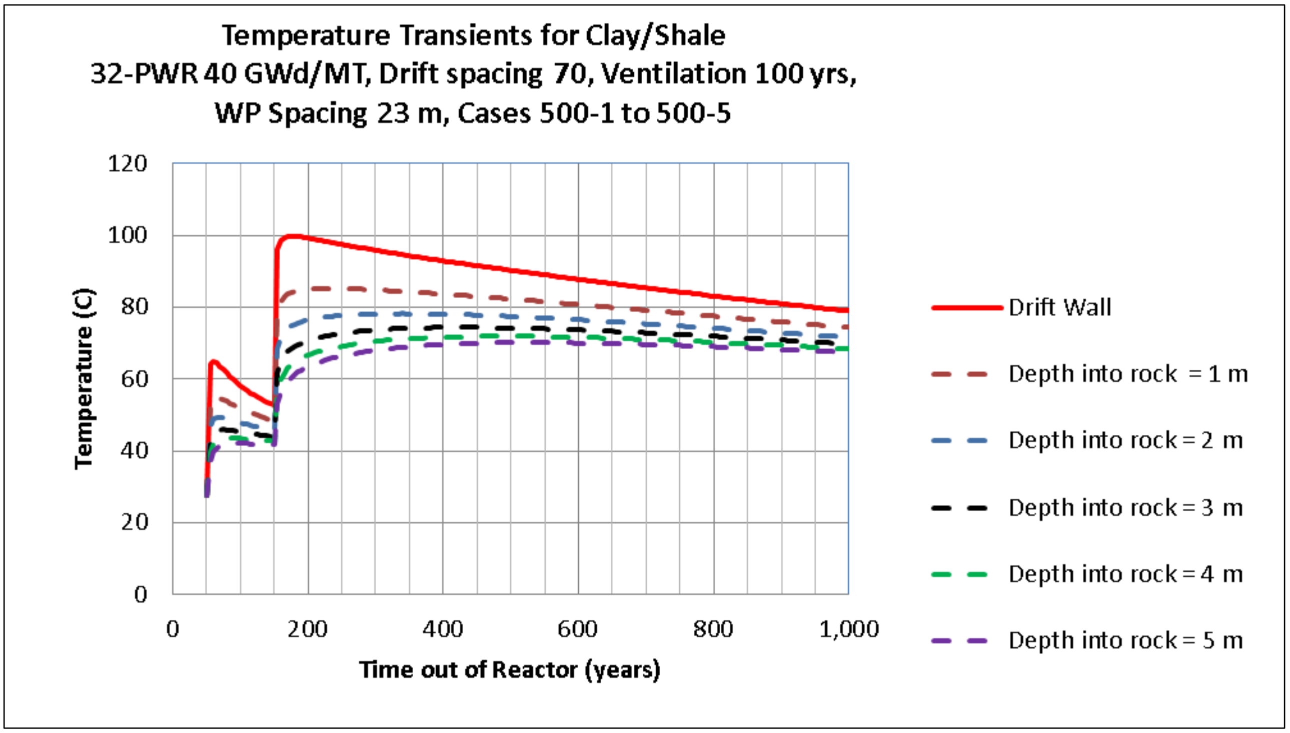


Figure 5-9. Temperature histories for values of depth into the rock, for the sedimentary open concept, for 23-m package spacing (32-PWR size packages, 40 GW-d/MT burnup)

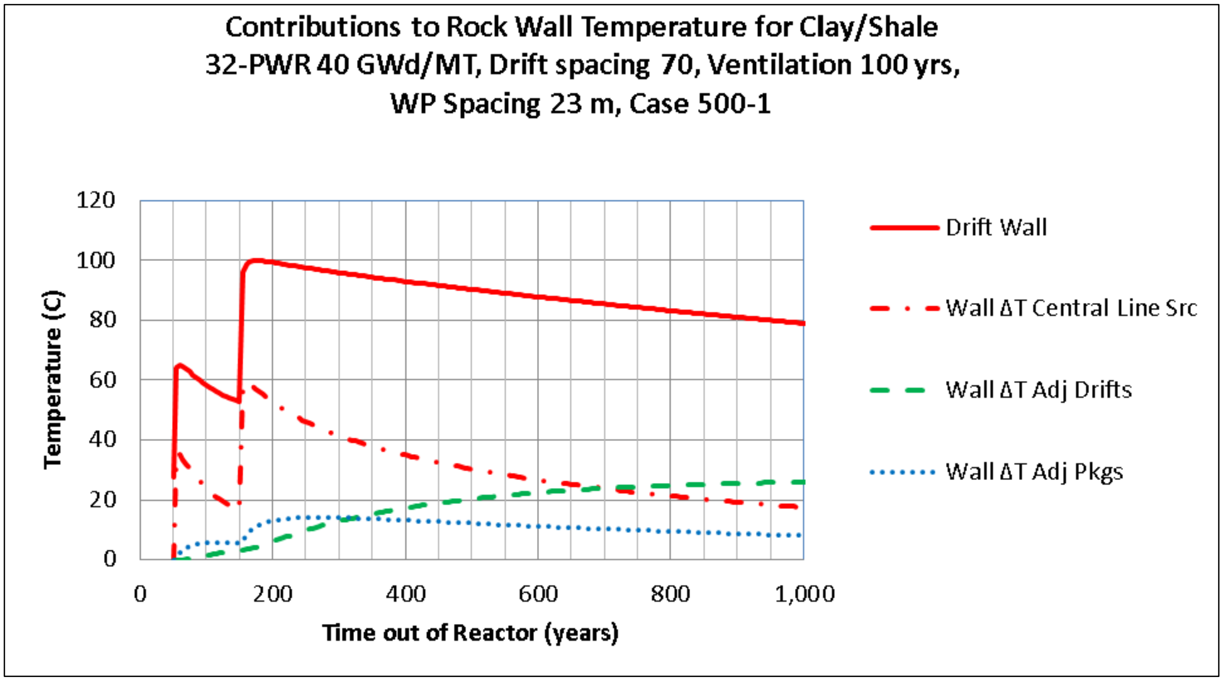


Figure 5-10. Contribution of heat sources to drift-wall temperature, for the sedimentary open concept, for 23-m package spacing (32-PWR size packages, 40 GW-d/MT burnup)

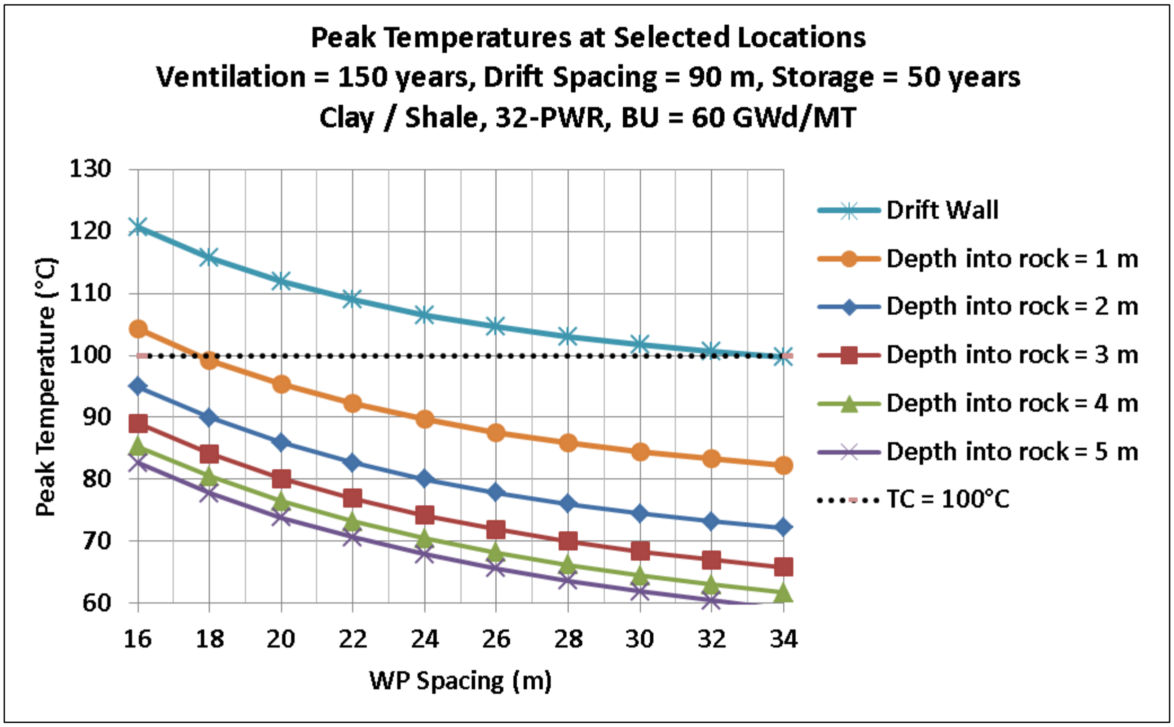


Figure 5-11. Peak temperature vs. package spacing for values of depth into the rock, for the sedimentary open concept, (32-PWR size packages, 60 GW-d/MT burnup, 90-m drift spacing)

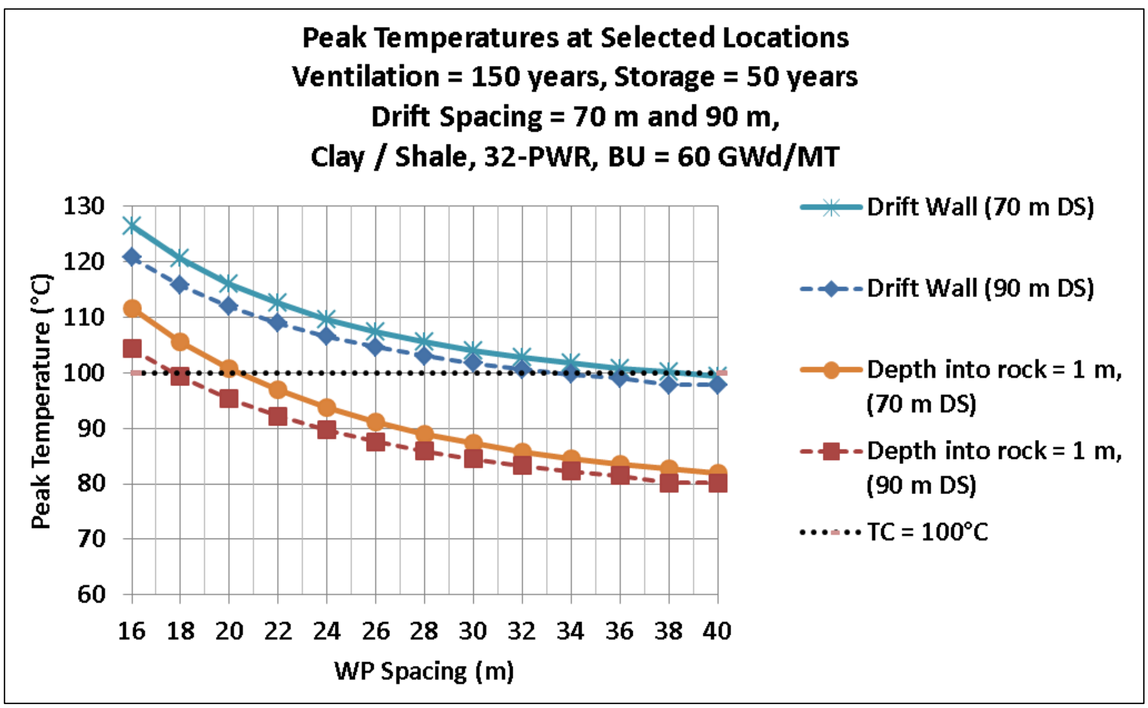


Figure 5-12. Peak temperature vs. package spacing for the drift wall and 1 m into the wall, for the sedimentary concept, comparing 70 and 90-m drift spacings (32-PWR size packages, 60 GW-d/MT burnup)

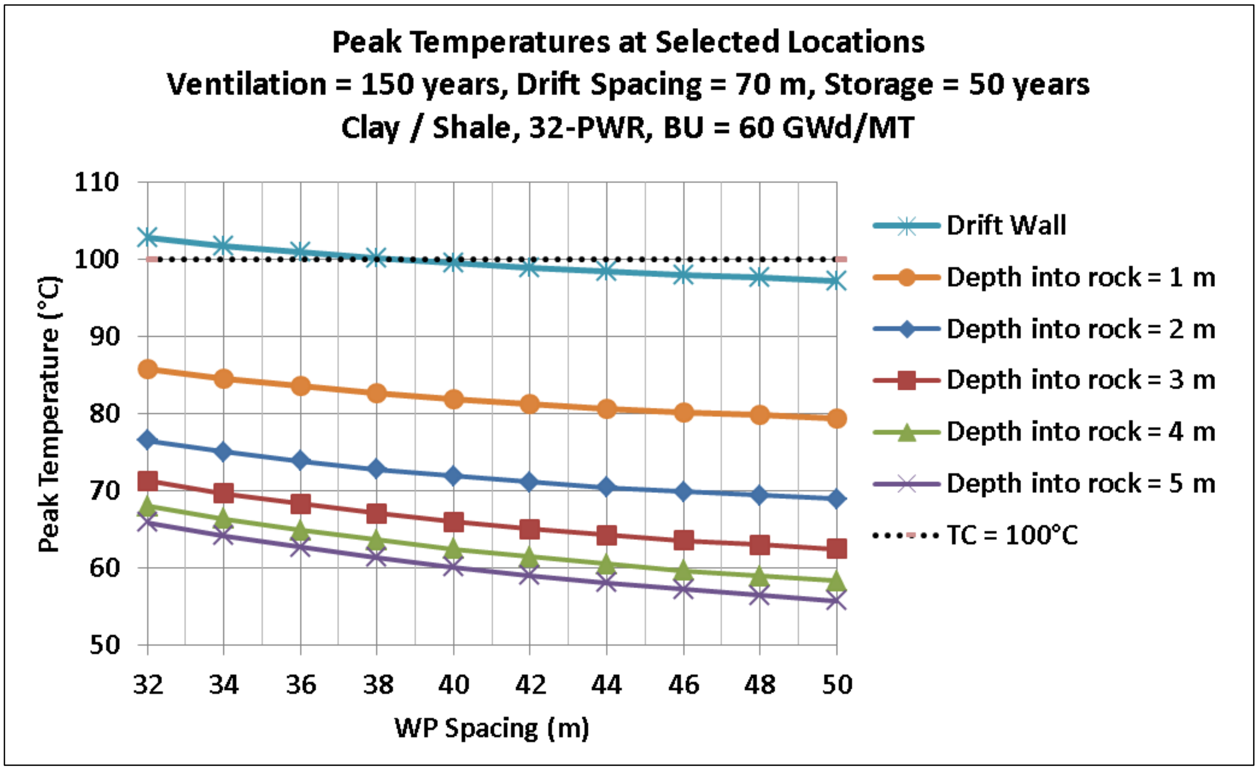


Figure 5-13. Peak temperature vs. package spacing for values of depth into the rock, for the sedimentary open concept (32-PWR size packages, 60 GW-d/MT burnup, 70-m drift spacing)

Table 5-2 presents the depth into the drift wall of the 100°C peak temperature isotherm, as a function of waste package spacing, for two cases of ventilation time and drift spacing. This is a measure of the extent of the host rock that would be affected by temperatures above the target. The table indicates that package spacing of 34 m (drift spacing of 90 m) or 38 m (drift spacing of 70 m) would ensure the peak drift wall temperature is at or below 100°C.

Table 5-2. Depth into the drift wall for which peak temperature equals the 100°C target value (60 GW-d/MT)

	Package Spacing (m) [B]						
	26	28	30	32	34	36	38
Ventilation Time (yr) and Drift Spacing (m) [A]	**Depth (into drift wall) to Meet T=100°C**						
150 yr / 90 m	0.27	0.18	0.10	0.04	0.00	0.00	0.00
150 yr / 70 m	0.46	0.34	0.24	0.17	0.10	0.05	0.01
[A] After 50 years decay storage. [B] SNF burnup 60 GW-d/MT.							

Figure 5-9 shows the temperature histories at the drift wall and at various depths into the host rock, for package spacing of 23 m (drift spacing of 70 m). Figure 5-10 shows that the relative contributions to the peak drift wall temperature from heating by adjacent waste packages in the same drift, and by adjacent drifts, are relatively small compared to the contribution of the central waste package. This indicates that increasing the spacings would have only a small effect on the peak drift wall temperature.

Summary – Table 5-3 summarizes the parametric spacing study results for sedimentary open concepts with a peak drift wall temperature of 100°C. These results show that to accommodate disposal of large DPCs in sedimentary media (e.g., clay/shale) both large drift spacings (between 70 m and 90 m) and large WP spacings (between 20 m and 40 m) would be needed to limit peak drift wall temperature to 100°C, for SNF burnup up to 60 GW-d/MT. The larger spacings would be for higher burnup SNF (e.g., greater than 40 GW-d/MT). If the peak drift wall temperature were to be extended to 120°C, then significantly smaller spacings could be possible. Calculated temperature profiles show that the region of the host rock where temperature is greater than 100°C is limited in extent (e.g., approximately 1 m into the drift wall).

Table 5-3. Summary of parametric spacing study for sedimentary open concepts with peak drift wall temperature of 100°C

Burnup (GW-d/MT)	**40**	**60**	**60**
WP spacing (m)	**23**	**38**	**34**
Drift spacing (m)	**70**	**70**	**90**
Repository total area (km^2)	14.4	23.4	26.9
Areal mass loading (m^2/MT)	102.8	166.9	192.5
Total emplacement drift length (all SNF emplaced with same spacings; km)	209	339	305
Decay heat per WP at emplacement (W)	10,163	15,824	15,824
Avg. areal total heat load (all SNF locally w/ same age, burnup; W/m^2)	6.6	6.3	5.5
SNF age out-of-reactor at closure (yr)	150	200	200
Decay heat per WP at closure (W)	3,882	4,173	4,173
Areal heat load at closure (all SNF locally w/ same age, burnup; W/m^2)	2.5	1.7	1.4
32-PWR size, DPC-based waste packages			

References for Section 5

BSC (Bechtel-SAIC Co.) 2008. *Postclosure Analysis of the Range of Design Thermal Loadings*. U.S. Department of Energy, Office of Civilian Radioactive Waste Management. ANL-NBS-HS-000057 REV 00. January, 2008.

Carter, J., A. Luptak, J. Gastelum, C. Stockman and A. Miller 2012. *Fuel Cycle Potential Waste Inventory for Disposition.* FCR&D-USED-2010-000031 Rev. 5. U.S. Department of Energy, Used Fuel Disposition R&D Campaign. July, 2012.

DOE (U.S. Department of Energy) 2008. *Yucca Mountain Repository License Application for Construction Authorization*. DOE/RW-0573. Washington, D.C.: U.S. Department of Energy.

Greenberg, H.R., J. Wen and T.A. Buscheck 2013. *Scoping Thermal Analysis of Alternative Dual-Purpose Canister Disposal Concepts*. LLNL-TR-639869. Lawrence Livermore National Laboratory. June, 2013.

Greenberg, H.R., M. Sharma, M. Sutton and A.V. Barnwell 2012. *Repository Near-Field Thermal Modeling Update Including Analysis of Open Mode Design Concepts*. LLNL-TR-572252. Lawrence Livermore National Laboratory. August, 2012.

Hardin, E., J. Blink, H. Greenberg, M. Sutton, M. Fratoni, J. Carter, M. Dupont and R. Howard 2011. *Generic Repository Design Concepts and Thermal Analysis (FY11).* FCRD-USED-2011-000143 Rev. 2. December, 2011. U.S. Department of Energy, Used Fuel Disposition R&D Campaign.

Hardin, E., T. Hadgu, D. Clayton, R. Howard, H. Greenberg, J. Blink, M. Sharma, M. Sutton, J. Carter, M. Dupont and P. Rodwell 2012. *Disposal Concepts/Thermal Load Management (FY11/12 Summary Report).* FCRD-USED-2012-000219, Rev.1. U.S. Department of Energy, Used Fuel Disposition R&D Campaign.

Hardin, E. 2013. *Temperature-Package Power Correlations for Open-Mode Geologic Disposal Concepts*. SAND2013-1425. Sandia National Laboratories. Albuquerque, NM. February, 2013.

Huff, K.D. and T.H. Bauer 2012. *Benchmarking a New Closed-Form Thermal Analysis Technique Against a Traditional Lumped Parameter, Finite-Difference Method*. FCRD-UFD-2012-000142. U.S. Department of Energy, Used Fuel Disposition R&D Campaign. July, 2012.

Jove-Colon, C. et al. 2012. *Evaluation of Generic EBS Design Concepts and Process Models: Implications to EBS Design Optimization.* FCRD-USED-2012-000140. U.S. Dept. of Energy, Office of Used Nuclear Fuel Disposition. June, 2012.

OCRWM (Office of Civilian Radioactive Waste Management). (1999). *License Application Design Selection Report*. B00000000-01717-4600-00123 REV 01 ICN 01. August, 1999. U.S. Department of Energy, Yucca Mountain Site Characterization Office, Las Vegas, Nevada.

Sutton, M., J.A. Blink, M. Fratoni, H.R. Greenberg and A.D. Ross 2011. *Investigations on Repository Near-Field Thermal Modeling.* LLNL-TR-491099 Rev. 1. Lawrence Livermore National Laboratory. December, 2011.

THIS PAGE INTENTIONALLY LEFT BLANK

6. Safety Strategies for Disposal Concepts

The purpose of this section is to describe the major elements of the postclosure safety case for each alternative geologic disposal concept for direct disposal of spent nuclear fuel (SNF) in dual-purpose canisters (DPCs). The roles of engineered barrier system (EBS) components, and the natural barrier system (NBS) in the proposed generic DPC direct disposal concepts (Section 4) are identified.

6.1 Safety Strategy Discussion

Safety strategies are a first step in the process of *performance allocation*, defined as an allocation of protection to a performance barrier (i.e., to engineered and natural barriers) that would be relied on to isolate waste from the environment. It allows for the relative importance of each component contained in a multiple barrier system for a repository to be identified and the uncertainties in the expected performance to be evaluated and tested. The performance allocation process is part of an iterative repository design process that allows for modifications to the disposal system design in order to increase regulatory confidence that the disposal system would meet the anticipated performance objectives. The performance allocation process developed in an earlier repository program (DOE 1988) consists of several steps:

- Develop a repository safety strategy
- Identify performance measures or performance objectives
- Identify information needs
- Develop a modeling and testing strategy to improve confidence and reduce uncertainty in the performance of key barriers

The safety strategy is the high-level approach adopted for achieving safe disposal (OECD 2004). An important aspect of achieving safe disposal is the concept of relying on multiple barriers. The use of multiple barriers offers defense-in-depth by providing a diverse set of features and processes that act collectively, and often independently, to minimize the likelihood of radionuclide migration. In this context, a barrier is defined as any material, structure, or feature that prevents or substantially delays the rate of movement of water or radionuclides from the repository to the accessible environment, or prevents the release or substantially reduces the release rate of radionuclides from the waste (10CFR60.2 and 10CFR63). The IAEA Safety Standards for Geological Disposal of Radioactive Waste (IAEA 2006) includes concepts of multiple safety functions:

"The natural and engineered barriers shall be selected and designed so as to ensure that post-closure safety is provided by means of multiple safety functions. That is, safety shall be provided by means of multiple barriers whose performance is achieved by diverse physical and chemical processes. The overall performance of the geological disposal system shall not be unduly dependent on a single barrier or function."

The following sections provide overview of the disposal concepts for consideration, identify multiple barriers, and discuss the barrier safety functions. The approach relies on *Generic Deep Geologic Disposal Safety Case* (Freeze et al. 2013) which provides information on system components and performance measures that broadly describe the safety of any repository concept in the types of geologic media being investigated by the Used Fuel Disposition R&D campaign.

At a high level, the geologic disposal facility consists of three components: 1) an engineered barrier system, 2) a natural barrier system, and 3) a biosphere (Freeze et al. 2013). The EBS consists of the source term and a significant portion of the near-field environment. The major components that encompass the EBS are the waste form (e.g., source term), waste package, buffer and/or backfill, and seals and/or liner. The natural barrier system consists of a portion of the near-field environment, specifically the disturbed rock zone, and the far field which consists of the host rock and the surrounding geologic units. The biosphere is where the potential receptor, typically defined by regulations resides. The biosphere encompasses the surface—which defines the receptor's surroundings—the receptor, the receptor's lifestyle, and the characteristics of the environment where the receptor resides.

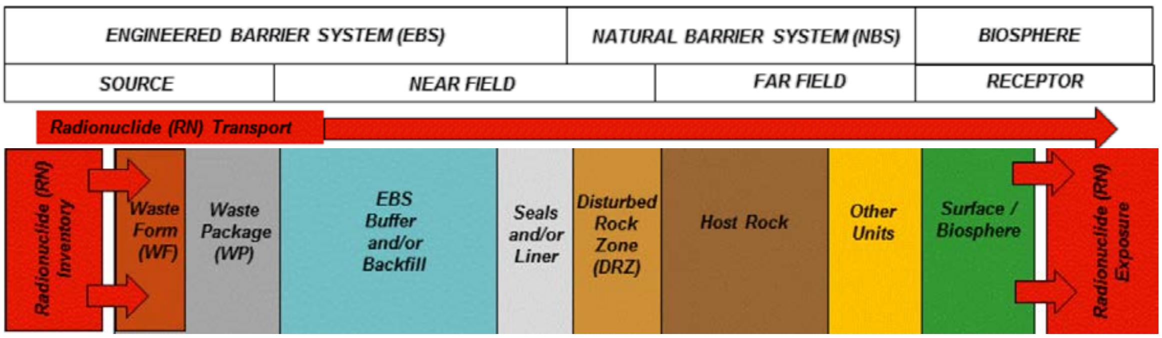


Figure 6-1.Components of a generic disposal system (from Freeze et al. 2013)

Depending on the disposal system, each barrier component (i.e., EBS, NBS, and biosphere) provides a specific safety function in the context of the multi-barrier system. Although the terminology used to describe these safety functions is different in different countries, these functions can be categorized into the following three general areas of stability/isolation, containment, and limited/delayed release (Bailey et al. 2011).

Stability/Isolation (e.g., geologic/hydrologic setting) – Isolating the waste from non-anthropogenic future events and climate changes, contributing to the stability of near-field conditions and to the longevity of the natural barriers. This forms a boundary condition that ensures that the other safety functions could fulfill their role, and reduces the probability for, and consequences from anthropogenic events such as future human actions that might result in inadvertent intrusions into the sealed repository.

Containment (e.g., waste package or seals design) – Preventing groundwater from coming into contact with the waste. In the case of disposal in crystalline rock or clay/shale formations this safety function would be provided by the EBS. In the case of disposal in salt formations much of the containment function may be provided by the NBS.

Limited and Delayed Releases (e.g., geomechanical and geochemical setting) – These safety functions begin to dominate once the containment functions deteriorate, for example, when waste packages would be breached as a result of corrosion. This is a major function of the NBS as well as components of the EBS, and provides for the long-term performance of geologic disposal.

In current U.S. Nuclear Regulatory Commission repository regulations the concept of "important to waste isolation" is used to define features that contribute to postclosure barrier safety functions. A barrier feature is important to waste isolation if it meets two conditions (SNL 2008): 1) the feature is associated with one or more processes or characteristics classified as important to barrier capability; and 2) the feature is a significant contributor to the barrier capability relative to the other features of the barrier. In addition, a feature is classified as important to waste isolation if it is one of the engineered features of the geologic repository whose function is to prevent or mitigate the consequences of potential disruptive events. This last classification criterion is particularly relevant to the DPC feasibility study with respect to issues related to criticality events. Criticality is considered a disruptive event.

In the discussions that follow, no credit is taken for cladding performance for the disposal concepts. While the thermal loading studies included a temperature constraint to protect the cladding, design, burnup, aging, and site specific handling issues suggest that, in the absence of detailed information on the condition of specific DPCs, it is appropriate to assume that the cladding has failed. This assumption results in a conservative performance assessment, and, should specific information eventually become available, additional barrier performance can be allocated.

The disposal concepts considered here are described in Section 4. Of the 13 concepts considered initially (Table 4-1) six were identified as general concepts for DPC direct disposal (Table 4-3). This report focuses on generic safety strategies for those six concepts, which include:

- Salt
- Hard rock, open (unbackfilled and backfilled)
- Sedimentary, open (unbackfilled and backfilled)
- Cavern-retrievable

Some specific details that distinguish each disposal concept are provided in Table 6-1.

The sections to follow are organized in the context of the three main components of the disposal system (e.g., EBS, NBS, and biosphere) along with the anticipated safety functions identified for specific subcomponents of the disposal system design to which performance has been allocated. A cross walk that relates the generic disposal system design performance allocation for direct disposal of SNF in DPCs is provided in each section, along with a brief overview of each disposal concept.

6.2 Barrier Reliance in Previous Safety Assessments

Previous safety assessments, or performance assessments, have indicated those repository components on which performance would be likely to depend for the different media examined. Performance assessments have been identified for salt, clay, and granite media.

Salt Safety Assessments - Germany has investigated the salt dome at Gorleben as a possible site for a geologic repository since 1977 (Weber et al. 2011). The safety concept for the Gorleben repository does not rely on the waste containers to act as an engineered barrier and there are no plans for additional measures to retain radionuclides in the event that brines reach the waste. Two engineered barriers—seals and backfill—are included in the safety concept. Hansen and Leigh (2011) proposed a framework for evaluating disposal of thermal waste in salt. The study discussed two general categories for possible scenario development: an isothermal “cool” repository and a thermally “hot” repository. Important conclusions from the study showed that radionuclides would not be expected to migrate from the disposal horizon based on thermal, hydrologic, and geochemical considerations, especially reducing conditions.

Table 6-1. Data for generic disposal concepts suitable for DPC direct disposal

Host Geologic Media/Concept	Salt	Hard-Rock Unsaturated Unbackfilled Open	Hard-Rock Backfilled Open	Sedimentary Unbackfilled Open	Sedimentary Backfilled Open	Cavern Retrievable
Depth	500 m	200 to 500 m	200 to 500 m	200 to 500 m	200 to 500 m	200 to 500 m
Hydrologic setting	Saturated	Unsaturated	Unsaturated or saturated	Nominally saturated	Nominally saturated	Unsaturated or saturated
Host Medium	Domal or bedded salt	Granite, tuff, or other competent rock type	Granite, tuff, or other competent rock type	Sedimentary rock (e.g., mudstone, claystone, shale)	Sedimentary rock (e.g., mudstone, claystone, shale)	Granite, tuff, or other competent rock type
Ground Support	Rockbolts	Rockbolts; shotcrete as needed	Rockbolts; shotcrete as needed	Shotcrete and steel supports, or pre-cast concrete, or steel liner	Shotcrete and steel supports; or pre-cast concrete; or steel liner	Rockbolts; shotcrete as needed
Seals and Plugs	Shaft & tunnel plugs and seals	Shaft & ramp plugs and seals	Shaft & ramp plugs and seals	Emplacement drift plugs; shaft & ramp plugs and seals	Shaft & ramp plugs and seals	Shaft & ramp plugs and seals
Emplacement Mode	Horizontal, alcove or in-drift	Horizontal, in-drift	Horizontal, in-drift	Horizontal in-drift	Horizontal in-drift	Vertical, in-drift (or vertical or horiz. borehole)
WP Target Capacity	Up to 32-PWR (or larger)	Up to 32-PWR (or larger)	Up to 32-PWR (or larger)	Up to 32-PWR (or larger)	Up to 32-PWR (or larger)	Up to 32-PWR (or larger)
Package Size	≤ 2 m D x 5 m L	≤ 2 m D x 5 m L	≤ 2 m D x 5 m L	≤ 2 m D x 5 m L	≤ 2 m D x 5 m L	Casks approx. 3.5 m D x 6.5 m H with lid
Drift Diameter	5 m (nominal; alcoves)	5.5 m (drifts)	4.5 m (drifts)	4.5 m (drifts)	4.5 m (drifts)	Approx. 8 m high x 6 m wide
Area (m^2/MTHM)	~60	<100	$\geq$100	~100	$\geq$100	$\geq$50
Overpack	Steel	Corrosion resistant	Corrosion resistant	Steel	Steel	Storage cask/vault
Backfill and/or Buffer	Crushed salt	None	Low permeability backfill in all drifts	None	Low permeability backfill in all drifts	Low permeability backfill/buffer in all drifts
Additional EBS Components		Option for engineered water diversion barriers		Low-permeability backfill in non-emplacement openings		

THIS PAGE INTENTIONALLY LEFT BLANK

Intact salt is impermeable and the fractures would be self-healing. The most recent Waste Isolation Pilot Plant safety assessment (DOE 2009) showed that no potentially disruptive natural events or processes would be sufficiently likely to occur to warrant inclusion in the undisturbed or the disturbed performance scenarios. The undisturbed performance scenario is used for compliance calculation. The assessment includes a human intrusion scenario because it cannot be assumed that institutional controls prevent its occurrence. Over time salt creep would tend to heal fractures and reduce the damaged zone permeability. The scenario analyses in the 2009 safety assessment conservatively assumed that the damaged zone does not heal and that pathways created for fluid flow between the repository and overlying and underlying beds continue to exist indefinitely. Accordingly, brine in the repository would be expected under most conditions. For the Waste Isolation Pilot Plant performance assessment, under the disturbed conditions, the expected performance of the repository would be initially the same as for the undisturbed conditions.

Clay Safety Assessments - A scenario of long-term, diffusive transport through the host clay upward from the emplacement boreholes (Hansen et al. 2010) was considered in the generic safety assessment, using a one-dimensional advective-dispersive model. The evaluation showed that geologic barrier played a significant role in delaying or halting radionuclide transport; radionuclides that do enter the clay host rock travel very slowly by diffusion because of the extremely low permeability. Moreover, chemically-reducing conditions would limit radionuclide solubility, and sorption onto clays would also retard transport. Under the conditions modeled, a clay repository could achieve total containment, with no releases to the environment in undisturbed scenarios. Nagra (2002) presented its safety case for the disposal of spent fuel, high-level radioactive waste, and various types of intermediate-level radioactive waste in a Swiss clay formation. The canisters could provide absolute containment for an extended time period because they would be mechanically strong and corrosion resistant in the expected environment. Flow would be dominated by diffusion rather than advection providing a strong barrier to radionuclide transport, and the chemical environment favors the immobilization and retardation of radionuclides and the long-term stability of the engineered barriers. In the French safety case for a repository in clay, waste packages would be expected to maintain their integrity for several thousand years; subsequently, water contacting the glass and spent fuel assemblies causes them to dissolve over a period of several hundreds of thousands of years. Travel by diffusion through the engineered barrier system would take at least 100,000 years.

In the French repository program the emphasis has been on clay, taking reversibility into consideration (Andra 2005). In the Safety Case, three barriers act to prevent or delay the release and transport of radionuclides. Waste packages are expected to maintain their integrity for several thousand years; subsequently, water contacting the glass and spent fuel assemblies causes them to dissolve for several hundreds of thousands of years. Travel by diffusion through the engineered barrier system will take at least 100,000 years, during which time most of the radionuclides will disappear through decay. Once in the geosphere, most of the radionuclides will become trapped by one of two mechanisms. First, large amounts of smectite will tend to immobilize dissolved species. Second, the chemistry of the interstitial water causes some radionuclides to precipitate. The radionuclides that do not become trapped will migrate very slowly by diffusion.

The Belgian repository program has been focused on examining a clay formation (ONDRAF/NIRAS 2001) in which the packages and overpacks would last for up to several tens of thousands of years. A clay-based backfill material would be expected to maintain its integrity for several thousands of years, in large part because the design avoids exposure to excessive temperature increases. It is the clay formation itself that plays the dominant role in terms of preventing or delaying radionuclide transport, occurring primarily through the slow process of molecular diffusion. Sorption by the clay minerals or by organic materials in the clay would halt the transport of many radionuclides.

Granite Safety Assessments - The Swedish repository program is examining a granite repository using the KBS-3 method (SKB 2011), which specifies that the spent fuel be contained inside copper canisters with a cast iron insert; the canisters would be surrounded by bentonite clay. The waste package is the principal barrier to radionuclide release in the safety assessment. A safety assessment accompanies the license application in Finland (Posiva 2010) for the construction of a disposal facility at a site with metamorphic (mostly gneiss) and igneous (mostly granite) host rock. Groundwater flow occurs primarily through fractures and fracture zones; the more transmissive fractures tend to be located in local zones of abundant fracturing. The key barriers that prevent or delay radionuclide transport and release include the copper-iron canister, which after being breached could still provide some level of barrier performance for a period of time, a bentonite buffer, with low transport of radionuclides, backfill of the tunnels, and host rock properties including low groundwater flow rates and slow transport of radionuclides through the host rock.

The Canadian program has conducted multiple safety assessments for the behavior of a hypothetical repository located in the Canadian Shield granitic rock. For the Third Case safety assessment, the hypothetical repository was located in a fractured, granitic rock of intermediate permeability (Gierszewski et al. 2004). While the containers would be the primary barrier to radionuclide release in this study, the other engineered and natural barriers would be effective in preventing or delaying the release of most radionuclides. The Fourth Case safety assessment (Kremer et al. 2011) considers a different hypothetical site with a shallower repository and revised designs for the underground facilities, floor emplacement, and used fuel containers. The copper and steel containers were designed to withstand mechanical stresses while providing a corrosion-resistant barrier. The buffer material surrounding the containers consists of compacted bentonite clay. In both case studies, robust containers would be important barriers to isolating waste effectively in the repository and preventing or significantly delaying radionuclide transport to the biosphere. In sensitivity cases in which all the containers were assumed to have failed, the multiple barrier system would be capable of retaining radionuclides or delaying their transport.

Generic U.S. studies of hypothetical granite repositories have place reliance on slow advective flow (Vaughn et al. 2011).

Summary – The information about barrier reliance in published safety assessments is compiled, summarized, and generalized in Table 6-2. It illustrates that generally, salt repositories place little reliance on engineered barriers or natural barriers, and that releases tend to be associated with human intrusion. Clay repositories, on the other hand, rely principally on diffusive transport to ensure long-term safety, although some programs do take credit for engineered barrier components. The international granite program safety assessments, which represent sites being licensed, rely on very long life waste packages for long-term safety, with lesser reliance on other engineered and natural components. Generic U.S. studies of hypothetical granite repositories place reliance on slow advective flow. While most of the international program safety assessments assume that because of repository depth or the geologic setting, human intrusion would not be an issue, that is not likely to be the case for a repository in the U.S. based on regulatory precedent. While human intrusion is not a barrier, it is included in the table because it is an anthropogenic event that acts to prevent a barrier from limiting or delaying release. This summary supports assignments of barrier reliance shown in Tables 6-3 through 6-8 below.

Table 6-2. Summary of barrier reliance from previous media-specific assessments

	Stability/ Isolation	Containment		Limited or Delayed Release				
	Geology/ Hydrologic Setting	Long-Life Waste Package (corrosion resistant)	Waste Package (corrosion allowance)	Buffer/ Backfill	Sorption	Diffusion Dominated Transport	Advective or Fracture Flow	Human Intrusion
Salt	√√		√	√				√√
Clay/Shale	√		√	√	√	√√		√
Crystalline	√	√√		√	√		√	
			Minor reliance	√	Major reliance	√√		

6.3 Design Concepts for Direct Disposal of DPCs

Of the 13 design concepts considered by Hardin and Voegele (2013) six were deemed potentially viable for direct disposal of SNF in DPCs (Section 6.1). Performance allocation strategies for those six concepts are discussed below.

6.3.1 Salt Concept

Performance would be allocated to the waste package and far-field seals/plugs to prevent water intrusion during the period of time needed for the disturbed rock zone (DRZ) to heal. This performance could be sought through the use of a corrosion resistant waste package or seals/plugs or a combination.

A robust corrosion allowance waste package would be used to protect the DPC during handling and transport, maintain containment during repository operations, support compliance with regulatory retrievability requirements, and isolate waste during the thermal period (a few hundred to thousands of years) during which there may be fracture flow paths in the near-field host rock. Note that the impact of temperature on the DRZ in salt depends on many factors such as thermal loading, excavation design, overburden load, salt rheological properties, etc.

The DRZ and backfill would reconsolidate over hundreds to thousands of years after closure, which would be important to performance. Corrosion of waste package materials such as carbon steel and stainless steel may stop or slow when moisture becomes sufficiently scarce due to consumption by corrosion reactions, and limited moisture mobility after cooling and reconsolidation. Moisture is already scarce in domal salt, and in bedded salt it would become depleted in the near-field environment due to brine migration that could occur in the first few hundred to thousands of years, but then stops. The performance allocation for DPC direct disposal in salt is summarized in Table 6-3.

Table 6-3. Salt barrier safety functions

Region	Feature	Baseline Scenario Representation	Safety Function	Reliance on Barrier
Source	Inventory	Up to 32-PWR (or larger)	–	–
	Waste Form	2×10^{-5}/yr fractional degradation rate, no cladding credit	–	–
	Waste Package	Corrosion allowance (e.g., steel)	Containment	Low to Medium
	Waste Package Buffer	–	–	–
Near-Field	Disturbed Rock Zone	Reconsolidated	Containment or Limited/Delayed Release	High
	Near-Field Host Rock	Diffusive transport, no sorption	Containment or Limited/Delayed Release	High
Far-Field	Backfill	Reconsolidated	Containment or Limited/Delayed Release	High
	Seals / Liner	Shaft & tunnel plugs/seals	Containment or Limited/Delayed Release	Medium
	Far-Field Host Rock	Diffusive transport, no sorption	Containment or Limited/Delayed Release	High
	Aquifer	Dilution volume	–	–

6.3.2 Hard-rock Open Emplacement

The hard-rock emplacement mode concept consists of rock types that have relatively high strength (e.g., granites, basalt, and crystalline basement), corrosion-resistant overpacks, and placing the package on pedestals in open, ventilated drifts. The open emplacement mode allows extended ventilation of the repository for heat removal after waste emplacement, prior to permanent closure. In a previous comprehensive design study (OCRWM 1999), ventilation of the repository was performed with suction fans located at the ground surface and were assisted by the density difference between a hot and cold air column that creates a natural flow of warm air rising in the exhaust shaft (i.e., chimney or shaft effect). This concept combines the functions of surface decay storage (i.e., in fuel pools or dry cask storage) with geologic disposal in the same facility (Hardin et al. 2012).

Hard-rock unbackfilled, unsaturated, open concept – For this concept the safety strategy places high reliance on the corrosion resistant package (e.g., Hastelloy or titanium) to provide containment (Table 6-4). The environment would likely be oxidizing, at least in the unsaturated zone, so passive materials (protected by oxide layers) would be good candidates for the waste package outer layer. Other options include the use of amorphous metal and thermal-spray ceramic coatings that have excellent corrosion resistance (Blink et al. 2009). The need for a corrosion resistant waste package arises because of the oxidizing, CO_2 rich environment in the unsaturated zone (which accelerates SNF degradation) and the free-draining host rock. Another way to ensure reliable containment would be to add engineered water diversion features to the EBS (e.g., drip shields protected by a resistant coating) or multiple corrosion-resistant layers to the waste package. The primary function of these barriers would be to prevent or limit advective flow contacting the waste package or waste form, particularly during the thermal period. They could also protect the waste package from mechanical damage due to seismic ground motion and/or drift collapse.

Performance would be allocated to the SNF waste form, which would be expected to limit or delay the contaminant release. One option for increasing the reliance on this barrier would be to reduce the conservatism in waste form performance models by strengthening the scientific basis. Another option would be to better represent the partly degraded performance of the waste package, which if subjected to cracking or pinhole openings, could maintain chemically reducing conditions at the waste form (as demonstrated by Ferriss et al. 2009).

The far-field host rock (unsaturated zone and underlying aquifers) would also limit transport of radionuclides by sorbing contaminants, possibly combined with matrix diffusion. The medium reliance suggests that far-field transport pathways could be through aquifers that would be chemically reducing, with small specific discharge, and high specific surface area. Depending on site specific conditions, some radionuclides such as ^{129}I could be highly mobile in advective transport conditions, so that waste isolation performance would depend more on engineered barriers.

Table 6-4. Hard rock unbackfilled, unsaturated open concept barrier safety functions

Region	Feature	Baseline Scenario Representation	Safety Function	Reliance on Barrier
Source	Inventory	Up to 32-PWR	–	–
	Waste Form	2×10^{-5}/yr fractional degradation rate, no cladding credit	Limited/Delayed Release	Medium
	Waste Package	Corrosion resistant	Containment	High
	Waste Package Buffer	Not applicable	–	–
Near-Field	Engineered Water Diversion Barrier *	Long Lifetime (e.g., corrosion resistant)	Containment	Medium
	Disturbed Rock Zone	Free drainage	–	–
	Near-Field Host Rock	Free drainage	–	–
Far-Field	Backfill	Not applicable	–	–
	Seals / Liner	Not applicable	–	–
	Far-Field Host Rock	Advective transport with sorption	Limited/Delayed Release	Medium
	Aquifer	Dilution volume	–	–

* Unique to unbackfilled, unsaturated concepts.

Hard-rock backfilled, open concept – This safety strategy is similar to the hard-rock unbackfilled, unsaturated concept above, with the use of backfill instead of engineered water diversion barriers (Table 6-5). With backfill, the concept is intended to be suitable for saturated as well as unsaturated settings, and for host rock with limited permeability (and possibly chemically reducing conditions) as well as free drainage. Backfill would be important for saturated conditions, or when the unsaturated host rock is not free draining, to prevent preferential water movement throughout the network of repository openings (Hardin et al. 2012). Far-field plugs and seals would be needed to limit inflow.

The waste package outer barrier would be corrosion resistant, but additional materials would be available if the disposal environment is chemically reducing. An example would be the copper clad waste package proposed for the Swedish disposal concept (SKB 2011). The corrosion resistant materials mentioned previously (Hastelloy, titanium, amorphous metal and ceramic coatings) could also be used, for example, to accommodate uncertainty as to the future disposal environment.

The host rock may be reducing, and also the EBS if the backfill has low permeability and contains organic matter and other reducing chemical species. The SNF waste form degrades slowly under such reducing conditions.

Table 6-5. Hard-rock backfilled open concept barrier safety functions

Region	Feature	Baseline Scenario Representation	Safety Function	Reliance on Barrier
Source	Inventory	Up to 32-PWR	–	–
	Waste Form	2×10^{-5}/yr fractional degradation rate, no cladding credit	Limited/Delayed Release	Medium
	Waste Package	Corrosion resistant	Containment	High
	Waste Package Buffer	Not applicable	–	–
Near-Field	Backfill	Low permeability, diffusion dominated	Containment	Medium
	Disturbed Rock Zone	Potential advective transport pathway	–	–
	Near-Field Host Rock	Potential advective transport pathway	–	–
Far-Field	Backfill	Low permeability	–	–
	Seals / Liner	Barriers to advective flow, located away from thermal effects	Containment or Limited/Delayed Release	Medium
	Far-Field Host Rock	Advective transport in with sorption	Limited/Delayed Release	Medium
	Aquifer	Dilution volume	–	–

6.3.3 Sedimentary Rock Open Emplacement

The sedimentary open concept includes shale, claystone, mudstone and argillite (Hansen et al. 2010). This category of argillaceous sediments, referred to as clay/shale throughout this report (Sections 4 through 7), has been considered as host rock for geologic disposal throughout the world. Some of the reasons are because these are common rock types, they often have low bulk permeability and high sorption capacity, and they exhibit fracture healing behavior.

Currently, three countries are considering clay/shale media for the disposal of SNF and high-level waste. These countries are Switzerland (claystone at the Mont Terri site), France (argillite at the Tournemire site), and Belgium (plastic Boom clay at the Mol site). Assessments for these programs have shown that advective transport under reasonable hydraulic gradients would be insignificant over regulatory timeframes. Chemically-reducing conditions limit radionuclide speciation and solubility, and thus mobility. Sorption of many radionuclides onto clays with high specific surface area would retard transport for most radionuclides.

Sedimentary unbackfilled, open concepts – The sedimentary unbackfilled, open disposal concepts (low- and high-temperature variations) are described in Section 4.4. It would use in-drift disposal, and remain open and ventilated until the SNF age is 150 years out-of-reactor. For the low-temperature option, ventilation and decay of heat-generating nuclides, combined with drift and package spacings (Sections 4 and 5) could limit the peak host rock temperature to 100°C or less. For higher burnup SNF or to accelerate repository closure, the high-temperature option would heat the host rock near the emplacement drifts (and only at waste package locations) to temperatures greater than 100°C.

At repository closure the emplacement drifts would be compartmented using plugs and seals (but not backfill) to isolate segments containing a few waste packages (e.g., 10 packages per segment). The point is to avoid completely backfilling on the order of 200 km of emplacement drifts (for 10,000 DPC-based packages), which would have to be performed remotely at elevated temperature, in a radiation environment. Instead the drifts would be closed in segments, preventing preferential groundwater flow between segments.

High reliance for the safety case would be placed on the far-field host rock, which would limit/delay transport by providing diffusion dominated transport, radionuclide sorption, and reducing geochemical conditions (Table 6-6). Given the performance attributes of the far-field host rock, a corrosion allowance waste package might suffice. However, a major function of the corrosion-resistant package would be to limit consequences from inadvertent human intrusion by drilling. Two aspects of human intrusion would be beneficially affected: 1) package structural integrity for 10,000 years or longer helps ensure that a future driller would recognize an obstruction and desist (see 40CFR191, Appendix C); and 2) in the event that drilling introduces large amounts of fluid, package integrity prevents the mobilization of soluble radionuclides from one or more waste packages (e.g., all those within a sealed segment). Human intrusion could be the most important scenario leading to radionuclide releases to the environment in sedimentary geologic settings (Hardin et al. 2013).

In addition to the waste package and far-field host rock, performance would be allocated to the SNF waste form, and to the plugs and seals that isolate the emplacement segments (Table 6-6).

Table 6-6. Sedimentary unbackfilled, open concept barrier safety functions

Region	Feature	Baseline Scenario Representation	Safety Function	Reliance on Barrier
Source	Inventory	Up to 32-PWR	–	–
	Waste Form	2×10^{-5}/yr fractional degradation rate, no cladding credit	Limited/Delayed Release	Medium
	Waste Package	Corrosion resistant	Containment	Medium
	Waste Package Buffer	Not applicable	–	–
Near-Field	Backfill	Not applicable	–	–
	Disturbed Rock Zone	Potential advective transport pathway	–	–
	Near-Field Host Rock	Potential advective transport pathway	–	–
Far-Field	Backfill	Low permeability	–	–
	Seals / Liner	Compartment the emplacement drifts; located away from thermal effects	Containment or Limited/Delayed Release	Medium
	Far-Field Host Rock	Advective transport with sorption	Limited/Delayed Release	High
	Aquifer	Dilution volume	–	–

Sedimentary backfilled, open concept – The sedimentary backfilled, open concept is similar to the unbackfilled concept but adds backfill as a barrier in the near field (Table 6-7). All drifts would be backfilled prior to repository closure, and parts of the repository (i.e., panels) could be backfilled at any point during repository operations. The corrosion resistant waste package would be designed to maintain containment throughout the regulatory period including the thermal period, and the time required for the DRZ to stabilize and reconsolidate. Use of a corrosion resistant overpack with low-permeability backfill could help to ensure that fluid circulated by an intersecting borehole in the human intrusion scenario would impact the SNF contents of only one package.

The backfill would serve as a barrier to groundwater flow, and would retard some released radionuclides. It would also stabilize the host rock, limiting or preventing additional rock damage associated with eventual collapse of the openings. A range of potential backfill materials exists (Hardin and Voegele 2013, Appendix B) but swelling clay-based materials would be preferred for remote installation in this concept because: 1) they would be chemically compatible with clay/shale host rock; 2) they fill voids; and 3) the swelling pressure would stabilize the host rock preventing collapse. Backfill with higher temperature tolerance (150°C or higher) would be needed in order to close the repository at SNF age of 150 years out-of-reactor (Hardin et al. 2012).

Table 6-7. Sedimentary backfilled, open concept barrier safety functions

Region	Feature	Baseline Scenario Representation	Safety Function	Reliance on Barrier
Source	Inventory	Up to 32-PWR	–	–
	Waste Form	2×10^{-5}/yr fractional degradation rate, no cladding credit	Limited/Delayed Release	Medium
	Waste Package	Corrosion resistant	Containment	Medium
	Waste Package Buffer	Not applicable	–	–
Near-Field	Backfill	Low permeability	Containment or Limited/Delayed Release	Medium
	Disturbed Rock Zone	Potential advective transport pathway; not explicitly modeled	–	–
	Near-Field Host Rock	Potential advective transport pathway	–	–
Far-Field	Backfill	Low permeability	–	–
	Seals / Liner	Low permeability	–	–
	Far-Field Host Rock	Advective transport with sorption	Limited/Delayed Release	High
	Aquifer	Dilution volume	–	–

6.3.4 Cavern-Retrievable Storage and Disposal Concept

Development of this concept initially focused on implementation in saturated settings, however EPRI (2010) suggested that concept could be readily adapted to unsaturated hard rock settings, for either vertical or horizontal emplacement configuration. Low reliance on the waste package reflects the case where DPC surface-storage casks, which are not designed for disposal, would be emplaced directly underground, and the cask cools internally by convection. High reliance would be placed on the backfill, which would have sufficient thickness and properties to limit radionuclide release to the host rock, while stabilizing the opening to collapse. The storage casks would be shielded, allowing worker access to install a tight buffer of compacted dehydrated swelling clay, or a similar material. The far-field transport pathway would be relied upon as the natural barrier.

Table 6-8. Cavern-retrievable storage-disposal concept barrier safety functions

Region	Feature	Baseline Scenario Representation	Safety Function	Reliance on Barrier
Source	Inventory	Up to 32-PWR	–	–
	Waste Form	2×10^{-5}/yr fractional degradation rate, no cladding credit	Limited/Delayed Release	Medium
	Waste Package	Corrosion allowance	Containment	Low
	Waste Package Buffer	Not applicable	–	–
Near-Field	Backfill	Low permeability, swelling	Containment or Limited/Delayed Release	High
	Disturbed Rock Zone	Potential advective transport pathway	–	–
	Near-Field Host Rock	Potential advective transport pathway	–	–
Far-Field	Backfill	Low permeability	–	–
	Seals / Liner	Low permeability	–	–
	Far-Field Host Rock	Advective transport with sorption	Limited/Delayed Release	Medium
	Aquifer	Dilution volume	–	–

6.3.5 Summary of Performance Allocation

A summary of reliance on specific barrier components for six general DPC direct disposal concepts is presented in Table 6-9. All the concepts rely on multiple natural and engineered barriers. A corrosion allowance or corrosion resistant waste package is specified for each. Corrosion allowance packages would not be assigned high reliance because of their limited containment lifetime. Under appropriate environmental conditions, the waste packages could provide containment for an extended time period because they would be mechanically strong and corrosion resistant. Most of the concepts assign some reliance on waste form degradation rates and radionuclide solubility (except salt, for which undisturbed radionuclide releases would be typically zero). Where backfill is used, it would be relied on to fill a barrier safety function. The use of corrosion resistant waste packages "trades" with the backfill/buffer component (i.e., corrosion allowance packages would be more often used with backfill). Radionuclide attenuation in advective conditions would be important where advective transport dominates. In the case of salt, diffusion-dominated transport has simplified the safety strategy with respect to engineered components. Finally, while human intrusion is not a barrier, it is included in the table because it is an anthropogenic event that acts to prevent a barrier from limiting or delaying release.

As noted in Section 2, consistent with current U.S. regulations, inadvertent human intrusion would not be expected to be included in the probabilistic dose calculations. Rather, it would be assessed separately. The more current of the regulations considers a stylized intrusion scenario; there is a likelihood that this approach would continue to be used in new, generally applicable regulations (McCartin 2012). A meaningful basis for evaluating the importance of human intrusion for performance allocation strategies for the disposal of DPCs can be developed by combining attributes of the current U.S. repository regulations. It is assumed that a human intrusion event would be specified such that a single well bypasses a portion of the natural barrier system vertically above or below the repository, but the remainder of the natural barrier in the horizontal direction to accessible environment is retained (consistent with 40CFR197). The approach would also retain the aspects of the current generally applicable rule that specifies the number of penetrations to be assumed; one for hard rock sites and more than one for sedimentary sites (consistent with 40CFR191).

For the salt concept, a robust corrosion allowance waste package would be used to isolate the waste during the thermal period during which there could be fracture flow paths in the near-field host rock; these fractures would be expected to close due to salt creep. For the sedimentary unbackfilled, open concept a corrosion resistant waste package could help mitigate the consequences of a human intrusion event, because: 1) long-term package structural integrity would help to ensure that a future driller recognizes an obstruction; and 2) in the event that a human intrusion event introduces large amounts of fluid, integrity of all the packages in a segment would limit the mobilization of radionuclides. Human intrusion could be the most important scenario leading to radionuclide releases to the environment in sedimentary geologic settings (Hardin et al. 2013). For the sedimentary backfilled, open concept, the corrosion allowance waste package would be designed to maintain containment during repository operations, the thermal period, and the time required for the DRZ to stabilize and reconsolidate. In the event of human intrusion, the backfill would be relied upon to limit contact of mobilized fluid with other packages. This discussion indicates that for the salt and sedimentary concepts, human intrusion could be a dominant scenario for radionuclide release to the environment.

Table 6-9. Comparison of barrier safety functions

	EBS Components				NBS Components			
	Corrosion Resistant, Long-Life Package	Corrosion Allowance Package	Waste Form Degradation & Solubility	Buffer/Backfill	Sorption	Diffusion-Dominated Transport	Advective and/or Fracture Flow	Human Intrusion
	DPC Direct Disposal Performance Allocation							
Salt		√				√√		√√
Hard Rock Unbackfilled, Open	√√		√		√		√	
Hard Rock Backfilled, Open	√√		√	√	√		√	
Sedimentary Unbackfilled, Open	√√		√		√	√√		√√
Sedimentary Backfilled, Open		√	√	√	√	√√		√√
Cavern-Retrievable		√	√	√√	√		√	
	Major Reliance (High) √√				Minor Reliance (Medium) √			

References for Section 6

Andra (National Radioactive Waste Management Agency). 2005. *Dossier 2005, Andra Research on the Geological Disposal of High-level Long-lived Radioactive Waste: Results and Perspectives*. Paris, France: Andra. (www.andra.fr)

Bailey, L., D. Becker, T. Beuth, M. Capouet, J.L. Cormenzana, M. Cunado, D.A. Galson, L. Griffault, J. Marivoet and C. Serres 2011. *PAMINA (Performance Assessment Methodologies in Application to Guide the Development of the Safety Case): European Handbook of the State-of-the-Art of Safety Assessments of Geological Repositories*—Part 1. Deliverable (D-N°: 1.1.4). European Commission.

Blink, J., J. Farmer, J. Choi, and C. Saw 2009. "Applications in the nuclear industry for thermal spray amorphous metal and ceramic coatings." *Metallurgical and Materials Transactions A*. June, 2009. 40A, pp. 1344-1354.

DOE (U.S. Department of Energy) 1988. *Site Characterization Plan, Yucca Mountain Site, Nevada Research and Development Area, Nevada*. Nevada Operations Office/Yucca Mountain Project Office, Nevada, DOE/RW-0199. 9 vols.

DOE (U.S. Department of Energy) 2009. “Appendix SCR-2009: Feature, Event, and Process Screening for PA.” *Title 40 CFR Part 191 Subparts B and C Compliance Recertification Application for the Waste Isolation Pilot Plant*. DOE-WIPP 09-3432. Carlsbad, NM: Carlsbad Field Office. i–x, pp. 1–222.

EPRI (Electric Power Research Institute) 2010. *EPRI Review ,of Geologic Disposal for Used Fuel and High Level Radioactive Waste: Volume III – Review of National Repository Programs.* Final Report, December, 2010. Intera, Inc. for the Electric Power Research Institute. #1021614.

Ferriss, E.D.A., K.B. Helean, C.R. Bryan, P.B. Brady and R.C. Ewing 2009. “UO_2 corrosion in an iron waste package.” *Journal of Nuclear Materials*. V. 284, pp. 130-139.

Freeze, G., M. Voegele, P. Vaughn, J. Prouty, W.M. Nutt, E. Hardin, and D. Sevougian 2013. *Generic Deep Geologic Disposal Safety Case*. FCRD-UFD-2013-000146, Rev. 1. U.S. Department of Energy, Used Fuel Disposition R&D Campaign.

Gierszewski, P., J. Avis, N. Calder, A. D'Andrea, F. Garisto, C. Kitson, T. Melnyk, K. Wei and L. Wojciechowski 2004. *Third Case Study - Postclosure Safety Assessment*. Report No: 06819-REP-01200-10109-R00. Toronto, Ontario: Ontario Power Generation.

Hansen, F.D., E.L. Hardin, R.P. Rechard, G.A. Freeze, D.C. Sassani, P.V. Brady, C.M. Stone, M.J. Martinez, J.F. Holland, T. Dewers, K.N. Gaither, S.R. Sobolik and R.T. Cygan 2010. *Shale Disposal of U.S. High-Level Radioactive Waste*. SAND2010-2843. Albuquerque, NM: Sandia National Laboratories.

Hansen, F.D. and C.D. Leigh 2011. *Salt Disposal of Heat-Generating Nuclear Waste.* SAND2011-0161. Albuquerque, NM: Sandia National Laboratories.

Hardin, E., T. Hadgu, D. Clayton, R. Howard, H. Greenberg, J. Blink, M. Sharma, M. Sutton, J. Carter, M. Dupont and P. Rodwell 2012. *Disposal Concepts/Thermal Load Management (FY11/12 Summary Report)*. FCRD-USED-2012-000219, Rev.1. U.S. Department of Energy, Used Fuel Disposition R&D Campaign.

Hardin, E. and M. Voegele 2013. *Alternative Concepts for Direct Disposal of Dual Purpose Canisters*. FCRD-UFD-2013-000102, Rev.0. U.S. Department of Energy, Used Fuel Disposition R&D Campaign.

IAEA (International Atomic Energy Agency) 2006. *Geological Disposal of Radioactive Waste.* IAEA Safety Standards Series No. WS-R-4. Vienna, Austria: International Atomic Energy Agency and the Organization for Economic Co-operation and Development, Nuclear Energy Agency.

Kremer, E.P., J.D. Avis, T. Chshyolkova, F,. Garisto, P. Gierszewski, M. Gobien, N.G. Hunt, C.I. Kitson, C.L.D. Medri, T.W. Melnyk and L.C. Wojciechowski 2011. “Postclosure Safety Assessment of a Deep Geological Repository for Canada’s Used Nuclear Fuel.” ISBN: 978-0-89448-085-0. *Proceedings of the 13th International High-Level Radioactive Waste Management Conference (IHLRWMC) 2011*, Albuquerque, NM (April 10-14, 2011). American Nuclear Society.

McCartin, T. 2012 verbal communication. Presentation to the U.S. Nuclear Waste Technical Review Board, Spring 2012 Board Meeting. (Transcript from March 7, 2012; www.nwtrb.gov).

Nagra (National Cooperative for the Disposal of Radioactive Waste) 2002. *Project Opalinus Clay: Safety Report, Demonstration of disposal feasibility for spent fuel, vitrified high-level waste and long-lived intermediate-level waste (Entsorgungsnachweis)*. Technical Report 02-05. Wettingen, Switzerland: Nagra.

OCRWM (Office of Civilian Radioactive Waste Management) 1999. *License Application Design Selection Report*. B00000000-01717-4600-00123 REV 01 ICN 01. August, 1999. U.S. Department of Energy, Yucca Mountain Site Characterization Office, Las Vegas, Nevada.

OECD (Organisation for Economic Co-Operation and Development) 2004. *Postclosure Safety Case for Geological Repositories Nature And Purpose.* Nuclear Energy Agency No. 3679, ISBN 92-64-02075-6. Paris, France: Organisation for Economic Co-Operation and Development, Nuclear Energy Agency

ONDRAF/NIRAS (Belgian Agency for Radioactive Waste and Enriched Fissile Materials) 2001. *Technical Overview of SAFIR 2: Safety Assessment and Feasibility Interim Report 2.* NIROND 2001-05 E. Belgium: ONDRAF/NIRAS.

Posiva 2010. *Interim Summary Report of the Safety Case 2009.* POSIVA 2010-02, ISBN 978-951-652-173-5. Eurajoki, Finland: Posiva Oy.

SKB (Swedish Nuclear Fuel and Waste Management Co.) 2011. *Long-term Safety for the Final Repository for Spent Nuclear Fuel at Forsmark: Main Report of the SR-Site Project.* Technical Report TR-11-01 (3 volumes). Stockholm, Sweden: SKB. www.skb.se

SNL (Sandia National Laboratories) 2008. *Postclosure Nuclear Safety Design Bases*. ANL-WIS-MD-000024 REV 01. Prepared for the U.S. Department of Energy, Office of Civilian Radioactive Waste Management. February, 2008

Vaughn, P., et al. 2011. *Generic Disposal System Modeling Fiscal Year 2011 Progress Report.* FCRD-USED-2011-00018. U.S. Department of Energy, Used Fuel Disposition R&D Campaign. August, 2011.

Weber, J.R., S. Keller, S. Mrugalla, J. Wolf, D. Buhmann, J. Mönig, W. Bollingerfehr, J. Krone and A. Lomerzheim 2011. "Safety Strategy and Assessment for a German HLW-Repository in Salt." *Proceedings of the 13th International High-Level Radioactive Waste Management Conference (IHLRWMC) 2011*, Albuquerque, NM (April 10-14, 2011). American Nuclear Society.

THIS PAGE INTENTIONALLY LEFT BLANK

7. Performance Assessment Approach

The purpose of performance assessments (PAs) in evaluating the feasibility of direct disposal of spent nuclear fuel (SNF) in dual-purpose canisters (DPCs) such as those that currently exist, is to support conclusions as to whether or not disposal could be accomplished safely as defined by regulatory performance objectives. The focus of such performance assessments would be the potentially important aspects of DPC direct disposal performance that would be different from those associated with current waste package designs and disposal concepts. Models used to compare the safety of DPC direct disposal with alternative concepts would include appropriate FEPs, at sufficient detail to discern the differences. Potential differences between direct disposal of DPCs and disposal of the same SNF in packages (including canisters) designed for disposal and emplaced in the same host medium, could include emplacement mode and engineered barrier system design, thermal effects (comparing DPC direct disposal to other open modes), thermal effects (comparing DPC direct disposal to enclosed modes), the quantity of spent nuclear fuel, and the inner canister design. Performance assessments would consider the state of advancement of supporting models, among other factors. This section describes how the PAs that could examine potentially important aspects of DPC direct disposal performance would be conducted, including screening of features, events and processes (FEPs), and the definition of scenarios for the assessments.

Future PAs would model the implementation of safety strategies described in Section 6 for selected DPC disposal concepts (Section 4). They would evaluate significant differences, if any, in postclosure waste isolation performance for DPC direct disposal, as compared to re-packaging the same SNF in purpose-built containers and disposal in the same geologic setting. The hypothetical purpose-built containers could have any capacity; they likely would contain fewer SNF assemblies than DPCs because smaller containers facilitate handling and limit ventilation times needed to accommodate thermal constraints.

The PAs would be generic (i.e., not site-specific) consistent with the objectives of the Used Fuel Disposition (UFD) R&D program. This is reflected in the safety strategies, which identify and describe the functions for the engineered and natural barriers, for a total of six generic disposal concepts (Section 6). More concepts and variations would certainly be possible but these six, further simplified to three for purposes of this section, would likely be sufficient to support initial findings as to the safety aspects of feasibility evaluation for DPC direct disposal.

The following sections describe the DPC disposal concepts to be evaluated, the differences between DPC direct disposal and other alternatives, and the screening of FEPs for inclusion in the PA models. PA models would be developed for defined cases which would be grouped into scenario classes, also described. Scenario classes would be based on separate and independent initiating conditions or events (e.g., separate nominal scenario, human intrusion scenario, etc.) so that the probability weighted consequences for each scenario can be summed (or combined appropriately using annual event probabilities) to assess system performance. Finally, numerical strategies would be briefly discussed that would be sufficient for implementing a base case model, and then more detailed studies.

7.1 Disposal Concepts for Performance Assessment

The reduced set of disposal concepts recommended for PAs to support feasibility evaluation (Table 4-3) consists of:

- Salt (in-drift emplacement)
- Hard Rock Backfilled, Open Concept (in-drift emplacement)
- Sedimentary Backfilled, Open Concept (in-drift emplacement)

The salt concept is an enclosed emplacement mode, in which waste packages would be placed on the floor in long drifts, and covered immediately with crushed salt backfill (Section 4.2.1). The heat dissipation properties of salt, and its tolerance for temperature up to 200°C, allow emplacement and immediate backfilling after only a few decades of decay storage (Hardin and Voegele 2013). The other concepts would place packages in open drifts, and ventilate for decades to remove heat (Sections 4.5 and 4.6.2). At repository closure the drifts would be completely backfilled with low-permeability material. All of these concepts can dissipate heat from DPCs while maintaining peak temperatures at prescribed limits, with repository (panel) closure by the time the SNF age is 150 years out-of-reactor (see assumptions in Section 2).

As a simplification, backfill is represented as a base case, and the unbackfilled cases (hard rock and sedimentary) can be considered when additional FEPs (e.g., rockfall, drift collapse, mechanical damage, etc.) are included in PA models. As a further simplification, the cavern retrievable concept is represented by the hard rock backfilled concept.

The proposed base case includes: 1) waste package corrosion; 2) saturated, low-permeability host medium; 3) emplacement drift backfill; 4) disturbed rock zone (DRZ) development; 5) advective and/or diffusive transport of radionuclides; 6) insignificant colloid and biocolloidal mobility; and 7) insignificant effects on radionuclide mobility from thermally driven processes or repository introduced materials.

More advanced simulation cases suggested by Hardin et al. (2013) include additional model refinements to represent:

- Lumped parameter model for near-field chemistry
- Backfill options
- Thermally driven coupled processes
- Waste package degradation mechanisms and partial containment
- Consequences from seismic ground motion and drift collapse

The base case and simulation cases reference a disposal system architecture shown in Figure 6-1.

7.2 Potentially Important Aspects of DPC Direct Disposal Performance

Performance assessment models used to compare DPC direct disposal with alternative concepts should include appropriate FEPs, at sufficient detail to discern the differences, and could evaluate releases, defense in depth, engineering acceptance, and ancillary effects. Potential differences between direct disposal of DPCs and disposal of the same SNF in packages (including canisters) designed for disposal and emplaced in the same host medium, may include:

Emplacement Mode and EBS Design – With the exception of the salt concept, DPC disposal concepts would be open emplacement modes that use ventilation to remove heat. Enclosed modes in crystalline and sedimentary rock types (Section 4; also Hardin et al. 2012) have steel liners or low-permeability, clay-based swelling buffer materials directly contacting the waste package. For the open modes waste packages would be surrounded by void space, then by backfill or other engineered barriers installed at closure. After closure, void spaces would gradually fill with debris from rockfall and collapse. Backfill emplaced remotely in emplacement drifts would have less density and uniformity than manually emplaced buffer/backfill materials. Hence, comparisons of DPC direct disposal with other concepts involving re-packaging, would likely include effects associated with greater permeability and potential for radionuclide transport in the near field.

Thermal Effects (comparing DPC direct disposal to other open modes) – The most common DPC sizes currently in use have the capacity for 24- or 32-PWR assemblies (or BWR equivalent). Future DPCs may contain more SNF, such as the Magnastor 37-PWR canister recently marketed by NAC International. By contrast, previously in the U.S. the transport/aging/disposal (TAD) canister was designed to contain 21 PWR assemblies. Based partly on that selection, a 21-PWR reference package size was recommended for open emplacement modes (Hardin et al. 2012). The 21-PWR size gives a reduction in the number of packages required to dispose of the U.S. SNF inventory, compared to smaller packages used with enclosed modes, while taking advantage of thermal management by the extended ventilation that is possible with open modes.

Peak temperatures for larger-capacity packages could be controlled by decay storage and repository ventilation, but post-peak temperature would remain higher for hundreds to thousands of years (aging attenuates short-lived fission products, but larger packages contain more heat-generating actinides with intermediate half-lives, such as ^{241}Am). Thus, although peak drift wall temperature could be managed and might be equivalent for 21-and 32-PWR sizes, for packages containing more SNF the peak temperature further into the host rock may be greater, and elevated temperature is likely to persist longer. In the backfill, the extent (e.g., in the axial direction between packages) and duration of elevated temperatures would also be greater. These differences could eventually impact radionuclide transport if the controlling rock and backfill characteristics would be thermally sensitive. The direction of affected radionuclide transport could be radial or axial.

Thermal Effects (comparing DPC direct disposal to enclosed modes) – Enclosed emplacement modes in crystalline and sedimentary rock types were shown to require 4-PWR size waste packages (or smaller) to limit peak temperature at the waste package surface to less than 100°C (Hardin et al. 2012). The additional time required for 50-year old SNF to cool by 8-fold (the difference between 32 and four assemblies per package) is on the order of 400 to 1,000 years depending on burnup. It is impractical to cool DPCs that long before disposal. Thus, both the peak temperature throughout the near field, and the duration of elevated temperature, would generally be greater for in-drift disposal of DPC-based packages (with decades of repository ventilation) than for smaller packages using enclosed emplacement modes. Performance of enclosed modes would more closely resemble that analyzed for the Swedish (SKB 2011) and French (Andra 2005) SNF disposal concepts. In accordance with this discussion, assessment of DPC direct disposal should include thermal effects on the near-field environment, in sufficient detail for comparison with alternative enclosed modes.

A notable exception to the need for small packages with enclosed emplacement modes is the salt concept, which could accommodate SNF waste packages up to 32-PWR size or larger. Peak salt temperature is directly related to package thermal power at emplacement, and the power limit could be met by 32-PWR size packages with high burnup SNF, after decay storage of approximately 70 years (Hardin and Voegele 2013). This is a significantly larger disposal capacity than the reference 12-PWR package size that was selected for a previous study based on shaft hoist considerations instead of thermal performance (Hardin et al. 2012).

Quantity of SNF – Once a waste package breach occurs more SNF would be exposed to the disposal environment with DPCs, than with smaller containers. The difference would be greatest in comparing DPCs with 4-PWR size canisters used for enclosed emplacement modes. The onset of diffusive radionuclide transport is controlled by concentration, not total contaminant mass, so the greatest potential for differences in radionuclide mobility would initially result from advective transport. Advection is expected to be insignificant for low-permeability host media (except possibly for human intrusion scenarios) but might occur locally and/or in response to changes in boundary conditions. Even the most massive, homogeneous sedimentary host units could be traversed by through-going faults or fracture zones with greater permeability, that could act as advective pathways in and out of the emplacement drifts. The possibility for advective transport is a factor of interest in the safety of DPC direct disposal, related to quantity of SNF per package.

Inner Canister Design – Canisters purpose-built for disposal may have features not found in existing DPCs, such as thicker shells, plates and/or spacers to extend structural lifetime in corrosion environments; thicker neutron absorbing elements that could function after 10^4 years of degradation; and fillers that could exclude moderating groundwater after package breach. Existing DPCs do not include these features (assuming they cannot be reopened; see Section 2). The probability of criticality may be greater, and the consequences of a criticality event may need to be considered.

7.3 Performance Assessments for Direct DPC Disposal

A base case PA is proposed below together with a set of simulation cases that would include additional FEPs and/or address particular questions. The included FEPs should be sufficient to represent potentially important differences between DPC direct disposal and disposal of the same SNF in purpose-built packages, including distinguishing among effects due to size and to lack of features that a disposal package would have, in the same host medium. The topics addressed would be those discussed above: emplacement mode and EBS design, thermal effects (comparing DPC direct disposal to other open modes), thermal effects (comparing to enclosed modes), quantity of SNF, and inner canister design. A review of scenarios needed to implement total system PA shows that the base case representation of (backfilled) concepts would need: 1) nominal scenario; 2) human intrusion scenario; and 3) criticality scenario (unless postclosure criticality FEPs can be excluded based on probability of occurrence).

7.3.1 Features, Events and Processes

The recommended FEPs for inclusion in the base case are specified by Hardin et al. (2013) based on a comprehensive generic list developed by Freeze et al. (2011). The recommended base case FEP list (Hardin et al. 2013, Table 1) is limited to facilitate PA model development, but an extended list that could be supported by more detailed simulation cases is also provided.

The Freeze, et al. (2011) set of generic FEPs was developed from international sources and prior experience in the U.S. and is currently being used by the UFD campaign in the disposal R&D program. The generic FEP list was analyzed by Vaughn et al. (2011) who identified FEPs that were then included in generic performance assessments for repositories in clay/shale media, crystalline rock, salt, and the deep crystalline basement (deep borehole disposal concept). The results for these three generic mined-disposal PA models are similar to the recommended base case FEP list (Hardin et al. 2013, Table 1).

Among the generic FEPs, some were identified in a planning exercise as warranting more investigation because of the state of knowledge and the potential impact on waste isolation performance (Nutt 2011). A later analysis of FEPs considered which should be included in the EBS components of a next-generation PA (Hardin 2012). The list identifies a simpler case that could be readily implemented, and a more advanced case that could be used to evaluate impacts from additional processes and repository design features. The FEP list for the simpler case is compared with that recommended for DPC direct disposal PA (Hardin et al. 2013, Table 1). Given the extent of previous FEP studies, there is little need to reiterate, except to point out that the FEPs needed to evaluate DPC direct disposal safety are similar to those recommended previously by multiple studies. Also, that the associated PA effort can be approached iteratively starting with simpler models.

For comparison of DPC direct disposal with disposal of the same waste re-packaged in purpose designed containers using the same ***open*** emplacement mode, performance of the SNF waste form, waste package and other engineered barriers, and the far field would be similar in concept and safety strategy. Thermal effects would be potentially important in relation to waste package capacity. Impacts from inner canister design differences may be limited to the technical analysis that is relied upon to disposition the postclosure criticality FEP (either include or exclude). The quantity of SNF is not likely to be significant for comparing packages differing only slightly in size. The FEPs needed for comparing DPC direct disposal with similar ***open*** modes of emplacement would be limited to an appropriate base case that includes thermal effects and advective transport of radionuclides (as well as other transport processes).

For comparing DPC disposal to disposal of the same SNF in the same host medium but smaller canisters using an ***enclosed*** emplacement mode, more FEPs need to be included. For example, reference enclosed crystalline or clay/shale concepts for spent fuel use 4-PWR packages encapsulated in clay-based swelling buffer/backfill, which would involve buffer degradation. The additional FEPs needed to compare ***open*** emplacement of DPC-based packages (with backfilling at closure) vs. ***enclosed*** emplacement of smaller waste packages, would be included among those recommended for the base case (Hardin et al. 2013, Table 1).

All canisters, whether existing DPCs or purpose-built for disposal, would have disposal overpacks (Section 4). These would provide additional benefits to the safety of handling, transport and emplacement in the repository, and provide containment integrity for some period of time after emplacement. Disposal overpacks for existing DPCs might differ with respect to dimensions, shape and lifting features. However, for this study they could be assumed to have the same characteristics as disposal overpacks for purpose-built disposal canisters, such as material type, thickness, fabrication method, surface treatment, etc., that could affect waste isolation performance.

Various FEPs such as those representing multi-phase thermally driven processes, corrosion resistant packaging, and consequences from seismic ground motion, may be added to represent the other concepts discussed in Chapter 6.

7.3.2 Base Case

The base case is intended to be implemented using off-the-shelf system modeling software. A standardized model framework could be used for all disposal concepts, varying the connectivity of model components and the uncertainty distributions describing key parameters to represent concept-specific details. A base case model implemented this way would have limited dimensionality (e.g., 1-D or 2-D) and likely would rely on simplified mathematical models or abstractions, especially to represent processes in the near-field and EBS (Vaughn et al. 2011). Processes to be represented this way in the nominal scenario include waste package and waste form degradation, solubility controls, sorption, and radionuclide transport (head gradient, conductivity, diffusion or flow area, path length).

For the recommended base case model process detail would be limited, and certain FEPs would be excluded (as indicated in Table 1 of Hardin et al. 2013). Some of these simplifications and the supporting rationale are as follows:

- Preclosure events and processes do not affect postclosure performance or do not discriminate direct disposal of DPCs.
- Surface characteristics and surficial mechanical, hydrologic, chemical, biological, and thermal processes do not discriminate direct disposal of DPCs.
- Seismic and faulting events do not significantly impact the natural system and biosphere, and mechanical degradation and seismic ground motion do not significantly affect backfilled underground systems (Pratt et al. 1979). Faulting does not affect the repository, igneous activity has very low probability of disrupting the repository, and other long-term processes can be excluded. Climate change can be addressed by changing groundwater flow boundary conditions.
- Heterogeneity and co-location of the waste inventory can be addressed by considering a range of age and burnup without considering location in a repository.
- Cladding performance can be conservatively neglected although partial credit for cladding containment may be taken in a more complete simulation case.
- Early waste package failure and modes of waste package corrosion would be potentially significant for corrosion-resistant disposal overpacks, but much less so for corrosion allowance materials designed for shorter corrosion lifetimes. Evolution of flow pathways within failed packages can be neglected using a simplification that flow occurs there without restriction.
- Backfill is not degraded by mechanical or chemical processes although clay-based backfill and buffer materials may be eroded by groundwater flow.
- Radionuclide mobility is not affected by mechanical loading of waste forms, loading at interfaces, or other mechanical degradation processes in the EBS.
- Flow in the EBS can be represented using simplified approaches, but unsaturated flow and flow through far-field plugs and seals requires more detailed models.
- Chemical processes in the EBS can affect the rates of waste package and waste form degradation and radionuclide transport. Bounding approximations to those rates can be used, supported by sensitivity analyses.

- Colloids do not contribute significantly to radionuclide transport because of limited mobility in low-permeability engineered and/or natural media.
- Thermal effects on flow and transport can be neglected as relatively brief transients that occur before most radionuclides are released. Because disposal overpacks would be resistant to thermal degradation, or provide corrosion allowance, water is not likely to contact waste until after the thermal period.
- Seals and plugs would be located outside the influence of heating.
- Gas generation from corrosion of steels is a second-order influence on radionuclide mobility and transport.
- Radiation effects in the disposal environment can be taken into account by adjusting degradation rates and solubilities.
- Far-field flow and transport in the host rock formation and other units would be substantially unchanged by repository excavation, mechanical effects, unsaturated flow processes, dehydration, surface water discharge, and thermally driven processes. Thermal processes, gas generation, and nuclear criticality would not be significant in the far field. Chemical processes in the far field can be neglected if simplified bounding approaches to radionuclide transport are used.
- Future human actions would be limited to inadvertent human intrusion.
- Human behavior and biosphere characteristics do not discriminate direct disposal of DPCs.
- Biological processes do not significantly affect radionuclide transport in the far field.
- The disposal packages would be designed to preclude an in-package nuclear criticality by limiting the potential for water accumulation to flood breached waste packages and by the inclusion of neutron absorbers. Criticality in the EBS or near field is very unlikely because of homogeneous hydrochemical conditions that promote dissipation rather than concentration of fissile radionuclides, and/or moderator exclusion and the presence of neutron absorbers.

7.3.3 Simulation Cases

The simulation cases described below identify types of models that could provide additional process-level detail for PAs for DPC direct disposal. Additional FEPs that could be included for this purpose are included in the FEP list (Hardin et al. 2013, Table 1). The additional detail could be abstracted to augment the base case model, or a different approach could be used such as merging component models with a “host” multi-physics simulator (Hardin 2012).

Lumped Parameter Model for Near-Field Chemistry – A “mixing cell” approach for modeling water chemistry in the near-field EBS is appropriate for non-advecting or slowly advecting conditions. This simulation case would be used to further investigate the relationship between the quantity of SNF and advective transport conditions. The near-field EBS water composition serves as a concentration boundary condition for diffusive transport and as the composition of advective outflux, for example, associated with human intrusion.

Backfill Options – This simulation case could be used to quantify the performance consequences of different backfill functions, and the effectiveness of plugs and seals. Functions assigned to backfill may include isolating waste packages by limiting radionuclide transport, and stabilizing openings in the host rock. If the backfill function is limited to stabilizing openings, then other materials might be used in lieu of swelling clay.

Thermally Driven Coupled Processes – The thermal conditions of DPC direct disposal could be important if thermally driven processes cause long-duration changes in the EBS or host rock. Changes such as fracture or matrix porosity changes, dewatering, bulk rock shrinkage, mineral alteration, reduction of surface area, etc., could affect the chemistry and quantity of water flow in the EBS.

Waste Package Degradation Mechanisms and Partial Containment – Containment lifetime is an objective for most disposal concepts, whether to limit releases during the thermal period, or throughout the regulatory performance period. Corrosion-resistant outer barrier materials would be used in the disposal overpack for several concepts (Section 6), and can be represented using more detailed mechanistic models.

Seismic Ground Motion and Drift Collapse – For unbackfilled disposal concepts, rockfall and drift collapse would eventually degrade the emplacement openings, potentially impacting engineered barriers such as the waste package. Seismic initiation could increase the frequency of rockfall and drift collapse. Where waste package containment lifetime is an important part of the performance strategy, rockfall and drift collapse could be important to waste isolation performance.

Unsaturated Flow in the Host Rock – In the unsaturated zone, multi-phase flow and thermally driven flow processes would be potentially important. The controlling FEPs would be the same as for saturated flow, but the conceptual, mathematical and numerical models would be different. Simulation could be accomplished using existing thermal-hydrology process models, coupled as necessary with chemical and mechanical models.

Degradation of Far-Field Plugs and Seals – Degradation of far-field engineered barriers could change groundwater flow patterns and facilitate advective transport of radionuclides. Although these far-field components likely would not be affected by heat, their performance may be important in relation to the quantity of SNF if there is significant advective transport. Simulation of plug and seal performance could be accomplished using groundwater flow codes taking into account flow contributions from backfill, the DRZ, undisturbed host rock, and other flow features that may be active.

7.3.4 Scenarios

An overall assessment of risk typically combines probabilistically independent scenarios by summing the consequences calculated for each. As independent scenarios, they depend only on the initial state of the system and the class of event sequences that is unique to the scenario. The nominal scenario is the starting point for building a PA. It describes projected performance without disruptive events, and is required to include features and processes that could significantly contribute to system performance. The possibility that one or more waste packages could be defective leads to the early failure component of the nominal scenario. There would be many different ways that defects could arise, and predicting many of them is problematic, so an early failure component may be added based on a conservative, probability weighted consequence.

Human intrusion is a required part of the PA using a separate scenario (that could be, depending on the regulatory construct, stylized) to represent the dose consequence from future drillers inadvertently intersecting one or more waste packages and exposing the contents to transport by groundwater. The human intrusion scenario is defined and required by current regulations (§40CFR191 or 40CFR197), independently from other scenarios including seismic ground motion and postclosure criticality. The potential for changes in the probability of borehole interception of waste packages, associated with direct disposal of DPCs, was evaluated (Hardin et al. 2013, Appendix A) for sedimentary and hard-rock settings. The results indicate that the expected number of intersections is greater than 1, particularly for sedimentary concepts (including the salt concept), and for horizontal waste package orientation.

For a backfilled repository that is situated where there would be limited seismic and fault displacement hazards, it is reasonable to anticipate that seismic ground motion and its consequences could be excluded from PA so that no seismic scenario would be needed. For an unbackfilled repository subject to seismic hazard of sufficient likelihood, a mechanistic representation of ground motion effects would be needed to support a seismic consequence model.

Postclosure criticality is treated as a separate scenario that could be formulated, for example, based on an event tree approach (Rechard et al. 1996). Assuming the probability of volcanic disruption of the repository is less than 10^{-8} per year, seismicity is the only other disruptive event class that might have a high enough joint probability with criticality to be included in PA. However, if seismic FEPs can be excluded on low consequence for backfilled repositories, it is likely that the potential for criticality would not affected, so criticality could be considered in a separate scenario.

7.3.5 Numerical Implementation

The base case model described in this section can be implemented using off-the-shelf software to model waste form degradation, EBS performance, and radionuclide transport, incorporating parameter uncertainty and applying Monte Carlo methods to generate successive realizations of total system performance. The base case would use abstracted inputs such as temperature histories from supporting calculations, functions to represent waste package and waste form degradation, and uncertainty distributions for radionuclide transport parameters. Most of these inputs would be developed separately for different concepts.

A set of simple PA models was developed by Vaughn et al. (2011) for repositories in clay/shale media, salt, crystalline rock and deep boreholes. In particular, the clay/shale model is structured and versatile, and runs entirely within the GoldSim® software application. This software was developed to represent systems with both engineered and natural components, incorporating uncertainty and applying Monte Carlo methods. The clay/shale model includes "mixing cells" for waste form, waste package, a second engineered barrier, a DRZ, and the far field. Multiple cells would be used to discretize these features, and the cells would be connected by advective-diffusive pathways. Waste form and waste package degradation parameters (e.g., waste inventory, fractional degradation rate, package lifetime, temperature dependence, temperature histories), groundwater flow parameters (flow area and path length, flux, velocity) and radionuclide transport parameters (representing diffusion and sorption) would be input as uncertainty distributions. Parameters would be used to tailor individual elements and the connectivity between them, to represent different disposal concepts. The 2-D far-field part of the model could be used to represent axial and radial variation of a DRZ. With appropriate development of inputs, the approach is suitable for modeling the base case described in this report, especially with emphasis on comparisons between DPC direct disposal and alternative concepts involving re-packaging.

A more advanced numerical model architecture was described previously (Hardin 2012) and is the ultimate goal of model development. A similar architecture is being developed by the UFD R&D campaign. It would replace the off-the-shelf modeling software, such as GoldSim, with a detailed numerical grid and multi-physics simulation algorithms that could include multi-phase non-isothermal flow and reactive chemical transport. This would avoid abstractions to a great extent, and run all model components simultaneously in a high-performance computing environment. In this approach the PA model would always use such a numerical host simulation for the natural system, although it might have limited dimensionality and could even be 1-D if that satisfies the application requirements. The EBS would be represented as a subdomain for each waste package within the host simulation, where multi-physics couplings would be activated, specific to local processes such as corrosion of metallic components, evolution of EBS flow paths, sorption on corrosion products, etc. Uncertainty and successive realizations of the model would be managed at runtime using a shell such as DAKOTA (Freeze and Vaughn 2012).

References for Section 7

Andra (National Agency for Radioactive Waste Management) 2005. *Dossier 2005 argile – architecture and management of a geological disposal system*. December, 2005. http://www.Andra.fr/international/download/Andra-nternationalen/document/editions/268va.pdf.

Freeze, G.A., P. Mariner, J.A. Blink, F.A. Caporuscio, J.E. Houseworth and J.C. Cunnane 2011. *Disposal System Features, Events and Processes: FY11 Progress Report.* U.S. Department of Energy, Used Fuel Disposition R&D Campaign. FCRD-USED-2011-000254 Rev. 0. August, 2011.

Freeze, G. and P. Vaughn 2012. *Development of an Advanced Performance Assessment Modeling Capability for Geologic Disposal of Nuclear Waste: Methodology and Requirements*. SAND2012-10208. Sandia National Laboratories, Albuquerque, NM.

Hardin, E. 2012. *Generic Engineered Barrier System Model and System Architecture.* U.S. Department of Energy, Used Fuel Disposition R&D Campaign. FCRD-UFD-2012-000180 Rev.0. July, 2012.

Hardin, E., T. Hadgu, D. Clayton, R. Howard, H. Greenberg, J. Blink, M. Sharma, M. Sutton, J. Carter, M. Dupont, and P. Rodwell 2012. *Disposal Concepts/Thermal Load Management (FY11/12 Summary Report)*. FCRD-USED-2012-000219, Rev.2. U.S. Department of Energy, Used Fuel Disposition R&D Campaign. November, 2012.

Hardin, E. and M. Voegele 2013. *Alternative Concepts for Direct Disposal of Dual-Purpose Canisters.* U.S. Department of Energy, Used Fuel Disposition R&D Campaign. FCRD-UFD-2013-000102 Rev. 0. February, 2013.

Hardin, E., C. Bryan and M. Voegele 2013. *Features, Events and Processes and Performance Assessment Scenarios for Alternative Dual-Purpose Canister Disposal Concepts.* FCRD-UFD-2013-000172 Rev. 0. U.S. Department of Energy, Used Fuel Disposition Campaign. July, 2013.

Nutt W.M. 2011. *Used Fuel Disposition Campaign Disposal Research and Development Roadmap*. FCR&D USED- 2011-00065 Rev. 0. U.S. Department of Energy, Used Fuel Disposition R&D Campaign. March, 2011.

Pratt, H.R., D.E. Stephenson, G. Zandt, M. Bouchon and W.A. Hustrulid 1979. *Earthquake Damage to Underground Facilities.* Proc. of the 1979 Rapid Excavation and Tunneling Conference, Vol. 1. American Institute of Mining Engineers, Littleton, CO.

Rechard, R.P., M.S. Tierney, L.C. Sanchez and M.-A. Martell 1996. *Consideration of Criticality when Directly Disposing Highly Enriched Spent Nuclear Fuel in Unsaturated Tuff: Bounding Estimates*. Sandia National Laboratories, Albuquerque, NM. SAND1996-0866. May, 1996.

SKB (Swedish Nuclear Fuel and Waste Management Co.) 2011. *Long-term safety for the final repository for spent nuclear fuel at Forsmark: Main report of the SR-Site project.* (3 volumes). Technical Report TR-11-01.

Vaughn, P., et al. 2011. *Generic Disposal System Modeling Fiscal Year 2011 Progress Report.* FCRD-USED-2011-00018. U.S. Department of Energy, Used Fuel Disposition R&D Campaign. August, 2011.

THIS PAGE INTENTIONALLY LEFT BLANK

8. Criticality Evaluations for Existing Dual-Purpose Canisters

A general methodology for addressing postclosure criticality was developed for previous licensing-related activities (DOE 1998) and approved by the U.S. Nuclear Regulatory Commission (NRC 2000). This methodology was used to demonstrate that criticality could be excluded from repository performance considerations on the basis of low probability (NRC 2011). The performance specifications and loading requirements for the transportation, aging, and disposal canisters (TADs) were important to the low probability results. Also, because TADs have not been fully designed, licensed or deployed, there is a level of flexibility for future canister designs to reduce the potential for criticality, that existing DPCs do not have. As noted in Section 6, long-life overpacks could be used as a means to address criticality by limiting the ability of a moderator to enter the waste package over time.

The design of existing DPCs has not taken geologic disposal conditions into account, so they may not have performance capabilities that would support a low probability determination. Of particular relevance to criticality, DPC basket structures and neutron absorber materials were not designed to maintain their efficacy under geologic disposal conditions and time frames. This does not mean that DPCs cannot meet disposal objectives, but rather that some options are more limited and additional reliance on other engineered barrier system functions or repository aspects may be necessary to demonstrate repository system performance. Some limited studies have been conducted in the past to evaluate the feasibility of DPC direct disposal from a criticality perspective, and have concluded that while such a demonstration is technically feasible, it would also be technically challenging (BSC 2003; EPRI 2008) and acceptance by the regulatory is uncertain.

Performance of the neutron absorber material used in DPCs is one key factor in demonstrating subcriticality, even using full burnup credit in the analysis. The neutron absorber panel material used in the majority of existing DPCs is Boral®, which is composed of B_4C particles and Alloy 1100 aluminum hot rolled together to form a neutron absorbing core (Section 3). The neutron absorbing core is then bonded to two thin sheets of Alloy 1100 aluminum cladding. Various tests have been performed on this material because it is used in existing canisters, casks and spent fuel pools. Some corrosion tests, conducted under (borated) pool chemistry conditions showed a 0.00028 inch per year (7 μm/year) rate of cladding (sheathing) material loss, corresponding to about 40 years before degradation of the Boral core would commence (EPRI 2009). Additionally, tests under simulated vacuum drying conditions (as may be used to dewater fuel canisters) showed that blisters can form within the core (NRC 2011). Disposal time periods of interest are on the order of 10,000 years, so Boral would likely degrade if exposed to an aqueous environment.

Another consideration associated with the majority of existing DPCs is that the neutron absorber panels do not extend over the full length of the internal DPC cavity. Over time, in conjunction with fuel assembly material degradation, this can lead to regions within a DPC where locations within the fuel basket do not have sufficient neutron absorber (based on reactivity analysis).

In a repository performance assessment (PA) features, events and processes (FEPs) that could affect the repository are examined (Section 7). In developing a PA, FEPs that can significantly affect repository performance are screened for inclusion or exclusion. Options for excluding a FEP include criteria for low probability, low consequence and regulation. The purpose of the criticality evaluations presented here is a scoping evaluation of DPC reactivity changes that could occur under disposal conditions, and identification of system attributes that could be used to support future FEP screening justifications.

8.1 Nuclear Criticality Analysis

Licensed storage and transportation cask systems have well-defined assembly-loading criteria (e.g., specifications for "approved contents" in a storage cask system's Certificate of Compliance). These specifications define loading conditions and characteristics for which the cask system as described in the Safety Analysis Report complies with the applicable regulatory requirements. Canister systems certified for transportation have been analyzed under fully flooded (fresh water) conditions (§10CFR71.55). In such analyses the neutron absorbers and basket geometry are assumed to function as designed, and the systems are demonstrated to be subcritical under both normal and hypothetical accident conditions of transportation.

Note that if water can be excluded or significantly delayed from the repository or from entering a package, there is little potential for criticality. Hence, an important assumption for criticality analysis is that water enters a package at some point over the regulatory timeframe (e.g., 10,000 years). While the different geologic settings and material degradation mechanisms yield a large number of potential configurations, two stylized configurations are used in this study to assess DPC reactivity changes that can occur over repository timeframes:

- Total loss of neutron absorber from unspecified degradation and transport processes
- Loss of the internal basket structure resulting in elimination of assembly-to-assembly spacing

These general configurations are consistent with analyses performed previously (DOE 2008). In this report they are analyzed for DPCs flooded with fresh water and NaCl brine. Two types of DPCs are analyzed: the Transportable Storage Canister (TSC-24) from NAC International, and the Multi-Purpose Canister (MPC-32) from Holtec International. These designs represent major categories of existing DPCs (as of June, 2013 there are 185 MPC-32 and 220 TSC-24 loaded casks in dry storage). TSC-24 canisters deployed at the Maine Yankee site, and MPC-32 canisters at the Sequoyah plant are analyzed. Information regarding actual canister loading and some operating history data for the assemblies in these canisters were available to support detailed evaluations. The TSC-24 is a flux-trap design with dual neutron absorber panels between each fuel pair of adjacent assemblies, and the MPC-32 uses an egg-crate basket design with single neutron absorber panels between assemblies.

While existing DPCs do not have corrosion-resistant neutron absorbing materials, they were loaded conservatively, so that there may be significant reactivity margin to offset increases from flooding with basket degradation and/or loss of absorber. Scoping the available margin requires use of burnup credit and detailed canister-specific analysis.

8.2 Codes, Models, and Methods

All neutronics calculations were performed using the SCALE code system (ORNL 2011). Depletion calculations were loaded into KENO-VI criticality models (Clarity and Scaglione 2013). Fuel assembly designs used in this analysis include the Westinghouse (W) 17×17 Optimized Fuel Assembly (OFA), the Westinghouse 17×17 Standard Fuel Assembly (STD), and the Combustion Engineering (CE) 14×14 fuel assembly. The Westinghouse designs are stored in MPC-32 canisters and the CE 14×14 design is stored in TSC-24 canisters. Graphical depictions of the W 17×17 and the CE 14×14 fuel assemblies are shown in Figures 8-1 and 8-2, respectively, with dimensions provided in Tables 8-1 and 8-2.

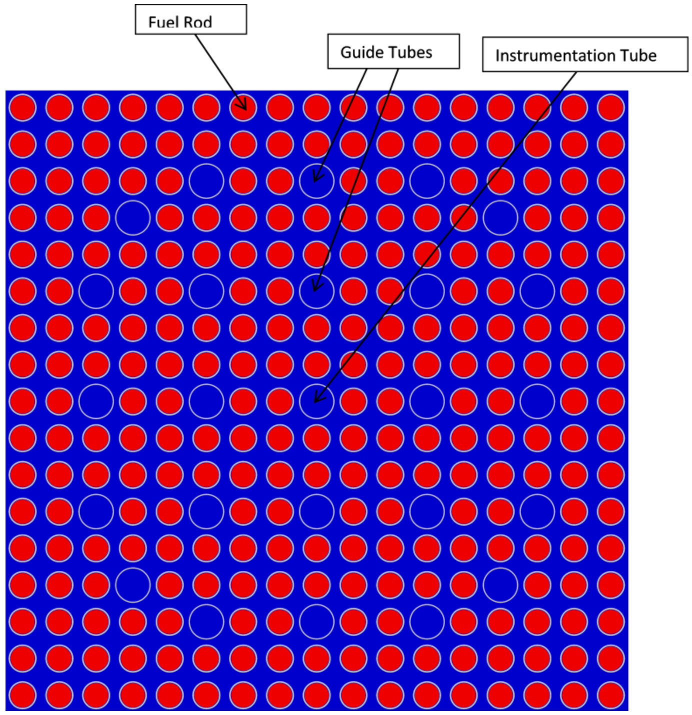


Figure 8-1. Cross-sectional model view of a Westinghouse 17 × 17 OFA and STD fuel assembly

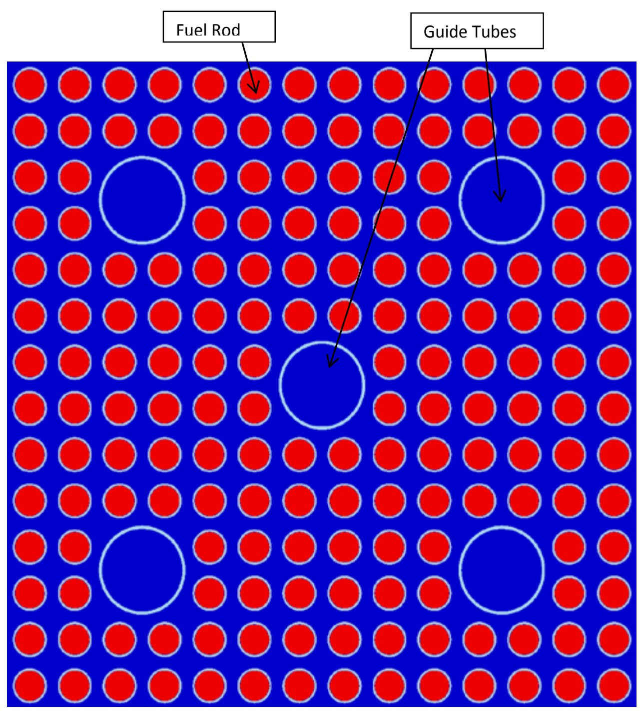


Figure 8-2. Cross-sectional model view of a Combustion Engineering 14 × 14 fuel assembly

Table 8-1. Westinghouse 17 × 17 OFA dimensions

Parameter (Source: Sanders and Wagner 2002)	OFA	STD
	Dimension (cm)	Dimension (cm)
Pellet outer diameter	0.7844	0.81915
Fuel rod outer diameter	0.9144	0.94996
Cladding thickness	0.0571	0.0571
Fuel rod pitch	1.2598	1.2598
Active fuel height	365.76	365.76
Guide/instrument tube outer diameter	1.204	1.204
Guide/instrument tube thickness	0.0407	0.0407
Fuel density (g/cm^3) (96% of theoretical density)	10.5216	10.5216
Number of fuel rods	264	264
Number of guide/instrument tubes	25	25

Table 8-2. Combustion Engineering 14 × 14 dimensions

Parameter (Source: Sanders and Wagner 2002)	Dimension (cm)
Pellet outer diameter	0.95361
Fuel rod outer diameter	1.1176
Cladding thickness	0.07112
Fuel rod pitch	1.34112
Active fuel height	347.98
Guide/instrument tube outer diameter	2.4079
Guide/instrument tube thickness	0.5080
Fuel density (g/cm^3) (96% of theoretical density)	10.5216
Number of fuel rods	176
Number of guide tubes	5

8.3 Reactivity Analysis of TSC-24 Canisters (Maine Yankee)

The TSC-24 canister is designed for storage and transportation of up to 24 PWR fuel assemblies. A cross section of the TSC-24 model is shown in Figure 8-3. Dimensions of the model are provided in Tables 8-3 and 8-4. For reactivity control, the fuel basket of the TSC-24 has a flux trap configuration. Flux traps are regions of water between neutron absorber panels, where neutrons that pass through one panel are thermalized and cannot readily return through either panel to interact with fuel.

Table 8-3. TSC-24 basket dimensions

Parameter (Source: Radulescu et al. 2012)	Dimension (cm)	Dimension (in.)
Tube wall thickness	0.121412	0.0478
Basket height	384.302	151.3
Cell inner dimension	22.352	8.8
Heat transfer disk (aluminum)	1.27 (thickness)	0.5
Support disk (steel)	1.27 (thickness)	0.5

Table 8-4. TSC-24 neutron absorber panel dimensions

Parameter (Source: Radulescu et al. 2012)	Dimension (cm)	Dimension (in.)
Panel width	20.9042	8.23
Boral core thickness	0.126992	0.050
Boral cladding thickness	0.31754	0.01250
Panel length	382.27	150.54
Wrapper thickness	0.045466	0.0179
Neutron absorber areal density (g $^{10}B/cm^{2)}$	0.025	0.025

8.3.1 Reactivity Analysis of MPC-32 Canisters (Sequoyah)

The MPC-32 canister is designed for storage and transportation of up to 32 PWR fuel assemblies. A cross section of the MPC-32 model is shown in Figure 8-4. Dimensions of the cask model are provided in Tables 8-5 and 8-6. For reactivity control, the internal basket of the MPC-32 uses a single neutron absorber panel between adjacent assemblies. This reduces the spacing between assemblies and accommodates more fuel. Burnup credit is used in the safety analysis to demonstrate criticality control for transportation.

Table 8-5. MPC-32 basket dimensions

Parameter (Peterson et al. 2013)	Dimension (cm)	Dimension (in.)
Wall thickness	0.714375	0.28125
Basket height	448.31	176.5
Cell inner dimension	22.7076	8.94

Table 8-6. MPC-32 neutron absorber panel dimensions

Parameter (Peterson et al. 2013)	Dimension (cm)	Dimension (in.)
Boral panel width	19.05	7.5
Boral core thickness	0.25654	0.101
Boral sheathing thickness	0.0254	0.01
Boral panel length	396.24	156
Wrapper thickness	0.1905	0.075
Neutron absorber areal density (g $^{10}B/cm^2$)	0.0372	0.0372

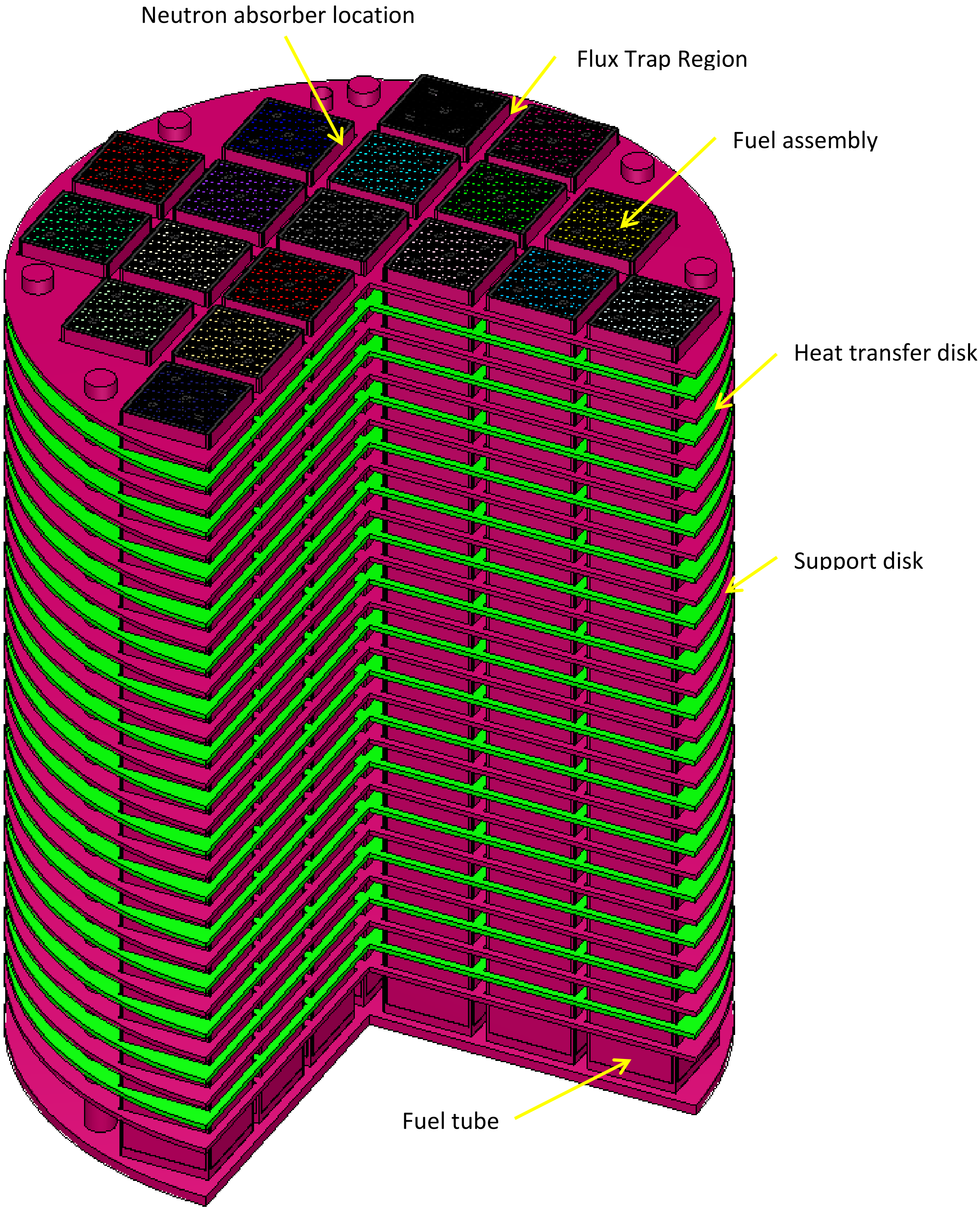


Figure 8-3. Cut-away view of the TSC-24 design basis KENO model

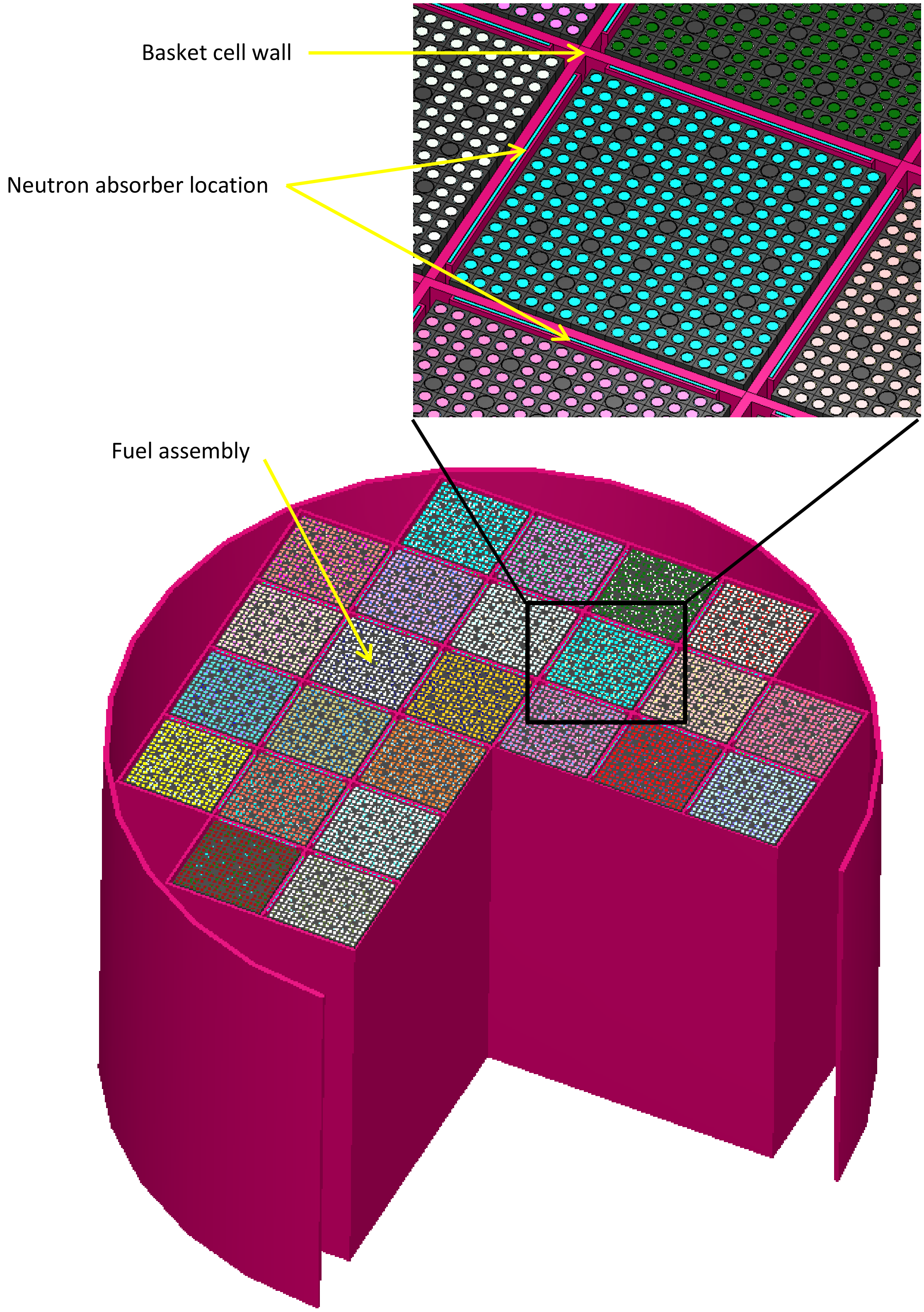


Figure 8-4.Cut-away view of the MPC-32 design basis KENO model

8.4 Burnup Credit

Assembly-specific burnup has been factored into each of the canister evaluations presented. The spent nuclear fuel (SNF) depletion and decay isotopic compositions were recently added to a library of fuel data (Peterson et al. 2013). This library and associated calculation tools (SNF-ST&DARDS) analyze each unique assembly design (e.g. Westinghouse 17×17 OFA or STD); and the initial enrichment, burnup, and decay time of each assembly; and generate explicit reactivity models for each canister using the specific loading pattern. For the reactivity calculations described here, burnup credit isotopic compositions are consistent with the actinide and fission product isotope data of NUREG/CR-7108 and NUREG/CR-7109 (Radulescu et al. 2012; Scaglione et al. 2012). The list of nuclides important to burnup credit evaluations is presented in Table 8-7.

Table 8-7. SNF isotope set: actinides plus 16 fission products

Actinides					
^{234}U	^{235}U	^{236}U	^{238}U	^{238}Pu	^{239}Pu
^{240}Pu	^{241}Pu	^{242}Pu	^{241}Am	^{243}Am	^{237}Np
Fission products					
^{95}Mo	^{99}Tc	^{101}Ru	^{103}Rh	^{109}Ag	^{133}Cs
^{143}Nd	^{145}Nd	^{147}Sm	^{149}Sm	^{150}Sm	^{151}Sm
^{152}Sm	^{151}Eu	^{153}Eu	^{155}Gd		

8.5 Disposal Configurations

As discussed previously, two configurations are evaluated for each DPC design type: the loss-of-absorber and degraded basket configurations.

8.5.1 Loss-of-Absorber Calculations

The loss-of-absorber configuration is represented by removal of the neutron absorber panels and replacement by moderator (flooding liquid). This configuration is hypothesized to result from a situation where the corrosion resistance of the fuel assemblies and of the stainless steel basket structural materials is greater than that of aluminum and Boral.

8.5.2 Degraded Basket Calculations

The degraded basket configuration is conceptually an extension of the loss-of-absorber configuration, continued until the stainless steel basket structure has completely lost its structural integrity. To represent loss-of-absorber and loss-of-configuration control, the assembly-to-assembly spacing is reduced uniformly and the assemblies are arranged in a cylindrical configuration. Corrosion products from the basket materials are completely displaced from the fuel assemblies, conceptually representing either accumulation at the bottom of the DPC or flushing from the system. Although not bounding per se, this configuration is conservative from a criticality perspective and provides an estimate of the amount of additional margin that could be needed to demonstrate subcriticality.

An additional set of calculations was performed to evaluate the impact from flooding with NaCl brine instead of fresh water. Natural chloride is about 75% ^{35}Cl, which has a moderate thermal neutron capture cross-section of ~44 barns. Chloride is a component of virtually any groundwater composition, and is especially abundant in salt formations and deep basement rock. In the analysis, the Na^+ counter-ion is used as a surrogate for other constituents that may be present that serve to reduce the concentration of H_2O molecules.

8.6 Results

Results presented in this section do not account for computational bias and uncertainty. A reasonable estimate for both would be approximately +2% (Δk_{eff}) resulting in a subcritical limit of $k_{eff} < 0.98$. Whereas reactivity increases gradually from approximately 100 years to 10,000 years and beyond due to radioactive decay, the reactivity at 10,000 years is roughly +0.003 to +0.005 (Δk_{eff}) greater than the 8000-year case. Peak reactivity occurs at roughly 20,000 to 30,000 years and is +0.02 to +0.035 (Δk_{eff}) greater.

8.6.1 Reactivity Calculation Results for TSC-24 Canisters (Maine Yankee)

When actinide and fission product burnup credit is used in conjunction with canister-specific loading, calculated k_{eff} values range from 0.61 to 0.81 for the 60 casks stored at Maine Yankee when flooded with pure water. Loss of absorber and degradation of the basket cause significant increase in reactivity. The results from a representative canister (TSC-5) are shown in Figure 8-5. The absorber loss and basket degradation cases produced reactivity increases on the order of +0.17 and +0.38 (Δk_{eff}), respectively. With a 1 molal solution of NaCl reactivity decreased -0.08 (Δk_{eff}).

Most of the Maine Yankee TSC-24 canisters (48 of the 60) were evaluated for the loss-of-absorber case and all are conditionally subcritical ($k_{eff} < 1.0$) provided burnup credit and canister-specific loading are incorporated (Figure 8-6).

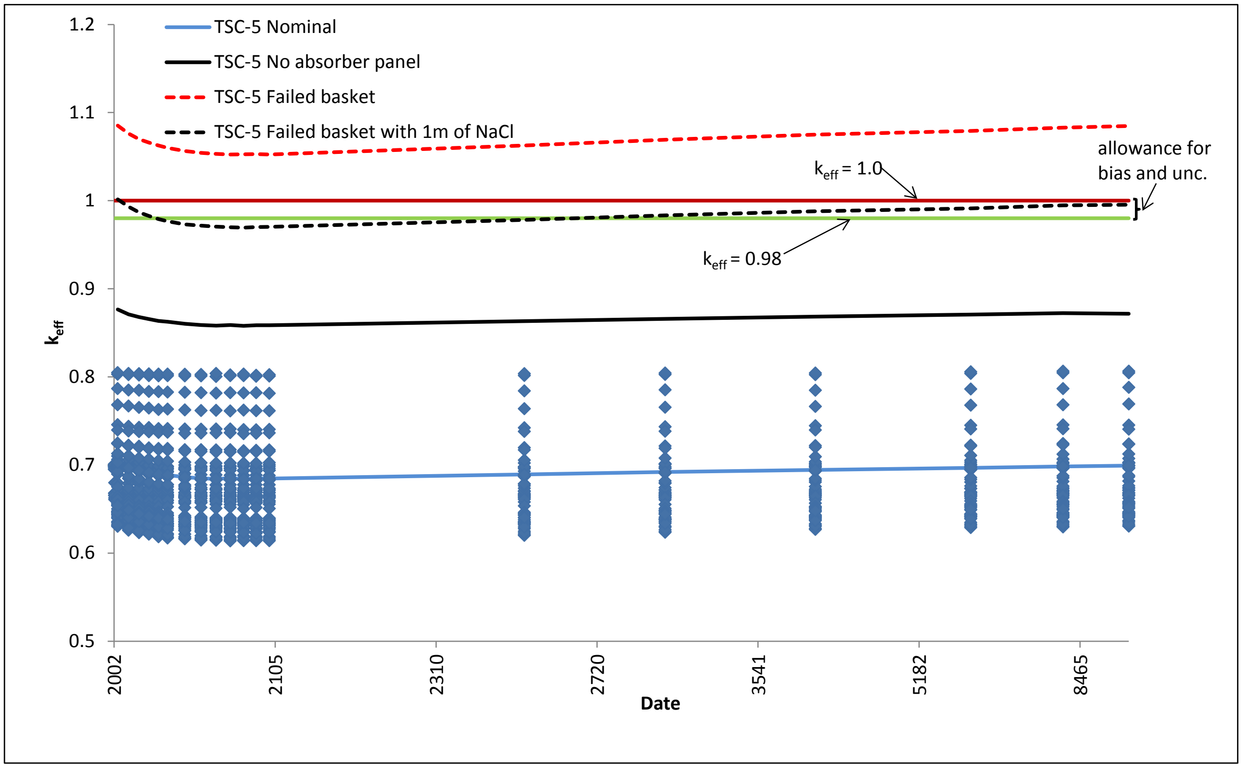


Figure 8-5. TSC-24 canister assessment of k_{eff} based on actual loadings at Maine Yankee

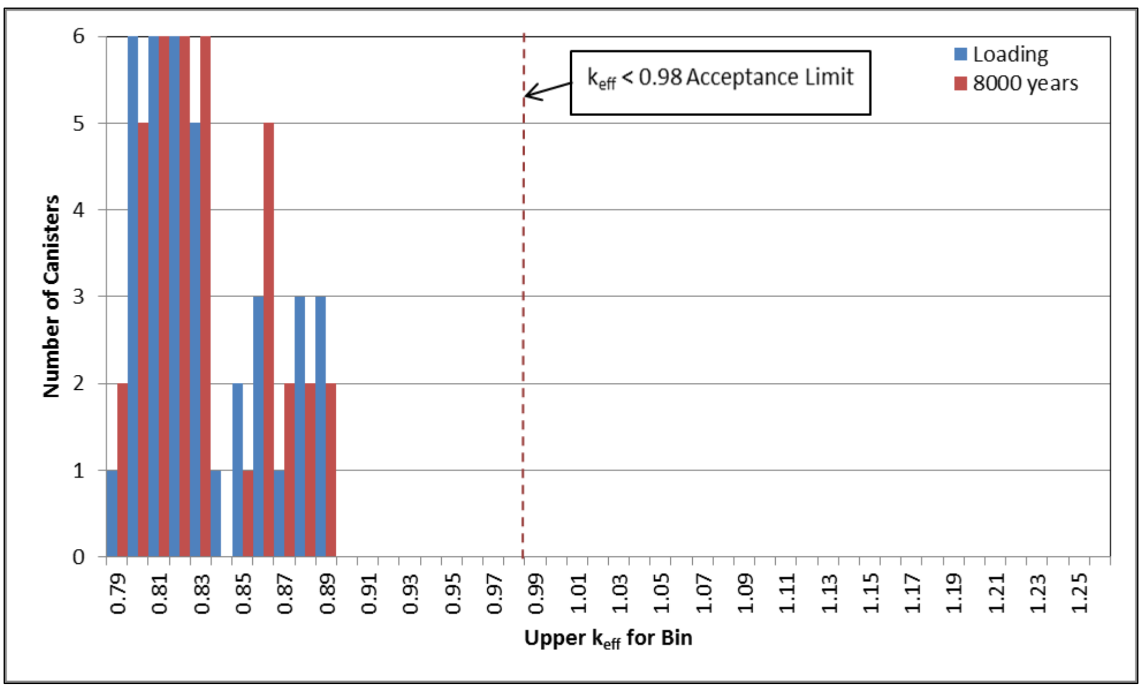


Figure 8-6. Distribution of k_{eff} for the no absorber case for Maine Yankee (fresh water)

For the degraded basket case 31 specific canisters were evaluated resulting in significant reactivity increases, greater than the decrease available from burnup credit and canister-specific loading. As shown in Figure 8-7 none of the canisters are subcritical for the degraded basket condition for either decay time. With the 1 molal NaCl brine, roughly half of the canisters analyzed are subcritical (Figure 8-8). With the brine concentration increased to 2 molal, most of the canisters analyzed are subcritical (Figure 8-9).

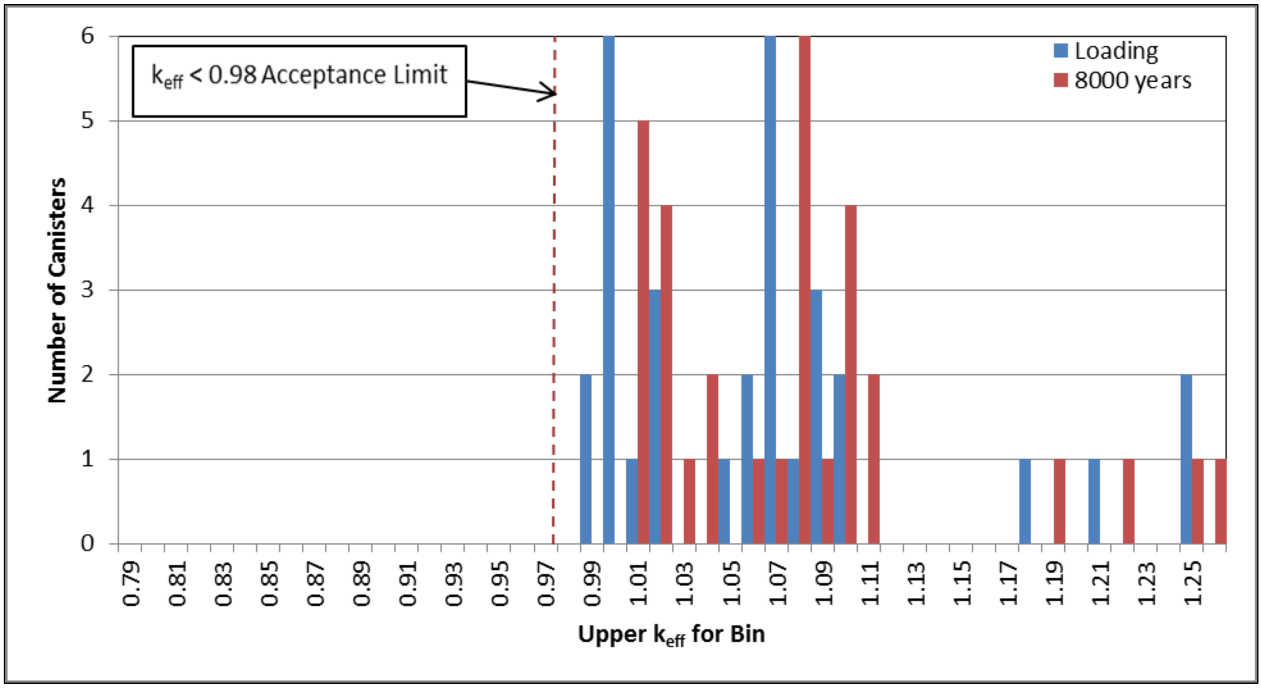


Figure 8-7. Distribution of k_{eff} for the degraded basket case for Maine Yankee (fresh water)

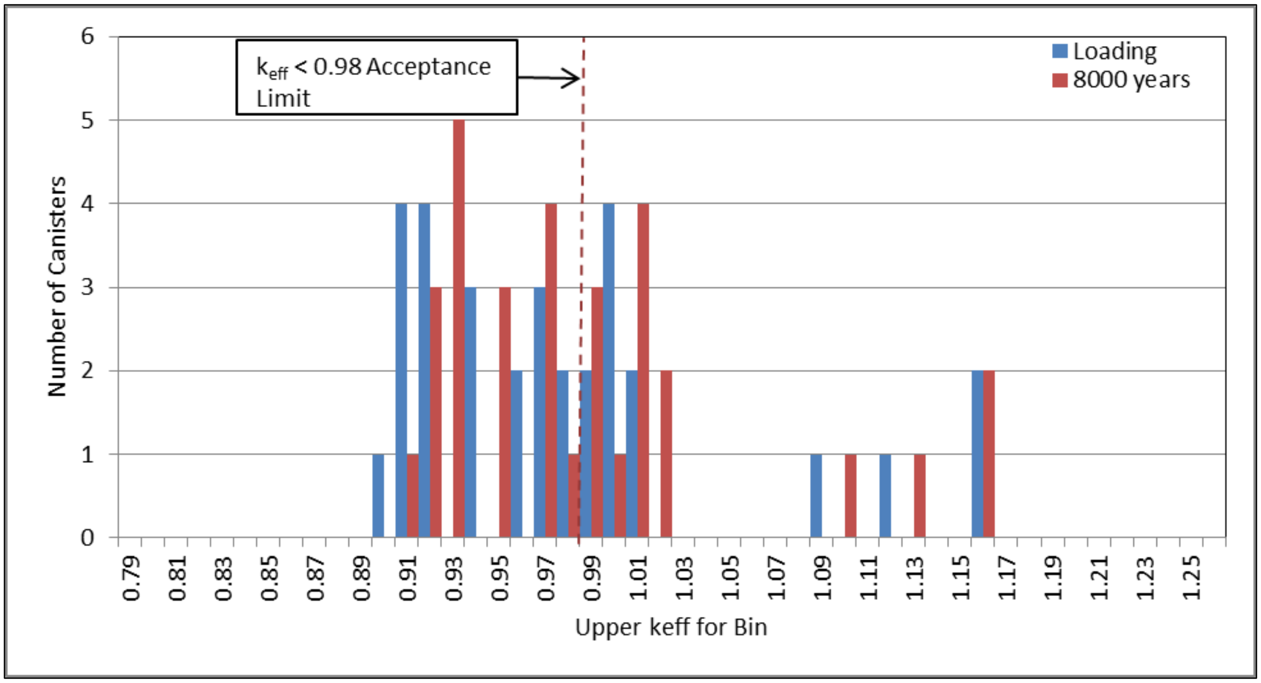


Figure 8-8. Distribution of k_{eff} for the degraded basket case for Maine Yankee canisters flooded with 1 molal NaCl

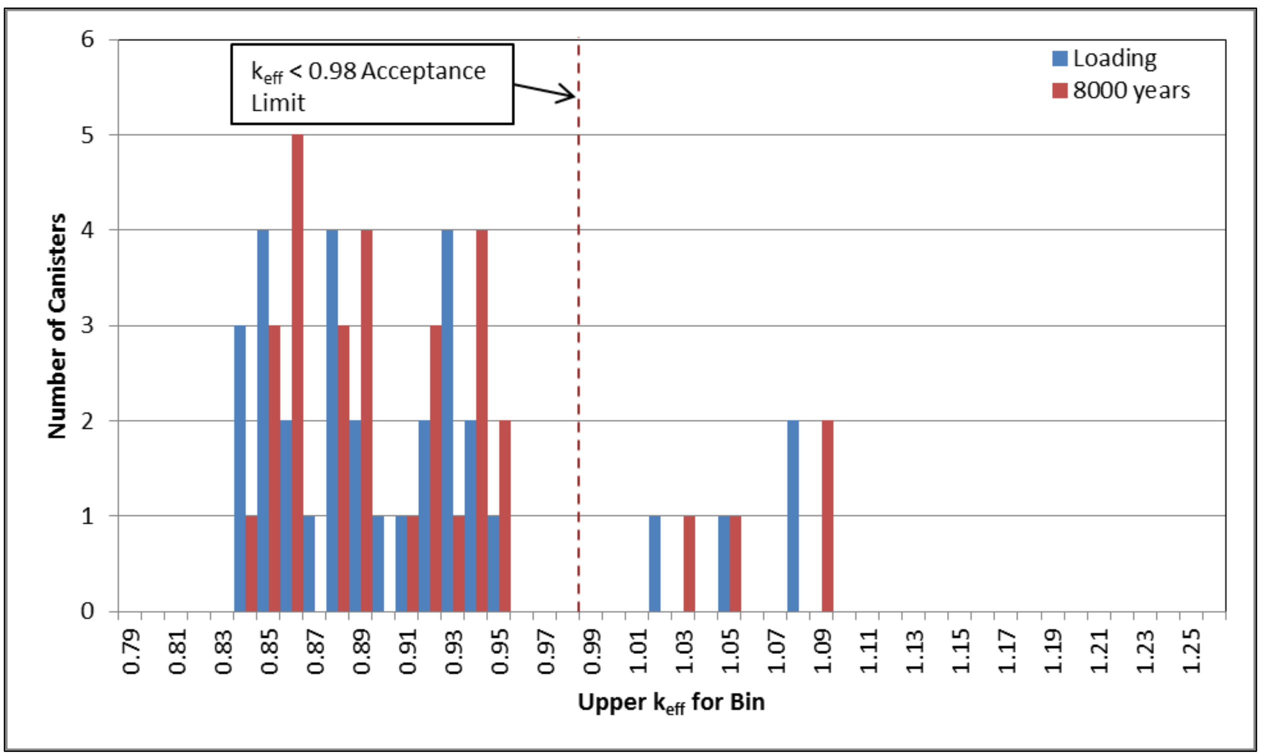


Figure 8-9. Distribution of k_{eff} for the degraded basket case for Maine Yankee canisters flooded with 2 molal NaCl

8.6.2 Reactivity Calculation Results for MPC-32 Canisters (Sequoyah)

When actinide and fission product burnup credit is used in conjunction with canister-specific loading, calculated k_{eff} values range from 0.80 to 0.88, when flooded with pure water, for the 26 canisters stored at the Sequoyah facility that were analyzed. The results from a representative canister (MPC-068) are shown in Figure 8-10. The loss-of-absorber and basket degradation cases produced reactivity increases on the order of +0.12 and +0.20 (Δk_{eff}), respectively. With a 1 molal solution of NaCl reactivity decreased -0.07 (Δk_{eff}).

The histogram for canisters analyzed (Figure 8-11) shows that about a third are subcritical when burnup credit and canister-specific loading are considered. Note that discharged burnable poison rod assemblies (BPRAs) are present in these 26 canisters but were not represented in these models. Discharged BPRAs could be credited for moderator displacement, and typically result in -0.02 to -0.03 (Δk_{eff}) reactivity decrease when modeled. Crediting the moderator displacement effect would make all 26 canisters conditionally subcritical with no absorber credit, provided that burnup credit and canister-specific loading are considered.

For the degraded basket case 26 specific canisters were evaluated resulting in significant reactivity increase that is greater than the decrease available from burnup credit and canister-specific loading. As shown in Figure 8-12 none of the canisters are subcritical for the degraded basket condition. With the 1 molal NaCl brine, only a few of the Sequoyah canisters analyzed are subcritical (Figure 8-13). With the brine concentration increased to 2 molal, all of the canisters analyzed are subcritical (Figure 8-14).

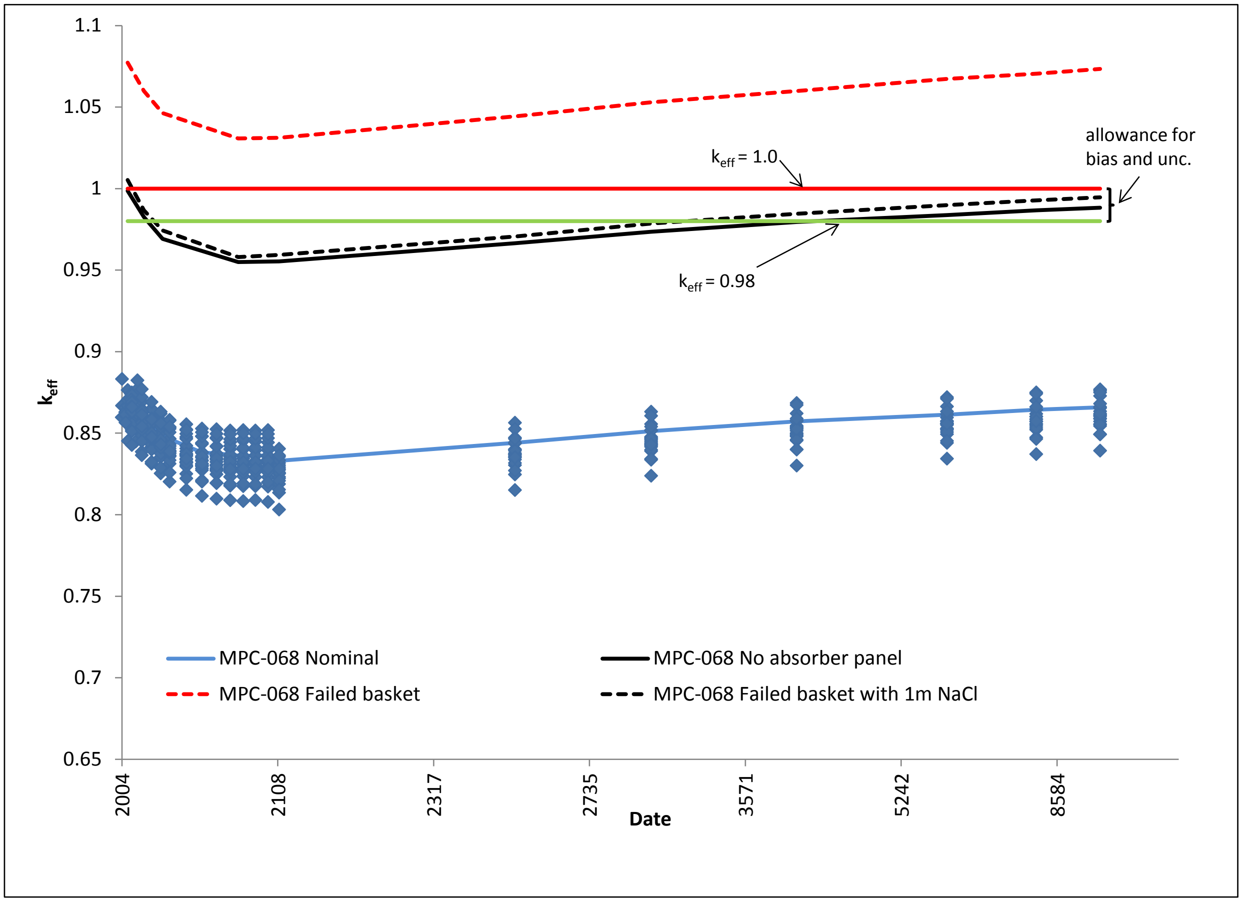


Figure 8-10. Reactivity vs. time calculations for representative canister MPC-068 (as loaded) at Sequoyah

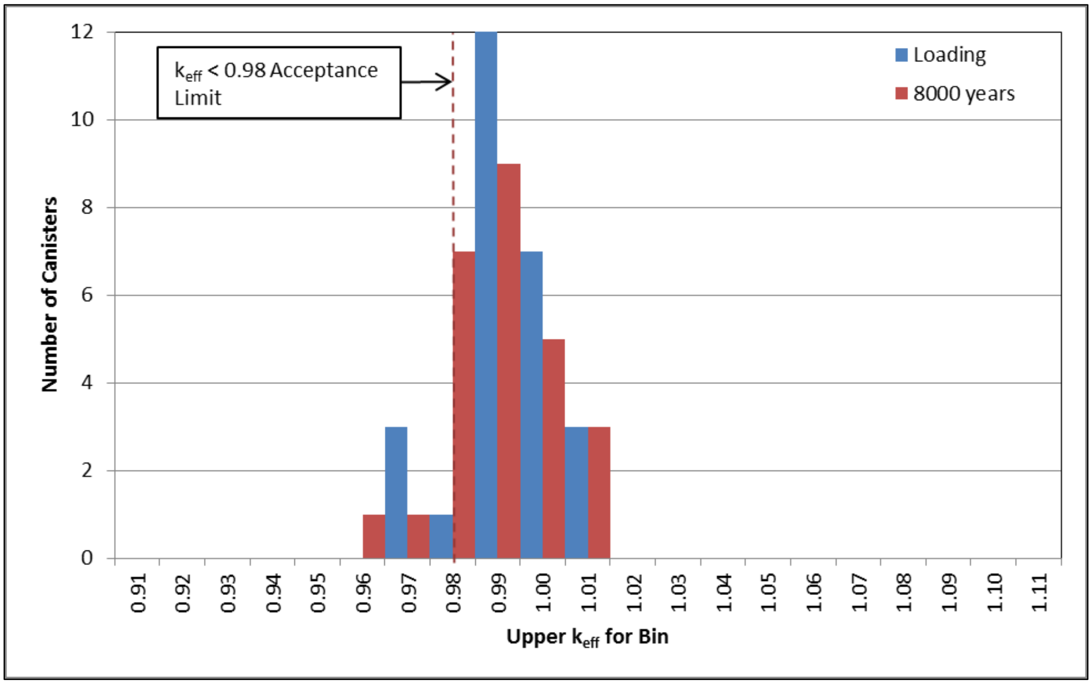


Figure 8-11. Distribution of k_{eff} for the no absorber case for Sequoyah (fresh water)

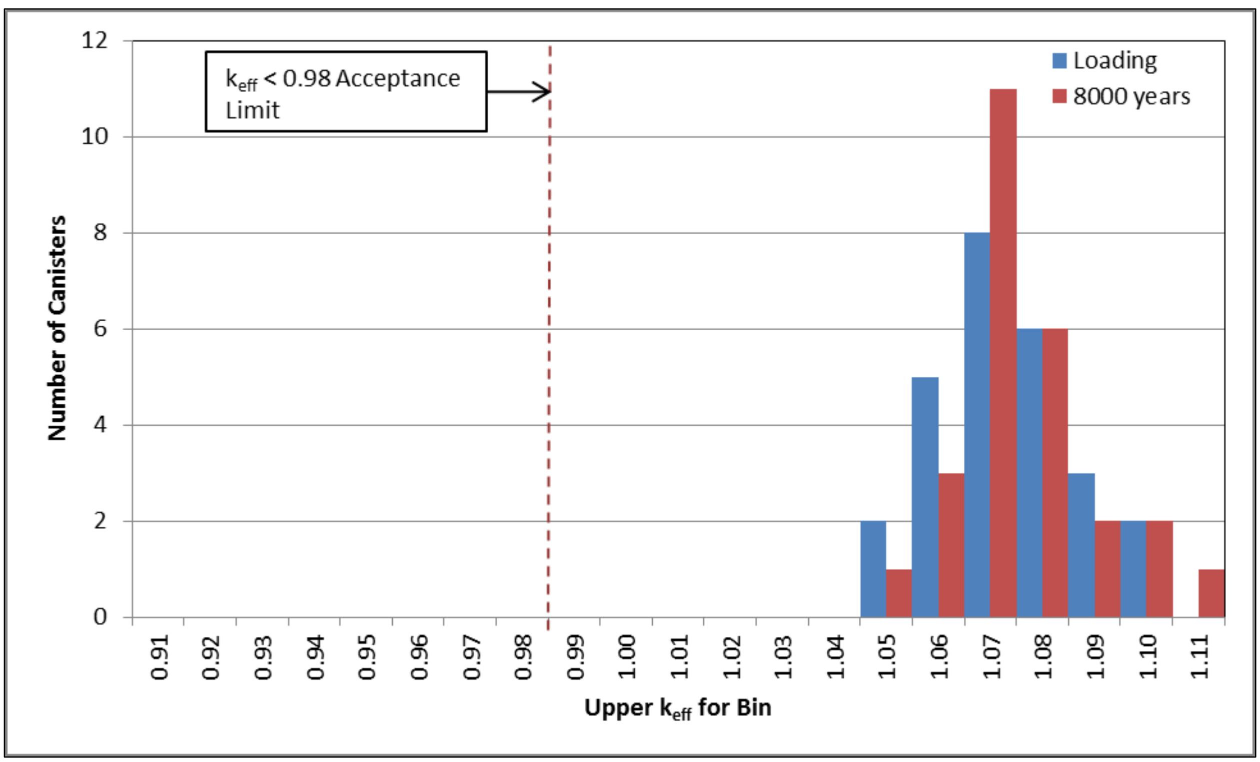


Figure 8-12. Distribution of k_{eff} for the degraded basket case for Sequoyah (fresh water)

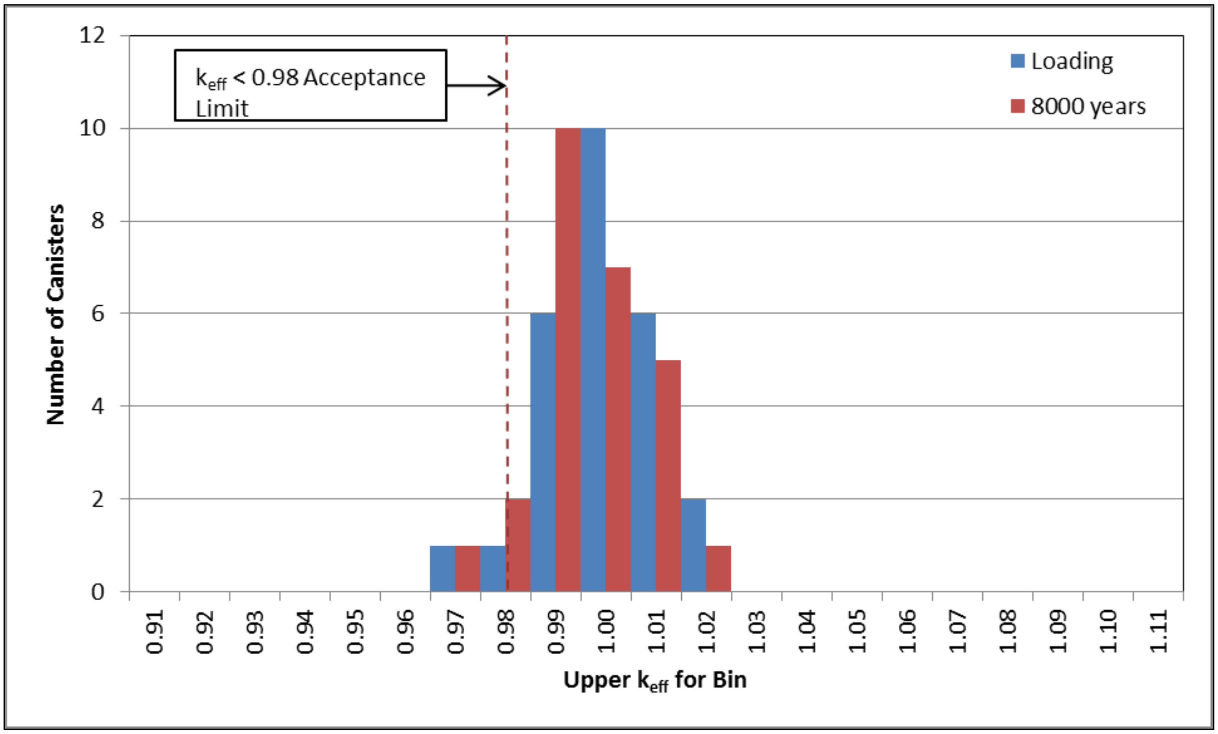


Figure 8-13. Distribution of k_{eff} for the degraded basket case for Sequoyah canisters flooded with 1 molal NaCl

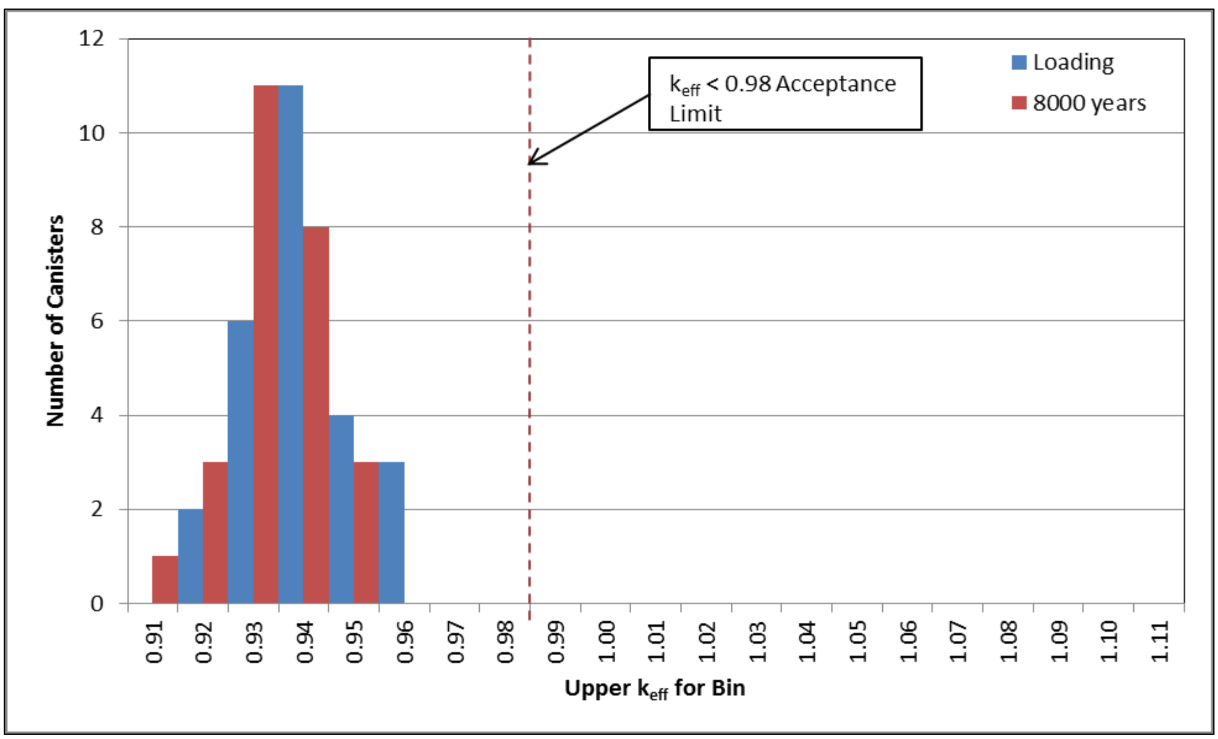


Figure 8-14. Distribution of k_{eff} for the degraded basket case for Sequoyah canisters flooded with 2 molal NaCl

8.7 Summary of Results

This scoping analysis assumes that waste package breach at a relatively early time in the repository system evolution allows flooding (or partial flooding) of the canister sufficient to produce significant moderation and degrade the internal components. Note that if water can be excluded or significantly delayed from the repository or from entering a package, there is little potential for criticality. Reactivity analyses were performed for two cases of canister degradation: loss-of-absorber and basket degradation (including loss-of-absorber), and two types of DPCs. For the more conservative basket degradation cases, subcriticality could be demonstrated for some but not all DPCs (results summarized in Table 8-8). Flooding with NaCl brine instead of pure water, demonstrated subcriticality for nearly all DPCs (1 molal and 2 molal NaCl brine solutions).

The sources of reactivity margin (relative to licensing and loading analyses) investigated in this report include:

- Burnup credit
- Use of actual as-loaded DPCs
- Radionuclide inventory decay
- Credit for composition of flooding liquid (chloride brine)

The results indicate that criticality screening analyses that take into account these attributes and the specific disposal environment, are feasible for many DPCs. This outcome depends on site-specific characteristics of the disposal environment, and the availability of detailed fuel inventory information (burnup, reactor performance data). The chemistry of moderator solutions is shown to be a particularly important attribute for some disposal environments.

Table 8-8. Summary of detailed analysis for DPCs at the Maine Yankee and Sequoyah sites

Case	Cooling time	Number of DPCs subcritical	Percentage of DPCs subcritical
Maine Yankee (48 no-absorber and 31 degraded-basket canister cases analyzed)			
No absorber	0	48	100%
	8000	48	100%
Degraded basket	0	0	0%
	8000	0	0%
Degraded basket with 1 molal NaCl	0	19	61%
	8000	17	55%
Degraded basket with 2 molal NaCl	0	27	87%
	8000	27	87%
Sequoyah (26 canisters analyzed)			
No absorber	0	4	15%
	8000	9	35%
Degraded basket	0	0	0%
	8000	0	0%
Degraded basket with 1 molal NaCl	0	2	8%
	8000	3	12%
Degraded basket with 2 molal NaCl	0	26	100%
	8000	26	100%

Significant improvement in the realism of the degradation cases is possible by reassessing the performance of basket structural materials (primarily stainless steel). A full performance assessment would include the probability and time of waste package and internal component failure as well as the probability of flooding or partial flooding. This information can substantially influence conclusions about the likelihood of one or more criticality events. Additional work is warranted to better understand the material degradation processes and rates, the potential impacts on fuel geometry, and the associated probabilities for use in FEP screening.

Further, the possibility of one or more criticality events is not by itself a definitive indication that direct disposal of DPCs is infeasible for a given geologic setting and disposal concept (see Section 2). Stylized analyses in the past have shown that the consequences of criticality, if properly weighted by the probability, may not have a an impact on repository performance that would exceed safety standards (Mohanty et al. 2004). The consequences of criticality, in the context of overall repository system safety performance, also warrant further analysis (see Section 10).

References for Section 8

BSC (Bechtel-SAIC Co.) 2003. *The Potential of Using Commercial Dual-Purpose Canisters for Direct Disposal*. TDR-CRW-SE-000030 REV 00. Office of Civilian Radioactive Waste Management. Las Vegas, NV.

Clarity, J.B. and J.M Scaglione 2013. *Feasibility of Direct Disposal of Dual-Purpose Canisters-Criticality Evaluations.* ORNL/LTR-2013/213. Oak Ridge National Laboratory, Oak Ridge, TN. June, 2013.

DOE (U.S. Department of Energy) 1998. *Disposal Criticality Analysis Methodology Topical Report*. YMP/TR-004Q Rev. 0. Office of Civilian Radioactive Waste Management. November, 1998.

EPRI (Electric Power Research Institute) 2008. *Feasibility of Direct Disposal of Dual-Purpose Canisters: Options for Assuring Criticality Control*. 1016629, Electric Power Research Institute, Palo Alto, CA.

EPRI (Electric Power Research Institute) 2009. *Handbook of Neutron Absorber Materials for Spent Nuclear Fuel Transportation and Storage Applications: 2009 Edition.* 1019110. Palo Alto, CA.

Mohanty, S , R. Codell, J. Menchaca, R. Janezka, M. Smith, P. LaPlante, M. Rahimi and A. Lozano 2004. “System-Level Performance Assessment of the Proposed Repository at Yucca Mountain Using the TPA Version 4.1 Code.” Center for Nuclear Waste Regulatory Analyses, San Antonio Texas. Revison 2. March, 2004.

NRC (U.S. Nuclear Regulatory Commission) 2000. “Safety Evaluation Report for Disposal Criticality Analysis Methodology Topical Report, Revision 0.” Letter from C.W. Reamer (NRC) to S.J. Brocoum (DOE/YMSCO), June 26, 2000, with enclosure.

NRC (U.S. Nuclear Regulatory Commission) 2011. *Technical Evaluation Report on the Content of the U.S. Department of Energy’s Yucca Mountain Repository License Application, Postclosure Volume: Repository Safety After Permanent Closure*. Office of Nuclear Material Safety and Safeguards. July, 2011.

ORNL (Oak Ridge National Laboratory) 2011. *Scale: A Comprehensive Modeling and Simulation Suite for Nuclear Safety Analysis and Design*. ORNL/TM-2005/39 Version 6.1. Available from Radiation Safety Information Computational Center at Oak Ridge National Laboratory as CCC-785. June, 2011.

Peterson, J. et al. 2013. *Used Nuclear Fuel Storage, Transportation, & Disposal Analysis Resource and Data System (SNF-ST&DARDS).* FCRD-NFST-2013-000117 Rev. 0. March, 2013. U.S. Department of Energy, Used Fuel Disposition R&D Campaign.

Radulescu, G., I.C. Gauld, G. Ilas and J.C. Wagner 2012. *An Approach for Validating Actinide and Fission Product Burnup Credit Criticality Safety Analyses – Isotopic Composition Predictions*. NUREG/CR-7108 (ORNL/TM-2011/514), prepared for the US Nuclear Regulatory Commission by Oak Ridge National Laboratory. Oak Ridge, TN. April, 2012.

Sanders, C.E. and J. C. Wagner 2002. *Study of the Effects of Integral Burnable Absorbers for PWR Burnup Credit*. NUREG/CR-6760 (ORNL/TM-2000/321), prepared for the U.S. Nuclear Regulatory Commission by Oak Ridge National Laboratory. Oak Ridge, TN. March, 2002.

Scaglione, J.M., D.E. Mueller, J.C. Wagner and W.J. Marshall 2012. *An Approach for Validating Actinide and Fission Product Burnup Credit Criticality Safety Analyses – Criticality (k_{eff}) Predictions*. NUREG/CR-7109 (ORNL/TM-2011/514). Prepared for the U.S. Nuclear Regulatory Commission by Oak Ridge National Laboratory. Oak Ridge, TN. April, 2012.

THIS PAGE INTENTIONALLY LEFT BLANK

9. Preliminary System Analysis of DPC Direct Disposal

A preliminary system analysis has been completed that assesses some of the potential impacts on cost and schedule, from directly disposing large dual-purpose canisters (DPCs) in a geologic repository (Nutt 2013). This analysis:

- Forecasts when DPCs loaded with spent nuclear fuel (SNF) from the existing fleet of nuclear power plants could be emplaced in a geologic repository, for five different emplacement thermal power limits (4, 6, 8, 10, and 12 kW/canister).
- Projects repository DPC acceptance rates at the repository for each emplacement thermal power limit considered.
- Estimates the incremental costs that would be required to store DPCs at an interim storage facility (ISF) for cooling to the emplacement thermal limit.
- Compares these incremental costs with estimates of the cost that would be required to re-package the SNF into smaller canisters for disposal.

9.1 Approach

The CALVIN portion of the Transportation Storage Logistics simulator (TSL-CALVIN) (Nutt et al. 2012a) was used to determine when DPCs loaded at the existing fleet of nuclear power plants would be sufficiently cool to meet emplacement thermal power limits. TSL-CALVIN was described and used previously to perform the SNF logistic analysis in *Used Fuel Management System Architecture Evaluation, Fiscal Year 2012* (Nutt et al. 2012b). For the present analysis, TSL-CALVIN was modified to constrain SNF shipments from storage facilities (at power plant sites or an ISF) to the repository, to occur when the DPCs cool to a repository emplacement thermal power limit. Disposable canisters of any type are not shipped to the repository until sufficiently cooled, when this feature is used.

The major assumptions used in this system analysis are summarized as follows:

- SNF will be generated at all currently operating power plants, with 20-year life extensions, and gradual increases in burnup (approaching what can be achieved with 5% enrichment; Carter et al. 2012). As power plants are decommissioned all SNF will be put into dry storage in current and projected DPCs. Discharges from future nuclear builds is not considered.
- The projected used fuel inventory from the existing fleet will be designated SNF, and the SNF will be transported off the power plant sites in large welded dry-storage canisters or bolted dry-storage casks.
- All dry storage canisters in use at power plant sites are transportable (whether or not they are currently certified for transport).
- An ISF will serve as the principal surface decay storage facility for DPCs (in addition to at-reactor storage prior to shipment). Shipment of DPCs from power plant sites to an ISF will begin in 2025, at a rate of either 3,000 or 4,500 MTHM per year (ISF receipt rate).
- A repository will open and begin to package and emplace DPCs underground in 2048. The inventory of SNF in the ISF at this time will be 69,000 MTHM or 103,500 MTHM, depending on the assumed ISF receipt rate (3,000 or 4,500 MTHM per year).

- Once the repository is operating, DPCs cool enough for disposal will be shipped directly from power plant sites, or from the ISF if none are available at the reactor sites.
- All dry storage canisters are disposable, once sufficiently cooled and loaded into disposal overpacks.
- Shipments to the ISF will continue after 2048, as needed to transfer fuel from decommissioned plants (subject to ISF receipt rate limits).
- Once the repository opens, power plant sites continue to load DPCs that are the same size and thermal capacity just as they did prior to repository operations.

This last assumption is important because it may drive the total duration of used fuel management system operations (i.e., operation of the independent spent fuel storage installation pads, the ISF, and a repository) for longer than otherwise necessary. For example, if, a repository is sited by the mid to late 2020's then designed and licensed by the mid 2040's, the maximum allowable waste package thermal output will be known by the early 2040's or sooner. Standard canisters designed in accordance with known repository requirements could be deployed for the remaining bare fuel inventory at that time instead of continuing the use of existing DPCs. Although these purpose-designed canisters could be smaller than existing DPCs, the change would facilitate more flexibility in the waste management system that is not reflected in the assumptions described in Section 2. More detailed logistical analyses that include both direct disposal of existing DPCs and future use of standardized canisters will be considered in future DPC feasibility study work (Section 10).

Additional details regarding at-reactor dry storage trends and TSL-CALVIN modeling assumptions are provided in *Used Fuel Management System Architecture Evaluation, Fiscal Year 2012* (Nutt et al. 2012b, Appendix A). For the purpose of this analysis all welded dry storage canisters and bolted dry storage casks are referred to as DPCs.

The system analyses also represent repository throughput, in order to evaluate the duration and cost of repository operations needed to dispose of all DPCs. The assumption that a repository will begin operation in 2048 is consistent with the *Strategy for the Management and Disposal of Used Nuclear Fuel and High-Level Radioactive Waste* (DOE 2013). From 2048 on a arbitrarily large MTHM per year repository acceptance rate is used in TSL-CALVIN to allow direct shipments from the power plant sites to the repository. This causes the entire inventory of SNF in storage at power plant sites to be considered for shipment to the repository. TSL-CALVIN then calculates the heat output of each DPC and compares it with the emplacement thermal power limit, to control which shipments to the repository actually occur. In this way the tool can be used to determine the amount of SNF that becomes available each year for disposal.

This analysis assumes that an ISF will serve as the principal surface decay storage facility for DPCs. The assumption that an ISF will commence full-scale operation in 2025, is also consistent with the DOE strategy document (DOE 2013). Two ISF acceptance rates are assumed: 3,000 MTHM/yr and 4,500 MTHM/yr. These result in ISF inventories of 69,000 MTHM and 103,500 MTHM, respectively, when repository operation begins in 2048.

A range of maximum repository acceptance rates for SNF in DPCs is also considered. The forecast repository acceptance rate for SNF in DPCs is then determined as:

$$SNF_{A,i} = \sum_{j=2048}^{i} SNF_{C,j} - \sum_{j=2048}^{i-1} AR_j$$

$$AR_i = MIN\left[AR_{max}, SNF_{A,i-1} + SNF_{C,i}\right] \qquad \text{Eq. 9-1}$$

where:

- AR = annual repository acceptance rate for SNF in DPCs (MTHM)
- AR_{max} = maximum annual repository acceptance rate for SNF in DPCs (3,000 and 4,500 MTHM)
- SNF_A = total amount of SNF in DPCs that are sufficiently cooled and available to be transported to the repository each year (MTHM)
- SNF_C = the amount of SNF in DPCs that become sufficiently cool in a given year (MTHM); from TSL-CALVIN
- i = year of repository acceptance (≥ 2048)

The forecast ISF inventory of SNF in DPCs is then determined as:

$$SI_i = MIN\left[SNF_T\ , \sum_{j=2025}^{i} AS_j\right] - \sum_{j=2048}^{i} AR_j \qquad \text{Eq. 9-2}$$

where:

- SI = inventory of SNF in the ISF (MTHM)
- AS = annual ISF acceptance rate for SNF in DPCs (3,000 and 4,500 MTHM)
- AR = annual repository acceptance rate for SNF in DPCs (MTHM)
- SNF_T = total amount of SNF that will be discharged by the reactor fleet (~140,000 MTHM)
- i = year of repository acceptance (≥ 2048)

As a baseline for comparing maximum ISF inventory (in MTHM of SNF) it is assumed that for a disposal scenario involving SNF re-packaging with the blending of hotter and cooler fuel assemblies the maximum ISF inventory would occur in 2048, when the repository begins operation, and that the repository and ISF acceptance rates will be equal. Specifically, it is assumed that:

- If re-packaging occurs at the ISF, then all SNF will have to be transported to the ISF, re-packaged, and transported away from the ISF at the same rate; or
- If re-packaging occurs at the repository, shipments to the ISF will cease and all shipments to the repository will be either from the reactor sites, the ISF, or both.

In either case the inventory at the ISF will remain constant at the 2048 value until insufficient fuel is available from the power plant sites to achieve the desired acceptance rates.

The additional SNF storage capacity required for DPC decay storage is then determined from the difference between the maximum forecast ISF inventory of SNF in DPCs (Equation 9-2) and the maximum ISF inventory for SNF when considering re-packaging/blending (69,000 MTHM or 103,500 MTHM).

As a baseline for comparisons of the time the ISF has to remain open, it is assumed that for a disposal scenario involving SNF re-packaging with the blending of hotter and cooler fuel assemblies, the ISF will remain operational over the entire period required to transfer the entire inventory of SNF (~140,000 MTHM) to the repository. It is further assumed that acceptance would be constant at the maximum acceptance rate. This yields ISF operational durations of 47 years past the start of repository acceptance for a 3,000 MTHM/yr acceptance rate and 31 years for an acceptance rate of 4,500 MTHM/yr.

Note that the maximum ISF inventory may be larger, and the ISF operational time may be longer for a disposal scenario involving SNF re-packaging with the blending of hotter and cooler fuel assemblies than determined above (69,000 MTHM or 103,500 MTHM). This is because the repository thermal constraints and the availability of sufficient SNF at the ISF could limit the ability to blend SNF, requiring additional ISF storage capacity and increased ISF decay storage time. However, the assumptions used in this analysis result in a larger incremental ISF capacity and a longer incremental ISF operating time when comparing scenarios involving the direct disposal of DPCs to those involving re-packaging/blending.

The incremental cost of extended decay storage is estimated by determining the capital cost required to deploy additional ISF storage capacity and the cost to continue ISF operations for an additional period of time to allow the DPCs to cool. The annual operating cost to operate the ISF is assumed to be $24 million per year (Nutt et al. 2012b, Table 6-4) for an analysis case in which the ISF only stores DPCs.

The capital cost to deploy additional storage capacity at the ISF is assumed to be $152,000 per MTHM. From the analysis case in which the ISF only stores DPCs (Nutt et al. 2012b, Table 6-5] the capital cost was $8.2 billion for a 3,000 MTHM acceptance rate with ISF and repository operations beginning in 2020 and 2040, respectively. The maximum ISF capacity for this scenario was approximately 60,000 MTHM with 6,000 MTHM built in the first two years of operation and the remaining 54,000 MTHM deployed incrementally ($8.2 billion ÷ 54,000 MTHM = $152,000 per MTHM).

9.2 Preliminary Results

This section forecasts when DPCs loaded with SNF will be cool enough for disposal, for five emplacement thermal power limits (4 kW/canister, 6 kW/canister, 8 kW/canister, 10 kW/canister, and 12 kW/canister). These limits span the range for various disposal concepts (Section 4). Thermal analysis of open disposal concepts (Hardin 2013) shows that emplacement power could be limited to a few kW in sedimentary host media, but could be 10 kW or greater for hard rock media (unbackfilled or high-temperature backfill). Similar analysis for the salt concept (Hardin et al. 2012) shows that emplacement power could be as high as 10 kW.

Projections of repository DPC acceptance rates for each emplacement thermal power limit considered are provided below. Incremental costs are estimated for DPC decay storage at an ISF, and compared with estimates of the cost to re-package and blend all SNF from DPCs into disposal canisters that meet repository thermal constraints.

9.2.1 DPC Availability and Acceptance

Forecasts for when DPCs (or packages the same size as DPCs) could be emplaced in a geologic repository for the five different emplacement thermal power limits are determined using TSL-CALVIN and are shown in Figures 9-1 through 9-5. These figures plot both the numbers of canisters (BWR and PWR) and the quantity of SNF (MTHM) that become sufficiently cool each year after repository operation begins in 2048. For comparison, the figures also include timelines representing the duration of emplacement operations assuming re-packaging, with constant repository acceptance rates of 3,000 MTHM/yr and 4,500 MTHM/yr.

Figures 9-1 through 9-5 show that for smaller emplacement power limits, the amount of SNF that becomes sufficiently cool each year is smaller. Correspondingly, the duration of repository operation (during which SNF would be accepted for disposal) is extended. The extended duration is driven by hotter DPCs loaded with PWR SNF in the 2040's and 2050's. Depending on the emplacement power limit, more than 200 years of decay storage could be required to dispose of the entire inventory of SNF from the existing nuclear fleet. As stated earlier (last assumption in Section 9.1) the long decay storage time to accommodate the entire inventory of SNF from the existing nuclear fleet could be mitigated by deploying purpose-built standard canisters when site specific repository waste package emplacement thermal limits are known.

The projected repository DPC acceptance rates for each emplacement power thermal limit and maximum acceptance rates of 3,000 MTHM/yr and 4,500 MTHM/yr are shown in Figure 9-6. The maximum annual acceptance rate can only be maintained for only a few years until the availability of sufficient cool SNF is depleted. The acceptance rate then depends exactly on the rate that DPCs cool sufficiently. This indicates that a repository for the direct disposal of DPCs could be designed for a lower acceptance rate (unless canisters that are purpose-built for disposal can be deployed after emplacement thermal limits are determined).

Figure 9-7 shows repository annual acceptance rates that are optimized to remain as constant for as long as possible while completing disposal when the last DPC is sufficiently cool to emplace. Higher emplacement power limits lead to higher acceptance rates and correspondingly shorter repository loading times.

Summary-level logistics information is provided in Table 9-1 for each combination of ISF and repository acceptance rate considered. Also provided are the maximum ISF and repository operation time for the scenario when re-packaging and blending of SNF occurs.

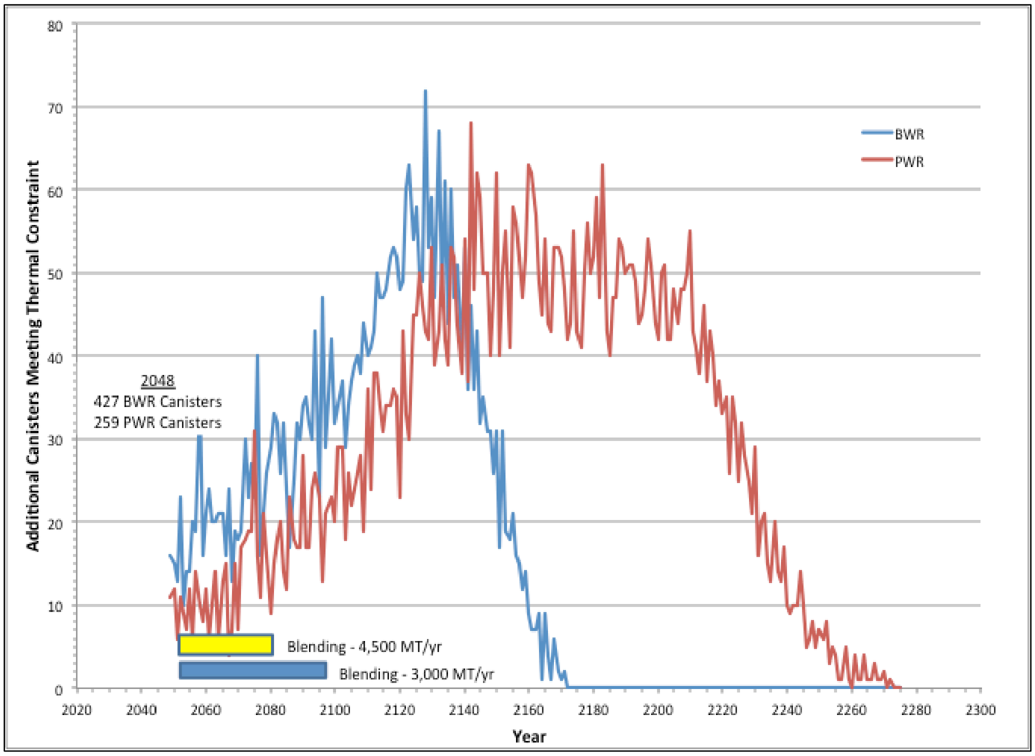
Table 9-1. Summary of system analysis results

a) 3000 MTHM/yr ISF Acceptance Rate

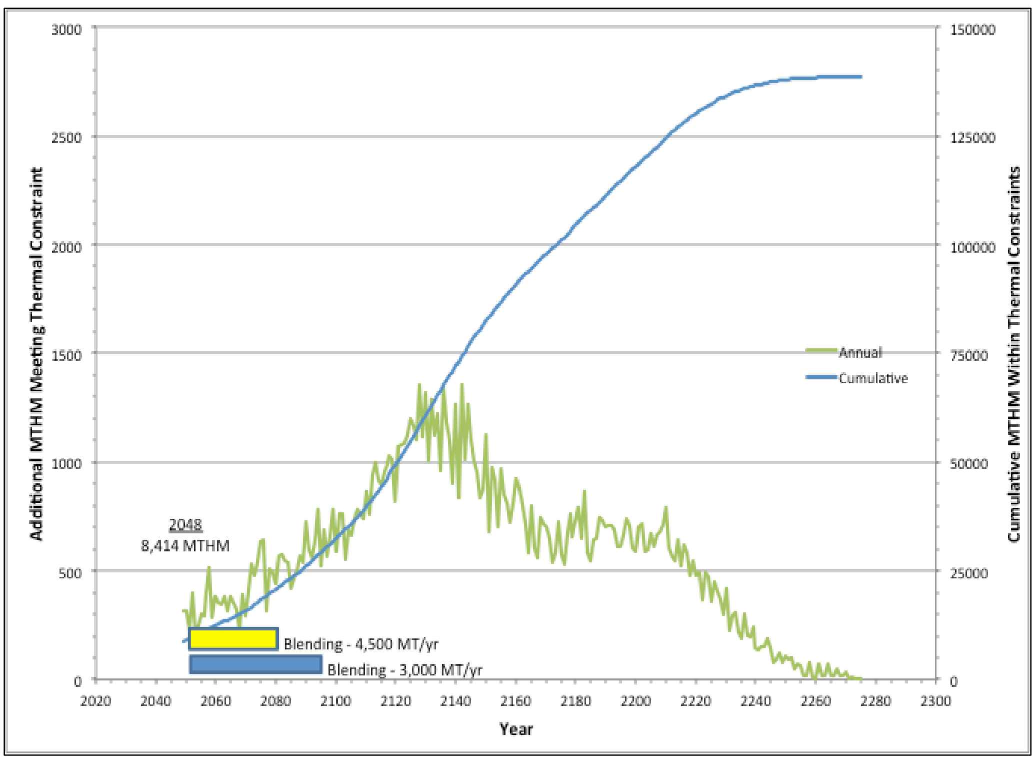
Repository Emplacement Thermal Limit (kW)	Repository Acceptance Rate (MTHM/yr)	Inventory in Storage at Start of Repository Operations (MTHM)	Repository Emplacement Time Required @ 3000 MTHM/yr with Re-Packaging/Blending (years)	Peak Inventory in Interim Storage (MTHM)	Additional Interim Storage Capacity Required (MTHM)	Repository Emplacement Time Required Due to Thermal Constraints (years)	Additional Repository Emplacement Time Required (years)
4	650	69000	47	123116	54116	224	177
6	1100			112700	43700	135	88
8	1400			105800	36800	102	55
10	1700			98900	29900	83	36
12	2000			92000	23000	70	70

b) 4500 MTHM/yr ISF Acceptance Rate

Repository Emplacement Thermal Limit (kW)	Repository Acceptance Rate (MTHM/yr)	Inventory in Storage at Start of Repository Operations (MTHM)	Repository Emplacement Time Required @ 3000 MTHM/yr with Re-Packaging/Blending (years)	Peak Inventory in Interim Storage (MTHM)	Additional Interim Storage Capacity Required (MTHM)	Repository Emplacement Time Required Due to Thermal Constraints (years)	Additional Repository Emplacement Time Required (years)
4	650	103500	31	128316	24816	224	193
6	1100			121135	17635	135	104
8	1400			116335	12835	102	71
10	1700			111535	8035	83	52
12	2000			106735	3235	70	39

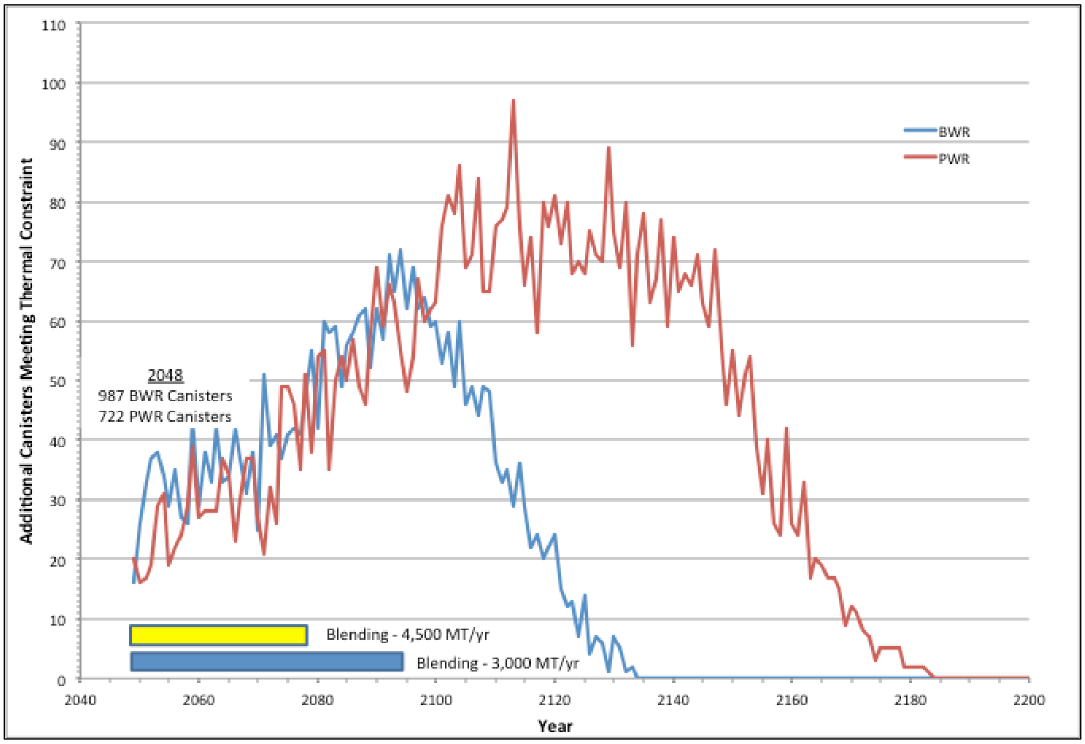


a) Number of canisters

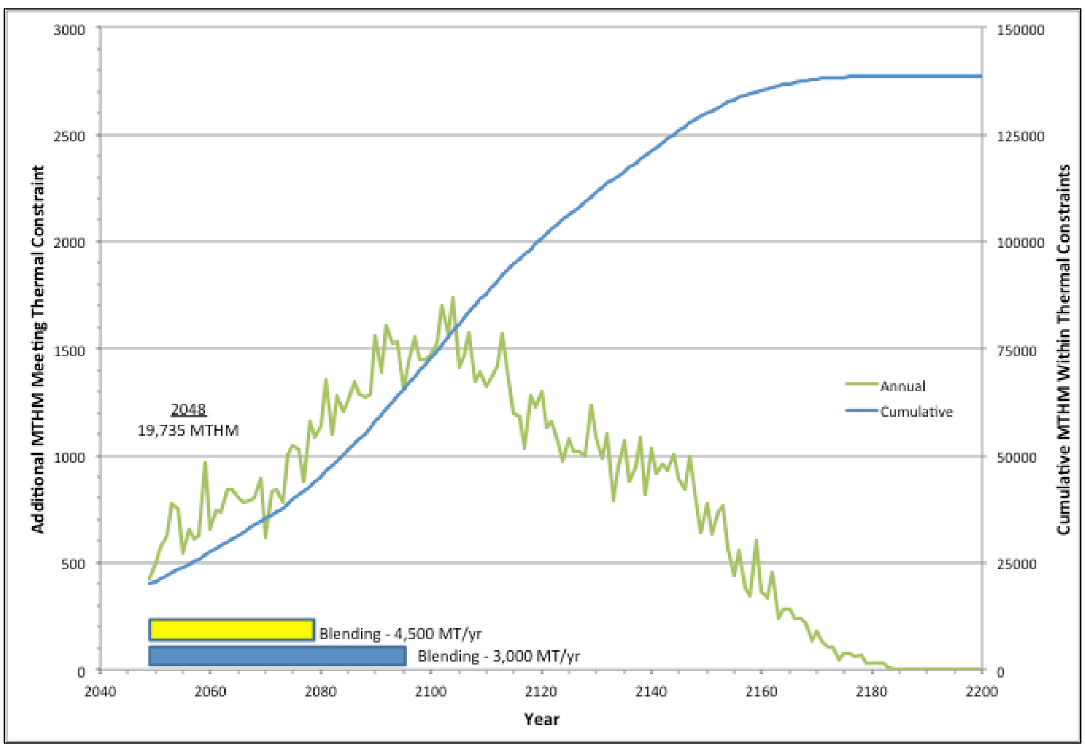


b) Amount of SNF

Figure 9-1. Projections for DPCs cooling to 4 kW each year

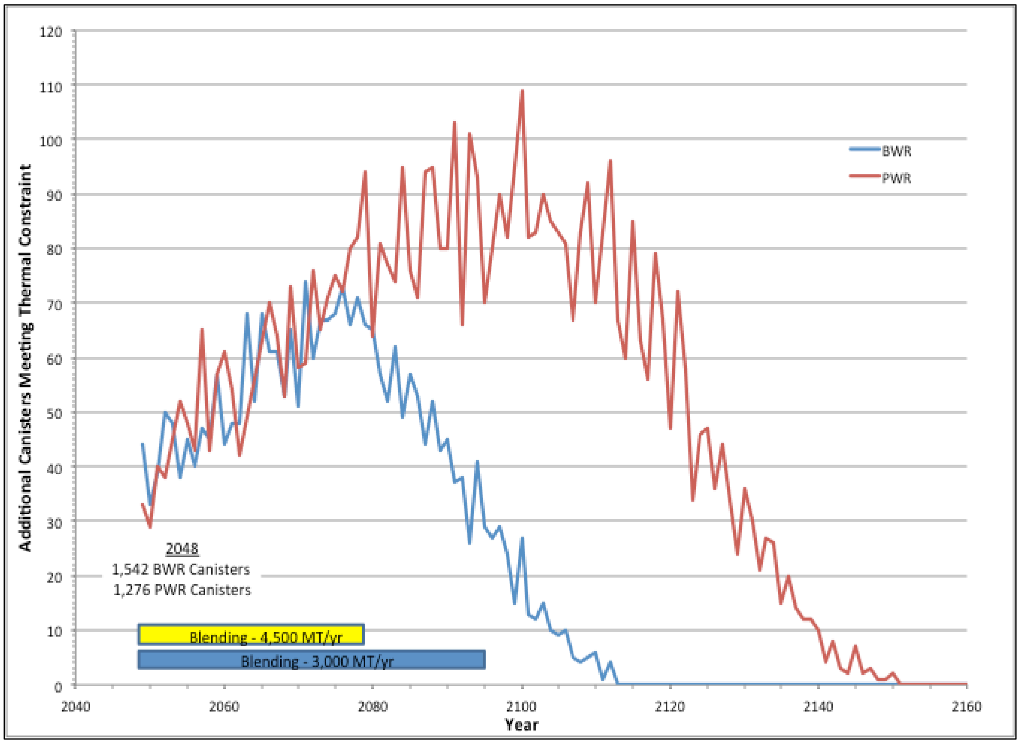


a) Number of canisters

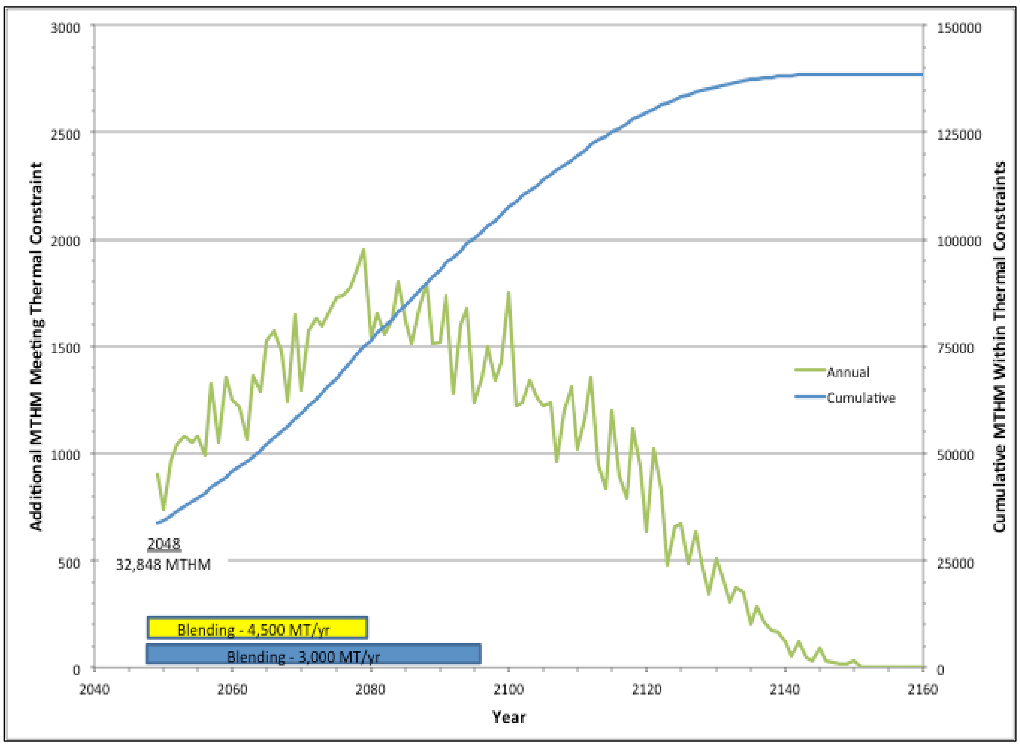


b) Amount of SNF

Figure 9-2. Projections for DPCs cooling to 6 kW each year

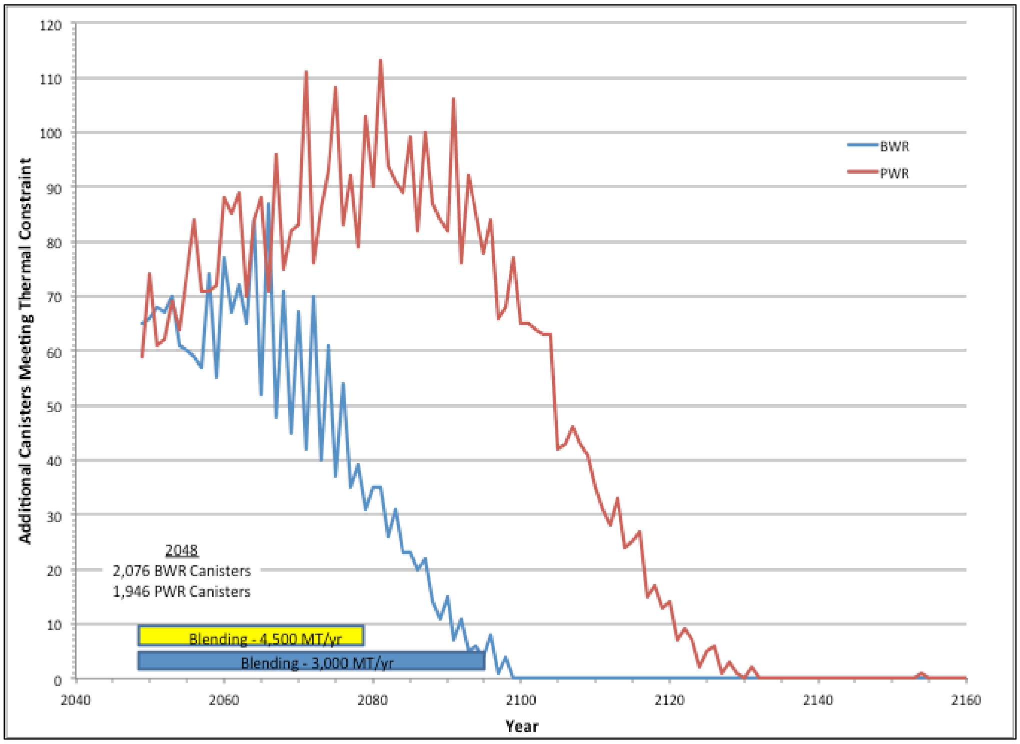


a) Number of canisters

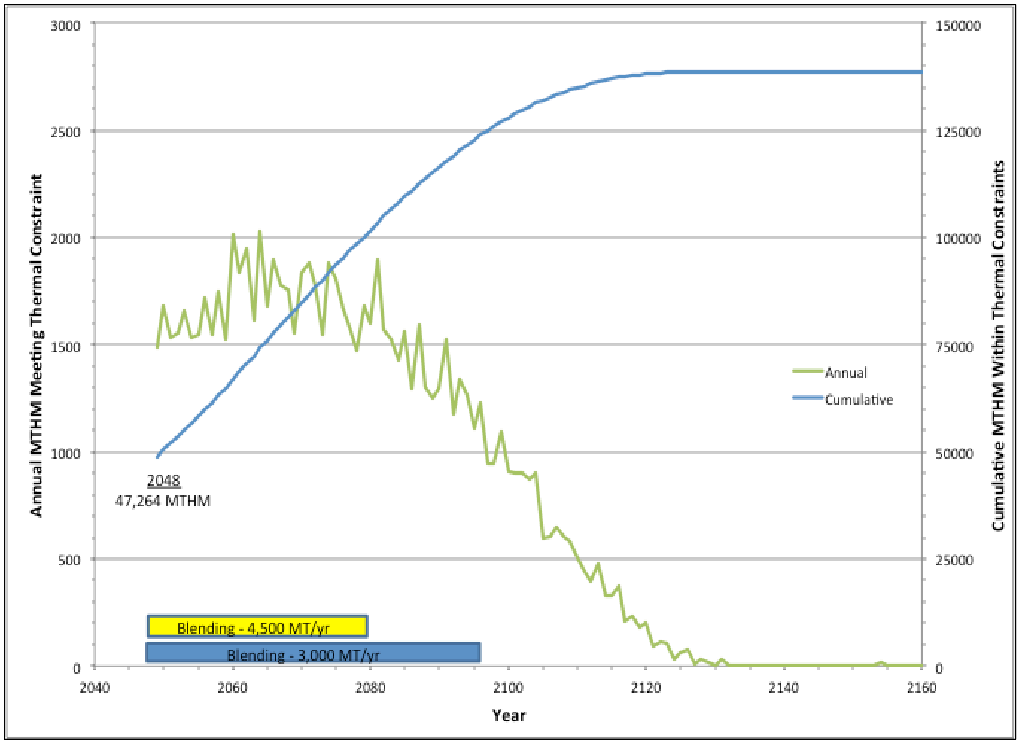


b) Amount of SNF

Figure 9-3. Projections for DPCs cooling to 8 kW each year

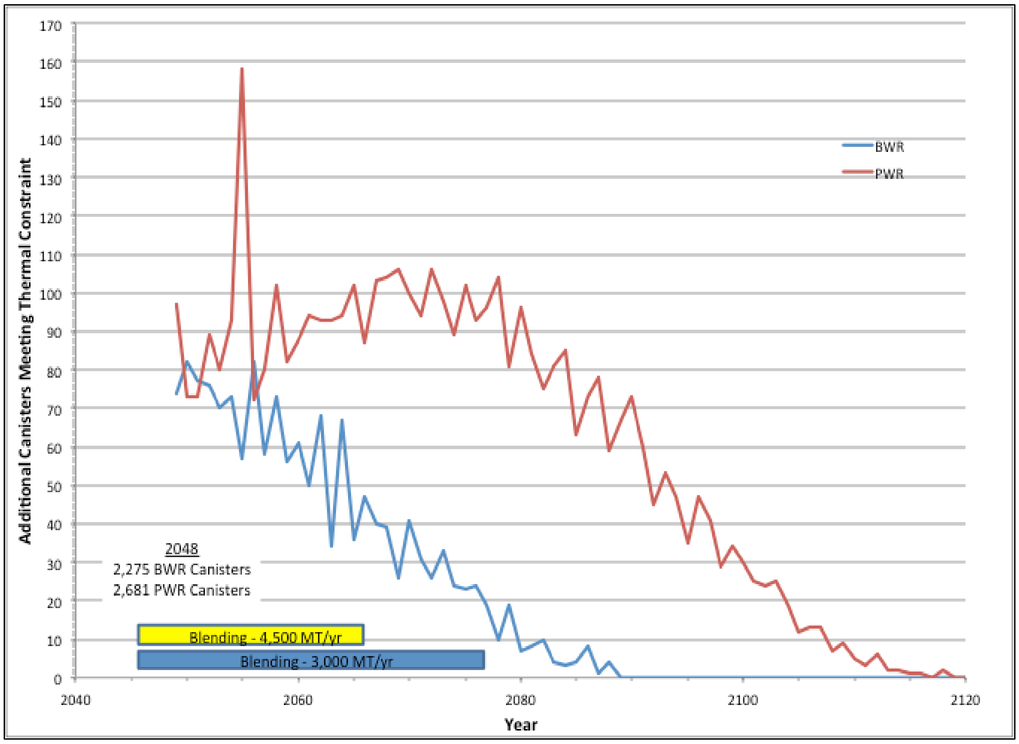


a) Number of canisters

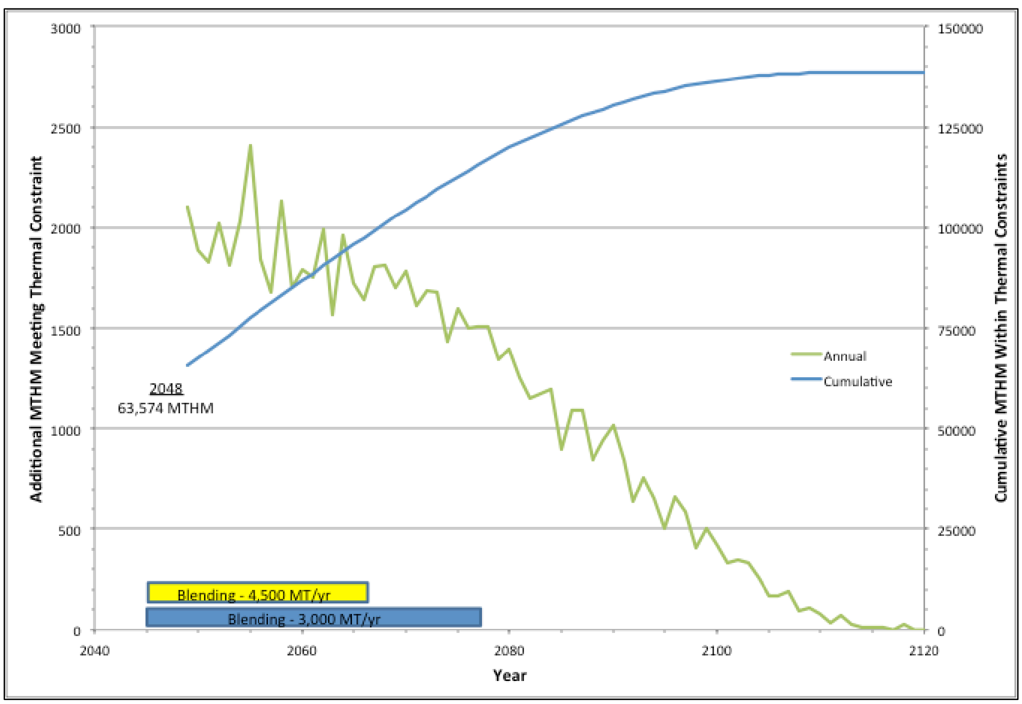


b) Amount of SNF

Figure 9-4. Projections for DPCs cooling to 10 kW each year

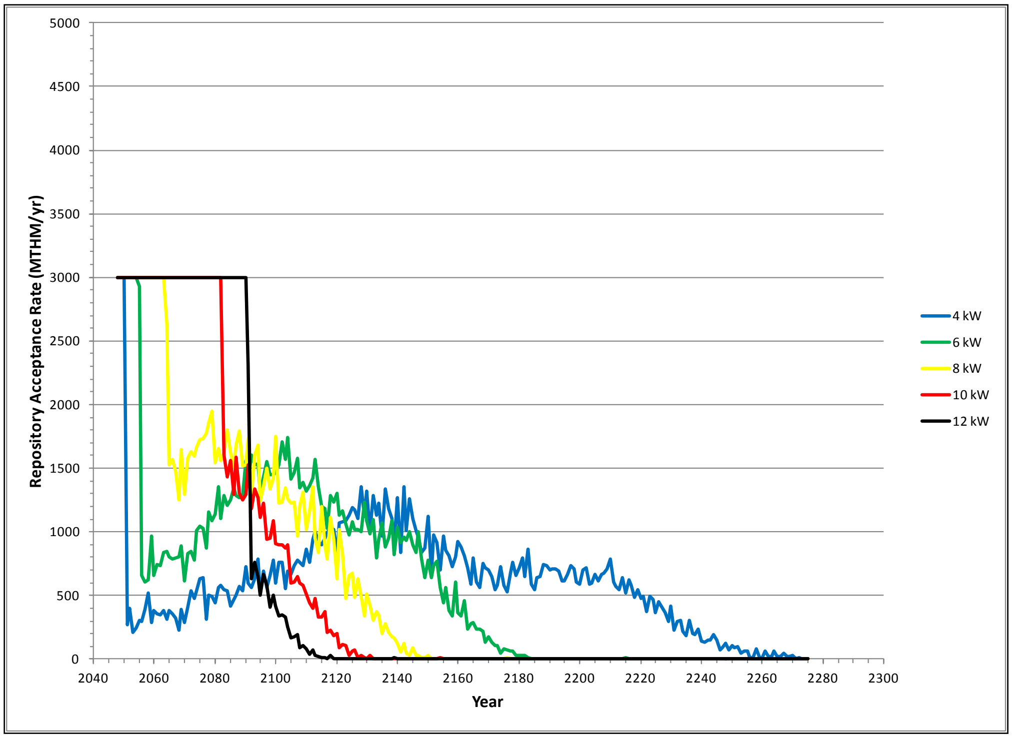


a) Number of canisters

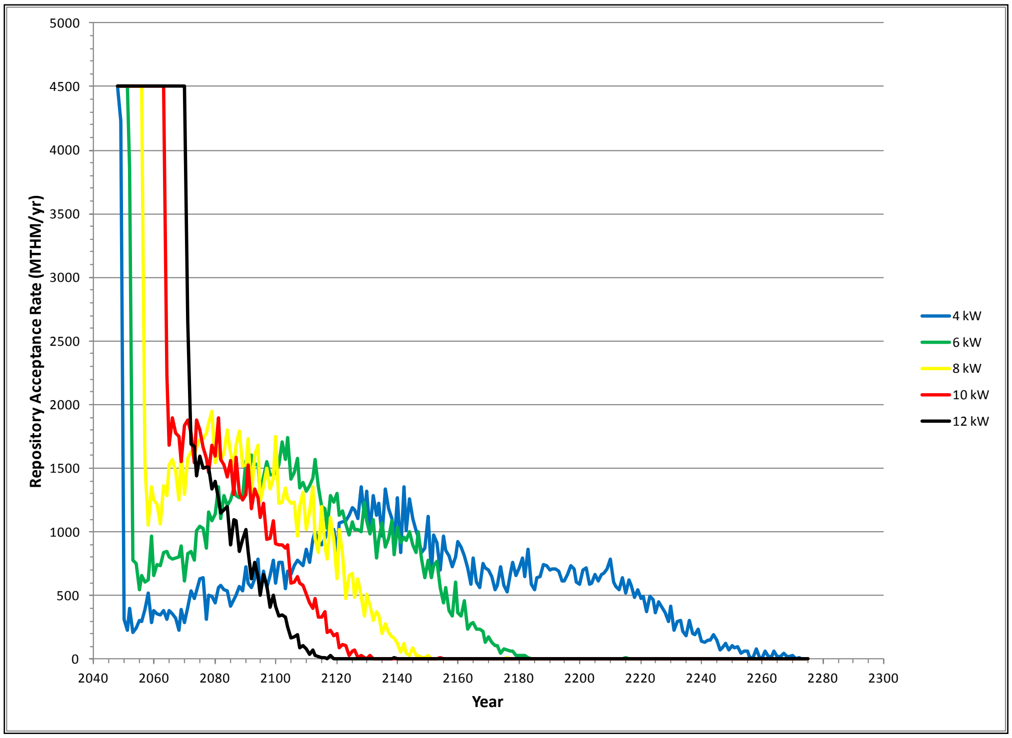


b) Amount of SNF

Figure 9-5. Projections for DPCs cooling to 12 kW each year



a) 3,000 MTHM/yr maximum repository acceptance rate



b) 4,500 MTHM/yr maximum repository acceptance rate

Figure 9-6. Projected Annual Repository DPC Acceptance Rates for Large Initial Acceptance

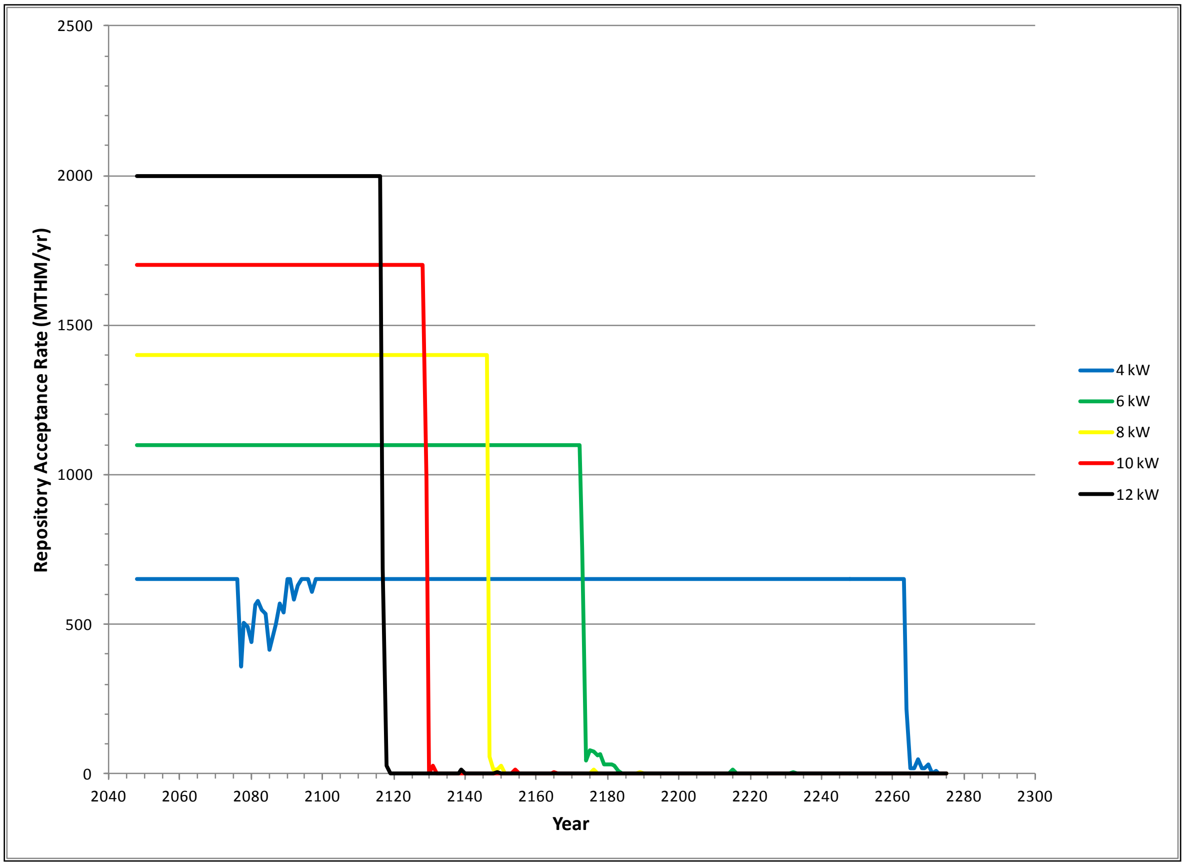


Figure 9-7. Optimized Annual Repository DPC Acceptance Rates

9.2.2 Incremental ISF Storage Costs and Re-Packaging Cost Comparison

The incremental additional cost of extended decay storage is shown in Table 9-2. Both the estimated incremental capital cost required to deploy additional ISF storage capacity and the cost to continue ISF operations for an additional period of time to allow the DPCs to cool sufficiently are shown. The results show that incremental ISF storage costs increase as the repository emplacement thermal power limit decreases. A higher ISF acceptance rate leads to lower incremental costs because less additional ISF capacity needs to be deployed (see Table 9-1).

For comparison with incremental storage life-cycle costs in Table 9-2, the life-cycle cost of deploying and operating a re-packaging facility was estimated for a range of different SNF management scenarios (Nutt et al. 2012, Table 6-10). The life-cycle cost for a 3,000 MTHM/yr re-packaging facility was estimated to range from \$6.5 billion to \$14.5 billion, depending on the size of the new disposal canisters (with smaller canisters resulting in greater cost). The life-cycle cost for a larger capacity facility is greater, approximately \$20 billion for 6,000 MTHM/yr. These re-packaging cost estimates do not include disposal (i.e., overpacks, repository construction and operation, etc.). When disposal costs are taken into account re-packaging of SNF from DPCs into smaller containers could add on the order of \$10B and possibly several times that to the total disposal cost for commercial SNF in the U.S., depending on the disposal concept and the number of waste packages (e.g., see Kalinina and Hardin 2012).

Note that the question of where re-packaging could occur (either at an ISF, at a stand alone facility, or co-located with a repository) is a topic of ongoing systems evaluations work and includes other system considerations such as transportation and shared infrastructure cost. Thus this is just one example of comparative analysis of system costs.

Table 9-2. Incremental ISF storage costs

a) 3000 MTHM/yr ISF Acceptance Rate

Repository Emplacement Thermal Limit (kW)	Repository Acceptance Rate (MTHM/yr)	Additional ISF Capital Cost ($ B)	Additional ISF Operating Cost ($ B)	Additional ISF Lifecycle Cost ($ B)
4	650	$8.2	$4.3	$12.5
6	1100	$6.6	$2.1	$8.8
8	1400	$5.6	$1.3	$6.9
10	1700	$4.5	$0.9	$5.4
12	2000	$3.5	$0.6	$4.1

b) 4500 MTHM/yr ISF Acceptance Rate

Repository Emplacement Thermal Limit (kW)	Repository Acceptance Rate (MTHM/yr)	Additional ISF Capital Cost ($ B)	Additional ISF Operating Cost ($ B)	Additional ISF Lifecycle Cost ($ B)
4	650	$3.8	$4.6	$8.4
6	1100	$2.7	$2.5	$5.2
8	1400	$2.0	$1.7	$3.7
10	1700	$1.2	$1.2	$2.5
12	2000	$0.5	$0.9	$1.4

9.3 Summary

A preliminary system analysis was conducted using simulation tool TSL-CALVIN, to assess some of the cost and schedule impacts from directly disposing of DPCs in a geologic repository. The analysis results show that:

- The operation of a geologic repository for the direct disposal of DPCs could be extended compared with a re-packaging strategy that proceeds at a higher rate of throughput (e.g., 3,000 MTHM/year). The duration of repository operations increases as the thermal power limit decreases and for lower emplacement thermal power limits (e.g., 4 kW/canister for

DPC disposal in sedimentary media) it could be more than 200 years. This could be mitigated around the time of repository licensing by deploying purpose-built standard canisters that meet site-specific requirements for disposability.

- Because of the cooling time that could be needed, a repository for DPC direct disposal could have a smaller throughput rate, for example, an optimal rate for a 10 kW emplacement thermal limit is 1,700 MTHM/year. Smaller capacity operating facilities could be deployed, partly compensating for the cost of extended operations. Alternatively, a higher steady-state acceptance rate could be achieved by deploying purpose-built standard canisters when waste package emplacement thermal limits are determined.
- The incremental cost of extended ISF storage (for cooling DPCs) is likely to be less than the life-cycle cost of building and operating a re-packaging facility. Depending on the emplacement thermal power limit, and the size of new canisters used in re-packaging, pre-disposal system cost savings on the order of $10B or more could be realized. Note that any additional costs of extended surface storage such as increased maintenance and inspection, canister refurbishment, etc., have not been quantified and could reduce the overall potential cost savings.
- For an emplacement power limit of 10 kW or greater (Figures 9-4 and 9-5) such as could be used for DPC direct disposal in salt or for hard rock (crystalline) options (Section 4), emplacement could be substantially complete by calendar 2130 (with a few outlying, high-burnup canisters). For the salt concept repository closure could soon follow, while for hard rock concepts a few decades of repository ventilation could be needed before closure. An optimal disposal acceptance rate of approximately 1,700 MTHM/year was calculated to complete disposal by 2130 (Figure 9-7).

Given the long operational time frames considered and uncertainty in the underlying assumptions and bases used for the schedule and cost analyses presented here, it is important to remember that the ranges of values presented are ROM and are only useful for relative comparison, not absolute values.

References for Section 9

Carter, J., A. Luptak, J. Gastelum, C. Stockman and A. Miller 2012a. *Fuel Cycle Potential Waste Inventory for Disposition.* FCR&D-USED-2010-000031 Rev. 5. U.S. Department of Energy, Used Fuel Disposition R&D Campaign. July, 2012.

DOE (U.S. Department of Energy) 2013. *Strategy for the Management and Disposal of Used Nuclear Fuel and High-Level Radioactive Waste*. January, 2013.

Hardin, E., T. Hadgu, D. Clayton, R. Howard, H. Greenberg, J. Blink, M. Sharma, M. Sutton, J. Carter, M. Dupont and P. Rodwell 2012. *Repository Reference Disposal Concepts and Thermal Management Analysis*. FCRD-USED-2012-000219 Rev. 2. November, 2012. U.S. Department of Energy, Used Fuel Disposition R&D Campaign.

Hardin, E. 2013. *Temperature-Package Power Correlations for Open-Mode Geologic Disposal Concepts*. SAND2013-1425. Sandia National Laboratories. Albuquerque, NM. February, 2013.

Kalinina E.A. and E. Hardin 2012. *Logistical Simulation of Spent Nuclear Fuel Disposal in a Salt Repository with Low Temperature Limits*. SAND2012-8109. Sandia National Laboratories. Albuquerque, NM. September, 2012.

Nutt, M., E. Morris, F. Puig, E. Kalinina and S. Gillespie 2012a. *Transportation Storage Logistics Model – CALVIN (TSL-CALVIN)*. FCRD-NFST-2012-000424. U.S. Department of Energy, Nuclear Fuel Storage and Transportation Planning Project. Washington, D.C. October, 2012.

Nutt, M., E. Morris, F. Puig, J. Carter, P. Rodwell, A. Delley, R. Howard and D. Guliano 2012b. *Used Fuel Management System Architecture Evaluation, Fiscal Year 2012*. FCRD-NFST-2013-000020 Rev. 0. U.S. Department of Energy, Nuclear Fuel Storage and Transportation Planning Project. Washington, D.C. October, 2012.

Nutt, W.M. 2013. *Preliminary System Analysis of Direct Dual Purpose Canister Disposal*. FCRD-UFD-2013-000184 Rev. 0. July, 2013. U.S. Department of Energy, Used Fuel Disposition R&D Campaign.

10. Engineering and Performance Research and Development Needs

This section identifies R&D needs for which further investigation could contribute to future evaluation of direct disposal of commercial spent nuclear fuel (SNF) in existing dual-purpose canisters (DPCs). It represents a team consensus as to which issues are potentially important and also amenable to resolution. This list addresses various aspects of feasibility including cost, but emphasizes technical issues.

This R&D "roadmap" describes at a high level, more than 30 activities, mostly technical, that could facilitate the completion of feasibility evaluation. It does not discriminate among them with respect to possible measures of utility, as was done previously for the Used Fuel Disposition (UFD) R&D program as a whole (Nutt 2011), but it recommends developing a process to do so.

The list was compiled from team member input and the previous issue survey for the UFD campaign (Nutt 2011). Various alternative disposal concepts for existing DPCs have been identified (e.g., salt, hard rock, sedimentary; Section 4) and some of the issues are specific to these concepts.

10.1 Waste Characteristics and Heat Generation

These R&D needs pertain to the condition of DPCs and SNF after extended storage, DPC heat generation and heat dissipation, and materials interactions.

1. **Condition of SNF and canisters to allow storage and transport up to 100 years after discharge, and disposal.**

   The feasibility evaluation assumes up to 100 years of decay storage (Section 2) during which there may be degradation of DPCs and the fuel they contain. Improved understanding of the likelihood and consequences of degradation is needed to plan steps leading to DPC direct disposal (including whether longer storage is feasible). Advanced technologies for evaluating the condition of canisters and the SNF they contain, could be developed to facilitate inspection.

2. **Capability to transport the canisters to the repository.**

   Weights for DPCs with transportation overpacks range up to approximately 165 MT (Rigby 2010). Transport of large, heavy DPCs in shielded transportation containers must be assumed for this study, but should be considered in more detail as part of the Used Fuel Disposition R&D campaign.

3. **Capability to dispose of DPCs with higher thermal loading (up to 37 PWR/ 89 BWR) and containing higher burnup (up to 60 GWd/MT) SNF.**

   Dry storage canisters deployed in the future may be larger than existing ones, with greater thermal power. Thermal and postclosure criticality analyses presented in this report would be extended to additional package sizes (both larger and smaller than 32-PWR size) and different burnup characteristics.

4. **Update database on existing DPCs.**

   Continue to compile and organize information on DPC construction, fuel loading, burnup characteristics of individual assemblies, control rod or poison rod loads, etc. This information could be useful for mapping which DPCs are considered to be disposable, as a function of the disposal concept and other external variables.

5. **Compatibility of DPC materials and overpack materials.**

   Physical and chemical compatibility of these materials must be assured. Interactions of degraded waste package materials (e.g., corrosion products) will influence the disposal environment, radionuclide transport, and mechanical interfaces.

6. **Thermal analysis.**

   In-package temperatures will likely be less than temperature limits for fuel and other components (e.g., 350°C to limit cladding creep; see Section 4) because of the temperature margin possible if the package surface is limited to 200°C. However, there are complicating factors that might be important including thermal convection in large packages which could produce a temperature differential between top and bottom.

   After loading, storage canisters are pressurized with helium to reduce cladding corrosion and to improve heat transfer. Helium may be lost over time (e.g., 50 to 100 years) due to slow leaks that may form on DPCs (e.g., in weld-affected zones). Heat output is likely to have decayed significantly if and when such leaks form. Also, while leaks will vent the pressure down to atmospheric, exchange with air could take much longer. Analysis is needed to determine the need for helium (i.e., the potential impacts from leakage) during handling, transport and emplacement operations at a repository, and after disposal.

10.2 Preclosure Operations, Performance and Safety

These R&D needs are driven by the additional size and weight of DPC-based waste packages.

7. **Shielding of DPCs during handling and transport at the repository.**

   The size, weight, and materials required for adequate shielding of handling equipment and transporters at the repository are needed to establish design requirements. As noted in Section 3.4, mature technology exists for shielding DPCs. Hence this is an engineering detail that will need to be addressed later, rather than a feasibility issue.

8. **Loading horizontal and vertical DPCs into disposal overpacks.**

   Assess the availability of engineering solutions for loading horizontal DPCs (e.g., NUHOMS) into disposal overpacks, then sealing the overpacks. Also, assess the loading of vertical DPCs (or other canister types) into overpacks for horizontal transport and disposal.

9. **Preclosure safety assessment for direct disposal of DPCs.**

   Waste packages and the operations required for handling and emplacement will have to meet preclosure safety requirements of applicable regulations (e.g., 10CFR60) which require safety assessment. Event sequences could be influenced by the size and weight of the waste package, particularly when shielded, and the number of repeated operations required in facilities. Events that could impact the DPC handling and transport include:

   a) Side or end impacts on waste packages

   b) Rockfall onto the waste packages

   c) Waste package horizontal drops, vertical drops and collisions

   d) Waste package drop by emplacement machines

e) Waste package impact onto a sharp object

f) Tip-over associated with vertical drop or seismic ground motion

g) Missile impact

h) Transporter runaway

i) Peak waste package temperature during off-normal events

j) Fire in a disposal container handling cell

k) Waste package nuclear reactivity

10. Transport and emplacement of shielded waste packages underground.

Payload weight (up to 175 MT) for typical DPCs in shielded enclosures would exceed the capacity of any existing shaft hoist by more than a factor of 2, and ramp transport has never before been licensed for SNF or HLW. This activity would define and analyze the engineering issues associated with these options, including preliminary safety analysis for different conveyances.

11. Stability of underground excavations.

Direct disposal of DPCs may require large excavations and stability for up to 100 years. The largest spans will be at intersections, transfer stations, etc. Site-specific excavation designs will address support requirements, the influence of thermal loading, and the potential for maintenance.

10.3 Postclosure Performance

These items may be important for direct disposal of DPC-based waste packages, but could also benefit other disposal concepts (i.e., the range identified in Table 4-1, where applicable) or the design of future DPCs. As a general example, higher temperature tolerance for engineered materials could make other disposal concepts more efficient, not just DPC direct disposal concepts.

12. Buffer/backfill performance.

Buffer and backfill materials will be engineered barrier system items important to safety. Responses to heat, water intrusion and chemical processes could affect waste isolation. Thermally driven processes that could impact buffer or backfill properties may be active at temperatures above or below 100°C (see Hardin and Voegele 2013, Appendix B). Temperature and hydration state in the buffer/backfill will exhibit strong gradients. R&D on the evolution of these materials would likely involve multiple technical disciplines, and laboratory investigations, simulation, and field-scale validation.

13. Buffer temperature tolerance and thermal conductivity.

Establish a higher temperature tolerance (e.g., up to 200°C or higher) for backfill material, for example, by proving the performance of smectite clay-based materials under controlled conditions (e.g., dry, and/or protected from exposure to water or steam), or finding a new material that provides similar performance. Also, identify backfill materials or admixtures that increase thermal conductivity (e.g., graphite), and evaluate their cost and other factors (e.g., toxicity) that could determine suitability for use.

14. **Heating of near-field host rock to higher temperatures.**

    This report has shown that drift and package spacings, and required ventilation times, could be reduced if the peak temperature target for the near-field host rock were increased (e.g., higher than 100°C). It also showed (Section 5.3) that the extent of such a temperature zone would be limited to the immediate vicinity of each package. Analysis of host rock performance, include mechanical stability (for unbackfilled conditions) and radionuclide transport could be useful for assessing whether higher temperature targets could be considered.

15. **Brine migration in salt.**

    The potential for brine migration toward heat sources in salt under gradients of stress and temperature has been demonstrated (Hansen and Leigh 2011). The rate of migration, and the total brine production before cooling and mechanical reconsolidation, are uncertain. Brine could corrode the disposal overpack, producing corrosion products and hydrogen gas (see next item). Ultimately, the release and transport of radionuclides under nominal conditions requires an aqueous phase.

16. **Potential for gas generation and its importance to waste isolation performance.**

    Gases generated include helium from alpha decay, hydrogen from corrosion of the canister or fuel, and residual moisture. For some disposal concepts gas pressure could affect the disposal environment, and radionuclide transport. Assessment of gas generation effects could impact the selection of overpack materials and other system details.

17. **Waste package vertical movement in salt.**

    Recent analysis has shown that vertical movement (sinking) of heavy waste packages in salt, associated with thermal expansion of the host rock and thermally activated creep, may not be significant (Clayton et al. 2013). Additional validation of the constitutive model used for intact salt, which could involve *in situ* thermal testing, would provide more confidence in this result.

18. **Postclosure features, events and processes (FEPs) that could be influenced by the size, heat output, and quantity of waste in DPCs.**

    The FEP analysis described in Section 7 would be conducted in more detail, using appropriate models. This would be an iterative process that incorporates model development and design-related information, in addition to site-specific information if available.

10.4 Postclosure Criticality Analysis

These R&D needs relate specifically to analysis of the potential for criticality to occur in degraded DPCs after disposal, breach and flooding by groundwater. They include analysis methodology, and the completeness of its application to the population of existing DPCs. Criticality consequence analysis is included for development and possible future use in a "layered" argument whereby various factors are considered that substantially reduce the probability of a criticality event, and the overall risk depends on its consequences. Also included are measures to re-work existing DPCs to mitigate the potential for criticality (e.g., by filling with inert material to displace groundwater in the event of flooding).

19. Criticality safety.

More analysis using as-loaded DPC data and fuel characteristics is needed for assurance that the repository system, including natural and engineering features, will preclude or limit post-closure criticality. Further R&D would increase the sample of DPCs and DPC types analyzed, particularly to include boiling water reactor (BWR) fuel assemblies and the TransNuclear DPCs. Used BWR fuel assemblies typically have less reactivity than PWR fuel and the TN basket incorporates a disc basket system that may provide better geometry control in degraded basket analyses.

20. Nuclear reactivity analysis methodology.

A review of the criticality analysis approach described in Section 8 is needed to ensure appropriate realism in the degradation cases, and avoid excess conservatism. This activity would review the approach used in Section 8, which is based on DOE (1998). Develop new stylized, representative cases as appropriate to represent potentially reactive, degraded configurations. Consider the quantity and fate of corrosion products (e.g., B_4C particles, and products from aluminum and stainless steel corrosion). Revisit the extent of absorber panel degradation and basket structural material degradation that are accounted for, using updated corrosion data. Analyze degraded configurations more closely to evaluate and develop assumptions representing the fate of BORAL in the disposal environment, over repository time frames.

21. Nuclear reactivity sensitivity analysis.

Identify and prioritize parameters in the reactivity analysis that have the greatest impact on criticality, to identify opportunities for reducing conservatism and/or increasing realism.

22. Neutronics model validation.

Evaluate use of enhanced neutronics model validation techniques to reduce computational model bias and uncertainty (see Section 8).

23. Criticality probability.

Perform event-tree analyses to estimate criticality probability.

24. Criticality consequence modeling.

Model the consequences of intermittent criticality on thermally driven processes in the repository (impact on radionuclide inventory has been evaluated).

25. Canister fillers.

Identify materials that could be used to fill existing DPCs (e.g., by exposing drain and vent ports; see Section 3), to prevent criticality neutron absorption and/or moderator displacement. The effects on waste isolation could be included in the analysis.

10.5 Development of DPC Disposal Concepts

For certain concepts there is relatively little literature or other information that would support a conceptual design. Among the concepts described in Section 4, this is especially true for the cavern-retrievable concepts which may include development of underground vaults for the purpose of ventilated long-term storage and ultimate disposal.

26. Cavern-retrievable concept thermal analysis.

Simulate temperature histories for cavern retrievable concepts (Section 4.7.1) to evaluate thermal performance of the storage casks and the surrounding EBS.

27. DPC disposal vault concept.

Develop a configuration for the subterranean storage system (Section 4.7.2) that accepts a range of existing DPC types, and optimizes heat transfer during preclosure and postclosure.

10.6 System Logistics

28. Detailed logistical analyses.

Analyses are needed to address possible future impacts related to DPC direct disposal, for example: 1) optimized dry storage duration for every DPC; 2) impact from regulation-driven shift from pools to dry storage (e.g., duration or other limits); 3) impact from a centralized storage operating time limit; 4) costs/benefits from alternative SNF selection criteria for canistering in DPCs (e.g., OFF, YFF); 5) impacts from timing of DPC direct disposal licensing; and 6) quantitative analysis of DPC direct disposal contributions to fuel management system flexibility. Additional discussion of analysis needs in this area is provided in Section 9.

10.7 Feasibility Evaluation Approach

29. Update assumptions and work plan.

The assumptions in Section 2 of this report and the work plan under which the work is performance (Howard et al. 2012) need to updated to reflect changes in approach and scope that have arisen thus far in performing the study.

30. Platform for supporting future decisions.

Identify possible future decisions that the results from this feasibility evaluation are intended to inform. Such decisions include down-selection among disposal concepts and media, standardized (storage-transport-disposal) canister design, changes to DPC designs (e.g., for disposability), system-level decisions on whether to store as bare fuel or in dry storage casks, etc.

31. Engage stakeholders.

Industry and the Nuclear Regulatory Commission staff, among others, would be engaged to inform and possibly reach common understanding of technical matters such as disposal concept development, thermal analysis and postclosure criticality analysis.

32. Plan to narrow the range of disposal concepts investigated.

The feasibility evaluation approach should plan to support a future selection process that could narrow the range of disposal concepts to be considered for DPC direct disposal, focusing resources on the most promising concepts. That approach could define and recommend specific R&D activities to support such a process.

33. **Develop technology readiness information for disposal concepts.**

    As part of a strategy for developing and applying feasibility information, the disposal concepts would be broken down into natural and engineered components for which the current (generic) state of knowledge, and the maturity of available technologies, could be assessed.

34. **Improve cost data for comparison among alternatives, including re-packaging strategies.**

    Improved cost data that link disposal costs with upstream fuel system activities in a consistent manner, would support more useful comparisons of alternative strategies, especially between direct disposal or re-packaging.

References for Section 10

Clayton, D.J., M.J. Martinez and E.L. Hardin 2013. *Potential Vertical Movement of Large Heat-Generating Waste Packages in Salt*. SAND2013-3596. Albuquerque, NM: Sandia National Laboratories. May, 2013.

DOE (U.S. Department of Energy) 1998. *Disposal Criticality Analysis Methodology Topical Report*. YMP/TR-004Q Rev. 0. Office of Civilian Radioactive Waste Management. November, 1998.

Hansen, F.D., and C.D. Leigh 2011. *Salt Disposal of Heat-Generating Nuclear Waste.* SAND2011-0161. OSTI ID: 1005078. Albuquerque, NM: Sandia National Laboratories. January, 2011.

Howard R., J. Scaglione, J. Wagner, E. Hardin and W. Nutt 2012. *Implementation Plan for the Development and Licensing of Standardized Transportation, Aging, and Disposal Canisters and the Feasibility of Direct disposal of Dual Purpose Canisters*. FCRD-UFD-2012-000106 Rev. 0. U.S. Department of Energy Fuel Cycle Technology Program, Used Fuel Disposition Campaign.

Rigby, D.B. 2010. *Evaluation of the Technical Basis for Extended Dry Storage and Transportation of Used Nuclear Fuel*. U.S. Nuclear Waste Technical Review Board. December, 2010. (www.nwtrb.gov)

Nutt, W.M. 2011. *Used Fuel Disposition Campaign Disposal Research and Development Roadmap*. FCR&D USED-2011-00065 Rev. 0. U.S. Department of Energy, Used Fuel Disposition R&D Campaign. March, 2011.

THIS PAGE INTENTIONALLY LEFT BLANK

11. Summary and Preliminary Feasibility Statement

When completed, this study is intended to be an early look at the generic possibilities for DPC direct disposal. The purposes of this report are to document a set of preliminary technical analyses, to identify additional R&D needs, and to recommend whether the feasibility evaluation should continue (i.e., to assess whether there are disposal concepts promising enough to warrant the additional R&D and feasibility evaluation effort).

The objectives for direct disposal of SNF in DPCs are the same as for any geologic repository—the safety of workers and the public, and long-term isolation of the radioactive materials from the biosphere. Achieving these objectives will involve: 1) thermal management; 2) engineering feasibility; 3) preventing or limiting criticality after waste emplacement; 4) operational and logistical feasibility; and 5) achieving acceptance by regulators and the public. R&D to advance scientific understanding of elevated temperature disposal environments and large waste packages (i.e., for DPC direct disposal) will enhance understanding of lower temperature repository systems. Preliminary information on each of these topics is presented below:

Disposal Concepts– The feasibility of DPC direct disposal depends on the geologic setting, and a wide range of geologic settings is being carried forward on a generic (non-site specific) basis (Nutt 2011). A range of possible concepts for DPC direct disposal was identified (Section 4) drawing on previous concept development. Concepts for evaluation include the salt concept, and emplacement in hard rock (i.e., crystalline) or argillaceous sedimentary rock. This set of concepts is not exhaustive, but it covers a range of behaviors potentially important to DPC direct disposal including thermal response, postclosure nuclear criticality, and long-term opening stability. Other factors such as ground support, waste package transport and emplacement, and shaft vs. ramp access, are also important and may depend more on site-specific characteristics and in some instances, local experience and preference.

The salt concept would be backfilled immediately after emplacement, while openings in hard rock and sedimentary rock would be ventilated for decades to remove heat. Hard rock formations exist that would have excellent long-term stability, heat dissipation properties, and environmental conditions conducive to waste isolation. Argillaceous (clay-bearing) sedimentary media (e.g., clay or shale formations) could have very low permeability and chemically reducing conditions, but are likely to have more restrictive thermal constraints to limit alteration of the clay, relatively low thermal conductivity and more limited long-term stability. Backfill is an option for hard rock and sedimentary open-mode concepts, and could provide an additional, redundant engineered barrier. However, the use of backfill would significantly elevate EBS temperatures as discussed below.

The cavern-retrievable storage and disposal concept was first proposed about a decade ago and remains a potentially important alternative that combines the heat removal performance of an open mode before closure, with an enclosed mode after closure. Shielded dry-storage casks would be emplaced or installed underground, ventilated for decades to remove heat, and closed by installation of an encapsulating buffer. The use of existing surface storage casks (or cask designs) could lower costs but would require development and testing of a buffer system to assure waste isolation.

Safety – Important factors that help to ensure postclosure safety for DPC direct disposal include: 1) diffusion-controlled radionuclide transport in the EBS and NBS; 2) near-field transport properties that are relatively insensitive to temperature, or for which temperature effects can be modeled with confidence; 3) limited radionuclide transport in backfill and the host rock (particularly the far field); and 4) attributes that limit potential postclosure criticality. These characteristics would actually benefit any geologic repository. When prospective repository sites are identified, site-specific data will support more resolution of differences in postclosure safety associated with DPC direct disposal.

Thermal Management – The salt concept and the unbackfilled hard rock concepts could accept SNF in 32-PWR size packages, with SNF burnup to 60 GW-d/MT, and approximately 50 to 100 years decay storage depending on burnup. These repository concepts could close within the 150-year timeframe adopted for this study (Section 2), while meeting target values for peak host rock temperature (200°C in both types of media).

In sedimentary media, which have lower thermal conductivity and a lower target value for peak rock temperature (100°C) lower burnup PWR SNF (and BWR SNF with similar heat output) could be accommodated within the 150-year timeframe, with repository package and drift spacings similar to the hard rock concept. For higher burnup SNF (e.g., greater than 40 GW-d/MT) a modified concept would be needed that uses some combination of: 1) longer decay storage plus ventilation; 2) much larger spacing (roughly doubling the repository plan area); and/or 3) peak host rock temperature target greater than 100°C.

When backfill is added to the hard rock or sedimentary concepts, and installed at repository closure, the waste package temperature increases significantly. The temperature rise within the backfill (i.e., at the waste package surface) is much greater than the differences in temperatures between the hard rock and sedimentary concepts. Clay-based low-permeability backfill materials could be sensitive to temperature, and better understanding of clay behavior, or alternative materials, is needed to facilitate backfill options for DPC direct disposal.

For a similar reason the enclosed emplacement modes such as those being pursued in Sweden or France for crystalline and sedimentary media, respectively, when applied to DPC direct disposal, cannot meet the peak buffer temperature target without decay storage much longer the assumed timeframe (150 years out-of-reactor). It is therefore important to continue R&D that would support relaxation of thermal constraints on argillaceous host media and backfill/buffer materials. Such research could benefit any disposal concept, even those involving re-packaging.

Engineering Feasibility – Handling and packaging of large DPCs in surface facilities at the repository or at upstream installations, are within the state of available technology and current practice. The operations needed to transfer each DPC to a suitable disposal overpack are similar to those used for DPC loading, storage and transportation. Moreover, handling and packaging would be similar for any DPC direct disposal concept, no matter where the repository is located or in what geologic host medium. Thus, although engineering details need to be worked out and options are needed for standardized or universal equipment to handle the wide variety of DPC systems, there appear to be no significant feasibility questions associated with repository operations until the waste is transported underground.

Several options exist for surface-to-underground waste package transport in shafts or ramps, including shaft hoists, funiculars, and rubber-tire or rail-mounted ramp transporters. These waste transport options are technically feasible although some systems, if implemented for DPCs, would be the largest of their kind. The choice is likely to depend on site-specific geology and local experience. Additional engineering is needed to develop systems for transport within the underground facility and for emplacement. Such systems have been demonstrated for the

Swedish repository concept using existing technology, and the systems needed for in-drift emplacement concepts presented here would be large, but relatively simple. Note that rail-mounted and rubber-tire shielded transport-emplacement vehicle (TEV) concepts were evaluated for a repository in unsaturated tuff, for underground transport and in-drift emplacement of waste packages weighing up to approximately 74 MT.

Criticality – Understanding the likelihood and consequences of in-package nuclear criticality for at least 10,000 years after disposal is important for evaluating the feasibility of DPC direct disposal. This study is focused on the potential for in-package criticality, and not external criticality, as a possible factor in determining feasibility of DPC direct disposal. Site characteristics and engineered system attributes that prevent or limit the probability of groundwater intrusion into failed waste packages are beneficial. Intrusion of brine (a possibility for the salt concept) is beneficial because natural ^{35}Cl is a neutron absorber. Preliminary analysis indicates that many, although not all existing DPCs would be sub-critical even if chemically and mechanically degraded in the disposal environment. Additional reactivity margin is available by using as-loaded assembly information, updated burnup credit, and taking into account groundwater salinity. With further analysis, existing DPCs can be categorized according to the potential for criticality in different disposal environments (i.e., different groundwater compositions). The consequences of criticality, conditioned on the probability of its occurrence, should also be evaluated as necessary and appropriate, as part of a complete postclosure safety analysis.

Waste Management Operational and Logistical Considerations – A waste management approach that uses DPC direct disposal to dispose of all SNF from existing or decommissioned nuclear plants in the U.S., would likely take longer to implement compared with a re-packaging approach that proceeds at a higher rate of throughput (e.g., 3,000 MTHM/yr). This is because of the decay storage time needed for DPC-based packages to cool sufficiently for disposal (e.g., cool to approximately 10 kW for emplacement in a repository in salt or hard-rock). One advantage of extended operations is that smaller capacity operating facilities could be deployed.

According to logistical simulations presented in Section 9, the fastest timeframe for disposal of approximately 140,000 MTHM (i.e., the salt disposal concept) could be comparable in terms of total duration of repository operations, to the schedule proposed previously for 70,000 MTHM (or equivalent HLW; DOE 2008). This is mainly because the salt concept would not need to be ventilated for decades, for heat removal. The analysis shows that a repository in salt for which emplacement of DPC-based packages begins in 2048, could be closed in fewer than 100 years (although monitoring might continue). Hard rock concepts would be loaded in about the same time but could require up to 100 years of additional ventilation (depending on SNF burnup).

Re-packaging could use smaller canisters containing less SNF, to reduce the cooling time, expediting disposal. When a repository is sited the emplacement thermal power constraints will be better known. At that time the potential for long decay storage times could be mitigated, and throughput increased, by loading bare fuel at the power plants into smaller, purpose-built canisters that could be disposed of sooner. Package size and other requirements could be adjusted to accommodate disposal conditions, such as a geologic settings with limited temperature tolerance and capacity for heat dissipation. More detailed evaluation of scenarios that compare direct disposal of existing DPCs with future re-packaging options is included in the list of R&D needs (Section 10).

Acceptance – Once technical feasibility, safety and cost have been evaluated, it is important to communicate analysis findings, collaborate with industry, discuss safety with regulatory bodies, and promote reviews by external stakeholders. The current, ongoing feasibility evaluation represents the beginning of that process.

Preliminary Feasibility Statement – There appear to be no significant technical feasibility questions associated with repository operations (handling DPCs) at the surface, by analogy to current practices at power plants and storage sites. For transport underground, as concluded in a review of underground transport technology (Fairhurst 2012): "the method of transfer of heavy [175 MT]…loads to the subsurface might not pose an insurmountable technical constraint on siting and design of a geological repository." A significant engineering effort would be needed to develop surface handling and packaging, and underground transport and emplacement capabilities for DPC-based waste packages.

The preliminary analyses presented in this report and summarized above indicate that DPC direct disposal could be technically feasible, at least for certain disposal concepts. Preliminary analysis also suggests that substantial cost savings might be realized compared to re-packaging DPCs, although further analysis is needed to understand the economic consequences associated with the many possible scenarios.

Further technical and logistical analyses are needed to support a more definitive future evaluation of feasibility, and this report provides a survey of topics that should be considered (Section 10). All of the DPC disposal concepts proposed here are probably not equally feasible due to limitations imposed by the geologic setting or engineered materials. Recommendations include steps to narrow the range of alternative concepts to be carried forward in the evaluation.

References for Section 11

DOE (U.S. Department of Energy) 2008. *Yucca Mountain Repository License Application for Construction Authorization*. DOE/RW-0573. Washington, D.C.: U.S. Department of Energy.

DOE (U.S. Department of Energy) 2013. *Strategy for the Management and Disposal of Used Nuclear Fuel and High-Level Radioactive Waste*. January, 2013.

Hansen, F.D., and C.D. Leigh 2011. *Salt Disposal of Heat-Generating Nuclear Waste.* SAND2011-0161. OSTI ID: 1005078. Albuquerque, NM: Sandia National Laboratories. January, 2011.

Hardin, E. 2013. *Temperature-Package Power Correlations for Open-Mode Geologic Disposal Concepts*. SAND2013-1425. Sandia National Laboratories. Albuquerque, NM. February, 2013.

Mariner, P.E., J.H. Lee, E.L. Hardin, F.D. Hansen, G.A. Freeze, A.S. Lord, B. Goldstein and R.H. Price 2011. *Granite disposal of U.S. high-level radioactive waste.* SAND2011-6203. Albuquerque, NM: Sandia National Laboratories.

Nutt W.M. 2011. *Used Fuel Disposition Campaign Disposal Research and Development Roadmap*. U.S. Department of Energy, Used Fuel Disposition R&D Campaign. FCR&D-USED-2011-00065 Rev. 0. March, 2011.

Appendix A. Peer Review Plan and Document Cover Sheets

A peer review of this report was conducted in accordance with *Fuel Cycle Technologies – Quality Assurance Program Document, Revision 2 (12/20/12).* A peer review plan was prepared and approved, and a facsimile is appended below (3 pages). The Fuel Cycle Technologies (FCT) document cover sheet was used as the record of the peer review, and two signed copies of this form (one for each reviewer) are also appended below (2 pages total).

Peer Review Plan - Deliverable: M2FT-13SN0816112 (QRL: 3)

Document Title: Preliminary Report on DPC Disposal Alternatives (30Aug2013)

Purpose of Review: Peer reviews shall include identification of the following: 1) work to be reviewed; 2) scope of the peer review; 3) size and required capabilities of the peer review team (there shall be at least two members on each peer review team); and 4) expected method and reporting schedule.

Comments specific to this document:

The work to be reviewed was performed in the Dual Purpose Canisters-SNL *work package in late FY12 and FY13, and documented in:* Preliminary Report on Dual-Purpose Canister Disposal Alternatives (FY13) (FCRD-UFD-2013-000171 Rev. 0). *The scope of peer review will include all information and analyses described in that report. The peer review team will consist of two members: Robert Jubin, technical staff with ORNL, and Thomas Cotton, Vice President, Complex Systems Group, LLC. Qualifications of these members are discussed below. The peer review will be conducted by email and teleconference, starting with distribution of the report and review instructions, followed by a teleconference to discuss any issues raised by the reviewers, submittal of comments, report revision, reviewer concurrence, and issuance of FCRD-UFD-2013-000171 Rev. 0 in final form.*

Scope of Peer Review:

The scope of Peer Review shall include the following considerations as they apply to the work being reviewed:

1. Determine the reasonableness of the assumptions and validity of inputs that were used as the basis for the research and analyses.

2. Verify the adequacy of experimental requirements and criteria (e.g., acceptance criteria from testing) including the use of any applicable national or international standards described.

3. Verify the appropriateness of the methods and implementing documents used to complete the work under review.

4. Determine if the software applications (e.g., simulation, or computer model) used to complete the work under review are appropriate and adequate.

5. Determine the accuracy of the calculations and final documentation.

6. Determine the reasonableness and validity of the conclusions.

7. Verify that the conclusions are clearly stated such that misinterpretation is minimized. Identify any different conclusions that can be drawn from the results presented.

8. Verify that any uncertainty in the results is clearly and adequately discussed.

Additional criteria may be defined by the team and shall be defined in the review criteria documentation.

Comments:

Experiments were not performed for this work, so item 2 does not apply. The document to be reviewed is a preliminary report, to be issued part-way through a multi-year study of DPC disposal feasibility. The review should emphasize whether the work reported so far is complete *in its approach and thorough in its identification of technical issues. Comments that involve work scope may be responded to by incorporating recommendations that they will be addressed in detail by future work.*

Qualification Requirements for Peer Reviewers:

Peer reviews shall be conducted by individuals who have independence from the work under review. Independence means that the individual was not involved as a participant, supervisor, or advisor in the work under review and is, to the extent practical, free from other conflicts of interest.

The number of reviewer(s) is commensurate with the complexity of the work to be reviewed, its importance to program objectives, the number of technical disciplines involved, and the degree to which the subject issue is considered controversial by stakeholders and differing viewpoints are strongly held within the applicable technical and scientific community concerning issues under review. The supervisor, manager, or NTD of the performer of the work shall select peer reviewer(s) based on the complexity of the work being reviewed. Peer reviewers are individuals who meet at least one of the following criteria as judged by the responsible manager:

- Have adequate academic education in the same technical discipline in which the work is performed or in a closely related field, or have adequate work experience and technical activity in a related discipline.
- Have demonstrated evidence of proposing and solving engineering, experimental, or theoretical problems that are recognized as valid by the community of technical peers.
- Have contributed to the body of knowledge within a technical discipline such as publishing research results in the proceedings of scientific meetings or in professional journals.

The supervisor, manager, or NTD of the performer of work being peer reviewed must verify that peer reviewer(s) are qualified in accordance with the requirements herein. FCT MOs may require approval of peer reviewers, which should be called out in applicable work packages or otherwise formally requested.

Comments:

The reviewers were not involved as participants, supervisors, or advisors in the work reviewed, and to our knowledge, are free from other conflicts of interest. Selection of two reviewers is commensurate with the complexity, importance, technical disciplines, and stakeholder interest in the subject matter, so long as the reviewers have broad knowledge and experience in the back end of the nuclear fuel cycle in the U.S., as these reviewers do.

Both reviewers have credentials in relevant disciplines: Dr. Cotton is Vice President of Complex Systems Group, LLC, and has 35 years experience in analyzing technical and policy issues in all areas of high-level radioactive waste and spent fuel management. He has a degree in Electrical Engineering, and a Ph.D. in Engineering-Economic Systems. Dr. Jubin is currently Project Manager for the Advanced Fuel Cycle Science and Technology program at Oak Ridge National Laboratory. He has a Ph.D. in Chemical Engineering and over 35 years experience in nuclear engineering R&D. Both have extensive experience in proposing and solving relevant problems in nuclear waste disposal and fuel cycle technology. Evidence of their previous recognition in these *areas is provided by extensive publication and presentation records. Accordingly, all three of the criteria listed above apply to both reviewers.*

Documenting Peer Reviews:

The Peer Review will be documented using the current version of the FCT document cover sheet to show signature approval by peer reviewers.

8/26/2013

Ernest Hardin, Work Package Manager

Robert MacKinnon for

Kevin McMahon, Responsible Manager (SNL)

FCT Quality Assurance Program Document

Appendix E
FCT Document Cover Sheet

Name/Title of Deliverable/Milestone	Preliminary Report on Dual-Purpose Canister Disposal Alternatives (FCRD-UFD-2013-000171 Rev. 0)		
Work Package Title and Number	**FT-13SN081611** Dual-Purpose Canisters - SNL		
Work Package WBS Number	1.2.08.16	Milestone Number	**M2FT-13SN0816112**

Responsible Work Package Manager
Ernest Hardin E Hardin
(Name/Signature)

30Aug2013
(Date Submitted)

Quality Rigor Level for Deliverable/Milestone	☒ QRL-3	☐ QRL-2	☐ QRL-1 ☐ Nuclear Data	☐ N/A*

This deliverable was prepared in accordance with Sandia National Laboratories
(Participant/National Laboratory Name)

QA program which meets the requirements of
☒ DOE Order 414.1 ☐ NQA-1-2000 ☐ Other:

This Deliverable was subjected to:
☐ Technical Review ☒ Peer Review

Technical Review (TR)
Review Documentation Provided
☐ Signed TR Report, or
TR Report No.:
☐ Signed TR Concurrence Sheet (attached), or
☐ Signature of TR Reviewer(s) below

Peer Review (PR)
Review Documentation Provided
☐ Signed PR Report, or
PR Report No.:
☐ Signed PR Concurrence Sheet (attached), or
☒ Signature of PR Reviewers below

Name and Signature of Reviewers

Robert Jubin

Thomas Cotton
Thomas Cotton 08/28/2013
(Name/Signature) (Date)

*Note: In some cases there may be a milestone where an item is being fabricated, maintenance is being performed on a facility, or a document is being issued through a formal document control process where it specifically calls out a formal review of the document. In these cases, documentation (e.g., inspection report, maintenance request, work planning package do
cumentation, or the documented review of the issued document through the document control process) of the completion of the activity along with the Document Cover Sheet is sufficient to demonstrate achieving the milestone. QRL for such milestones may also be marked N/A in the work package provided the work package clearly specifies the requirement to use the Document Cover Sheet and provide supporting documentation.

Page E-1

FCRD-TIO-2011-000050, Rev. 0

FCT Quality Assurance Program Document

Appendix E
FCT Document Cover Sheet

Name/Title of Deliverable/Milestone	Preliminary Report on Dual-Purpose Canister Disposal Alternatives (FCRD-UFD-2013-000171 Rev. 0)
Work Package Title and Number	**FT-13SN081611** Dual-Purpose Canisters - SNL
Work Package WBS Number	1.2.08.16
Milestone Number	**M2FT-13SN0816112**

Responsible Work Package Manager
Ernest Hardin E Hardin 30Aug2013
(Name/Signature) (Date Submitted)

Quality Rigor Level for Deliverable/Milestone	☒ QRL-3	☐ QRL-2	☐ QRL-1 ☐ Nuclear Data	☐ N/A*

This deliverable was prepared in accordance with Sandia National Laboratories
(Participant/National Laboratory Name)

QA program which meets the requirements of
☒ DOE Order 414.1 ☐ NQA-1-2000 ☐ Other:

This Deliverable was subjected to:
☐ Technical Review ☒ Peer Review

Technical Review (TR)
Review Documentation Provided
☐ Signed TR Report, or
TR Report No.:
☐ Signed TR Concurrence Sheet (attached), or
☐ Signature of TR Reviewer(s) below

Peer Review (PR)
Review Documentation Provided
☐ Signed PR Report, or
PR Report No.:
☐ Signed PR Concurrence Sheet (attached), or
☒ Signature of PR Reviewers below

Name and Signature of Reviewers

Robert Jubin R. T. Jubin 28 Aug 2013
Thomas Cotton

(Name/Signature) (Date)

*Note: In some cases there may be a milestone where an item is being fabricated, maintenance is being performed on a facility, or a document is being issued through a formal document control process where it specifically calls out a formal review of the document. In these cases, documentation (e.g., inspection report, maintenance request, work planning package do
cumentation, or the documented review of the issued document through the document control process) of the completion of the activity along with the Document Cover Sheet is sufficient to demonstrate achieving the milestone. QRL for such milestones may also be marked N/A in the work package provided the work package clearly specifies the requirement to use the Document Cover Sheet and provide supporting documentation.

Page E-1

FCRD-TIO-2011-000050, Rev. 0