A Salt Repository Concept for CSNF in 21-PWR Size Canisters

Fuel Cycle Research & Development

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This report reflects research and development to explore technical concepts. No inferences should be drawn from it regarding future actions by DOE. This is a technical report that does not take into account the contractual limitations under the Standard Contract for Disposal of Spent Nuclear Fuel and/or High-Level Radioactive Waste (Standard Contract) (10 CFR Part 961). Under the provisions of the Standard Contract, DOE does not consider spent nuclear fuel in multi-assembly canisters to be an acceptable waste form, absent a mutually agreed to contract amendment. To the extent discussions or recommendations in this report conflict with the provisions of the Standard Contract, the Standard Contract provisions prevail.

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Acronyms

ALARA	As low as reasonably achievable
BWR	Boiling-water reactor
CALVIN	CRWMS Analysis and Logistics Visually Interactive Model
CRWMS	Civilian Radioactive Waste Management System
CRCF	Canister Receipt and Closure Facility
CSNF	Commercial spent nuclear fuel
DBE	German Service Company for the Construction and Operation of Waste Repositories
DEAB	Direct Disposal of Spent Fuel Elements
DIREGT	Direct Disposal of Transport and Storage Canisters
DPC	Dual-purpose canister
EBS	Engineered barrier system
ENTRIA	Disposal Options for Radioactive Residues: Interdisciplinary Analyses and Development of Evaluation Principles
FEP	Feature, event or process
GNS	Gesellschaft für Nuklear-Service mbH
GTCC	Greater than Class C
HLW	High-level waste
ISF	Interim Storage Facility
KOSINA	Conceptual Design, Safety Concept for Bedded Salt in Germany
LS-LSR	Low-stress, low strain-rate
MD	Munson-Dawson multi-mechanism creep model
MT	Metric ton
NAS	National Academies/National Research Council
NIST	National Institute of Standards and Technology
NRC	U.S. Nuclear Regulatory Commission
NWPA	Nuclear Waste Policy Act
ONDRAF/NIRAS	Belgian Agency for Radioactive Waste and Enriched Fissile Materials
PWR	Pressurized-water reactor
RCRA	Resource Conservation and Recovery Act
R&D	Research and development
RMEI	Reasonably maximally exposed individual
ROM	Rough-order-of-magnitude (for cost estimates)
SNF	Spent nuclear fuel
SNL	Sandia National Laboratories
TAD	Transport-aging-disposal canister
TEV	Transport-emplacement-vehicle
TOM	Transportation Operations Model

TRU	Transuranic waste
TSC	Transportation and storage canister or cask
TSPA	Total system performance assessment
WIPP	Waste Isolation Pilot Plant
WRPF	Waste Receipt and Packaging Facility

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Abstract

The most straightforward concept for disposal of large, heavy packages containing commercial spent nuclear fuel (CSNF) in a repository in bedded salt, would be to emplace them directly on the floor in emplacement tunnels. In-tunnel axially aligned horizontal emplacement would minimize excavated volume and avoid drilling of large-diameter emplacement boreholes. A similar concept was proposed in Germany for direct disposal of POLLUX® canisters. The repository would be constructed at a depth of 500 to 1,000 m for isolation from the surface, and for sufficient overburden stress to ensure creep reconsolidation of repository openings. It could entail modular panels of emplacement tunnels arranged on headings oriented in cardinal directions from a central core, to accommodate the estimated 140,000 MTU total U.S. CSNF inventory. To do so, the overall area of the repository layout would be approximately 20 km^2. Many layouts are possible, but the approach should be modular, excavation should be deferred for as long as possible to avoid maintenance, and the emplacement areas should share support facilities and shafts. Vertical shafts would be used in accordance with mining practice in sedimentary basins. Large diameter shafts would be needed for ventilation exhaust and waste transport, with smaller shafts for waste salt removal, men & materials, and ventilation intake.

The spacing between disposal tunnels as estimated from thermal modeling, seeks to limit the maximum average areal thermal load in the panels to 11 W/m^2 to control long-term heat buildup in the host rock. Peak salt temperature would occur within a few years and would be dominated by each waste package locally, simplifying thermal management. There would be some flexibility to decrease the package spacing or increase the emplacement thermal power limit. Backfilling emplaced waste packages immediately with mine-run crushed salt would provide shielding and expedite reconsolidation. This arrangement would isolate adjacent waste packages from one another by the intervening backfill, especially after it reconsolidates and its properties approach those of intact salt. After the repository is fully loaded and the performance confirmation program is complete, activities to permanently close the repository would be initiated. During closure operations all openings in the host salt would be backfilled, then shafts would be sealed, and boreholes plugged. Plans for the Waste Isolation Pilot Plant (WIPP) show how sealing and plugging could be done. A monitoring program could continue for 50 years or longer after repository closure.

With an emplacement thermal power limit of 10 kW per waste package, nearly all the CSNF that is projected to be produced by the current fleet of reactors in the U.S. could be emplaced over a period of approximately 50 years starting in calendar 2048. No barriers to implementation in a reasonable timeframe have been identified from this generic analysis.

Engineering challenges include: 1) shaft construction; 2) a very-large capacity shaft hoist; 3) overpack design; 4) a transport-emplacement-vehicle (TEV) for transporting waste packages once they are underground and emplacing them remotely; and 5) remotely operated equipment for emplacing backfill. The method of shaft construction would depend on site-specific conditions, and could involve freezing the subsurface. A shaft hoist with payload capacity of 175 MT seems technically and economically feasible based on development work in Germany, and it would be the largest hoist of its kind. The function of disposal overpacks would be to provide reliable containment during repository operations, which could be accomplished using a corrosion allowance material such as a low-carbon steel. Development effort would be needed to determine overpack thickness (e.g., 7 to 20 cm) that can resist corrosion and loading from salt creep, to

provide containment throughout the repository operational period. The transport-emplacement-vehicle (TEV) would be similar to previous concepts, particularly one option proposed for a Yucca Mountain repository. It would move over a rough salt floor on independently driven and steered wheels, and carry heavy shielding in addition to a waste package.

By analogy to the safety case for the WIPP in New Mexico, human intrusion is likely to be the dominant mode of radionuclide release from the repository. Treatment of human intrusion for a CSNF repository in salt could depend on promulgation of site-specific changes in the regulations. Radionuclide release and migration would be quite limited for undisturbed conditions. There may be opportunities for improved understanding of salt performance with waste heating, based on future *in situ* testing in an underground salt research laboratory.

This report also discusses a developing area of salt rock mechanics that involves low-stress, low strain-rate creep that might cause large, heavy waste packages to slowly sink. Site-specific sampling, testing, and modeling would be used to determine if the mechanism is important enough to merit consideration in design, or inclusion in performance assessment.

Part of engineering design and postclosure safety assessment for a CSNF repository in salt would be to implement a methodology to show that the probability of a criticality event in the repository, when waste packages eventually breach and are flooded, is less than the probability screening threshold for performance assessment. In the methodology, a criticality analysis would be performed for waste packages in the repository, incorporating measures that could be introduced as needed to limit reactivity, for example using fuel selection and loading rules, and crediting the absorption of thermal neutrons by natural chlorine in the environment. A similar analysis has been underway for CSNF stored in dual-purpose canisters. Ideally the strategy would be developed prior to actually loading SNF assemblies into canisters used in waste packages for disposal.

1. Purpose and Scope

This report describes a disposal concept for commercial spent nuclear fuel (CSNF) deep underground in bedded salt, using fuel canisters similar in size and construction to previously described transport-aging-disposal (TAD) canisters (OCRWM 2008). This includes a review of previous literature on concepts (Section 2) for disposal of spent nuclear fuel (SNF) or high-level waste (HLW) in salt formations, and a recommended approach for safe and effective disposal of CSNF (Sections 5, 6 and 7). The recommended approach includes the waste package (disposal overpack), repository construction and layout, and the waste shaft hoist. CSNF is assumed to be packaged in sealed multi-purpose canisters containing 21 fuel assemblies from pressurized water reactors (PWRs) or equivalent fuel (in terms of volume and mass) from boiling water reactors (BWRs). The recommended approach could also be suitable for safe disposal of damaged fuel (in damaged fuel cans), high-level defense wastes, and low-level waste (LLW) from the disposal facility itself.

A number of engineering challenges may need to be addressed to successfully implement CSNF disposal in bedded salt, including shaft construction and hoist capacity, handling of large, heavy waste packages underground, shielding, waste package overpack containment longevity, and remotely operated equipment for emplacement and backfilling. In addition, potentially important postclosure (e.g., 10,000 years) performance issues include inadvertent future human intrusion, criticality control, and related features, events and processes (FEPs) (Section 4). Each of these is addressed with discussion and analysis (Sections 3 and 4).

Logistical analysis is included in this report mainly to estimate how CSNF cooling before emplacement would constrain the duration and throughput of repository operations (Section 8).

Assumptions for a CSNF Repository in Bedded Salt – The repository could be located in a geologic setting such as the Permian Basin, at a depth of approximately 500 to 1,000 m below ground surface. Access would be provided by mined shafts (in lieu of ramps; raise-boring may also be used for shafts).

CSNF for disposal would be of sufficient age to meet waste package emplacement thermal limits, for a range of initial enrichment (up to 5%) and burnup (up to 62.5 GW-d/MTU). Aging could take place in fuel pools, or in other canisters such as dual-purpose canisters (DPCs) before the fuel is transferred to 21-PWR size canisters suitable for storage, transportation, and disposal (which could also receive 44 BWR assemblies using an appropriate fuel basket).

CSNF would be canistered and weld-sealed at upstream facilities, in 21-PWR size canisters closely resembling the proposed Yucca Mountain TAD canister (OCRWM 2008). The facility for doing so could be co-located with the repository, or located elsewhere. Discussion of such a facility is beyond the scope of this report. At the repository surface facilities these sealed canisters would be further packaged into overpacks for disposal. The overpacks would also be weld-sealed, and would perform functions specific to the disposal environment. The sealed canister combined with a sealed overpack is referred to here as the waste package. The term “TAD-type canister” is used in a general sense to refer to 21-PWR size packaging solutions for a salt repository, using canisters that are closely similar to the Yucca Mountain TAD canister specification (OCRWM 2008).

For this report TAD canister characteristics are mainly limited to size and weight, but also include clearance gaps and materials of canister shell and basket construction (Sections 4 and 5).

Throughput and timing of CSNF disposal is analyzed in Section 8; including some scenarios in which CSNF in DPCs such as those that presently exist and represent a large fraction of the total CSNF inventory, are disposed of directly without re-packaging into 21-PWR size canisters. These DPC scenarios are intended as a check on the potential need for additional CSNF aging time, to cool DPCs that contain more CSNF (e.g., 32 to 37 PWR assemblies). DPCs are not licensed for disposal, but technical feasibility of direct disposal has been evaluated (Hardin et al. 2013).

For logistical analysis (Section 8) the repository is assumed to be open for emplacement operations in 2048, and could operate beyond calendar 2100 in order to dispose of the full U.S. CSNF inventory (~140,000 MTU). The nuclear power utilities are assumed to begin packaging and storing spent fuel in compliant 21-PWR size canisters in calendar 2025 (except for certain scenarios in which this begins in 2036).

Statutory/Regulatory Framework – The current framework for a future geologic repository for CSNF in salt begins with the *Nuclear Waste Policy Act* (NWPA, as amended), and regulations from the U.S. Nuclear Regulatory Commission (NRC) and the U.S. Environmental Protection Agency (EPA):

- 10CFR20 (Standards for Protection Against Radiation)
- 10CFR50 (Domestic Licensing of Production and Utilization Facilities)
- 10CFR60 (Disposal of High-Level Radioactive Wastes in Geologic Repositories)
- 40CFR191 (Environmental radiation protection standards for management and disposal *of spent nuclear fuel, high-level and transuranic radioactive wastes)*

Both 10CFR60 and 40CFR191 are still in force and in principle could be applied to future repositories. However, the evolution in strategy adopted by the EPA and NRC for site-specific regulations for a repository in tuff at Yucca Mountain, 40CFR197 and 10CFR63, would likely be adopted for a future repository. Any changes to the EPA standards for repositories other than at Yucca Mountain would likely change 40CFR191, and be reflected in corresponding changes to NRC regulation 10CFR60. In particular, EPA and NRC have evolved these regulations to rely on mean annual dose computed from total system performance assessment (TSPA), instead of subsystem requirements such as engineered barrier system (EBS) containment time, minimum ground water travel time to the accessible environment, and EBS fractional release rates. Consequently, NRC stated when promulgating 10CFR63 that the “generic Part 60 requirements will need updating” (Rubenstone 2012; NRC 2001). Furthermore, NRC staff have suggested in presentations to the Blue Ribbon Commission on America’s Nuclear Future and the Nuclear Waste Technical Review Board (McCartin 2010; 2012), that regulations for future repositories would likely look similar to 10CFR63.

The National Academies/National Research Council (NAS) recommendations for standards specific to a repository in unsaturated tuff developed pursuant to the *Energy Policy Act of 1992*, may be applicable to other repositories for SNF and HLW even though this act only addressed a repository at Yucca Mountain. If so, then licensing of future repositories will require demonstration of compliance with a peak dose standard for a period of geologic stability (~10^6 yr was recommended by the NAS).

Other examples of possible future changes to repository regulations for a site other than Yucca Mountain (if authorized by legislation) that could affect licensing of a CSNF repository in salt, include:

- The accessible environment for performance assessment of DPC disposal will be nominally 5 km away from the boundary of the repository (§63.302).
- In general, FEPs and scenario classes formed from FEPs will be retained or omitted based on their influence on performance in the first 10^4 years after repository closure (§63.114). The criterion for screening out FEPs and scenario classes based on probability will remain at 10^{-8} in any one year. Other events such as seismic ground motion and climate change may be projected beyond 10^4 years depending on site-specific requirements (§63.342).
- NRC requirements for barriers of the disposal system will remain similar: Licensee must identify components of the disposal system that are important for isolation and demonstrate their performance (§63.115). However, no subsystem containment requirements will be specified.
- Inadvertent human intrusion will not be included in the probabilistic dose calculations for evaluating system performance against performance objectives. Rather, individual dose to the reasonably maximally exposed individual (RMEI) will be assessed, conditioned on the intrusion. The calculation would be done as discussed in Section 4.3.

In addition, future regulations are likely to include certain provisions of 10CFR20 and 10CFR50, as current regulations do.

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2. Background on Salt Repository Concepts

A geologic repository for CSNF in bedded salt would require several tens of kilometers of tunnels, plus access and ventilation shafts, and a waste handling route. It would be constructed at a depth of approximately 500 to 1,000 m, and it would occupy a plan area of approximately 20 km^2 depending on the disposal capacity and layout. The hydrologic description of candidate salt formations would likely be brine saturated, but the host salt would have such low porosity, low permeability, and low brine content that it would behave as though mostly dry.

Underground salt tunnels are readily mined with mechanized equipment such as "roadheaders" (boom cutters). Underground equipment typically runs on rubber tires to minimize maintenance associated with floor heave. Mines in evaporite formations (e.g., potash mines) are typically accessed by vertical shafts that minimize exposure to incompetent and/or water-bearing strata. Freezing the overburden before and during shaft construction was used at Gorleben, and was proposed for repository site characterization at the Deaf Smith site (DOE 1988). In principle, an evaporite section could also be accessed by ramp but at significantly greater cost if special measures such as freezing are needed. The question of final closure and sealing also favors shafts because ramps comprise about 10 times the volume, and 10 times the area of exposure to any water-bearing rock units. A comparison of shafts *vs*. ramps for the Canadian nuclear waste management authority supports the use of shafts (Lee and Heystee 2014). The use of shafts depends also on the availability and safety of hoists with sufficient capacity, which would be developmental as payload capacity exceeds 50 MT (Fairhurst 2012). A conceptual design for such a hoist is featured in this report (Section 7).

Evaporite sequences typically include layers of halite, other chloride minerals, anhydrite, and clay. The depth would be selected to optimize the host halite unit thickness, for construction feasibility, and for isolation of the waste. Some combination of elevated temperature and burial depth is needed to ensure that the repository closes fully due to salt creep within a few decades (e.g., the WIPP is situated in a thick halite section of the Salado formation, at a depth of approximately 620 m where the *in situ* temperature is 27°C).

Salt is known to decrepitate at temperatures greater than about 250°C due to the internal pressure developed in fluid inclusions. Peak temperature for intact salt is typically limited to 200°C to avoid decrepitation, and to avoid degrading other minerals that may be adjacent to the host salt layer and have greater sensitivity to temperature than halite (Hardin et al. 2012; Bollingerfehr et al. 2013). Crushed salt is relatively insensitive to decrepitation (already degraded by comminution) which is helpful because its thermal conductivity is greatly reduced by crushing. Crushed salt backfill could be allowed to achieve peak temperatures greater than 200°C while maintaining this limit for intact salt or adjacent layers. The low thermal conductivity of crushed salt tends to increase the temperature near heat-generating waste packages, for a few tens to hundreds of years, but with creep consolidation the properties of crushed salt gradually return to those of unexcavated, intact salt.

The following brief review is modified from a previous report that discussed repository options for U.S. defense wastes (Matteo et al. 2016).

Repository Site Near Lyons, Kansas - A conceptual design for a repository for disposal of transuranic (TRU) waste and HLW was completed in 1971, to be implemented in a former domal salt mine where R&D (Project Salt Vault) had been conducted between 1963 and 1967 (Mora

1999). The disposal concept called for emplacement of vitrified HLW canisters in boreholes drilled vertically in the floor, in galleries constructed for the purpose. The repository project was cancelled a few years later because of the presence of oil and gas wells, some unmapped, and because of ongoing solution mining activity in the vicinity.

Waste Isolation Pilot Plant – The WIPP was initially conceived as a repository for TRU waste, with the technical possibility of extending the mission to defense HLW. Investigation of the geomechanical behavior of the WIPP salt, including heated tests, was planned in 1982 and performed during the 1980s as the WIPP underground facility was developed (Mora 1999). Borehole heater tests were performed to simulate disposal of heat-generating HLW using vertical and horizontal configurations. Concurrent with WIPP *in situ* tests, salt response was also investigated in domal salt underground at Avery Island, Louisiana (Stickney and Van Sambeek 1984). These experiments were eventually closed out after 1987 when the U.S. Congress shifted the disposal path for defense HLW to coincide with the repository for CSNF being sited at Yucca Mountain and developed under the *Nuclear Waste Policy Act of 1982*.

U.S. Salt Repository Project – A comprehensive conceptual design for a mined repository in salt was developed by the U.S. Salt Repository Project in the 1980s (DOE 1988; Fluor 1987). The concept was developed for bedded salt in the northern Permian Basin at a depth of about 790 m. The salt in this setting is interbedded with clastic sediments and anhydrite, and overlain by an extension of the Ogallala Aquifer. Access to the underground facility was to be through shafts excavated through the overlying sediments (some water-bearing) using a freezing method in conjunction with a liner of concrete or cast iron.

The repository level was to have been excavated using a roadheader (boom-cutter). Waste packages would have been emplaced in boreholes either vertically in the floor, or horizontally in pillars between access tunnels. Access tunnels would be large openings, e.g., 6 m wide and 7 m high for vertical emplacement, spaced approximately 36 to 51 m apart. Vertical emplacement boreholes would be just large enough for waste packages (about 1 m in diameter) with depth of approximately 6 to 8 m depending on package length.

Waste package concepts for the Salt Repository Project were developed in two configurations, for borehole emplacement, and as self-shielding packages for in-tunnel emplacement (Westinghouse 1982). For HLW emplacement in boreholes, overpacks were to consist of approximately 10 cm of low-carbon steel as a corrosion allowance material, with the objective to provide containment lifetime on the order of 1,000 yr, consistent with EBS performance requirements at §10CFR60.113(a)(ii)(A). A thin (2.5mm) outer corrosion-resistant layer of Ti alloy was also evaluated for additional assurance. Peak temperatures less than 100°C were calculated for the waste form, overpack, and salt. The self-shielding overpack concept consisted of gray cast iron or cast steel, with thickness of 30 to 47 mm, containing a single pour-canister of HLW glass.

German Reference Concepts – Three concepts for CSNF disposal in salt were presented in a recent German update (Bollingerfehr et al. 2013): 1) in-tunnel disposal of large, self-shielding POLLUX® waste packages containing consolidated fuel rods; 2) vertical borehole emplacement of smaller BSK canisters containing consolidated fuel rods (Filbert et al. 2010); and 3) direct disposal of self-shielding CASTOR® casks currently used for storage and transportation of intact fuel assemblies. Waste streams for disposal also included vitrified reprocessing waste, and incidental wastes such as fuel assembly structural parts resulting from rod consolidation.

The POLLUX cask was designed specifically for disposal, with capacity for rod-consolidated fuel from ten PWR assemblies, but the cask can also be fitted for waste from reprocessing. The reference in-tunnel concept would consolidate SNF rods into POLLUX casks, and emplace them horizontally on the floor in long disposal tunnels, backfilled immediately with crushed salt (Figure 2-1). Existing reprocessing waste (i.e., vitrified glass) would be packaged in 887 additional POLLUX casks (Bollingerfehr et al. 2013).

The CASTOR-V cask type typically contains 19 PWR assemblies (Graf et al. 2012). Approximately 2,632 CASTOR casks with intact SNF from commercial power and research reactors were proposed to be emplaced in tunnels and backfilled (DIREGT concept) if a shaft hoist with sufficient capacity was developed. Use of POLLUX casks would require 85 MT hoist capacity, whereas CASTOR-V casks would require 175 MT capacity (Hardin et al. 2013a).

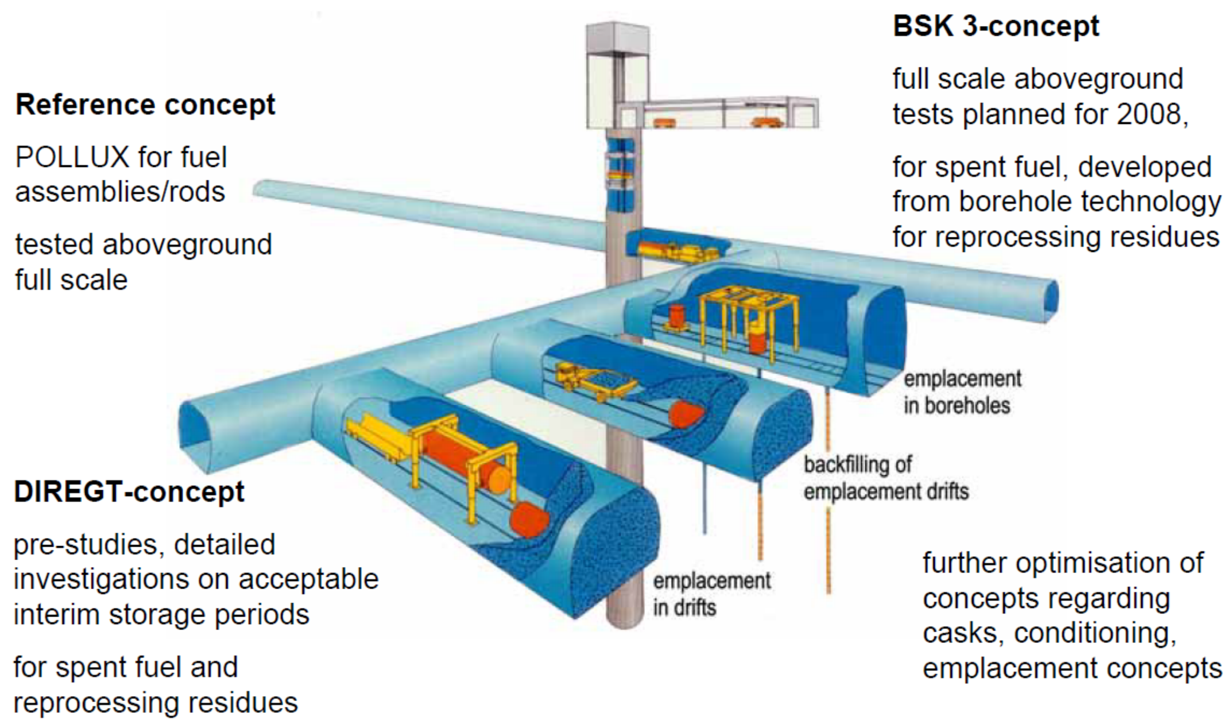

Figure 2-1. Gorleben disposal concepts for consolidated SNF rods in POLLUX casks (reference concept) and BSK casks (emplacement in boreholes), and for intact fuel in CASTOR casks (DIREGT concept) (from Bollingerfehr and Filbert 2010).

The alternative borehole concept would use the smaller BSK-3 canisters for all types of waste (Figure 2-1). These canisters would have an outer diameter of 52 cm and length of 5.06 m (Bollingerfehr et al. 2013). They would be emplaced vertically in 60-cm diameter, steel-lined boreholes. The BSK-3 canister is designed to contain SNF rods from three PWR assemblies, or three pour canisters of reprocessing waste. Emplacement boreholes would be drilled vertically downward from access tunnels at the repository level (e.g., starting at 870 m below the surface at Gorleben) to an additional depth of about 300 m. Approximately 50 canisters would be emplaced in each borehole, so that a total of about 200 boreholes would be needed for all waste types, occupying a plan area of 1 to 2 km^2. Note that this concept is for domal salt, and if applied to bedded salt the vertical borehole length could be limited by the host unit stratigraphy to a few tens of meters or less.

No decision has yet been made favoring any of the German disposal concepts reported for salt. Advantages of the POLLUX concept include some shielding, while minimizing size and weight, possible use for storage before disposal, and larger capacity so that fewer packages are needed (Bryan et al. 2011). Disadvantages include the extra cost of rod consolidation, extra hoisting capacity for the waste shaft (*vs*. smaller waste packages) and greater heat output which requires some attention to decay storage. Advantages of the BSK-3 concept include lower heat output, and similarity to packaging for HLW. Disadvantages also include the cost of rod consolidation, the need for re-packaging after storage, lack of shielding, difficulty of retrieving ~50 canisters from each borehole if required, and limitation of the BSK concept to domal salt or very thick sequences of bedded salt.

A variant concept was also generated whereby large POLLUX or CASTOR-V casks would be emplaced in horizontal boreholes instead of on the tunnel floor (Bollingerfehr et al. 2013). The motivation for borehole emplacement would be to provide more immediate, verifiable consolidation of the host salt around the packages. Experience with remote-handled waste canister emplacement at the WIPP has shown dimensional stability of emplacement boreholes to be problematic during emplacement operations (Nelson and White 2008). The concept calls for a novel mechanism to facilitate sliding packages into slightly inclined emplacement boreholes (discussed in Section 3.2).

The recent KOSINA (*Conceptual Design, Safety Concept for Bedded Salt in Germany*) project represents a change in emphasis for repository siting in Germany, to include bedded salt (project guidelines are described by Matteo and Hansen 2016). At the same time the independent ENTRIA (*Disposal Options for Radioactive Residues: Interdisciplinary Analyses and Development of Evaluation Principles*) project was undertaken to investigate siting methods that include non-technical factors. This includes a new emphasis on long-term monitoring to support future retrieval decisions. A new general concept would allow both in-tunnel emplacement of larger packages, and borehole emplacement of smaller ones, with long-term monitoring supported by a network of tunnels on a separate level of the repository (Figure 2-2).

Generic Salt Repository Study – A scoping study was performed in 2009-2011 to support planning for a closed, commercial nuclear fuel cycle that would involve reprocessing and disposal of vitrified HLW (Carter et al. 2011). Glass pour canisters with diameter of 61 cm and length of 2.74 m would be directly disposed of using an alcove mode of emplacement. The concept is essentially similar to the German reference in-tunnel concept described above, but without shielding or overpack on the HLW canisters, and using an alcove layout. Each access tunnel or room would be 3.3 m wide, 3.0 m high, and 135 m long, accommodating 16 alcoves, each containing one HLW canister. A panel would total 236 alcoves, and 80 panels would be filled during 40 years of operation (18,800 canisters). Other aspects of the layout such as service tunnels, ventilation, and shafts would be similar to the present design for the WIPP. Waste packages would be emplaced on the floor and covered immediately with crushed salt. The heat output of each HLW canister was assumed to be as much as 8.4 kW at emplacement, based on fresh HLW glass from the Savannah River Site. The alcove layout was selected to help dissipate this heat, spreading the packages out on a 12-m grid (repository-average maximum thermal load of 39 W/m^2). Thermal analysis showed that a 200°C peak temperature limit could be met in the host salt, but that HLW canister surface temperatures would approach 300°C (Clayton and Gable 2009).

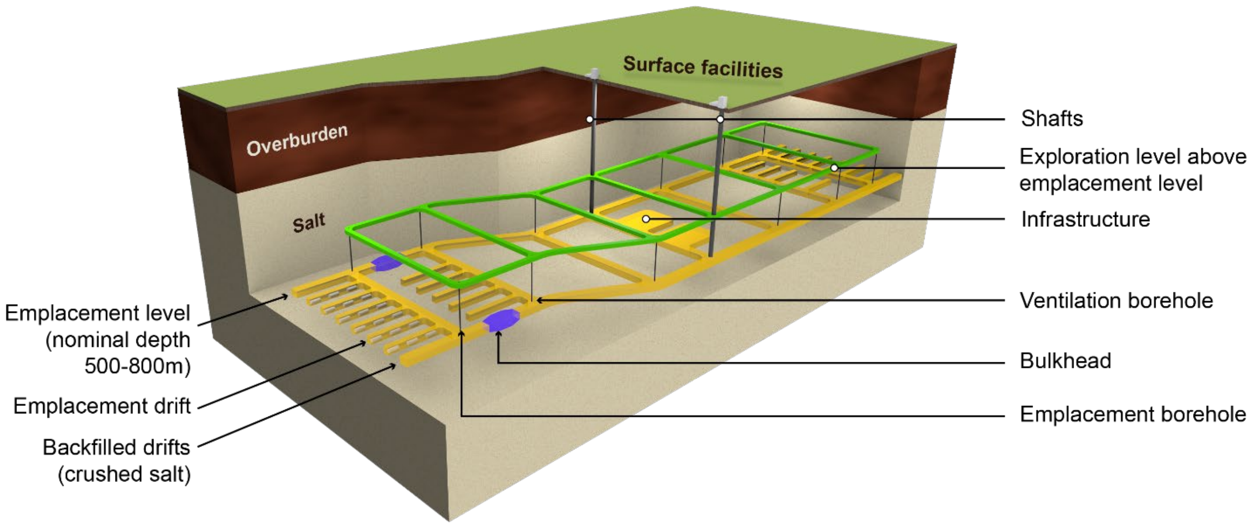

Figure 2-2. Bedded salt repository concept with monitoring from a separate level, developed for the ENTRIA project (https://www.entria.de).

Remotely operated and shielded equipment would be developed to move HLW canisters into position at the back of each alcove, situate them transversely on the floor, and then cover them with crushed salt. In addition, low-heat output waste from reprocessing could also be placed in the same alcoves and covered during backfilling. The access tunnel between alcoves could remain open for some time but would eventually require backfilling (in a few years) as host rock heating caused the rate of salt creep to increase.

Carter et al. (2011) were the first in the U.S. to propose direct disposal of HLW pour canisters without disposal overpacks and without emplacement boreholes, and using crushed salt for shielding after emplacement. The concept could be used in domal salt if sufficient salt dome volume is available to accommodate the extensive layout. Alternatively, the HLW could be aged longer before emplacement so the layout could be more compact. The report authors stated a preference for bedded salt because of the more extensive layout that would be feasible. This generic salt repository concept was adapted later for disposal of larger, heavier waste packages containing CSNF (Hardin et al. 2012).

Concept Refinement – Further refinement of the generic salt repository concept proposed using long linear disposal tunnels instead of alcoves, substantially reducing the excavated volume per each waste package. The emplacement scheme calls for fully excavating a disposal tunnel, then emplacing waste packages and backfill starting at the far end and retreating until the tunnel is full. Tunnel dimensions would provide sufficient clearance for transverse emplacement of 4.6-m long waste packages. Spacing between packages would be varied from approximately 1 to 10 m or more depending on heat output of HLW glass pour-canisters and other waste forms. Crushed salt for backfilling each tunnel would come from excavating the next one. The need for remotely operated and shielded equipment would be similar to the alcove concept discussed above. Shielding by crushed salt is analyzed in Section 6.4.

Further analysis of repository concepts for defense wastes, including in bedded salt, was presented by Matteo et al. (2016). They concluded that in-tunnel emplacement of waste packages would be viable considering both preclosure operational safety and postclosure waste isolation.

Emplacement of Large, Heavy SNF Waste Packages – In-tunnel axial emplacement was proposed in a subsequent study as a solution for large waste packages weighing 80 MT or more (Hardin et al. 2013). The advantage of axial emplacement is that the transporter can be more compact, straddling each waste package instead of projecting it ahead (Figure 2-3). This applies to smaller waste packages as well, to limit transporter size and weight.

The 2013 study also proposed to emplace each waste package into a semi-cylindrical cavity mined into the tunnel floor (Figure 2-3), to enhance heat transfer and lower the peak salt temperature. These features could decrease the peak salt temperature by more than 50°C, or conversely, reduce the duration of thermal aging required (Hardin et al. 2013, Figures 5-3 and 5-5).

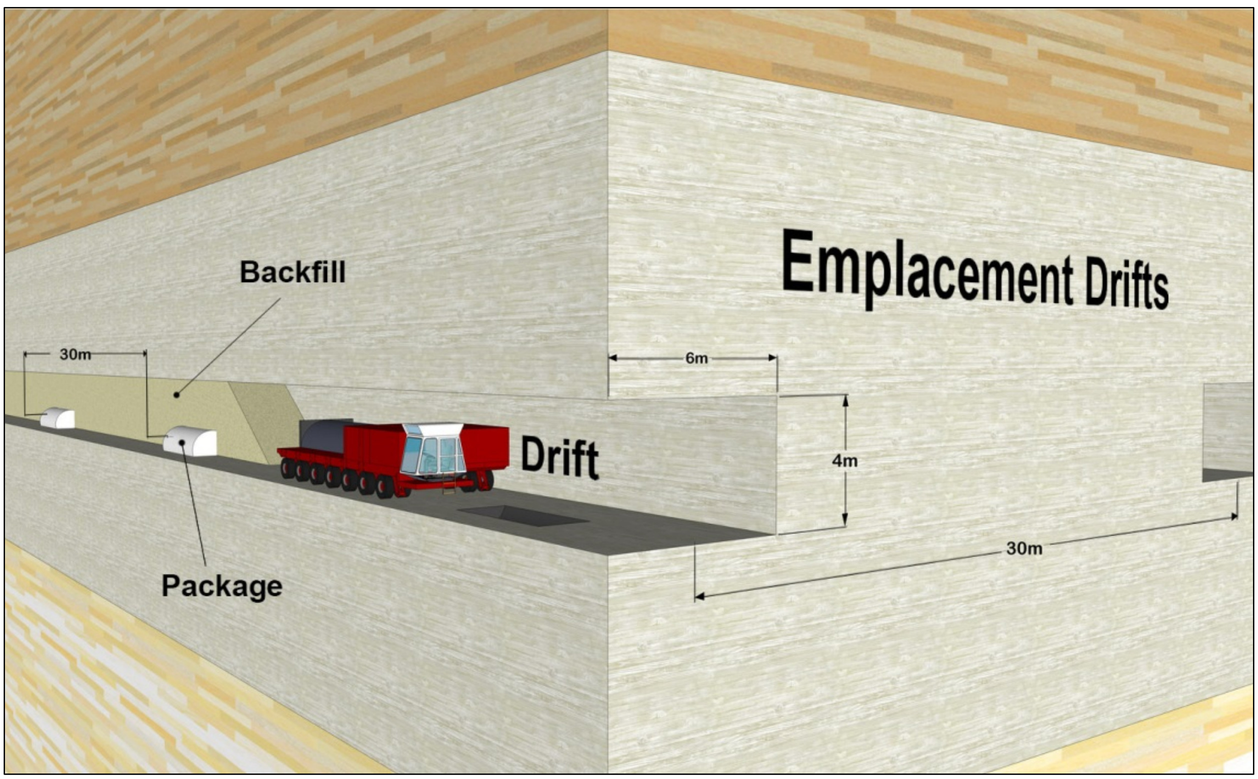

Figure 2-3. Schematic of in-tunnel axial emplacement of large, heavy waste packages containing SNF, showing shielded emplacement transporter, and semi-cylindrical cavities in the tunnel floor.

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3. Engineering Challenges with Disposal of CSNF in Salt

This section addresses technical questions that have been raised regarding disposal of CSNF in large, heavy waste packages in a salt repository. The information is presented as background for the disposal concept recommendations in Sections 5 and 6.

3.1 Waste Shaft Hoist

The following discussion applies to Koepe winder type friction hoists, which is the only type with sufficient payload capacity for heavy waste packages. A brief survey of friction hoists in service or presently under construction includes:

- **Esterhazy K3 Potash Project** (Mosaic Co., Saskatchewan): approx. 1,000 m lift, Koepe hoists with 55 MT skip load capacity, 15.8 MW hoist power train, headframe overall height 384 ft, likely to be the world's largest when completed ~2020 (North shaft) (https://im-mining.com/2017/06/06/).
- **Jansen Potash Project** (BHP Billiton, Saskatchewan): four new shaft hoists are under construction, Koepe production hoists intended to be world's largest when completed sometime after ~2021, approx. 1,000 m lift, 14 MW power train for each (payload to be determined) (https://im-mining.com/2017/09/).
- **Ma JiaLiang Coal Project** (Shanxi province, China): Koepe main hoist operational, payload 45 MT, hoist power train 7 MW (https://im-mining.com/2011/12/07/).
- **Waste Isolation Pilot Project** (U.S. Department of Energy, Carlsbad, NM): Koepe waste shaft hoist, 41 MT payload (largest in the world in 1986), hoist deck footprint 2.87 m wide x 4.67 m long (http://www.wipp.energy.gov/science/UG_Lab/WasteHoistNew.html).

Koepe winders, as opposed to drum winders, use an "endless" cable configuration looped through large driving wheels. The cables are driven by a wheel at the top of the shaft and looped back up in the shaft sump (i.e. the deepest portion of the shaft, which would extend at least 40 m below the base of the disposal level shaft station).

The cage and a counterweight are fixed to the cables so that as one moves up the other moves down at the same time. This arrangement provides the friction needed between the cables and the winder pulley at the surface. The use of a counterbalanced configuration produces a much lower starting torque than other hoist types, which in turn allows Koepe winders to use comparatively smaller motors. The counterweight can be specified so that motive power is required to run the hoist cage down the shaft (instead of up), as long as the payload is a fraction of capacity. With a full payload the hoist cage would not be fully counterbalanced and would descend with minimal power input. Once unloaded at depth the cage would become overbalanced and ascend with minimal power input.

Employing multiple cables of smaller size reduces the required diameters of individual pulleys, winder torque, and power requirements.

The following discussion describes hoisting of heavy waste packages, with shielding, on a rail-mounted transport cart. The TEV used for operations underground would be much heavier, so it would not be part of the hoist payload. This necessitates a transfer station underground, with transition from rail-mounted to rubber-tire transport.

Project DEAB Shaft Hoist (85 MT Capacity)

Anticipating a need for greater hoist capacity, the German agency DBE (German Service Company for the Construction and Operation of Waste Repositories) in the mid-1990s carried out the DEAB project (*Direct Disposal of Spent Fuel Elements*) for a potential repository at Gorleben. The purpose of DEAB was engineering demonstration of key components of a hoist system with payload capacity of 85 MT. This conceptual design was included in a U.S. review of disposal concepts (Hardin et al. 2013).

During waste transportation, the payload (e.g., waste package, shielding, and transportation cart) would be fixed inside the hoist cage. The maximum hoist velocity was set to 5 m/sec. Whereas some mining hoists may operate faster, lower velocity reduces the torque, power and braking requirements and is inherently safer. Because only a small number of waste transport operations would be anticipated to occur daily, hoist velocity would not significantly impact operational efficiency or repository system throughput.

The key factors determining the design of the hoisting system are the size, shape, and weight of the payload. The 85 MT design was based on a single POLLUX® waste canister (with shielding) and transportation cart. The POLLUX canister is a cylindrical container with a length of 5.5 m and an outer diameter (including trunnions) of 1.96 m (Bollingerfehr et al. 2012).

The DEAB project involved prototype engineering and testing of hoist components and safety features, and substantially increased the technical maturity of the 85 MT Koepe hoist concept. The DEAB design was subsequently adapted for an 80 MT payload capacity hoist for use in a Belgian repository, for the waste management authority ONDRAF/NIRAS. Further information on hoist specification and engineering, focused on extending the DEAB design to a payload of 175 MT, is provided in Section 7 and Appendix E.

Hoist Safety

The key concern related to shaft hoisting of nuclear waste payloads is a failure that could result in the uncontrolled fall of a waste shipment down the shaft and the associated potential for release of radioactive materials. The reliability of a hoist for transporting 85 MT payloads was investigated in the DEAB project, which demonstrated the technical feasibility and constructability of such a heavy hoist, and assessed its safety in repository applications, under the relevant German mining and nuclear licensing requirements.

A large-capacity shaft hoist for a U.S. repository would comply with preclosure safety requirements (e.g., 10CFR60 or 10CFR63), with a goal that the aggregated probability of an uncontrolled fall of a waste shipment down the shaft is "beyond Category 2" (i.e., less than 10^{-4} chance of occurring before permanent closure of the repository). Safety requirements differ in the German regulatory environment, tending to focus on annual probabilities rather than the total facility lifetime. A summary of operational safety for the DEAB concept, using four catastrophic event sequences, was extracted from DBE publications for a previous report (Hardin et al. 2013) and is not repeated here. The overall result was a total probability of 1.3×10^{-6} per year, which for a 50-year facility lifetime would give 6.5×10^{-5} per repository (i.e., beyond Category 2).

Discussion of repository waste hoist safety should include mention of the waste hoist freewheeling events that occurred at the WIPP in 1987 (Greenfield 1990). Uncontrolled movements of approximately 10 m and 90 m occurred successively during maintenance activities, without impact to material or personnel. Accident investigation found deficiencies in the hydraulic system that controlled hoist braking, and deficiencies in emergency controls, operating documentation, and administrative controls. A solenoid-operated valve was improperly plumbed (per design) into the hydraulic braking system and became a single point of failure. Emergency controls depended on this same valve and were inoperative. Maintenance activities to replace this valve, believed to be faulty, were incorrect and caused the events. Modification of the hoist and improvement of documentation and operating controls were implemented shortly after the incident.

3.2 Underground Transport and Emplacement

Equipment for underground transport and final emplacement of waste packages falls into two general categories: borehole deposition machines (vertical and horizontal) and in-tunnel transport-emplacement-vehicles. Vertical borehole deposition machines have been physically demonstrated: 1) at the Äspö crystalline rock underground laboratory for 4-PWR size waste packages (Mützel et al. 2001); and 2) at Gorleben for the BSK packages described above (Bollingerfehr et al. 2013). Horizontal borehole emplacement has been demonstrated for: 1) the KBS-3H disposal concept (Halvarsson 2008); 2) remote-handled TRU waste canisters at WIPP (DOE 2007); and 3) smaller, lighter canisters that would contain HLW glass, by the French program (ANDRA 2013).

Borehole emplacement is generally more complex than in-tunnel emplacement whereby packages are simply placed on the floor. For vertical borehole emplacement the waste package is upended and supported while being lowered into final position. For horizontal borehole emplacement a means must be provided to support the distal end of the package while pushing it into place. A water bearing has been used (Halvarsson 2008), and friction against the host rock or a steel liner (DOE 2007; ANDRA 2013). The German concept (never demonstrated) for emplacement of CASTOR® casks in horizontal boreholes would require “sliding elements” of unspecified design, to limit friction (Figure 3-1) (Bollingerfehr et al. 2013).

Another distinction between types of underground transport and emplacement equipment is the manner of transport along tunnels to the final emplacement location. For salt the impracticality of maintaining steel rail means that the transporter should have wheels or tracks and ride directly on the floor in service tunnels and disposal tunnels. Examples of both rail and rubber-tire transport (Figure 3-2) were generated in preparation for the Yucca Mountain License Application (DOE 2008). The use of wheeled platforms with many independently driven and steered wheels is common in shipyards and other industrial settings where large, heavy objects are moved. With many wheels independently suspended, such transporters can tolerate rough or compliant surfaces as would be encountered in salt tunnels. They could readily be disassembled for transport up and down the waste shaft. By comparison, rail systems ensure low rolling resistance and precise alignment of loads, but they use large quantities of steel and are expensive to install and maintain.

3.3 Shielding

Spent fuel requires shielding to protect workers from gamma and neutron radiation. Gamma radiation from fission product Cs-137/Ba-137 is especially intense during the first ~100 years out-of-reactor. Shielding is needed during transport to protect operators during normal operations, and during recovery from off-normal events. Self-shielded waste packages are an option that permits worker access during all phases of repository development, to an extent limited by collective dose, but with significant additional weight and packaging cost.

For the axial in-tunnel emplacement mode (see Section 2) mobile shielding is needed for shaft transport and the underground TEV. During emplacement, crushed salt backfill would provide a shielding function. Possible mobile shielding configurations, and the effectiveness of crushed salt backfill, are analyzed for representative gamma and neutron fields in Section 6.4 and Appendix A. The importance of shielding as an issue for engineering design as well as operational safety, arises primarily because of the weight penalty for shaft hoist design and the TEV. In Appendix A, shielding performance is analyzed for ranges of shield thickness and weight, showing that contact dose rate objectives can be achieved that are consistent with standard practice, regulatory guidance, and the payload capacity objective (175 MT) for a large shaft hoist.

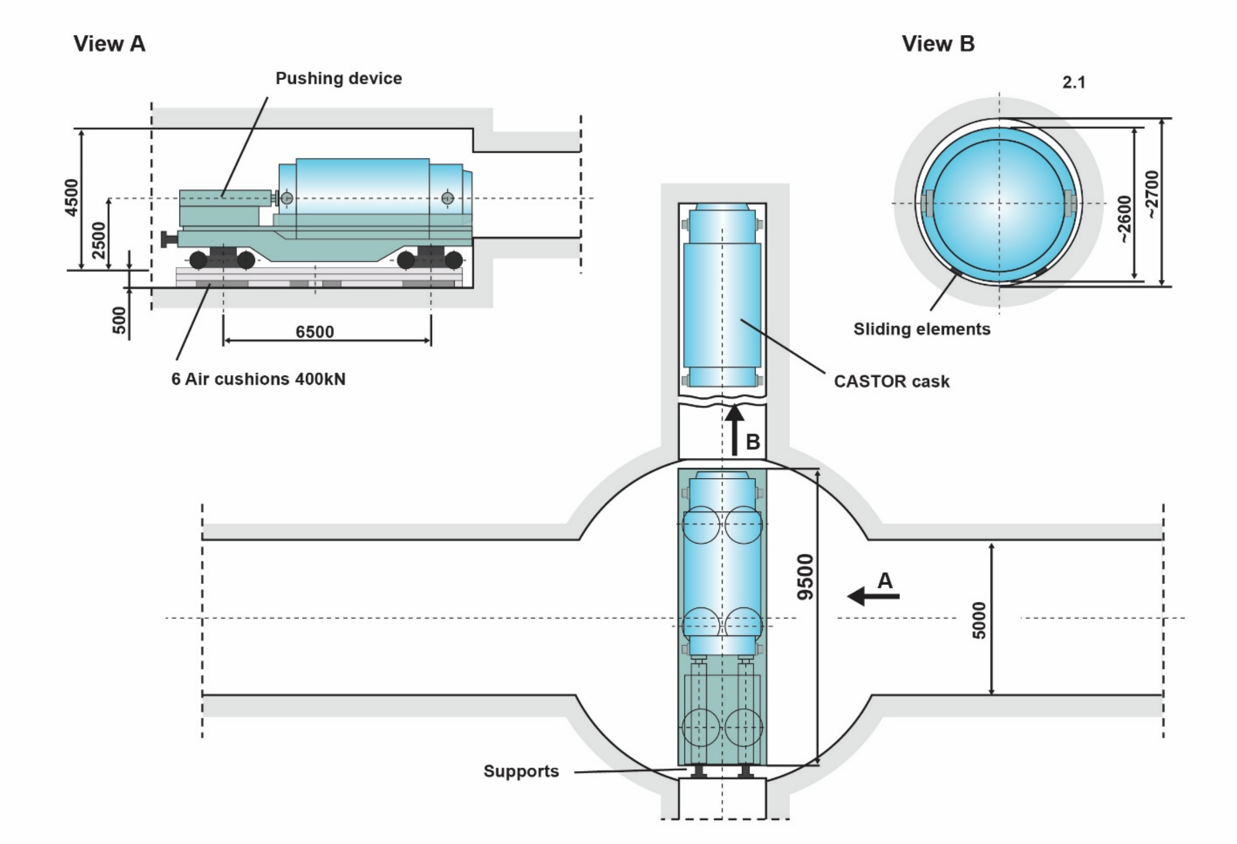

Figure 3-1. DIREGT concept for emplacement of CASTOR casks containing intact SNF, in large-diameter horizontal boreholes in salt (from Bollingerfehr et al. 2013).

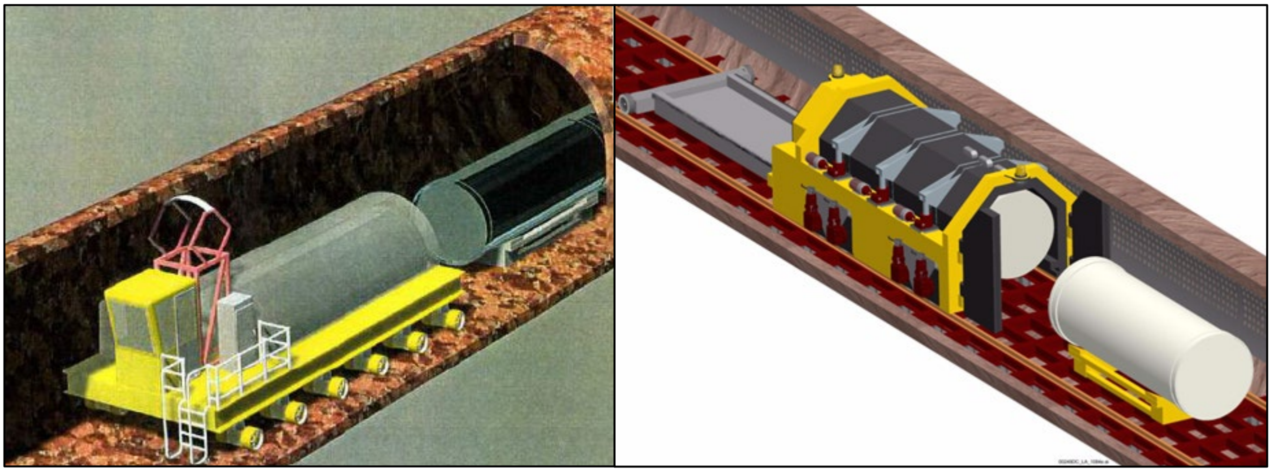

Figure 3-2. (left) Underground waste transporter supported by independent wheel trucks, running on compacted ballast (www.wheelift.com). (right) Rail-mounted TEV proposed in the Yucca Mountain License Application (DOE 2008).

3.4 Overpack Loads During Salt Reconsolidation

The functions of the overpack (Section 5) include full containment of the waste during repository operations (regardless of the condition of the sealed waste canister within). As shown in Section 8, loading of the waste packages into a repository would take approximately 50 to 75 years. When waste packages are first emplaced there would be an open head space on top of the stacked backfill in the disposal tunnel that could transport airborne contaminants. The tunnel opening would slowly creep closed and begin to impinge on the backfill within a decade or two. At this stage there is still open, connected porosity in the backfill that could transport airborne contaminants, although slower than in the headspace, while waste emplacement activities are still underway in other parts of the repository. Integrity of waste packages emplaced earlier in the operational period would become important during later operations. Containment integrity of all waste packages throughout the period of repository operations would help to ensure that the underground area remains a "clean" facility.

Processes that could degrade waste packages in a salt repository are mainly corrosion and mechanical loading of the overpack. Corrosion can be limited and predicted through use of a corrosion allowance material such as low-alloy steel. Corrosive penetration rates of hundreds of microns per year (see Section 4.1) could be tolerated for 50 to 75 years without package breach, given overpack thickness of a few centimeters. However, waste packages are expected to be hot and dry during the preclosure period when waste package containment would be required (using the performance strategy recommended here) so little or no corrosion could occur. Localized corrosion of carbon steel in brine has been observed but is also unlikely to be active for the hot, dry conditions. Steel corrosion has been extensively tested in brines and deliquescent humidity conditions (see for example, King 2007) although more environmental corrosion testing could be needed to verify that waste package penetration by corrosion does not occur in the preclosure timeframe.

On the other hand, mechanical loading of the overpack by salt creep has not been tested and is addressed here as a potentially important process that should be addressed in design. A mechanical analysis was undertaken to study deformation of waste package overpacks due to salt repository creep (Appendix B). The results show that a steel overpack of any practical thickness and material would likely yield during reconsolidation, if placed on the tunnel floor and surrounded by backfill. Compressive stress in the overpack was greater than 800 MPa for the largest wall thickness simulated (20 cm). Substantial yielding of low-alloy steel might not lead to package breach, but yielding is adopted here as a conservative indicator of the loss of containment integrity, pending availability of additional information.

The analysis showed that overpack yielding could be prevented, possibly with a smaller wall thickness, by emplacing the packages into semi-cylindrical cavities excavated in the floor. This was originally proposed to enhance heat transfer, but it also serves to reinforce the overpack during loading caused by salt reconsolidation. With the constitutive models used in the analysis, the timing is such that salt deformation and overpack loading are nearly complete after about 30 years (i.e., during the repository operational period). Emplacing waste packages into floor cavities could help ensure that breach does not occur during this process.

Because of the timing, overpack loading during reconsolidation is treated here as an engineering challenge with preclosure implications, rather than as a postclosure process in the next section.

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4. Key Postclosure Features, Events and Processes

The repository safety case would rely on multiple barriers including the geosphere, engineered borehole/shaft/tunnel seals, tunnel backfill, the waste container, and the waste form. The waste package containment lifetime would likely be limited (e.g., 10^2 to 10^3 years, subject to corrosion and loading discussed in Section 3.4). However, the host salt has extremely low permeability and geomechanical stability, such that no releases from the repository are expected. Slow degradation of the CSNF waste form could further extend isolation performance.

The only geologic repository for nuclear waste that is currently operating worldwide is the WIPP, a repository for TRU waste in bedded salt. For a spent-fuel repository, like at WIPP, results from total-system performance assessment could be dominated by future inadvertent human intrusion as discussed below (DOE 2014, Section 33).

The following sections summarize the technical status of key events and processes that could affect long-term performance of a repository in salt. The information is presented as background for the disposal concept recommendations in Sections 5 and 6.

4.1 Waste Package Corrosion

Overpack corrosion is included here because it represents an area of investigation that could enhance the assurance of repository safety. The following discussion focuses on behavior of low-alloy steel during the thermal period, which is the material recommended in Section 5 for disposal overpacks. Other materials could be considered, for example stainless steels could be used for longer structural lifetime, although its containment lifetime could be limited by localized corrosion.

Many reviews have been done of experimental corrosion rate data for carbon steel and cast iron (e.g., King 2007). The key environmental conditions for a salt repository during the first few thousands of years after closure are temperature, humidity, and brine composition. The temperature of a 21-PWR waste package would peak at approximately 150 to 200°C within a few years after emplacement, then decline rapidly for a few decades as the crushed salt backfill consolidates and its thermal conductivity increases (see thermal management discussion in Section 6.3). Temperature would then gradually decay back to ambient temperature over thousands of years, following the exponential decay of waste package heat output (Hardin et al. 2012, Table C-4, Backfill Consolidation Study). Relative humidity (RH) at the package surface would approach zero as the temperature peaks, then gradually increase over thousands of years to 76% (NaCl deliquescence at ambient temperature).

Steel corrosion can occur at RH as low as 53% in a NaCl environment (Schindelholz et al. 2014). The RH threshold for corrosion can be lower (e.g., 10 to 20% RH) if divalent (Ca, Mg) chloride salts are present. Importantly, the temperature does not have to be much higher than ambient to drop the local RH and turn off corrosion. For example, if the absolute humidity at the waste package surface is controlled by background ambient conditions (e.g., T = 27°C, RH = 76%), then reduction to 10% RH would require a temperature of about 62°C or greater. Waste packages containing 21 PWR/44 BWR assemblies would take roughly 1,000 years to cool to 62°C. Thus, corrosion may not begin for 1,000 years and possibly longer depending on the composition of salt contacting the overpack, and whether the local conditions are hydrologically unsaturated.

This simple analysis focuses on conditions for onset of corrosion, based on the chemical activity of H_2O (equivalent to RH). Even if the backfill porosity consolidates and becomes fully saturated with brine, the above discussion applies at least while the package surface temperature is greater than the dryout temperature for brine (e.g., 112°C for NaCl). This condition will exist for several decades after emplacement (see Section 6.3) depending on the SNF characteristics (age, burnup) and backfill properties.

A similar analysis was done for the Salt Repository Project using environmental conditions and laboratory corrosion data for steel in anaerobic brine (Diercks et al. 1988) to evaluate whether steel overpacks would meet the EBS subsystem containment requirement of 300 to 1,000 years at §10CFR60.113(a)(ii)(A) as applicable to a salt repository at that time. They concluded that 1,000-year containment could be difficult to achieve with steel using a corrosion allowance approach, but that a realistic representation of brine migration toward waste packages could potentially reduce the predicted rate of corrosion.

A different approach that could be beneficial to understanding waste package lifetime concerns the multi-process thermal-hydrologic-mechanical-chemical (THMC) coupled behavior of moisture in the host rock and near field. As the backfill and disturbed rock zone (DRZ) around the tunnel reconsolidate and heal, the transport properties will approach those of intact salt. The rate of reconsolidation (which may take tens to hundreds of years) depends on moisture content and temperature (Wieczorek 2017). Very low permeability to gas and brine (e.g., under hot, dry conditions) could inhibit any further transport of moisture from the host rock toward the package. Anaerobic corrosion of steel consumes 1.33 moles of H_2O for each mole of Fe, so corrosion of a heavy steel overpack has the potential to dewater the surrounding salt (which has typical moisture content of 1% w/w or less due to heating), thereby slowing the rate of corrosion. Studying the dewatering of the near field by heat, followed by the reconsolidation to very low permeability, is the primary purpose of recently proposed *in situ* testing in salt (DOE 2012).

Recent reviews of corrosion behavior of waste package materials have evaluated steel as an overpack material, stainless steel used in canister shells and baskets, and borated or Gd-containing stainless steels as the most corrosion resistant neutron absorbers (Bryan et al. 2014; Ilgen et al. 2014). Literature indicates that borated stainless steel corrodes faster than non-borated grades of stainless steels, for similar environmental conditions (Ilgen et al. 2015). Neither of these neutron absorbers is expected to resist corrosion in chloride brines, but corrosion test data are lacking and would be needed to evaluate borated stainless (included in the TAD canister specification for a Yucca Mountain repository; OCRWM 2008) for postclosure criticality control in a salt repository. Control of internal criticality during postclosure conditions is discussed further in Section 4.4.

4.2 Radionuclide Migration

Aqueous transport may occur by advection and hydrodynamic dispersion, and diffusion along chemical potential gradients, although rates of transport in a salt repository may be insignificant. A stylized analysis of waste isolation using CSNF inventory showed no significant releases from a salt repository for undisturbed conditions (Freeze et al. 2013).

Transport of released radionuclides in a repository for CSNF would be similar to the description for TRU waste in the WIPP performance assessment (DOE 2014, Appendix SCR, Section 6.6.1). One difference is that the EBS concept proposed here (Section 5) would involve roughly 20 m of reconsolidated salt backfill between adjacent packages throughout the repository, which would further impede transport. Additional attenuation would arise from tunnel and shaft seals

(Section 6). Also, the coupled multi-physics dryout behavior discussed above, could produce impervious conditions in the near field around each waste package.

4.3 Human Intrusion

Human intrusion would be part of the performance assessment for a repository in salt. The nature of the human intrusion assessment requirement is somewhat uncertain at present. For the Yucca Mountain License Application, in compliance with 40CFR197, human intrusion was represented using a separate calculation of the dose to the RMEI from future drillers inadvertently intersecting a waste package and exposing the contents to transport by groundwater. The human intrusion calculation is defined differently by the two regulations currently applicable to deep geologic disposal (specifically, by 40CFR191 and 40CFR197).

The original rule 40CFR191 currently applies to any HLW/SNF repository except Yucca Mountain. For calculating the frequency of inadvertent interception of waste packages by future boreholes, recognizing the variability and uncertainty in drilling rates, 40CFR191 states:

> *...the likelihood of such inadvertent and intermittent drilling need not be taken to be greater than 30 boreholes per square kilometer of repository area per 10,000 years for geologic repositories in proximity to sedimentary rock formations, or more than 3 boreholes per square kilometer per 10,000 years for repositories in other geologic formations.*

The potential for borehole interception of waste packages containing the full inventory of CSNF was calculated for sedimentary and other settings (e.g., hard rock) (Hardin et al. 2013, Appendix A). The results indicate that the expected number of interceptions is greater than 1, particularly for sedimentary concepts (including salt) and for horizontal waste package orientation.

Consistent with the more current Yucca Mountain rule, and depending on future changes in the regulations, inadvertent human intrusion would not be expected to be included in the probabilistic dose calculations for a repository in salt. Rather, it would be assessed separately. There is a likelihood that the approach used in 40CFR197 and 10CFR63 would continue to be used in new regulations for a repository somewhere other than at Yucca Mountain (McCartin 2012).

Human intrusion could be the most important scenario leading to radionuclide releases to the environment from a CSNF repository in salt (by analogy to the WIPP; Hardin et al. 2013). If regulated according to 40CFR191 the number of inadvertent borehole-waste package intersections would be estimated, and the assessment would include dose pathways such as “cuttings and cavings” (DOE 2014). However, this would not show “…how well a particular repository site and design would protect the public at large…” (10CFR63, 66 FR 55732, p. 55761, Supplementary Information, 3.10 Human Intrusion Standard). Hence, changes could be expected to the regulations applicable to a repository in salt.

4.4 Internal Criticality

Postclosure criticality is considered in FEP screening for performance assessment, a process in which the features, events, and processes that could affect repository performance are evaluated and excluded from further consideration if the probability of occurrence is low enough or the consequences are not significant (§63.342). A general methodology for addressing postclosure criticality was developed for previous licensing-related activities (DOE 2003) and reviewed by the NRC (2000). The methodology was used to demonstrate that criticality could be excluded from repository performance considerations on the basis of low probability, for the proposed Yucca

Mountain repository (NRC 2011). The performance specifications and loading requirements for the TAD canister (OCRWM 2008) were important to this low-probability result.

More recent work on the technical feasibility of direct disposal of existing DPCs (not licensed for disposal) included postclosure criticality (summarized by Hardin et al. 2015). That analysis concluded that many DPCs are directly disposable, especially in unsaturated hard rock or salt media. The criticality analyses for that assessment used two degraded canister configurations: 1) loss-of-absorber in which neutron absorber plates in the baskets dissolve and are replaced by groundwater, and 2) basket degradation in which the entire basket is removed and the fuel assemblies are moved together. Analyzing DPCs in their as-loaded state, using canister design and fuel burnup information, showed that sufficient reactivity margin exists for many (not all) existing DPCs to remain subcritical for both degraded configurations.

Postclosure criticality control with TAD-type canisters in a salt repository would encounter technical questions similar to the DPC direct disposal analysis. Whereas the current fleet of DPCs uses neutron absorber plates made from aluminum-based materials, the TAD canister specification calls for plates of borated 304 stainless steel, 11 mm thick (OCRWM 2008). The use of borated stainless steel in the TAD canister is considered to satisfy longevity requirements in the Yucca Mountain analysis, but the longevity in brine is probably less (Section 4.2). Hence, no credit is taken for the plates in the scoping discussion below.

A neutron-transport model of a 21-PWR TAD canister developed for a previous project (Sandia 2007) was modified to evaluate reactivity in a degraded configuration in a salt repository. The model used the Monte Carlo N-Particle code (MCNP5 V. 1.40). For scoping it was assumed that brine flows through a failed overpack and failed canister shell, into the fuel basket where it dissolves the neutron absorber plates and fuel tubes, bringing the PWR fuel assemblies into contact (making the array more reactive). The degraded configuration is shown in Figure 4-1, where magenta regions correspond to brine, and assemblies are shown as square arrays of fuel rods. The assemblies are brought together without separation, but the pitch between fuel rods is unchanged (collapsing the rod pitch would make the configuration less reactive). A range of dissolved salt densities was evaluated (Hardin et al. 2014a).

In the repository host formation the chloride concentration in brine would be high, potentially saturated with respect to halite (approximately 6 molal NaCl at ambient temperature). Absorption of thermal neutrons by natural Cl-35 (~75% of total Cl abundance) can help maintain subcriticality (Hardin et al. 2014a). Salado and Castile formation brines from the WIPP site have chloride concentrations close to saturation with respect to NaCl (Rechard et al. 1999, Table 1). Other salts present in the evaporite sequence, particularly divalent Mg- and Ca-chlorides, could drive chloride concentration even higher in brines.

Note that it is no longer current practice in the U.S. to drill oil-and-gas wells with diesel fuel as drilling fluid. Diesel could provide neutron moderation if it flooded a waste package, and would not generally contain chloride, but it would not significantly corrode absorber plates or rods made from aluminum, stainless steel, or other resistant materials.

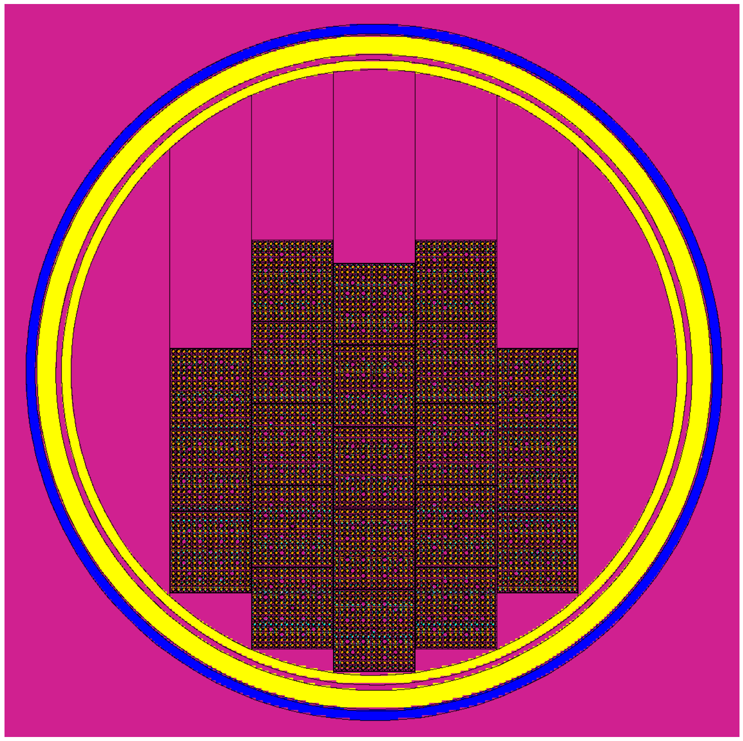

Figure 4-1. Modeled configuration of fuel assemblies with degraded basket.

The maximum neutron multiplication factor k_{eff} for a 21-PWR package with degraded basket and no absorber plates, flooded with any brine up to NaCl saturation, is greater than 1 for fresh fuel (i.e., 5% w/w U-235 enrichment with zero burnup). This result is presented along with a set of calculations for ranges of burnup and brine strength, and without any solid neutron absorbers (Figure 4-2). The calculations show that crediting burnup for as-loaded canisters can substantially reduce k_{eff} such that a 21-PWR waste package could remain subcritical provided a combination of salt concentration (at least ~3 molal) and fuel burnup (at least ~20 GW-d/MTU with 5% initial enrichment). This is without neutron absorber plates, and with assemblies moved together at closest possible incidence.

Using reactivity margin in this manner tends to reduce the margin available to accommodate error in subcriticality results. Margin is desired to account for uncertainties in the analyzed configuration and the composition of fluid, as well as numerical predictive uncertainty. Some margin may also be consumed by misload calculations, which incorporate the probability of human error in loading fuel assemblies into canisters. The topical report (DOE 2003) and previous misload analyses (Sandia 2008) indicate that the impacts of misloads should be considered.

Because TAD canisters have not been fully designed, licensed, or deployed there is flexibility for future canisters to provide any desired margin of subcriticality. For example, enhancements such as disposal control rods can be added (EPRI 2008; Hardin 2013). Such measures are not needed for disposal in a Yucca Mountain repository but could be added at the time that canisters are loaded, if there is a significant possibility that a canister would be disposed of in another medium such as salt. Additional corrosion testing of borated stainless steel (the neutron absorber in the TAD canister specification) in representative chloride brine could also be useful.

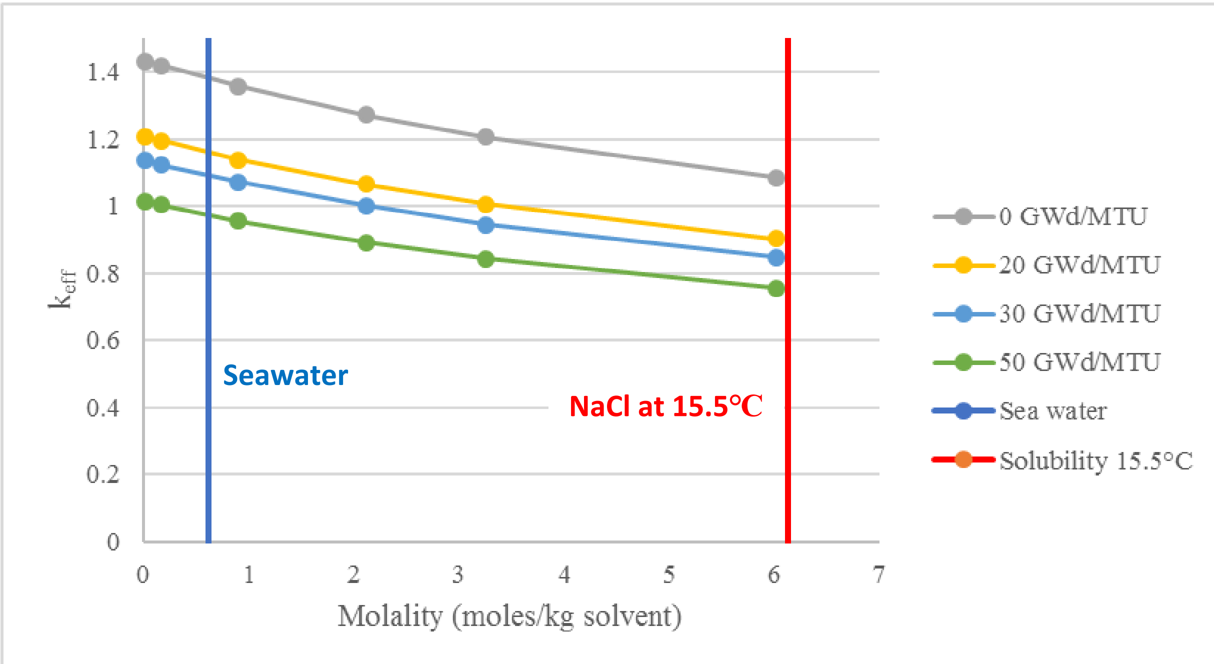

Figure 4-2. Effective neutron multiplication factor *vs*. salt concentration (molal).

4.5 External Criticality

The external criticality analysis for a CSNF repository in bedded salt would resemble that performed for WIPP (DOE 2014). External criticality refers to the possibility that fissile isotopes could be mobilized from waste packages by groundwater, and re-deposited somewhere along the transport pathway in sufficient quantity to cause a criticality event. The WIPP analysis considered migration of U-235 and Pu-239, mobilized by chloride brine that was released into the repository due to inadvertent future human intrusion by drilling from the surface (Rechard et al. 1999). The screening argument depends on limited solubility of fissiles in brines, the abundance of neutron-absorbing chloride, and limited distribution of porosity available for deposition. Colloidal transport of Pu was also considered.

A similar approach would be applied to a repository for CSNF. The analysis would also consider the chemical transport behavior and reactivity contributed by other isotopes of Pu and other actinides in CSNF, and by radioactive decay (Wagner and Parks 2003). Other chemical constituents contributed by the waste packages would be scarce (e.g., fission products) and/or immobile (e.g., chromium and iron from waste packaging). Such an analysis is beyond the scope of this report.

4.6 Thermally Activated, Low Strain-Rate Creep

Waste packages in a salt repository for CSNF will sink slightly during a postclosure performance period of 10,000 years or longer. The amount of sinking might be significant to repository performance assessment, or not, depending on the constitutive model used for creep modeling (and other models used in the performance assessment). The following discussion is taken from Hardin et al. (2014b) and presents emerging understanding of a low strain-rate creep mechanism that is not represented in many widely used salt creep models, but could contribute to waste package sinking, and is under active investigation (Reedlunn 2016; 2018).

Vertical movement of large, heavy waste packages has been identified as a potentially important process in salt repository performance (Winterle et al. 2012; OECD/NEA 2000). Extensive sinking (e.g., more than 1 m per 10^4 years) could move waste packages out of the host unit in bedded salt, where they could be exposed to different strata and conditions affecting waste isolation.

Salt creep models have been developed and conditioned on laboratory data, and on multi-year observations of room and borehole closure. However, the strain rates associated with these observations are generally orders of magnitude greater than could be associated with vertical movement of waste packages over 10^4 years. A mechanism for salt creep that prevails at low temperature and low-stress, low strain-rate (LS-LSR) conditions was recognized decades ago (Munson et al. 1984) but only in the past few years have attempts been made to measure its effects (Bérest et al. 2005, 2008, 2012). In these tests, total strains on the order of 10^{-4} are produced at rates on the order of 10^{-12} sec^{-1}. The mechanism is thought to involve pressure solution because it is similar to certain behavior at greater strain rates that is known to depend on moisture content.

Scoping calculations for viscous sinking of waste packages in salt are described in Appendix D, with the result that strain rates of 10^{-10} to 10^{-12} sec^{-1} can be expected, leading to sinking velocity on the order of 10^{-12} m/sec (0.1 to 1 m per 10^4 years). In terms of strain rate, package sinking could be similar to diapir penetration processes and long-term closure of larger man-made openings (mined or solution cavities).

Effective Viscosity

Effective viscosity is useful to describe the potential for salt creep, because: 1) salt typically does not exhibit yield strength or cohesion behavior (Jackson and Talbot 1986); and 2) power-law creep laws can be recast in terms of effective viscosity (Weinberg 1993). Examples of effective viscosity in natural salt formations were reviewed by Hardin et al. (2014). The LS-LSR creep tests on core reported by Bérest et al. (2005, 2012) show highly compliant viscous behavior, which can be expressed as effective viscosity of 10^{16} Pa-sec and compared with natural salt creep.

A recent numerical analysis (Clayton et al. 2013a) used the Munson-Dawson multi-mechanism deformation (MD) creep model (Munson 1997) both directly and as the source function for temperature-dependent viscosity in a viscoelastic model formulation. Using the MD model the effective viscosity at low stress and low (ambient) temperature is on the order of 10^{20} Pa-sec, and package movement is predicted to be very slow. The same analysis confirmed that waste package movement of 1 m per 10^4 years corresponds to effective viscosity of approximately 10^{16} Pa-sec.

LS-LSR Creep Mechanism

Bérest et al. (2005) published results from long-term (22 months) creep tests on cores, that were performed underground in a remote part of a salt mine with stable temperature and humidity conditions. The results were interpreted using a creep power law defined in two ranges (bilinear on a log $\dot{\varepsilon}$ *vs*. log σ plot; Bérest et al. 2012). The low-stress range is Newtonian (n=1) while the higher stress range is a power law (n=4) (Figure 4-3). This is a significant departure from previous creep laws that do not include the Newtonian branch.

The calculated normal stress magnitude in the salt, developed only from the waste package weight (corrected for salt density) is similar to loading conditions in the Bérest et al. creep tests. Ongoing LS-LSR tests include effects from temperature and confining pressure, on creep of WIPP salt (Hansen and Popp 2015). Results so far corroborate the Bérest et al. observations and relevance of the bilinear creep law. Re-analysis of room creep data from the WIPP has also suggested a LS-LSR mechanism (Reedlunn 2016). The action of a LS-LSR mechanism in simulations, produces a subtly different deformation regime in the repository host salt (Appendix D).

This assessment does not take into account the effects of heating, which last about 2,000 years in a repository but could increase strain rates by 2 to 3 orders of magnitude at temperatures 100 to 200°C above ambient, respectively (activation energy of 50.2×10^3 J/mol-K for dislocation creep; Weinberg 1993). Temperature effects on package movement were shown to be small with the MD constitutive model (Clayton et al. 2013a,b) but may contribute more to deformation caused by the LS-LSR mechanism (presently under investigation).

It is important to note that analysis of heavy waste packages sinking in salt would not be applicable to the WIPP which does not involve such packages.

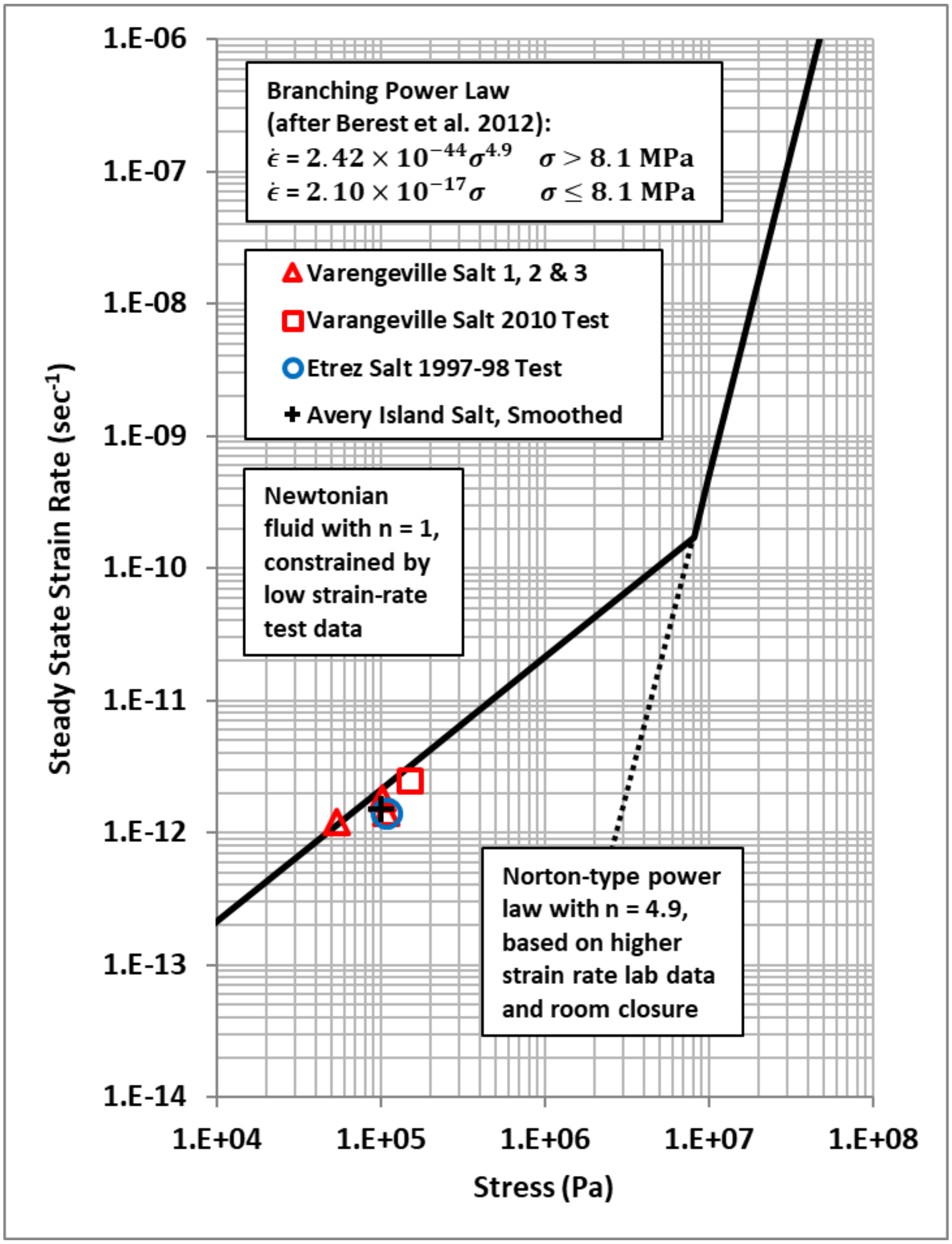

Figure 4-3. Bilinear creep law of the type proposed by Bérest et al. (2012).

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5. EBS Concept for CSNF Disposal

The principal function of disposal overpacks would be to maintain containment integrity throughout repository operations including handling, aging, transport at the surface and underground, emplacement and backfilling, and after backfilling prior to final closure of the repository. The disposal overpacks would be capable of transport in vertical or horizontal orientation at the surface, and horizontal transport and emplacement underground. Packages would be unshielded (or partially shielded) and additional temporary shielding would be used for transport and emplacement operations (Section 6.4). Overpacks would be right-circular cylinders with inner diameter sufficient to accommodate the 1.71-m TAD canister outer diameter (allowing a few centimeters for diametral clearance). Wall thickness would be approximately 7 cm as discussed below (possibly allowing yielding at the bottom of the overpack; Appendix B).

Disposal overpacks would be of welded construction, made from corrosion allowance material. Low-carbon steel is recommended because of the relatively low cost and ease of fabrication, ductility, and relatively well understood corrosion behavior in brine. This is consistent with the original conceptual design for the Salt Repository Project (Fluor 1987), with a generic reference case that considered corrosion (Sevougian et al. 2013), and with a review of overpack materials for DPCs (Ilgen et al. 2014). Criteria for selecting mild steel for the overpack material were discussed in detail by ONWI (1988, Section 2.3.2). The CSNF waste would be isolated initially from its surroundings by two welds, those sealing the TAD-type canister and the overpack.

Fuel canisters would maintain containment integrity during all operations prior to sealing in overpacks. They would maintain cladding temperatures during preclosure operations, and during the peak thermal period after emplacement, as prescribed in the TAD canister specification (OCRWM 2008). The overpack would perform with the same maximum temperature calculated for the canister. The fuel canister fuel basket could contribute to postclosure criticality control, in addition to the chloride effect (Section 4.4), but technical justification for longevity of neutron absorber materials in brine (e.g., corrosion testing) would be helpful.

Waste packages with 21-PWR size TAD-type canisters would be transported down the waste shaft in a horizontal configuration, supported by a cart on rails, then transferred to a rubber-tire TEV at a transfer station near the shaft. The packages would be transported to the emplacement area in the shielded TEV, then emplaced on the floor of the current working disposal tunnel into semi-cylindrical cavities (Figure 2-3). The cavities would be prepared just prior to waste emplacement using a specialized drum-shaped mining machine. The TEV would straddle the cavity to position the waste package for lowering. The overpack could be of a skirted design at each end to provide lifting points for the TEV. The TEV would have mobile shielding and could therefore have an operator during transport. However, for waste package transfer or for emplacement, remote operation would likely be used.

The purpose of the semi-cylindrical cavities would be to facilitate heat dissipation and provide structural reinforcement to the overpack. They would also stabilize package movement during emplacement and backfilling operations. The structural response of the overpack to reconsolidation of the host salt and backfill is addressed in Section 3.4. Mechanical loading of the overpack by salt would be non-uniform, but the upper arc of the overpack cross-section forms an arch when constrained from the sides (in the floor cavity), minimizing bending moments that produce high compressive stress. Scoping calculations indicate that with cavity emplacement the maximum compressive stress in the overpack could be limited such that a low-alloy steel grade could be used without exceeding the yield stress (e.g., 13-cm overpack result in Appendix B).

The type and weight of shielding needed for transport and emplacement operations, and shielding provided by the backfill, are discussed in Section 6.4 and Appendix A. After emplacement, each waste package would be covered with crushed salt backfill for shielding and to promote more complete and uniform reconsolidation (Figure 2-3). Backfilling would leave a small headspace in the disposal tunnel (a few tens of centimeters), that would conduct a small air flow toward the ventilation exhaust shaft until closed by reconsolidation.

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6. Repository Design and Layout

Previous concept development for geologic repositories in salt, and experience from the WIPP, support viable geologic disposal concepts for CSNF. A salt repository would be situated at sufficient depth (500 to 1,000 m below ground surface) to ensure that creep closure proceeds to completion (i.e., complete reconsolidation of the backfill and near-field host salt). Salt beds form in thick sedimentary sequences or basins that include layers of other evaporites (anhydrite, polyhalite, carnallite, etc.) or clay, that can affect rockmass deformation, heat transfer, and brine movement. A host salt interval would be selected for halite thickness and tested *in situ* to determine suitability for a repository. The overlying rock layers could contain permeable units (i.e., aquifers) that would be accommodated during initial construction, and during plugging/sealing of shafts when the repository is closed.

Sedimentary basins that have been previously characterized to evaluate suitability of bedded salt for a geologic repository include the Palo Duro (part of the Permian; DOE 1986), Delaware (part of the Permian; DOE 1980), and the Paradox (DOE 1984). Technical criteria for selection recommendations have included salt thickness, low seismicity, absence of major known reserves of petroleum, relatively few boreholes that penetrate the salt, lack of significant mineral deposits (except salt), absence of evidence for salt dissolution at depths greater than 300 m, and no recognized geologic, hydrologic, environmental, or socioeconomic factors that preclude study (ONWI 1983). Similar criteria would likely be used in the technical part of any future siting process.

This section offers a high-level discussion of repository surface facilities, shafts, and repository excavations. Key design questions are discussed including thermal management and shielding for CSNF waste.

Previous salt repository concepts (Section 2) have included generic concepts for HLW from reprocessing (Carter et al. 2011), defense HLW (Matteo et al. 2016), and CSNF (Hardin et al. 2012). A repository for CSNF could be several times larger than one for HLW from reprocessing and/or defense activities, so the scale of the underground facility for CSNF described here is adapted from the latter source. Description of surface facilities is taken from the HLW reports, and from the Yucca Mountain License Application (DOE 2008).

6.1 Surface Facilities

The repository would be divided into surface facilities, shafts, and repository-level underground excavations. The surface facilities would support underground operations, and receive, package, and transport waste (Carter et al. 2011, Section 4). Major surface facilities include a waste receipt and packaging facility (WRPF), repository systems and services, and balance-of-plant facilities.

Waste Receipt and Packaging Facility – The WRPF would be similar to the Canister Receipt and Closure Facility (CRCF) proposed for a Yucca Mountain (YM) repository (DOE 2008, Section 1.2.1.2). The YM plan would actually use three identical CRCFs to receive and package approximately 60,000 MTU of CSNF over an operational lifetime of ~30 years. This portion of the CSNF received at YM would consist of more than 6,000 sealed TAD canisters. Each would be placed into a disposal overpack, which would then be welded-sealed (dry operation) involving multiple steps, and then transported underground and emplaced.

A salt repository for CSNF would require at least one WRPF, and possibly two or more depending on total inventory for disposal and required throughput. Maximum throughput of 3,000 MTU per year (approximately 320 waste packages annually, of 21-PWR capacity) is used in logistical simulations to estimate how long the repository would need to operate (Section 8). That throughput corresponds to three modular WRPF facilities, by analogy to the YM CRCF concept of operations.

Each WRPF would be designed with stages for receipt and inspection, canister transfer, waste package closure welding and inspection, and preparation for transport underground. Buffer capacity would be included for temporary shielded storage of completed waste packages waiting for transport underground. Empty shipping casks would be inspected and shipped back to the waste generator or CSNF interim storage facility.

The WRPF would also include integral equipment for remote handling, cranes, ventilation, emergency services, and so on. The building would essentially be a large hot cell for handling sealed canisters, with pre- and post-processing stations. The receipt station would be accessed by standard rail or heavy-haul truck, while completed waste packages would be dispatched on rail-mounted carts, in horizontal orientation. The architecture of the WRPF would provide shielded work stations separated by isolation doors, with separate ventilation.

Note that packaging or re-packaging of CSNF into TAD-type canisters is not included in this description, and would be done at: 1) reactor sites; or 2) an upstream centralized handling/storage facility; or 3) an additional, large waste handling facility at the repository; or 4) some combination of these.

Repository Systems and Services – These support systems would include electrical supply, centralized monitoring and communications, fire suppression, security, radiological protection, environmental monitoring, and training (Carter et al. 2011).

Balance-of-Plant – Other support functions include heating, ventilation and air conditioning for non-radiological facilities, emergency response and medical, central engineering, control, and administration, analytical support, railroad operations, locally-generated waste management, warehouse and maintenance, and roadways (Carter et al. 2011).

6.2 Repository Layout and Construction Sequence

The underground repository would consist of a large array of modular disposal panels, arranged around a central support area (Figures 6-1 and 6-2). Each panel could consist of approximately 10 parallel disposal tunnels roughly 1.2-km long, with service and ventilation tunnels around the periphery. Tunnel lengths and the sequence of construction would be optimized using workflow studies that include long-term stability of openings. Some or all of the service tunnels could be used for waste emplacement after the adjacent panels are loaded, and after their support functions are no longer required. Several panels would be constructed along an outward heading from the central support area (Figures 6-1 and 6-2 show ten panels accessed on one heading).

With disposal tunnels spaced 30 m apart (center-center) the extraction ratio would be 20%. With waste packages spaced 30 m apart (center-center) each disposal tunnel would hold up to 39 packages, and each panel could hold 390 packages (with a small contingency). Package spacings and tunnel spacings would be controlled by thermal management, discussed in Section 6.3, in addition to opening stability and other geologic criteria. A repository for 140,000 MTU of CSNF in 21-PWR size packages would require approximately 456 km of disposal tunnels, and approximately 100 km of service tunnels. The total excavated volume, not including shafts, would be on the order of 15×10^6 m^3. These estimates scale approximately to the capacity, so a repository for 70,000 MTU would require 228 km of tunnels, with total excavated volume on the order of

8×10^6 m^3. Note that estimates of this nature are for scoping purposes only and can be compared only at the level of one significant figure.

An important constraint on the repository layout is the need to limit the length of time that tunnels need to stay open. Thus, a panel or group of panels would be mined, loaded, and backfilled in a few years, before maintenance on the service tunnels increased due to salt creep. Service tunnels would be mined first, then disposal tunnels starting with those most distant from the central support area. When all the panels and service tunnels are loaded and backfilled, any remaining open tunnels would be backfilled back to the central area. Emplacement along that heading would then be complete, and a new set of panels would be excavated on a new heading. The sequence should be designed so that any mishap in a disposal tunnel (e.g., roof collapse, or radiation leak) could be isolated with backfilling and ventilation control, so that further emplacement and other underground operations could continue in a "clean" environment. The layout concept shown in Figures 6-1 and 6-2 could achieve this by: 1) physical separation of tunnels used for construction and waste handling, from ventilation exhaust; 2) limiting panel and heading size, to limit the required duration of access to service tunnels; and 3) locating ventilation exhaust tunnels on the outside of the array so that in the event one becomes unstable or contaminated, a new one could be excavated parallel to it.

As drawn (Figure 6-1) service tunnels would complete a perimeter around each panel, and would be doubled in the direction of the heading (i.e., upstream and downstream from each disposal tunnel). This layout would allow reconnaissance of the host salt layer before starting construction of each panel. It would also increase access to the exhaust side of the panel for maintenance. Other service-tunnel arrangements are possible such as the reticulated array of tunnels proposed for the Salt Repository Project by Fluor (1987) and implemented at the WIPP. The reticulated arrangement is flexible but has certain disadvantages such as routing options for airflow, and the extent of service tunnels that need to be maintained for decades.

Shafts – A repository for CSNF in salt would be accessed by vertical shafts, which minimize exposure to ground water in overlying beds, and facilitate construction and plugging/sealing. Shafts can be constructed using the freezing method to penetrate water-bearing strata, but this would be more difficult for a ramp, with far less construction experience to draw from.

A single waste transport shaft would be capable of transferring up to three waste packages to the repository horizon per 10-hour shift, and could handle the 3,000 MTU per year emplacement throughput rate (Hardin et al. 2012). The finished diameter of the waste shaft would be approximately 9 m to accommodate CSNF waste packages with shielding, in horizontal orientation. The design operating life for shafts would be at least 50 years.

Shaft liners would be designed to meet the site geological and hydrological conditions, and would include sections of cast iron, reinforced concrete, and non-reinforced concrete or other materials as appropriate. Shaft collars and stations would be constructed from reinforced concrete. Head frames, hoist houses, and related facilities would be designed to withstand site-related design-basis seismic and weather loads.

A single ventilation exhaust shaft would be needed with finished diameter of approximately 9 m, air flow capacity of up to 500,000 cfm, and filtration capability at the surface. Additional, smaller diameter (5 to 7 m finished) shafts would be constructed for men-and-materials (and emergency egress), waste salt removal, and ventilation air intake. Over the operational lifetime of the repository additional intake shafts could be constructed to serve emplacement operations on the different headings. The first shafts to be constructed would be the men-and-materials and waste salt shafts, followed by the ventilation exhaust, waste transport, and ventilation intake shafts. Note that because ventilation would not be used to remove heat from a salt repository, fewer ventilation shafts would be needed, with less power expended on forced ventilation, than with other repository concepts (Hardin et al. 2012).

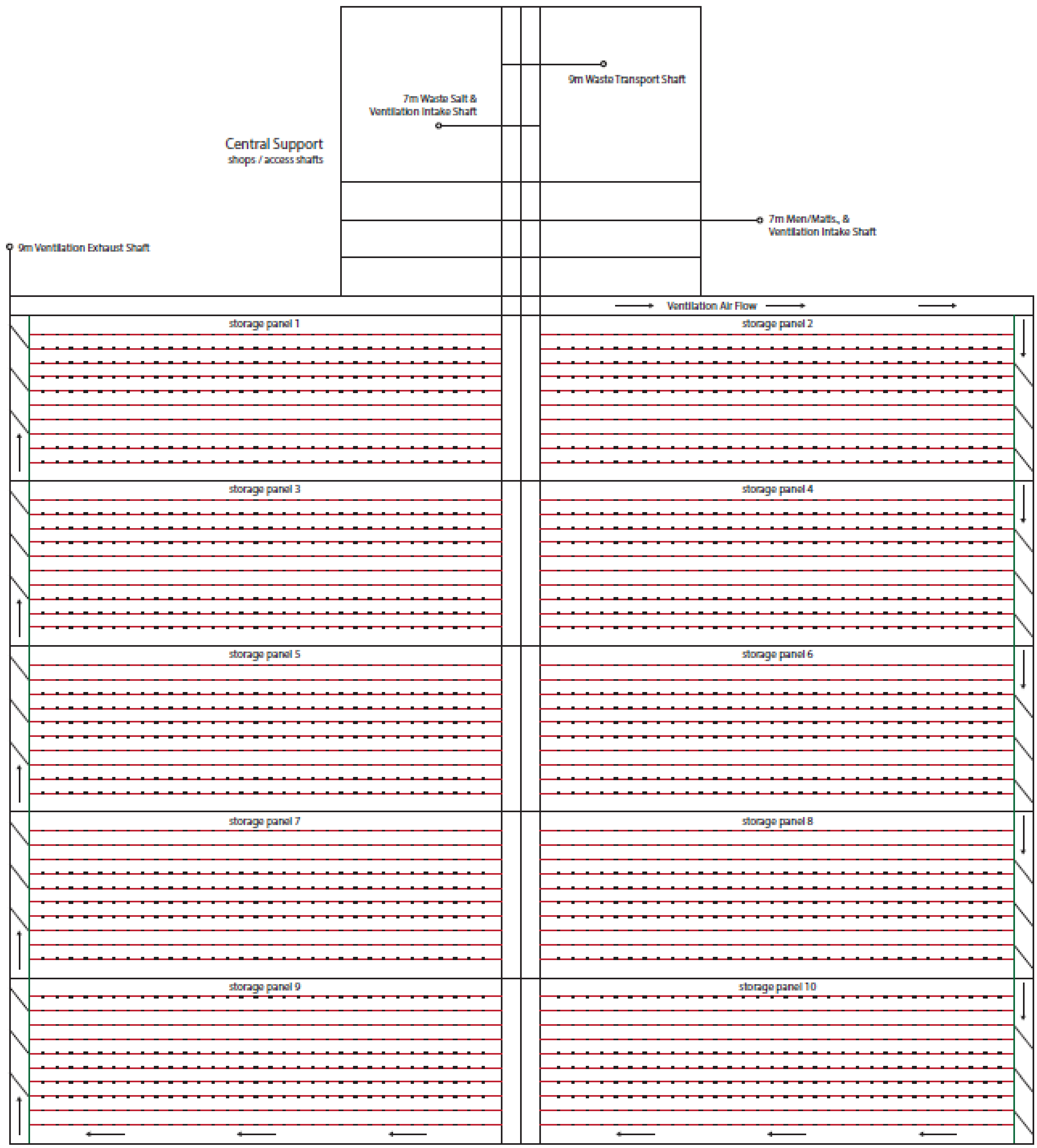

Figure 6-1. Plan view of repository layout, with ten panels developed in only one cardinal direction (schematic, not to scale).

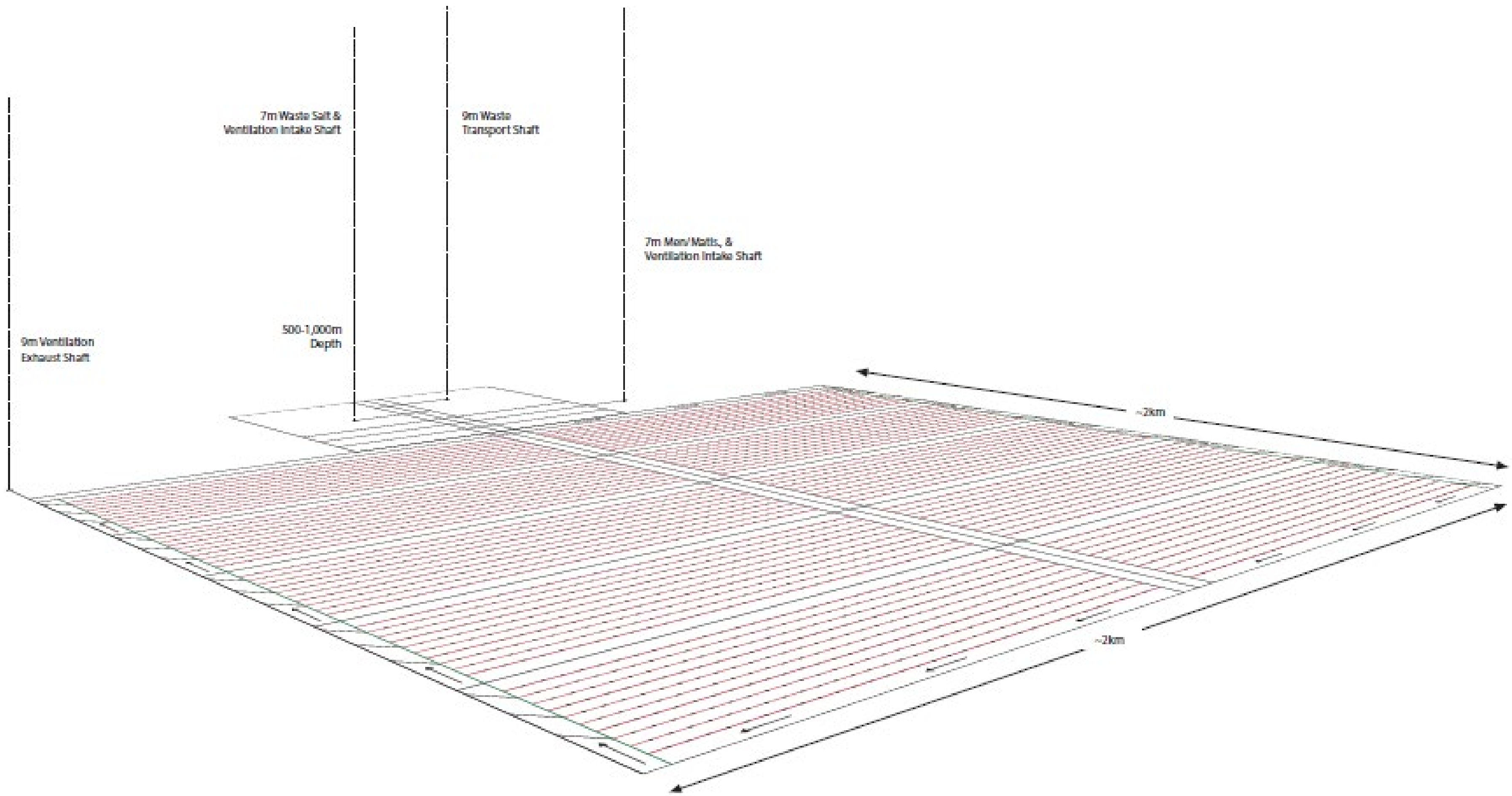

Figure 6-2. Perspective view of repository layout, with ten panels developed in only one cardinal direction (not to scale).

Emplacement, Access and Service Tunnels – Underground mining in the host salt interval would be mechanized and fast, using a roadheader (boom-cutter). This equipment is ubiquitous in salt and potash mines. Only minimal ground support would be used; the use of more robust ground support (e.g., rigid liners) is futile in plastic salt. Excavated openings would be reasonably stable for a few years, although room closure deformations of tens of centimeters could occur during this time. Installation of rock bolts and wire cloth or mesh in traffic areas would serve to protect workers from rockfall (but not roof collapse which is predicted and managed so that underground activities can be conducted safely).

Ground support for all openings would consist of rock bolts with wire mesh or cloth where needed to limit rockfall for worker access. The futility of applying robust ground support to resist salt creep is demonstrated by the creep consolidation calculations in Appendix B, which show how a thick-walled steel overpack could be deformed in a few decades. Ground support would be maintained frequently for service tunnels that are expected to stay open for 50 years or longer. Long-term maintenance would involve removing the ground support, recutting the opening, and re-installing ground support. Ground support is discussed further in Section 6.6.

Crushed salt backfill would consist of "mine-run" material produced from excavating the next disposal tunnel. Due to bulking, approximately 35% of all excavated salt would be hoisted to the surface and stored permanently in a landfill facility for tailings. Creep reconsolidation of the backfilled disposal tunnels would be complete in approximately 30 years (Appendix B). The central support area and shaft pillars would be maintained in stable condition by frequent maintenance, throughout the operational lifetime of the repository. Model predictions of opening stability in salt depend on the overburden stress, temperature (e.g., heat dissipated from emplacement areas), tunnel layout, and salt characteristics.

Sequence – Repository infrastructure (highways, rail, utilities) would be started first, followed by surface facilities construction. Modular facilities such as the WRPF described above would be constructed one at a time, ramping up disposal throughput in the early years of operation. The WRPF concept is simple (e.g., no fuel pool) which would expedite completion, but functions such as bare fuel transfer would need to be performed by other facilities.

Shaft sinking and underground support services would commence next, followed by underground facility development. When the entire disposal system is ready to deliver waste packages underground, mining would begin on the first disposal heading. Construction and backfilling activities would continue throughout the operational period.

Waste receipt, packaging, and emplacement operations, supported by additional underground development, would proceed for approximately 50 years (Section 8). Emplacement could be complete around calendar 2100 if it starts in 2048, which could involve special measures to manage the heat from the youngest and hottest fuel available near the end of repository operations.

After completing emplacement and performance confirmation, and obtaining authorization to close the repository, the remaining openings, boreholes, and shafts would be plugged and sealed. Closure activities have been estimated to take 10 years (Hardin et al. 2012). A monitoring program would likely be performed for at least several decades after permanent closure.

Secondary Waste From Repository Operations – Sources of secondary waste from repository operations could include (from Hardin et al. 2012):

- Cask, facility and equipment decontamination activities

- Tooling and clothing
- Facility ventilation filtration
- Chemical sumps
- Carrier and transporter washings

Only waste classified as LLW would be generated by normal repository operations, and no greater-than-Class-C (GTCC) or mixed waste types are anticipated. Waste generation would be minimized by handling CSNF in canisters that are sealed in upstream facilities. Based on excavated volume, repository excavations would have ample capacity for disposal of LLW generated by all repository operations.

6.3 Thermal Management

Thermal constraints in bedded salt (peak temperature limited to approximately 200°C) are related to halite decrepitation from flash-boiling of small amounts of pore water, and also from decomposition of hydrous minerals such as clays and polyhalite that could occur in layers above or below the host interval. Thermal constraints are the basis for the layout dimensions discussed above.

Two studies of CSNF disposal in bedded salt have been published recently: for CSNF in 12-PWR size packages (Hardin et al. 2012), and DPC-based waste packages containing up to 32 PWR fuel assemblies (or BWR equivalent; Hardin et al. 2013). These studies have shown that the peak thermal output of each package determines the local peak salt temperature, and that the waste package diameter is of secondary importance (within the diameter range evaluated). Explanations for this include: 1) waste package spacing is large enough that radiogenic heating in each package has decreased by the time significant heat arrives from adjacent packages; and 2) waste packages are generally large enough that heat flow in the near field where peak temperatures occur is only weakly radial. The latter result allows the waste package emplacement power limit to be specified independently from waste package size. Note that increasing emplacement tunnel and waste package spacings cannot lower peak package temperature, or allow the emplacement thermal power limit to increase, if peak salt temperature is generated by local heating.

The relationship between CSNF burnup and age out-of-reactor, and the 10 kW emplacement power limit for a salt repository, is shown in Figure 6-3. Even high-burnup CSNF (here 60 GW-d/MTU) can be disposed of in 21-PWR size packages after approximately 50 years of decay storage.

An example 3D finite element simulation of waste packages containing 32 PWR assemblies with 60 GW-d/MTU burnup, aged 70 years out-of-reactor, is shown in Figure 6-4. This calculation uses temperature-dependent host salt thermal conductivity (WIPP salt properties), and a porosity-conductivity history for backfill based on fully coupled simulations (Hardin et al. 2013, Section 5).

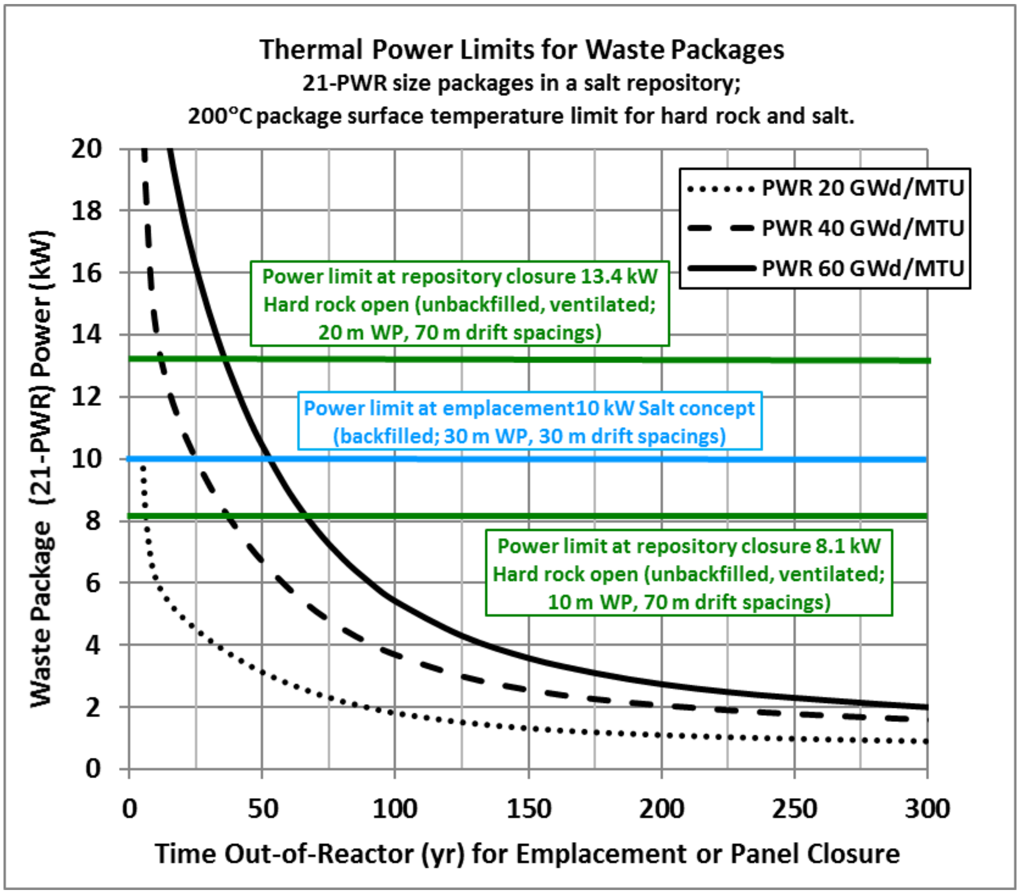

Figure 6-3. Burnup *vs*. cooling time to meet a 10 kW emplacement power limit, for 21-PWR size waste packages.

The effect from semi-cylindrical floor cavities is to lower the peak temperature in early time by more than 50°C, comparable to decades of additional aging before emplacement (Figure 6-4). These calculations bound the long-term behavior of 21-PWR size packages (compared to 32-PWR size packages) but accurately represent the short term peak temperature which depends on the emplacement thermal power limit (10 kW).

These calculations also show the basis for the 30-m package spacing and 30-m disposal tunnel spacing. The calculations assume a thick host salt unit, but decreased vertical heat dissipation through adjacent non-halite strata could be mitigated by additional decay storage before emplacement. With floor cavities there would be flexibility to decrease the package spacing or increase the emplacement power limit. With cooler, older CSNF the spacings might also be reduced, especially the package spacing which would be more cost effective. However, if spacings are reduced too much (e.g., areal average heating rate exceeds 10 kW / (30 m × 30 m) = 11 W/m^2) then there is increasing possibility of more complex thermal histories whereby the maximum temperature at a package occurs tens to hundreds of years after emplacement, from the combined effects of adjacent packages. This double-peak effect could be managed to reduce the repository footprint, but it would complicate loading of the repository (requiring a history of CSNF deliveries) and is not considered further in this scoping report.

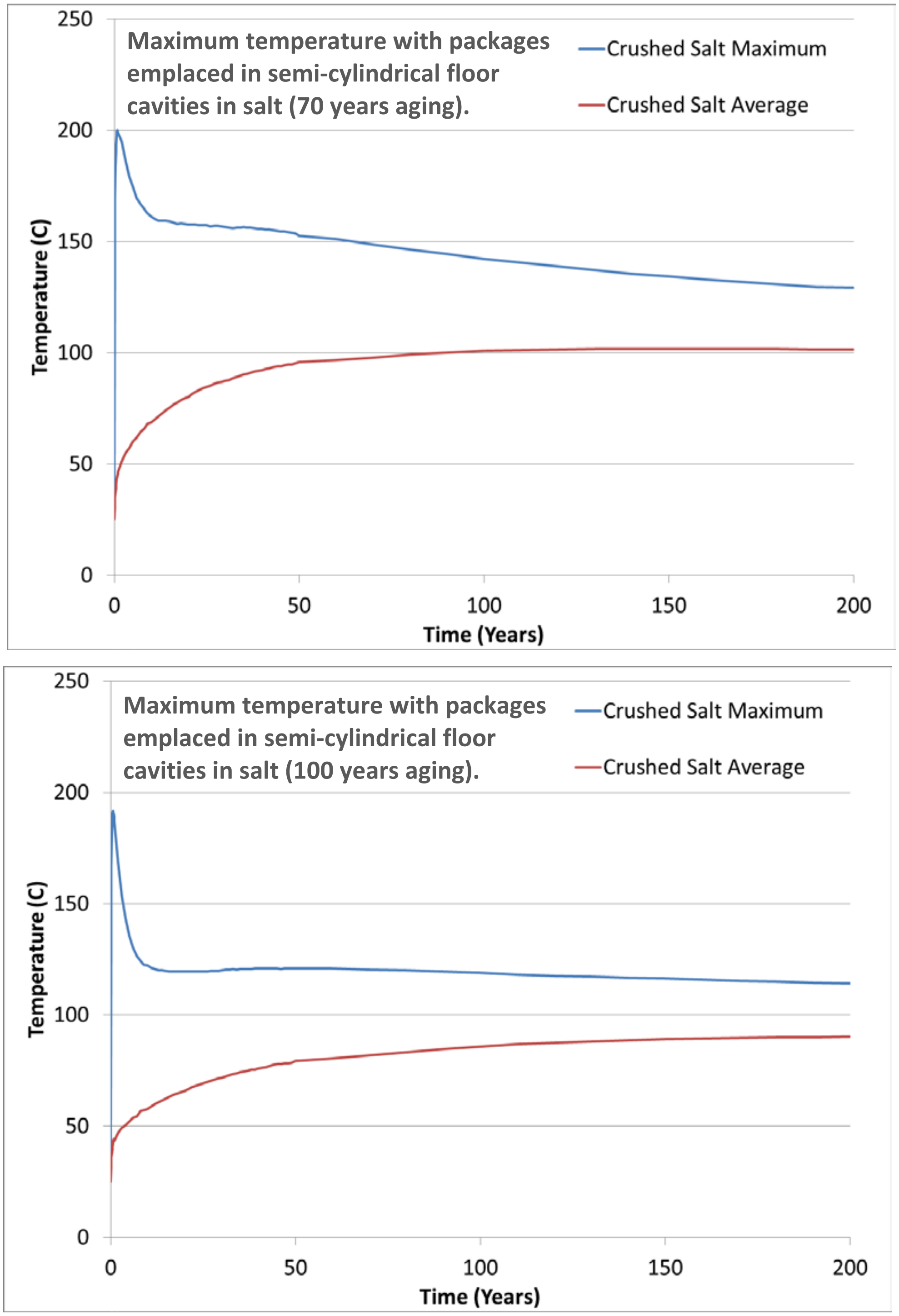

Figure 6-4. Temperature histories for high burnup SNF in 32-PWR size packages (from Hardin et al. 2013) showing ways to lower the maximum waste package surface temperature ("crushed salt maximum") with in-tunnel emplacement: (upper) floor cavities, and (lower) 100 yr *vs*. 70 yr of decay storage.

6.4 Shielding

Spent fuel requires shielding from neutron and gamma radiation to protect workers in surface facilities (outside of shielded hot cells), on or near waste transport vehicles, and during all underground operations. Waste packages would have heavy metallic wall thickness that provides some shielding, however they would not be sufficiently shielded for worker protection. Thus, supplemental shielding is required for "unshielded" waste packages in the repository operations area. The potential weight of shielding for underground transport including down the waste shaft, is an important engineering concern.

Some international concepts have incorporated self-shielding waste packages, such as the NAGRA (2002; 2009) reference concept for CSNF in which each packages would contain 4 PWR fuel assemblies. Others such as the Swedish KBS-3 concept (SKB 2011) have heavy-walled, but unshielded spent fuel packages. Self-shielding greatly increases weight and becomes especially problematic for larger capacity waste packages (i.e., 21-PWR) that generate stronger radiation output.

The weight of shielding needed for worker access near waste packages in transit tends to drive the payload capacity of the waste shaft hoist, and the overall weight of transporters such as that described in Section 3. Accordingly, an analysis of shielding mass is provided here. Another question is also addressed: the extent of shielding provided by crushed salt backfill that would be used to cover unshielded waste packages during emplacement operations.

Mass of Supplemental Waste Package Shielding for Surface and Underground Transport

Repository waste packages would not generally shield CSNF enough to reduce dose to safe levels. Thus, additional shielding would be needed for surface and underground operations. However, the mass of this extra shielding would be balanced against the ability to safely lift and transport the waste package. This is especially important if the repository is accessed using vertical shafts, so that the need for extra hoist capacity needs to be balanced against the need for reduced dose rates for worker access. To evaluate that trade off, this section correlates shielding mass with neutron and gamma dose rates.

The geometry assumed here is that the waste package would be transported down the shaft in a cylindrical shield with an operable end-door, then transferred underground to the TEV which has a mailbox shaped shield with both an end-door for loading and a bottom door for emplacement. The waste package would be loaded and sealed in vertical orientation, then lowered into horizontal orientation for transport, underground transfer, and emplacement.

To provide context, a variety of transportation and transfer overpacks have already been developed. Overpack weights (Greene et al. 2013) are shown in Figure 6-5 with a 50 MT payload, equivalent to the typical mass of a loaded TAD canister (OCRWM 2008) for purposes of comparison.

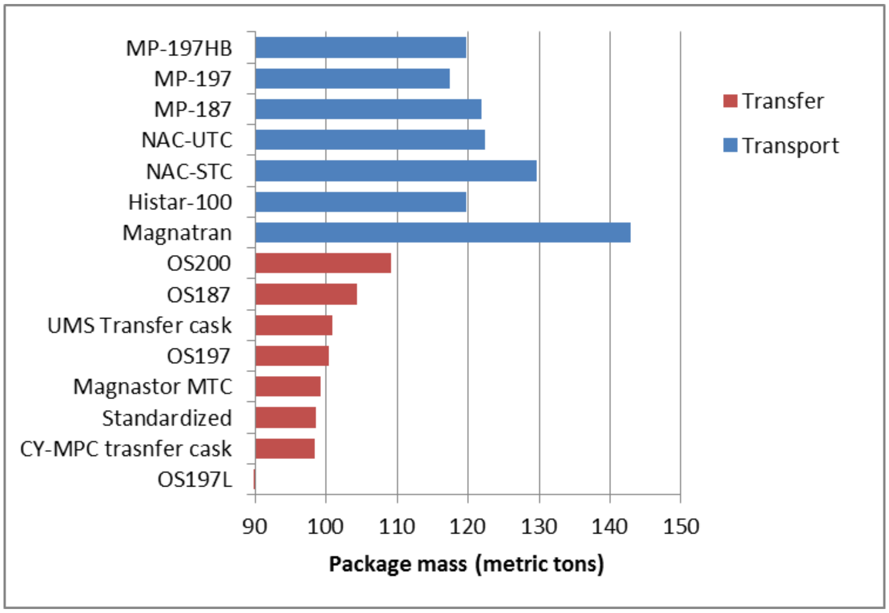

Figure 6-5. Masses of existing overpacks with a 50 metric ton payload without impact limiters.

Using the SCALE code package, layers of shielding were applied to a 21-PWR TAD canister housed in a 7-cm thick carbon steel emplacement overpack. The source term was assumed to be Westinghouse 17x17 PWR assemblies with 45 GW-d/MTU burnup and 20-year age out-of-reactor. This bounds a TAD canister loaded with similar BWR fuel, and approximately represents the 10 kW emplacement thermal power limit for waste packages in a salt repository.

The supplemental shield was assumed to be constructed of stainless steel with a lead gamma shield layer of uniform thickness, and a neutron shield layer of uniform thickness. The neutron shield material was assumed to be a vinyl ester resin known as VyalB (Issard 2009). It is a representative neutron shielding material that is used in licensed transportation casks. Its use for this analysis does not obviate selection of different materials in design.

Figure 6-6 shows surface dose rates on the cask mid-plane plotted against total weight including the loaded TAD canister, overpack, and supplemental shielding. Mass was evaluated assuming an inner shield liner of stainless steel with thickness of 12.7 mm, a steel liner between the gamma shield and neutron shield with thickness of 6.3 mm, and an outer sheath of stainless steel with thickness of 6.3 mm. Note that addition of an extra 6.3 mm to the outer sheath thickness would add approximately 3 MT to the shield mass.

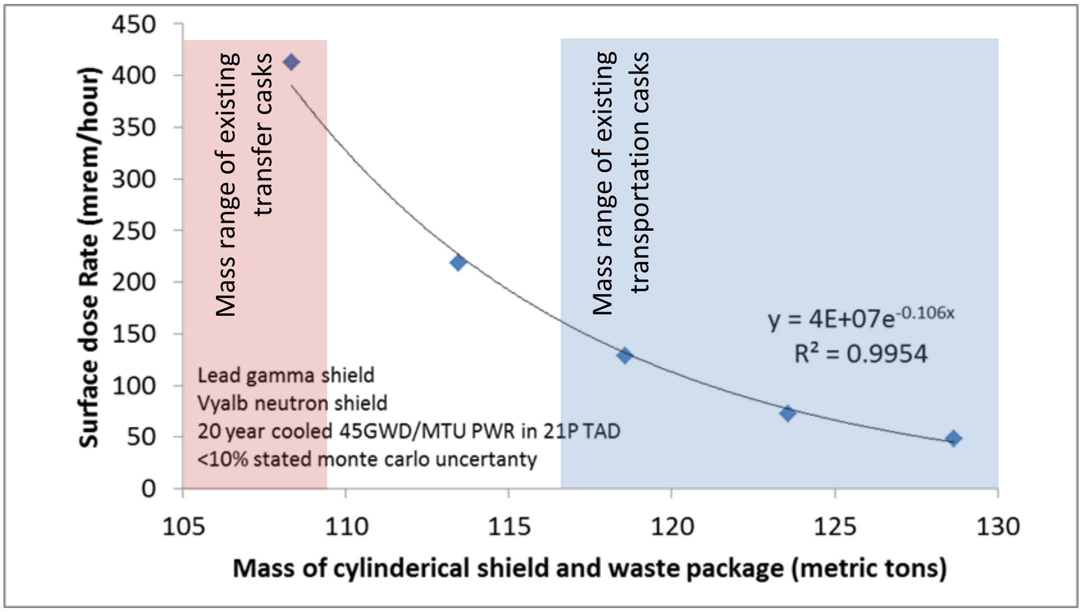

Figure 6-6. Surface dose rate *vs*. package weight.

To provide context, the background of Figure 6-6 shows ranges of weights for DPC transfer and transportation overpacks loaded with 50 MT payloads. For the transportation casks, the maximum permissible surface dose rate is 1,000 mrem/hour, and the maximum dose rate for publicly accessible surfaces is 200 mrem/hour (10CFR71). Transfer casks do not have maximum dose rates but are subject to ALARA. The annual total effective dose equivalent for workers is 5 rem (10CFR20).

Points in Figure 6-7 represent optimal dose configurations obtained from Figure 6-6, which shows the impact of varying the proportions of neutron and gamma shielding. Note that the mass of stainless steel liners is not included in the total masses in Figure 6-7.

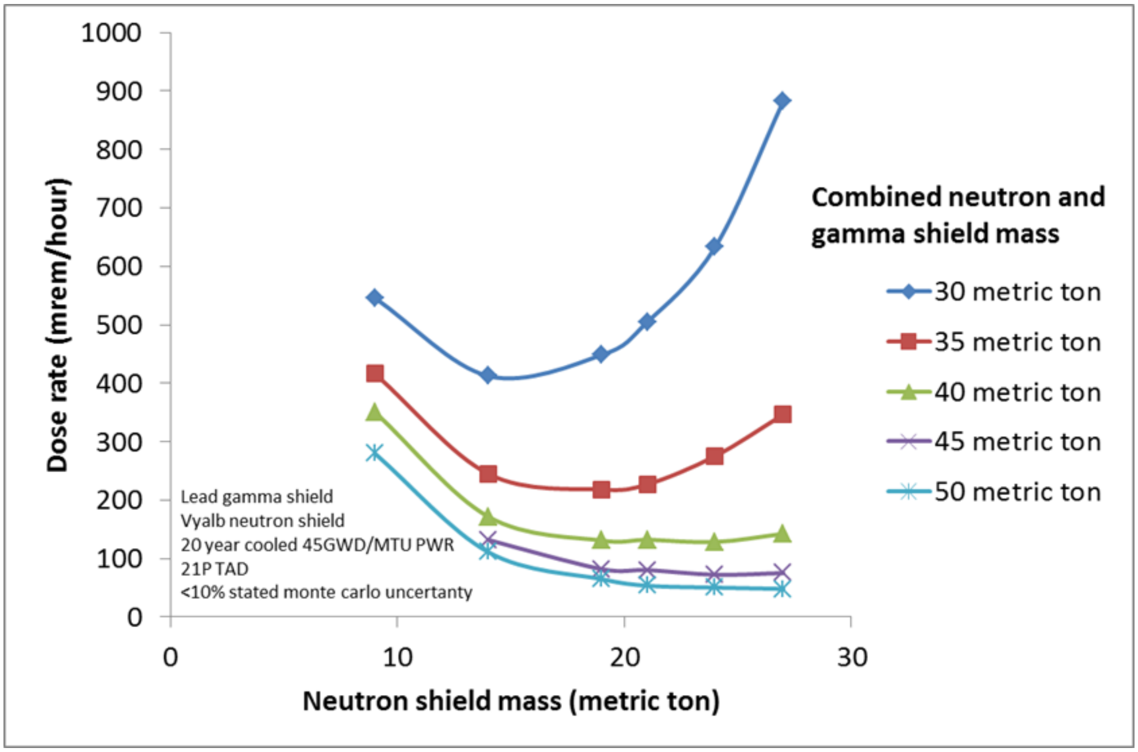

Figure 6-7. Dose rates *vs* neutron shield mass.

An alternative supplemental shield configuration was considered that uses a "mailbox" shape (Figure 6-8) to facilitate loading and unloading a waste package. This results in approximately a 7 MT mass penalty (Figure 6-9). This number might be reduced slightly as there are nulls near the corners of the supplemental shielding. With the mailbox configuration, to produce dose rates less than 100 mrem/hour at the surface (a reasonable limit for normal and off-normal operations) would require a total weight of 120 to 130 MT. This is the heaviest configuration evaluated, and consistent with the shaft hoist payload discussed in Section 7.

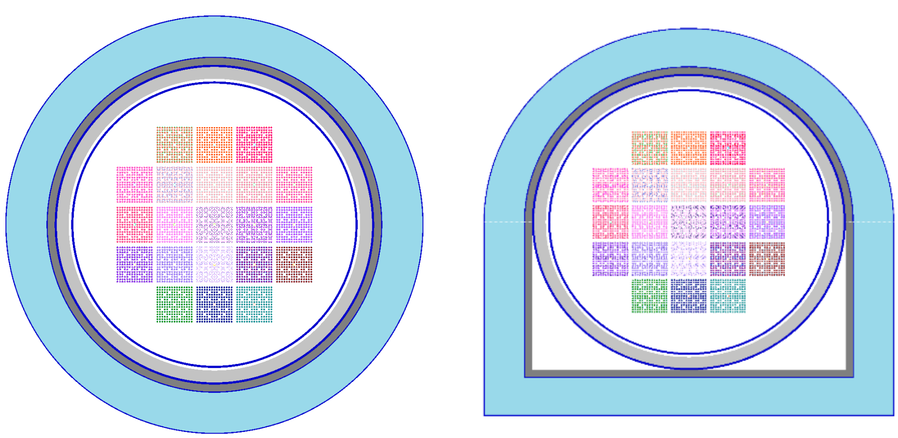

Figure 6-8. Alternative supplemental shield geometries.

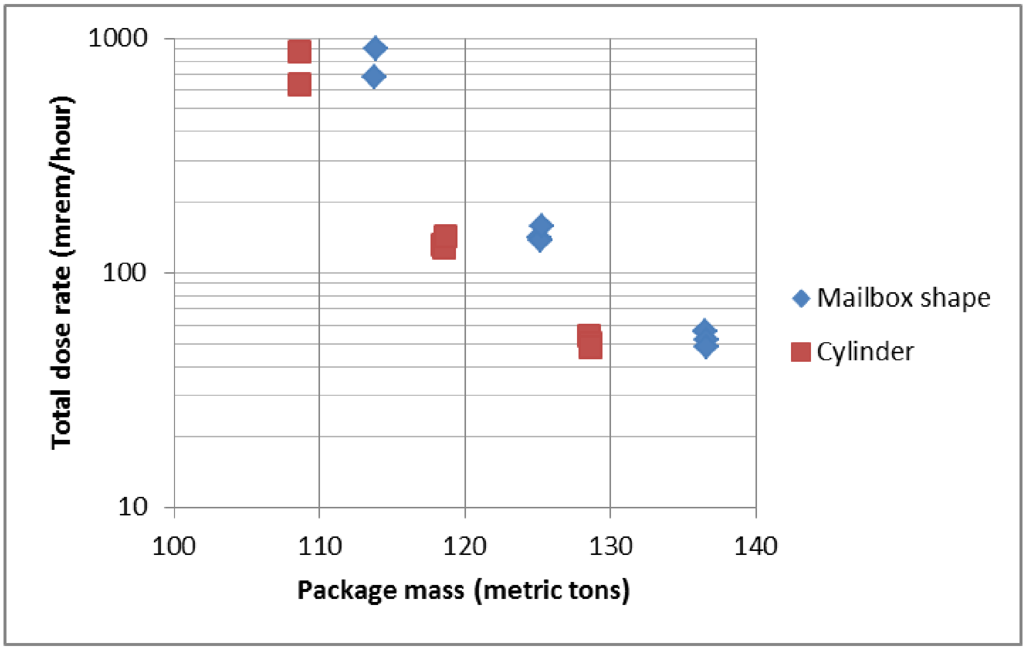

Figure 6-9. Calculated masses *vs.* dose rates for packages of comparable shield thicknesses.

Shielding by "Run of Mine" Crushed Salt Backfill During Emplacement Operations

Another shielding case was evaluated to examine the dose rate in the headspace remaining after disposal tunnel backfilling operations, directly above a package. The tunnel was assumed to be 6 m wide by 4 m high (Figure 6-10). Backfill was assumed to be 3 m deep with porosity of 40%. Waste packages were assumed to be sufficiently spaced (30 m center-to-center) that dose rate is not affected by other waste packages. Dose rate is dominated by neutrons. Total dose rate would be approximately 50 mrem/hr in the space above the backfill (Figure 6-11).

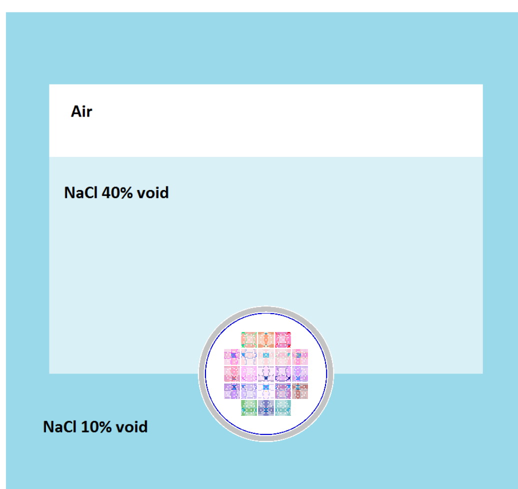

Figure 6-10. Arrangement for crushed salt backfill shielding analysis.

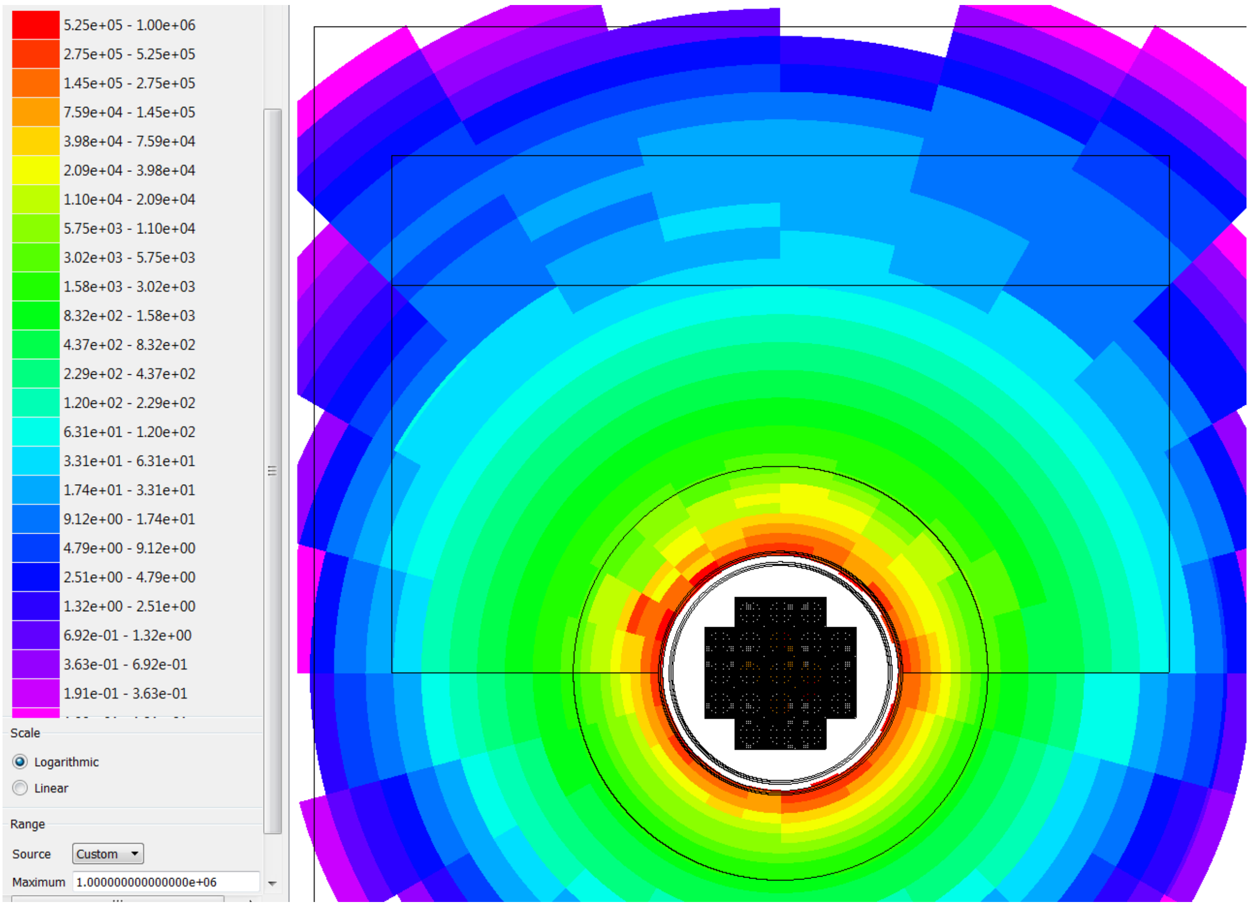

Figure 6-11. Total dose rate (mrem/hour) above crushed salt backfill (tunnel and backfill cross-sections shown by black lines).

6.5 Plugs and Seals

A CSNF repository in salt would have three types of seals and plugs installed at or before permanent closure: shaft seals, panel closures, and borehole plugs. Each of these has been addressed for the WIPP certification application (DOE 2014; 2004). Implementation of seals and plugs in a new repository would be based on site specific information, and the specifications could be refined as more information is collected during characterization and monitoring.

Shaft Seals – The functions of shaft seals would be to prevent radionuclides from reaching the accessible environment, and to restrict groundwater flow. Seals would be constructed from engineered materials that are readily available and possess good long-term stability. The seal system design would protect against structural failure of system components, minimize subsidence, and resist accidental entry. An example shaft sealing concept developed for WIPP is shown in Figure 6-12. The performance analysis, and the field and laboratory testing used to verify the design, are described in WIPP documentation (DOE 2004; Hansen and Knowles 2000).

The shaft liner system would be removed prior to, or during shaft seal installation to expose the rock layers. The liner (mainly cast iron or concrete) would be removed and grout chipped away, except in intervals that require liner support to stay open. The liner would be left in place in low-strength units, and especially in aquifers (which may have been frozen prior to shaft construction).

Shaft sealing would be accomplished by installing alternating lifts of engineered materials, using construction methods specific to each material:

- Salt-conditioned concrete
- Compacted clay
- Asphalt components
- Compacted salt
- Cementitious grout
- Earthen fill

For some of these sealing elements such as the concrete elements, keyways would be excavated into the shaft wall in certain layers to enhance mechanical support of overlying elements, and promote hydraulic sealing. Field and laboratory tests on sealing materials, and demonstration of construction methods were discussed by Hansen and Knowles (2000).

Panel Closure

The 1998 WIPP certification application specified that waste panel entries would be sealed using "Option D" which involves installation of a concrete-block wall, removal of the DRZ along a 30-m section of the panel access, and emplacing a monolithic concrete plug (rigid inclusion) in that interval. The concrete mix ("Salado Mass Concrete") would consist of a Portland-cement based aggregate concrete with sufficient NaCl added so that is it "salt saturated" and does not dissolve halite beds where in contact. This concept remains the baseline panel closure design at present, although DOE and EPA correspondence in 2002 and again in 2007 proposed changing the closure design to rely on crushed salt reconsolidation rather than a rigid inclusion.

For a CSNF repository in salt, organic solvents or other volatile chemicals would not be emplaced with the fuel, so that panel closure function would only limit postclosure radionuclide migration (and not control any other hazardous waste). There would be some redundancy with the function of crushed salt backfill in all the emplacement tunnels. Requirements for panel closures, and their design, would be worked out in the repository design process. This discussion serves to show that viable options are available.

Borehole Seals

Exploratory and monitoring boreholes near the repository would be plugged following applicable State and Federal regulatory requirements for plugging and abandoning wells (DOE 2014, Section 44). These requirements typically involve fully cementing every borehole to the surface, with chloride salts added to the cement in evaporite intervals.

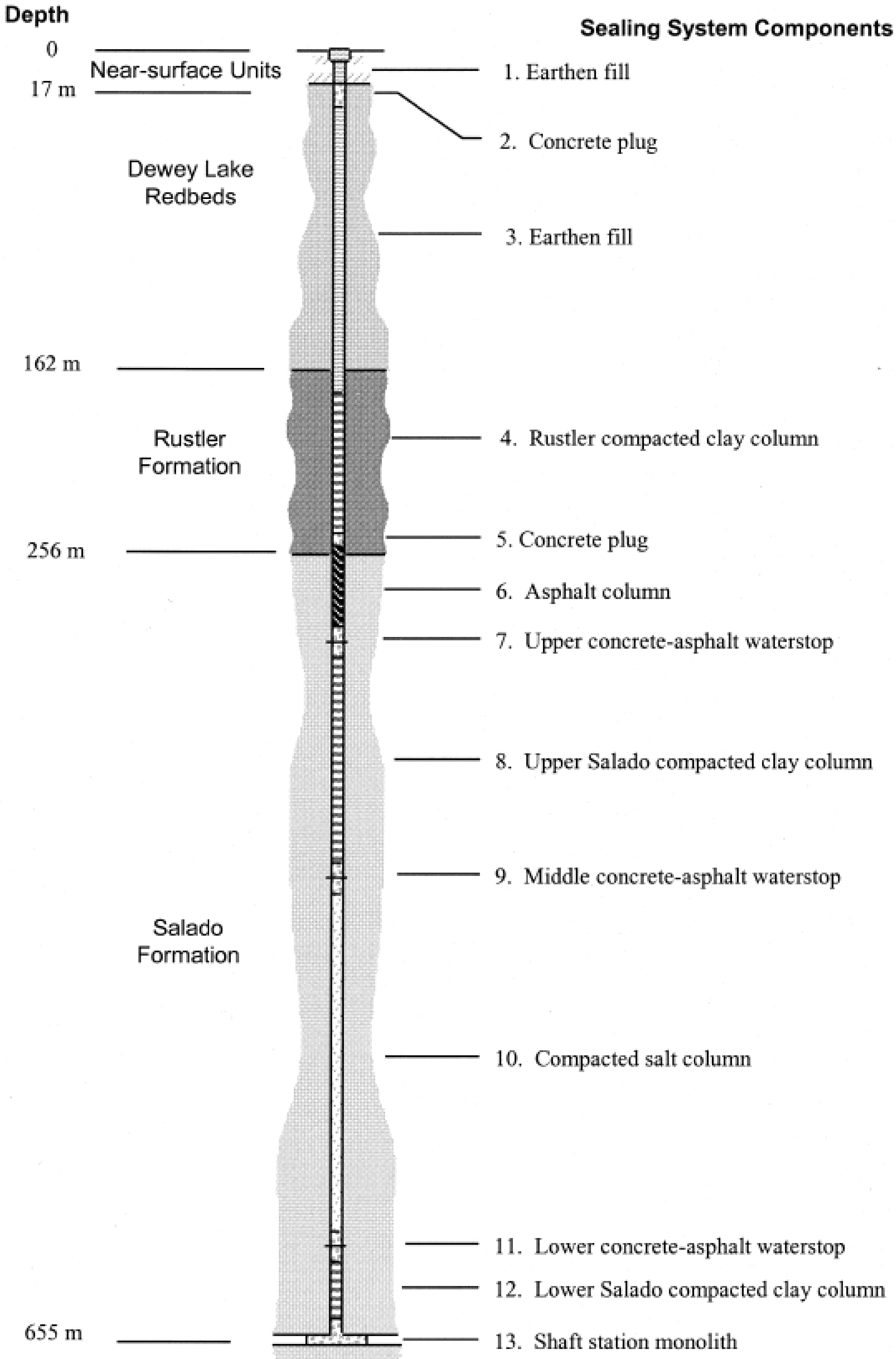

Figure 6-12. Shaft sealing plan for WIPP (from Hansen and Knowles 2000).

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7. Shaft Hoist

Sandia National Laboratories (SNL) contracted DBE Technology GmbH (DBE Tec, now BGE Tec) to describe the technical basis for a shaft hoisting system suitable for the transportation of payloads up to 175 MT. As noted in Sections 3.1 and 6.4, this capacity would be sufficient for lowering a 21-PWR size waste package with additional shielding and a transport cart, into a deep repository.

German radioactive waste management organizations have invested considerable effort during the past 30 years in developing the technology for safe shaft transport of canisters and waste packages for disposal. The DBE Tec report (Nieder-Westermann and Bertrams 2017) is a current summary of work to design and test hoisting systems that would have payload capacities 2 to 3 times the largest hoists used in mining. The DIREGT project (*Direct Disposal of Transport and Storage Canisters*) is described, which was tasked with designing a hoist system for payloads of up to 175 MT, that would accommodate CASTOR® shielded bare-fuel casks currently used for SNF transportation and storage in Germany. The now-completed DIREGT project was an R&D effort to evaluate the feasibility of direct disposal of SNF in CASTOR® casks, which can each weigh up to 160 MT. A transport cart would constitute the other 15 MT. The target disposal site was a potential repository in the Gorleben Salt Dome at a depth of approximately 870 m below ground surface.

The DIREGT project was an advancement of the earlier DEAB project that developed the technology and engineering solutions for shaft transport of POLLUX canisters with maximum weight of 65 MT, plus a transport cart weighing 20 MT. It was executed in three phases from 2006 to 2014:

- DIREGT I examined overall feasibility, basic design and repository layout requirements, and long-term effects from direct disposal of TSCs, including criticality and thermal-mechanical analyses.
- DIREGT II focused on technology and design development, including shaft hoisting requirements, operational considerations including failure mode analyses, and refined criticality and thermal-mechanical calculations.
- DIREGT III developed a detailed conceptual design including subsurface waste transportation systems with special emphasis on licensable shaft hoisting, waste emplacement, and confirmation of thermal-mechanical and criticality considerations analyses.

The DBE TECHNOLOGY report to Sandia (Nieder-Westermann and Bertrams 2017) focused on conceptual design for the shaft hoisting system. The 175 MT system is an evolution of the more mature design for an 85 MT system, discussed previously. Both the DIREGT and the DEAB concepts were included in the concept of operations developed for *Preliminary Safety Assessment Gorleben* (Bollingerfehr et al. 2012a; 2012b) completed in 2013. In addition, the DEAB design has been adapted for an 80 MT hoisting load capability for use in the Belgian repository program under contract from ONDRAF/NIRAS.

The report also describes a safety analyses conducted to confirm that the shaft hoisting concept would be expected to meet the accepted value for residual risk as applied in German nuclear power plants (Nieder-Westermann and Bertrams 2017). Repository safety regulations in the U.S. (e.g.

10CFR63) have different requirements, and preclosure safety analysis for a large-capacity shaft hoist would be an appropriate part of conceptual design for a U.S. repository in salt.

Finally, a rough-order-of-magnitude (ROM) cost estimate was provided for both hoist systems based on anticipated European material costs. The material and installation costs for a large shaft hoist would be a small fraction of the overall development cost for a repository regardless of the geologic medium.

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8. Logistical Analysis of Repository Operations

Disposal system logistical analysis was undertaken to examine the timing, facility throughput, thermal aging, and other aspects associated with disposal of standardized 21-PWR size multi-purpose canisters in a repository constructed in bedded salt. For this analysis, sealing of canisters into disposal overpacks was assumed to be done at the repository. Packaging of CSNF into 21-PWR/44-BWR size TAD-type canisters, and re-packaging from DPCs into TAD-type canisters if needed, is assumed to be done at upstream facilities. Note that the modeling described here may allow packaging of CSNF into TAD-type canisters at the repository, but this does not affect disposal timing which is controlled by throughput and thermal cooling. In the model, sufficient resources were assumed that packaging or re-packaging in TAD-type canisters was not critical path.

The logistics study used the simulation tool TSL-CALVIN, like previous studies including Jarrell et al. (2015), Kalinina (2014), and Nutt (2013). The tool couples the legacy Civilian Radioactive Waste Management System (CRWMS) Analysis and Logistics Visually Interactive model (CALVIN) and Transportation Operations Model (TOM) into a framework for evaluating the entire system for managing SNF. Within the transportation-storage logistics (TSL) tool, TSL-CALVIN simulates the logistics (and costs, if requested) associated with managing SNF across various facilities within the system (reactors, storage, and disposal facilities). This includes the schedules for fuel selection and shipping between facilities. TSL-TOM then uses the shipping schedule to analyze the transportation portion of the waste management system, however, no TSL-TOM analysis was performed for this study.

CALVIN is a system-level tool capable of simulating a broad set of potential future scenarios for managing CSNF (Nutt et al. 2012a). It can represent a range of back-end fuel management scenarios involving at-reactor storage, storage at one or more off-site interim storage facilities, and ultimate disposal with package-size and thermal constraints specific to geologic settings. The TSL tool can be used to evaluate different SNF pick-up scenarios within broader overall scenarios, and model the logistics of transportation, storage, re-packaging (as needed), and disposal. CALVIN was originally developed in the early 2000s and subsequently replaced by other tools specific to a Yucca Mountain repository. More recently, TSL-CALVIN was developed by leveraging the existing capabilities of CALVIN, adding additional capabilities to enable the simulation of a broader set of waste management scenarios.

Historical data on the fuel inventory comes from the UNF-ST&DARDS database (Lefebvre et al. 2017). The time step used in CALVIN is one calendar year. SNF is stored and transported in CALVIN only when it meets the storage or thermal limit for the canister being used. Waste package emplacement is done in CALVIN when the total heat output for a particular canister is below the repository emplacement thermal limit.

The CSNF management and disposal scenarios used for this analysis are:

- Scenario 1 (961): Base case with 10 kW repository emplacement thermal power limit, and transition to loading TAD-type canisters (either 21-PWR or 44-BWR size) at reactor sites starting in 2025. CSNF that is loaded into DPCs prior to 2025 would be re-packaged into TAD-type canisters.
- Scenario 2 (962): Scenario 1 with direct disposal of half the total inventory of CSNF that is in DPCs in the year 2025, if those DPCs are made from materials that are assumed to

be not readily degradable (i.e., without aluminum, carbon steel, etc.; Hardin et al. 2015). CSNF from other DPCs (not directly disposable) and from pools, would be loaded into TAD-type canisters starting in 2025.

- Scenario 3 (963): Scenario 1 with 8 kW repository emplacement thermal limit.
- Scenario 4 (964): Scenario 1 with 12 kW repository emplacement thermal limit.
- Scenario 5 (965): Scenario 2 with 8 kW repository emplacement thermal limit.
- Scenario 6 (966): Scenario 2 with 12 kW repository emplacement thermal limit.
- Scenario 7 (967): Scenario 1 using canisters loaded with either 21-PWR or 44-BWR assemblies, at reactor sites, with the transition from loading DPCs to loading TAD-type canisters at reactor and fuel storage sites, delayed until 2036. CSNF that is loaded into DPCs prior to 2036 would be re-packaged into TAD-type canisters.
- Scenario 8 (968): Scenario 2 using canisters loaded with the transition from loading DPCs to loading TAD-type canisters at reactor and fuel storage sites, delayed until 2036.

Note that 2036 is the year when it is assumed that the disposal site and any associated constraints are known, consistent with the repository opening for disposal operations in 2048.

The following DPC types are considered to be directly disposable (in Scenarios 2, 5, 6, and 8): DSC-24PTH, DSC-32PT, DSC-32PTH, DSC-61BT, DSC-61BTH, MPC-24, MPC-24EEF, MPC-32, MPC-68, MPC-HB, DSC-7P, CY-MPC, MPC-LACBWR, TSC-24, and Yankee-DPC.

The base scenario (961) makes the following assumptions:

- All reactors operating in 2025 begin to load CSNF in TAD-type canisters (21-PWR or 44-BWR) instead of DPCs used previously.
- All SNF passes through the Interim Storage Facility (a full-scale ISF opens in 2025).
- Repository begins disposal operations in 2048.
- Repository emplacement thermal power limit of 10 kW per waste package.
- The ISF and the repository have a 3,000-MTHM/year CSNF throughput rate.
- No DPCs are assumed to be directly disposable (this assumption is modified in Scenarios 2, 5, 6 and 8).
- Packaging of CSNF in canisters is done upstream of the repository in a manner that does not impact the schedule for disposal.
- Packaging of canisters in disposal overpacks is done at the repository.
- A total of 138,000 MTU of CSNF is generated over the operating lifetimes of all existing and shut down reactors (a value of 140,000 MTU is used elsewhere in this report).

Additional assumptions and conventions used in TSL-CALVIN are discussed in Appendix A of *Used Fuel Management System Architecture Evaluation, Fiscal Year 2012* (Nutt et al. 2012b).

Year-by-year breakdowns of waste package throughput (limited to 3,000 MTHM per year) are calculated for each of the analyzed scenarios. Output files containing information about CSNF arriving at the repository were also produced. Comparing Scenarios 1 and 2 (Figures 8-1 and 8-2) emplacement is substantially done by 2110, and more re-packaging (Scenario 1) can be done without additional delay of disposal operations (in accordance with assumptions used in the analysis) although it would affect costs.

The modified repository emplacement thermal limits (Scenarios 3 through 6, plotted in Appendix C) cause the completion of disposal to be delayed to 2125, or expedited to approximately 2100, for thermal limits of 8 and 12 kW, respectively. Delaying the transition to loading TAD-type canisters until 2036 (Scenarios 7 and 8) does not appreciably change the schedule for disposal, which is expected because the CALVIN algorithm allocates re-packaging resources as needed for these cases so that disposal proceeds as the fuel cools.

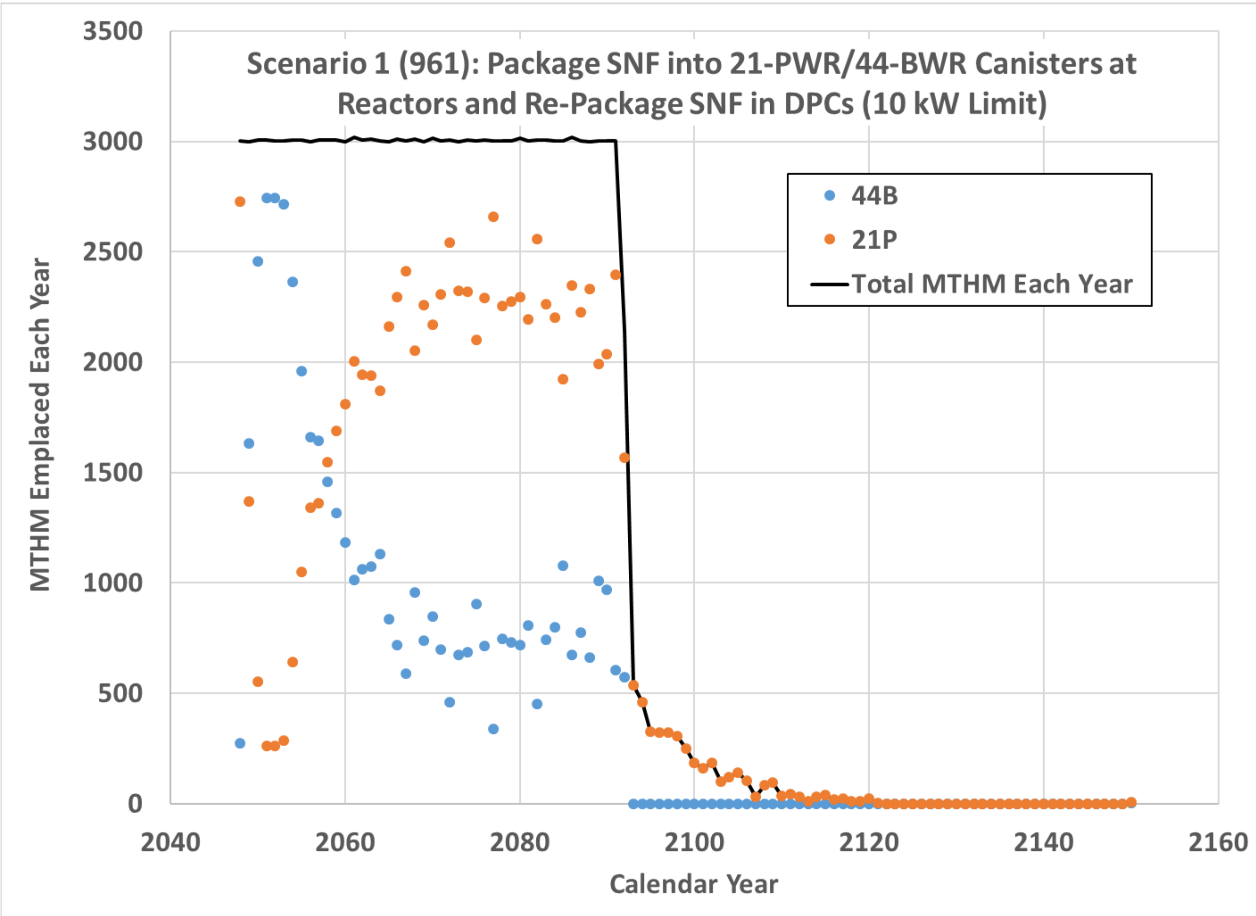

Figure 8-1. Simulation of repository throughput for Scenario 1, in which all CSNF is originally packaged, or re-packaged, into 21-PWR/44-BWR size canisters (10 kW thermal limit; loading of 21-PWR/44-BWR size TAD-type canisters begins in 2025).

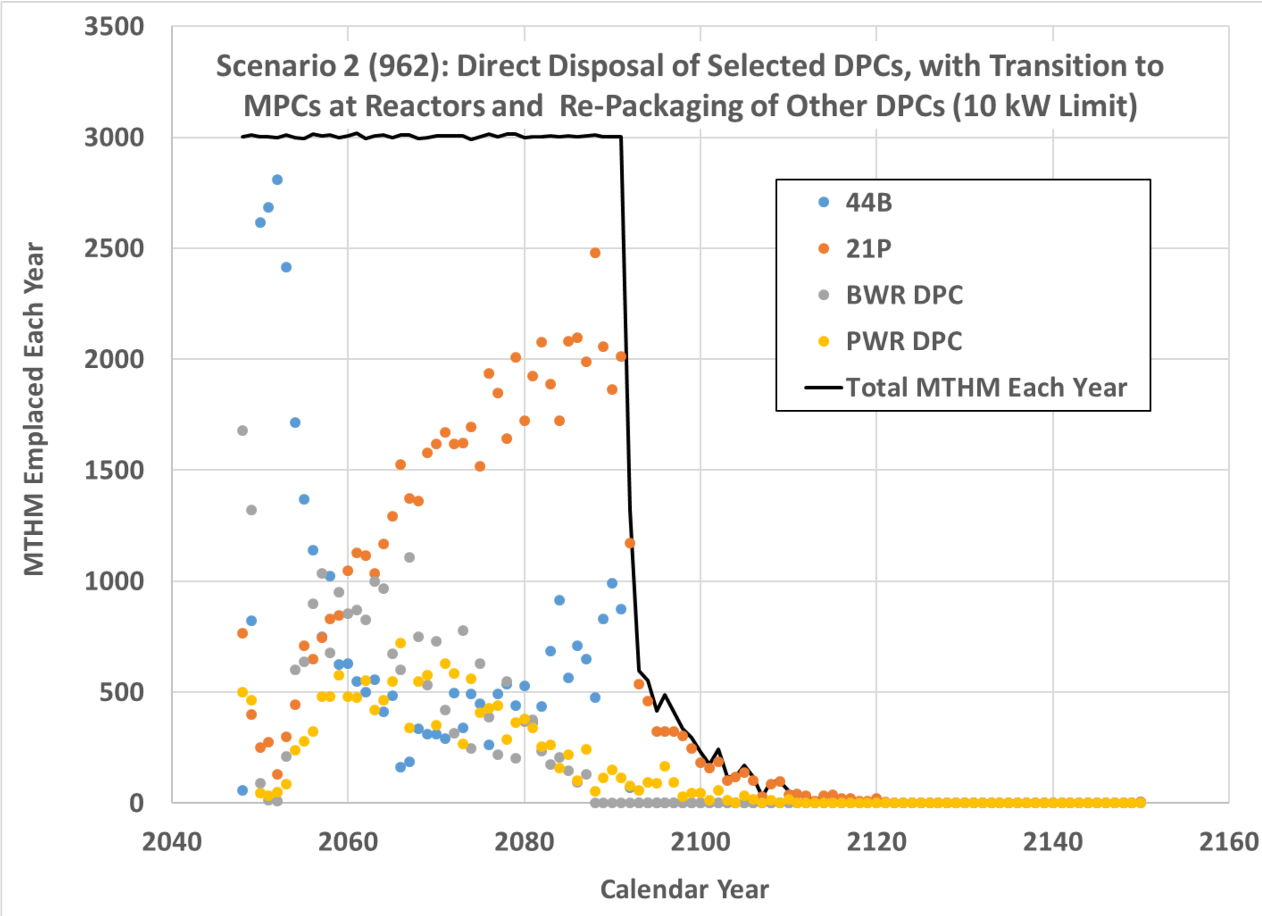

Figure 8-2. Simulation of repository throughput for Scenario 2, in which approximately selected DPCs are disposed directly, and the remainder CSNF is either originally packaged, or re-packaged if not in DPCs deemed disposable, into 21-PWR/44-BWR size canisters (10 kW thermal limit; loading of 21-PWR/44-BWR size TAD-type canisters begins in 2025).

These results show that the disposal could be completed in the time frame 2100 to 2125 constrained principally by thermal cooling, if the CSNF is packaged into 21-PWR/44-BWR size canisters suitable for disposal. Direct disposal of CSNF in DPCs could save costs associated with re-packaging, but would not accelerate the disposal schedule (which is constrained by cooling) if sufficient re-packaging resources are available.

The results in this study were produced using CALVIN Version 0.4.5.6.6. The CALVIN Access Database file names are *TSL-CALVIN Base Database (2017)_Generic_Salt_Repo_DPC_962_965_966_968.mdb* and *TSL-CALVIN Base Database (2017)_Generic_Salt_Repo_961_963_964_967.mdb*. These databases are consistent with the FY15 standardization study (Jarrell et al. 2015). The Excel files containing the output data are *Results_Rev2.xlsx* and *Results_all_Rev2.xlsx*.

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9. Summary and Conclusions

Previous development of concepts for geologic disposal in salt has come mainly from programs in Germany and the U.S. The most straightforward concept for disposal of large, heavy packages containing CSNF would be to emplace them directly on the floor in emplacement tunnels. In-tunnel axially aligned horizontal emplacement would minimize excavated volume and avoid drilling of large-diameter emplacement boreholes, and is recommended. A similar concept was proposed in Germany for direct disposal of POLLUX® canisters (Section 2).

Repository Concept of Operation – The repository would be constructed at a depth of 500 to 1,000 m to provide isolation of the waste from the surface, and for sufficient overburden stress to ensure creep reconsolidation of repository openings. As described here (Section 6) each modular panel would consist of 10 parallel disposal tunnels, 1.2 km long, encircled by service tunnels for construction access, waste transport, and ventilation. A single disposal tunnel would contain up to 39 waste packages (spaced 30 m apart), and a panel would contain up to 390. Each disposal tunnel would be backfilled during emplacement operations using crushed salt from the next disposal tunnel excavated at the same time, thus minimizing hoisting of waste salt to the surface.

With disposal tunnel spacing of 30 m, the total area of each panel would be 0.43 km^2. Ten such panels containing 3,840 waste packages (up to 36,288 MTU PWR fuel or 30,412 MTU BWR fuel) would be constructed along an initial heading from a central core service area. Three more headings could be constructed sequentially along cardinal directions from the central core, to accommodate the estimated 140,000 MTU total U.S. CSNF inventory. As each heading is completely loaded, additional waste packages would be emplaced in the service and ventilation tunnels as needed (total capacity of these would be equivalent to an additional panel on each heading, or an additional 10%). The overall area of the layout would be approximately 20 km^2. Many layouts are possible, but the approach should be modular and expandable, excavation should be deferred as long as possible to avoid maintenance, and the layout should share support facilities and shafts.

The repository would be accessed by vertical shafts (in lieu of ramps) because this is efficient and is the state of mining practice in sedimentary basins. A large diameter shaft would be needed for ventilation exhaust, with smaller shafts for waste salt removal, men & materials, and ventilation intake. The waste transport shaft would also have large diameter to accommodate a CSNF waste package in horizontal orientation, with additional shielding.

The spacing between disposal tunnels is estimated from thermal modeling, and seeks to limit the maximum average areal thermal load in the panels to 11 W/m^2. With waste packages set on a 30 m × 30 m grid, and exponential decay of CSNF heat output, the temperature history at each package would peak immediately after emplacement and decline uniformly after that (no double-peak behavior). Peak salt temperature would thus be dominated by each package locally, simplifying thermal management. Emplacing packages into semi-cylindrical cavities milled in the floor (Figure 2-3) would enhance heat dissipation and also provide mechanical support to the overpack (Section 6.3). There would be some flexibility to change spacings or increase the emplacement thermal power limit. Backfilling immediately with mine-run crushed salt would provide shielding during operations (Section 6.4) and expedite reconsolidation of the salt formation around the waste. After reconsolidation this arrangement would ensure isolation of adjacent waste packages from one another, by the intervening backfill.

After the repository is fully loaded and the performance confirmation program is complete, all openings in the host salt would be backfilled, then shafts would be sealed, and boreholes plugged. Plans made for WIPP show how sealing and plugging could be done. A monitoring program could continue for 50 years or longer after repository closure.

Logistical Simulations – With an emplacement thermal power limit of 10 kW, nearly all the CSNF that is projected to be produced by the current fleet of commercial reactors could be emplaced in approximately 50 years starting in calendar 2048. Near the end of this loading period, hot CSNF could be blended with cooler fuel, or other measures taken for thermal management to complete the repository. Note that this is a generic result (non-site specific) and that siting and licensing are not included in logistical estimates.

Engineering Challenges – These include: 1) shaft construction; 2) large capacity shaft hoist; 3) overpack design; 4) a TEV for transporting waste packages once they are underground and emplacing them remotely; and 5) remotely operated equipment for emplacing backfill. The method of shaft construction would depend on site-specific conditions, and it could involve freezing the subsurface through aquifers or weak layers prior to shaft sinking. The lining system would also be site-specific, and could use cast iron and reinforced concrete, with injection grouting, to ensure liner performance throughout disposal and closure operations.

A shaft hoist with payload capacity of 175 MT seems technically and economically feasible based on development work for a design with 85 MT capacity (Section 7). It would be the largest hoist of its kind anywhere by a factor of 2 or 3 in terms of payload. Friction hoists are in use around the world, but engineering of a scaled-up hoist would start from a lower state of technical maturity.

The function of disposal overpacks would be to provide reliable containment during the repository operational period. This could be accomplished using a corrosion allowance material such as low-carbon steel. Development effort would be needed to determine overpack thickness (e.g., 7 to 20 cm) that can resist potential corrosion from brine in heated salt, and can resist the non-uniform loading from thermally activated salt creep during the first few decades, without breaching. Ductility of the steel, and mechanical support by the floor cavities, could be important in optimizing overpack thickness and material selection (Section 5).

The TEV would be similar to previous design concepts, particularly a design alternative that was proposed for a Yucca Mountain repository (the only other repository concept that has been intensively studied, and would have similarly large, heavy waste packages). The TEV would move over a rough salt floor on many independent wheels, and would carry heavy shielding in addition to a waste package.

Postclosure Performance Issues – By analogy to the safety case for WIPP, human intrusion is likely to be a dominant mode of radionuclide release from the repository. Generic regulations (10CFR60 and 40CFR191) have different human intrusion assessment requirements than site-specific ones (e.g., 10CFR63 and 40CFR197) have used. Treatment of human intrusion for a CSNF repository in salt could depend on promulgation of site-specific changes in the regulations.

Radionuclide release and migration from a CSNF repository in salt would be quite limited for undisturbed conditions, again by analogy to the WIPP safety case. There may be opportunities for improved understanding of salt performance with waste heating, based on future *in situ* salt testing in an underground research laboratory.

This report also describes a developing area of salt rock mechanics that involves a mechanism for low-stress, low strain-rate creep that might cause large, heavy waste packages to sink as much as 1 m in 10^4 years. The consequences of sinking to isolation performance, and the salt properties that control it, are likely site-specific. Site-specific sampling, testing, and modeling would be used to determine if the mechanism is important enough to merit consideration in design, or inclusion in performance assessment.

Part of engineering design and postclosure safety assessment for a CSNF repository in salt would be to implement a methodology to show that the probability of a criticality event in the repository when waste packages eventually breach and are flooded (i.e., with brine), is less than the probability screening threshold for performance assessment. In the methodology, a criticality analysis would be performed for waste packages in the repository, incorporating measures that could be introduced as needed to limit reactivity, for example using fuel selection and loading rules, and crediting the absorption of thermal neutrons by natural chlorine in the environment. A similar analysis has been underway for CSNF stored in dual-purpose canisters. Ideally the strategy would be developed prior to actually loading SNF assemblies into canisters used in waste packages for disposal.

Appendix A. Shielding Analysis Additional Notes

These notes are related to work to minimize the shielding mass needed to transport CSNF casks in a salt repository. The general approach is to produce shielding mass *vs*. dose rate curves to provide options to those investigating repository designs.

Material Selection

First, several materials were evaluated as gamma-only shielding. Gamma shielding efficacy per unit mass monotonically increases with atomic number as shown below, so materials such as lead and depleted uranium were selected. In the below figure, the area of interest, are within the red box.

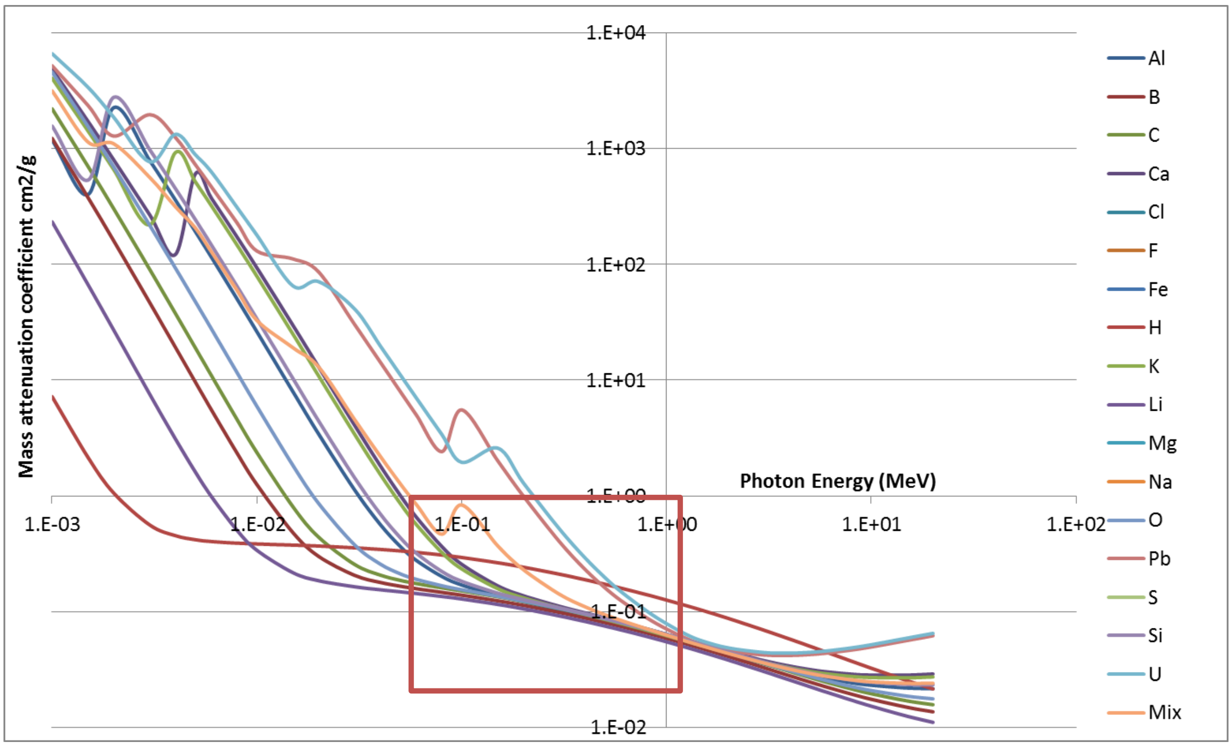


Figure A-1. Photon mass attenuation coefficients of elements for photons of various energies (NIST data).

For some higher energy gamma rays where Compton scattering dominates attenuation, hydrogen presents a larger attenuation coefficient per unit mass than lead or depleted uranium. This is due to hydrogen only having one nucleon per electron as opposed approximately two nucleons per electron found in most elements. Thus, hydrogenous materials, namely polyethylene have a relatively high attenuation per unit mass for gamma rays. Hydrogen is also a good moderator of neutrons, which is necessary for neutron shielding.

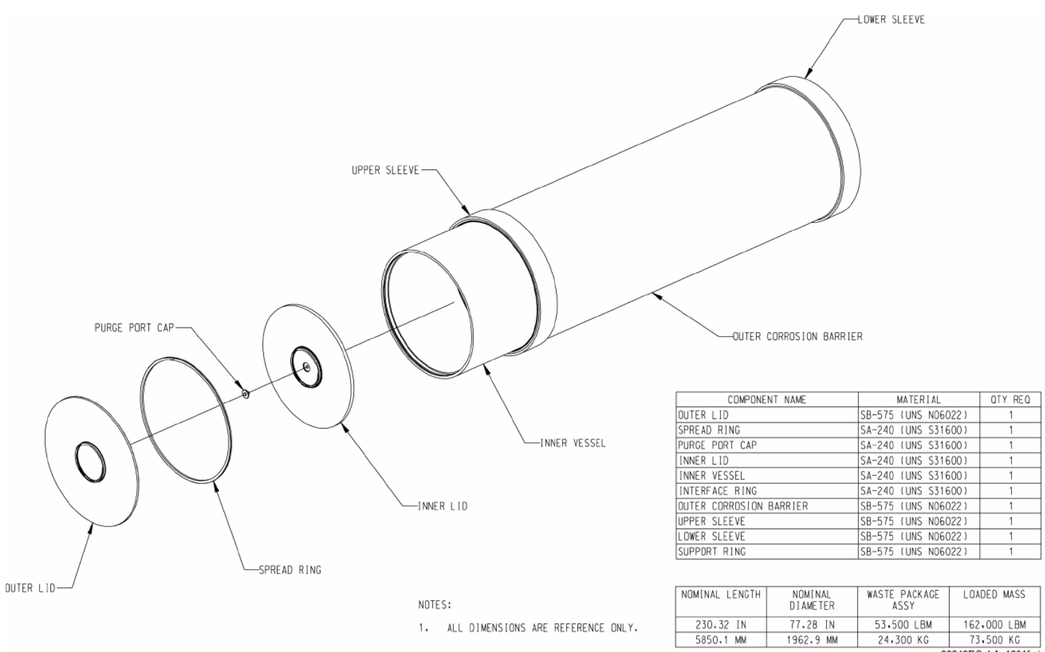
Of the hydrogenous materials, polyethylene with its two hydrogen atoms per carbon atom presents a good neutron shield. Other resins do not have as much hydrogen per unit mass, and exotic materials such as LiH and BeH_2 tend to be less stable in an oxygen atmosphere. However, polyethylene may also be consumed in a fire, so a self-extinguishing, vinyl ester, thermoset plastic impregnated with flame retardants and boron was assumed. The composition assumed goes under the tradename VYALB. It is selected as a representative neutron shielding material, since compositions were readily available, and it is used in the licensed transportation cask. Its section for this work should not hamper the use of other materials in later stages of design.

Due to the relatively poor radiation resistance of hydrogenous materials in comparisons to metals, lead was chosen as an inner gamma shield for this study.

Payload

The TAD canister specification lists a maximum loaded mass of 50 MT, nominal length of 212 inches, and stainless steel wall thickness of 0.5 inches (OCRWM 2008). Certain dimensions of the canister can be varied to accommodate different types of CSNF. The TAD specification does not prescribe a specific shield plug, but rather states dose rate limits. The YM License Application states 1,000 mrem/hour maximum, and 800 mrem/hour average dose rates over the top of the canister (DOE 2008, p. 1.5.1-24). Since the closure lid of the cask is a relatively small portion of the cask surface area, no shield plug is assumed here. If a shield plug were included the needed shielding mass would be slightly less.

A two-layer carbon steel overpack with outer diameter 74.08 in, intermediate diameters of 72.08 in and 71.7 in, inner diameter of 67.7 in, and length of 218.5 in is assumed. These dimensions correspond to the inner vessel and outer corrosion barrier of the YM waste package (Figure A-2). The total overpack thickness used in the analysis was 7 cm. The gamma and neutron shielding discussed in the following paragraphs was in addition to the canister and overpack.



COMPONENT NAME	MATERIAL	QTY REQ
OUTER LID	SB-575 (UNS N06022)	1
SPREAD RING	SA-240 (UNS S31600)	1
PURGE PORT CAP	SA-240 (UNS S31600)	1
INNER LID	SA-240 (UNS S31600)	1
INNER VESSEL	SA-240 (UNS S31600)	1
INTERFACE RING	SA-240 (UNS S31600)	1
OUTER CORROSION BARRIER	SB-575 (UNS N06022)	1
UPPER SLEEVE	SB-575 (UNS N06022)	1
LOWER SLEEVE	SB-575 (UNS N06022)	1
SUPPORT RING	SB-575 (UNS N06022)	1

NOMINAL LENGTH	NOMINAL DIAMETER	WASTE PACKAGE ASSY	LOADED MASS
230.32 IN	77.28 IN	53.500 LBM	162.000 LBM
5850.1 MM	1962.9 MM	24.300 KG	73.500 KG

Figure A-2. Waste package for TAD canisters from YM License Application (page 1.5.2-61).

Source Term

The source term was assumed to be a 21-PWR CSNF canister similar to that described in the TAD canister specification. However, it was evaluated at a burnup of 45 GW-d/MTU, aged 20 years out-of-reactor, which roughly corresponds to the 10 kW per waste package emplacement thermal limit. A 44-BWR TAD canister was found to have a smaller source term, so it was not analyzed.

Performance of Cylindrical Shields

Dose rates were taken from points along the mid-plane of the cask and averaged. Uncertainty due to Monte Carlo sampling is gauged by taking the standard deviation of five points pulled from the output. Performance was evaluated by varying the thicknesses of the gamma and neutron shields, to evaluate shielding performance at the cask surface mid-way up the cask, in the area where dose is highest. A sample dose rate map, plots, and tabulated data are presented below.

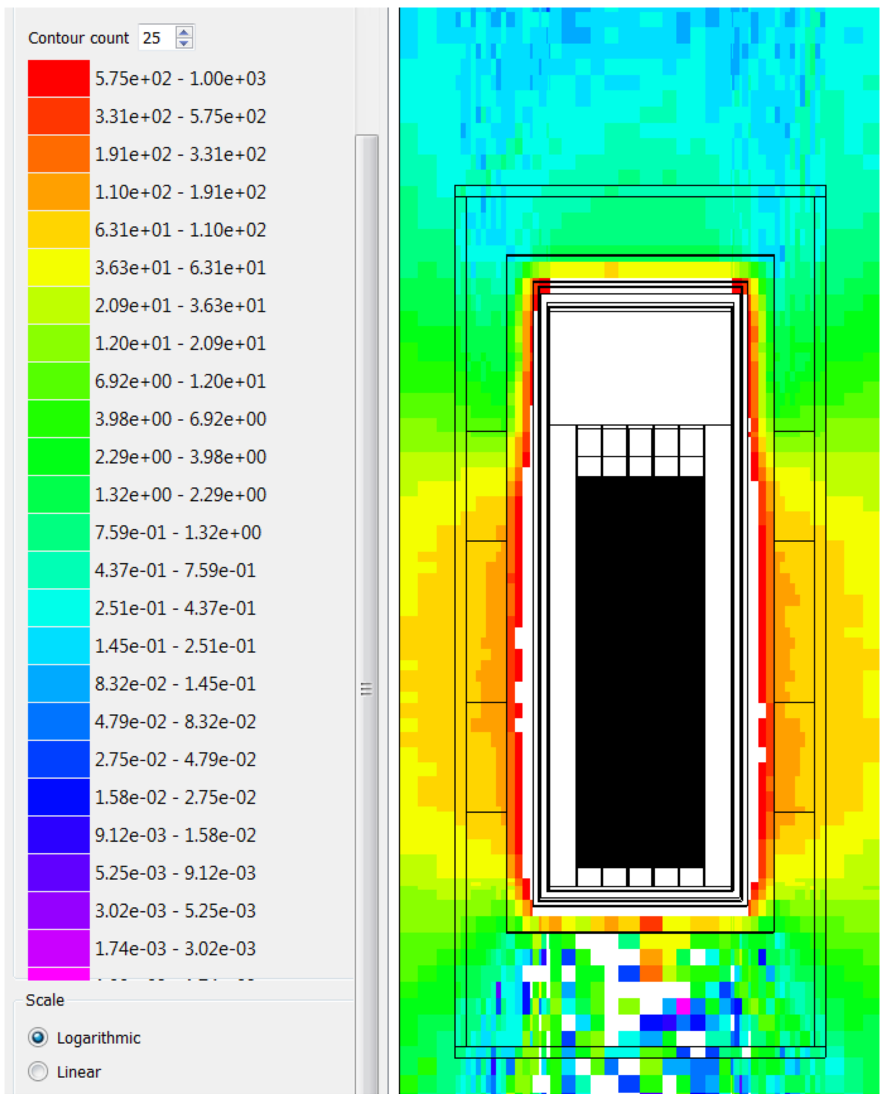


Figure A-3. Total dose rate map (mrem/hour) for cask with 19-MT gamma shield, and 21-MT neutron shield.

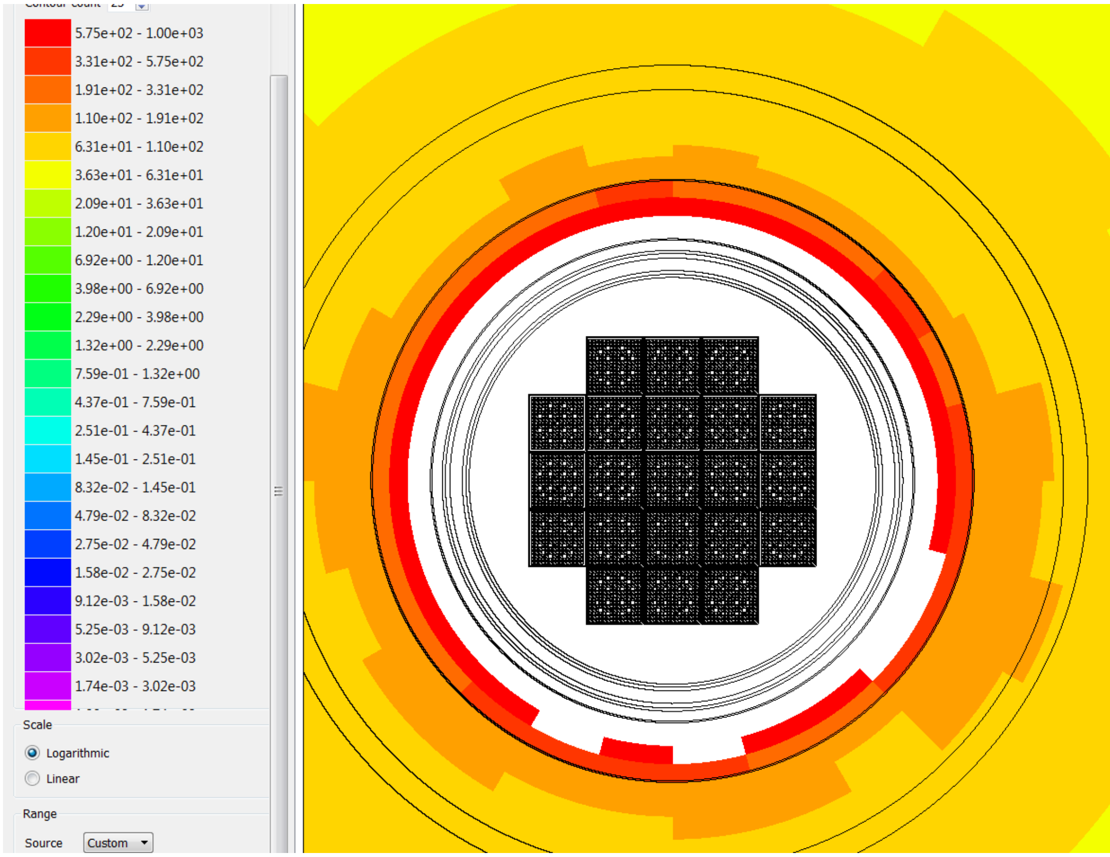


Figure A-4. Total dose rate map (mrem/hour) for cask with 19-MT gamma shield, and 21-MT neutron shield

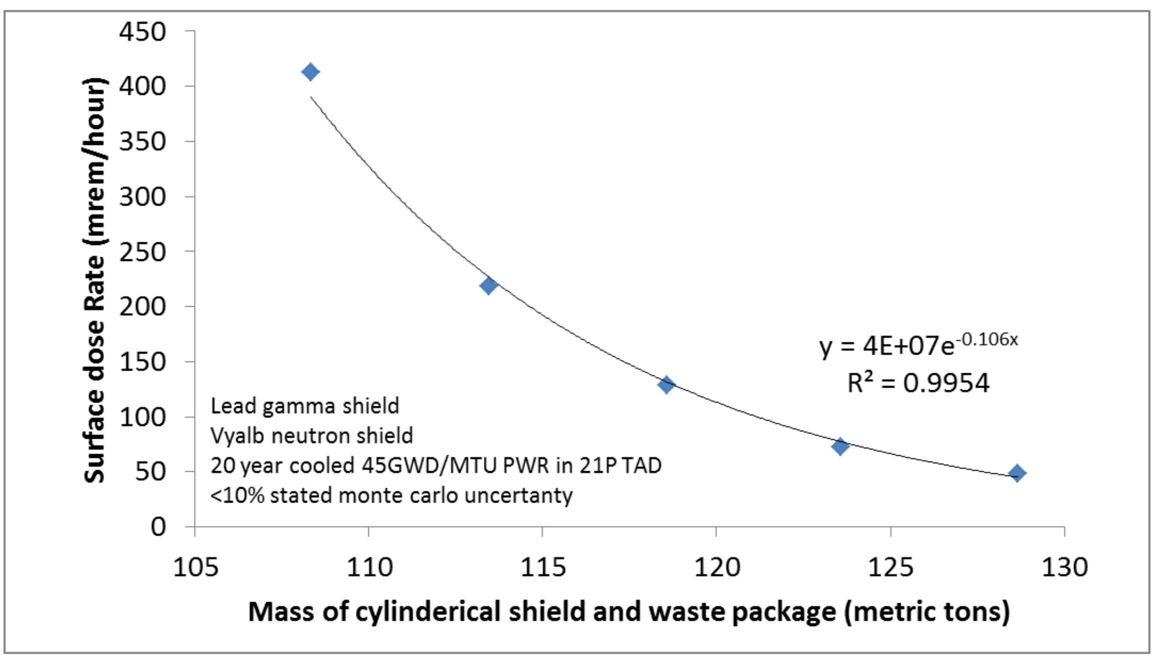


Figure A-5. Total dose rate at the surface *vs*. total mass (cylindrical shield).

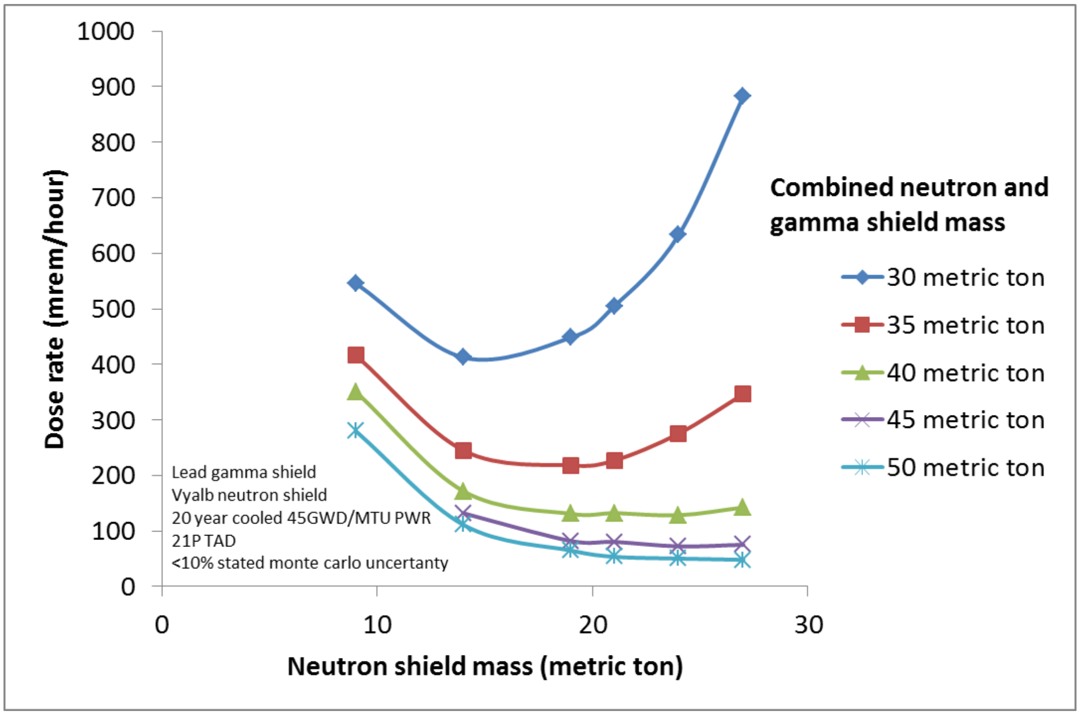


Figure A-6. Total dose rate at the surface *vs*. mass of neutron shield (cylindrical shield).

Table A-1. Calculated data and statistics for dose rate plots (cylindrical shield).

Total (mrem/hour)	Standard deviation of tally points	Gamma shield mass (MT)	Neutron shield mass (MT)	Neutron Dose Rate (mrem/hour)	Standard deviation of tally points	Gamma dose (mrem/hour)	Standard deviation of tally points	Package mass (MT)	Gamma shield thickness (cm)	Neutron shield thickness (cm)
48.23	3.82	23	27	15.62	1.78	32.62	3.38	128.65	4.99	29.07
50.45	4.37	26	24	23.40	3.14	27.04	3.03	128.58	5.62	26.13
53.80	3.72	29	21	28.64	1.83	25.17	3.24	128.51	6.24	23.14
64.91	3.20	31	19	39.68	1.16	25.23	2.98	128.47	6.65	21.11
72.57	5.40	21	24	20.35	2.14	52.22	4.96	123.58	4.57	26.44
75.31	9.85	18	27	13.57	1.43	61.74	9.74	123.65	3.94	29.40
80.64	8.50	24	21	30.02	2.85	50.62	8.01	123.51	5.20	23.41
82.10	4.79	26	19	40.72	1.52	41.38	4.54	123.46	5.62	21.36
111.17	2.70	36	14	87.90	2.41	23.27	1.23	128.35	7.66	15.89
128.57	8.39	16	24	21.80	1.81	106.77	8.19	118.58	3.51	26.75
130.97	8.19	21	19	46.45	2.03	84.53	7.93	118.46	4.57	21.62
132.03	3.55	31	14	93.87	2.58	38.16	2.44	123.35	6.65	16.09
132.23	6.95	19	21	35.00	2.82	97.23	6.35	118.51	4.15	23.69
142.54	11.31	13	27	15.43	1.54	127.11	11.20	118.65	2.87	29.75
171.81	7.61	26	14	100.65	6.84	71.16	3.33	118.34	5.62	16.29
218.18	11.14	16	19	48.10	6.12	170.08	9.31	113.46	3.51	21.89
227.05	17.21	14	21	37.94	4.38	189.12	16.64	113.50	3.08	23.99
245.05	10.03	21	14	104.92	6.05	140.14	8.00	113.34	4.57	16.50
275.55	26.70	11	24	29.95	3.13	245.60	26.51	113.57	2.43	27.07
280.13	13.55	41	9	252.98	9.80	27.16	9.36	128.23	8.66	10.46
346.68	25.44	8	27	18.54	2.67	328.14	25.30	113.64	1.78	30.10
350.89	20.99	31	9	293.08	15.24	57.81	14.44	118.22	6.65	10.73
412.41	28.95	16	14	110.66	5.30	301.75	28.46	108.33	3.51	16.71
416.56	47.02	26	9	293.50	13.13	123.06	45.14	113.21	5.62	10.87
448.47	16.62	11	19	48.65	6.29	399.82	15.38	108.45	2.43	22.17
504.60	27.69	9	21	38.32	3.83	466.28	27.43	108.50	2.00	24.29
546.05	57.28	21	9	287.41	40.00	258.64	41.00	108.21	4.57	11.02
633.25	47.80	6	24	26.29	4.84	606.96	47.56	108.57	1.34	27.41
882.83	48.51	3	27	17.51	5.24	865.33	48.23	108.64	0.67	30.47

Mailbox Shape

A mailbox shape shield was evaluated for use with a transporter, resulting in a ~10 MT mass penalty. Dose points were evaluated at the top, bottom, and sides of the shield along the mid-plane, and are therefore comparable to dose rates for cylindrical casks of the same thickness. The figure below has the same shield thicknesses as in the previous dose rate map from a cylindrical cask.

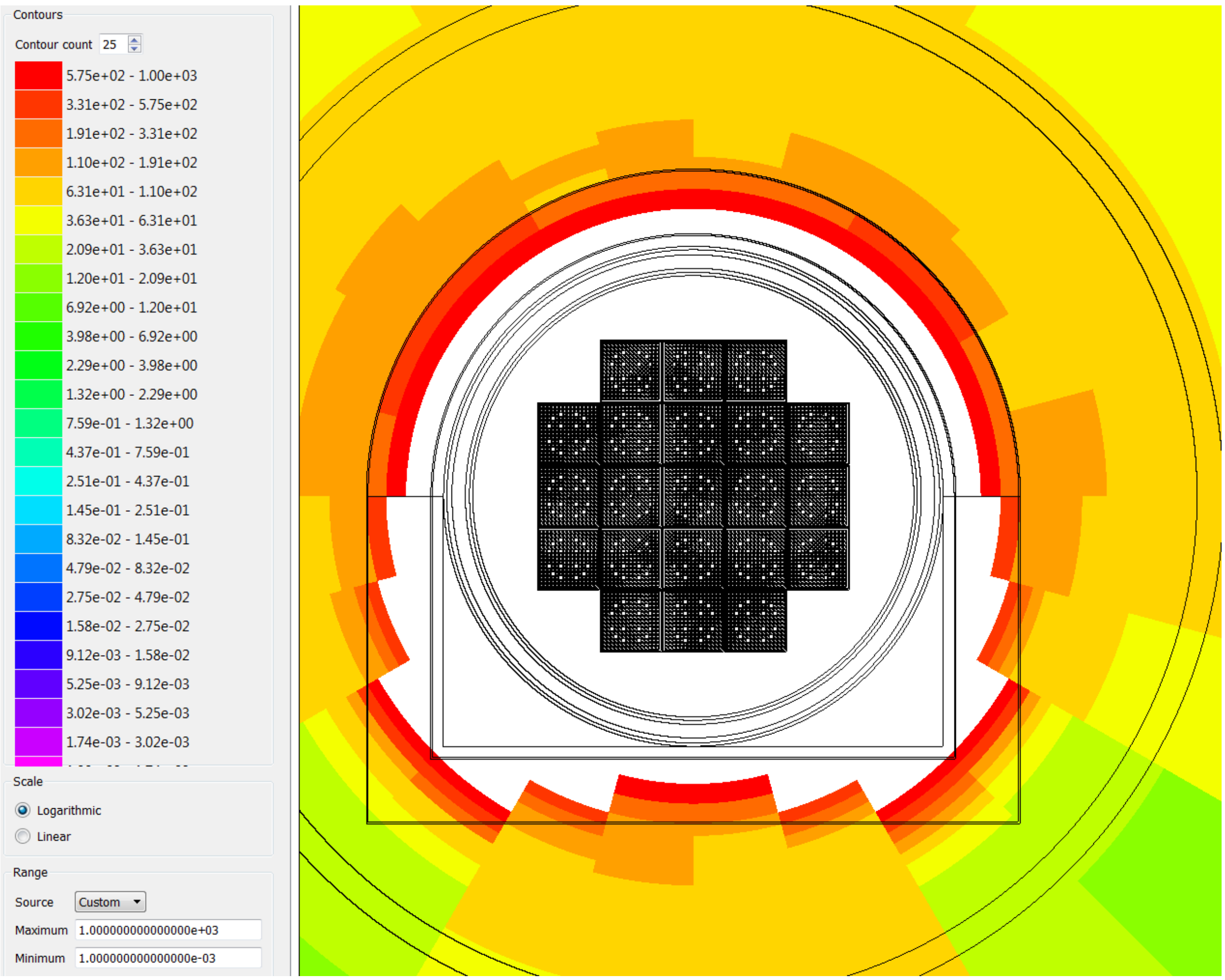


Figure A-7. Total dose rate map (mrem/hour) for cask with 19-MT gamma shield, and 21-MT neutron shield.

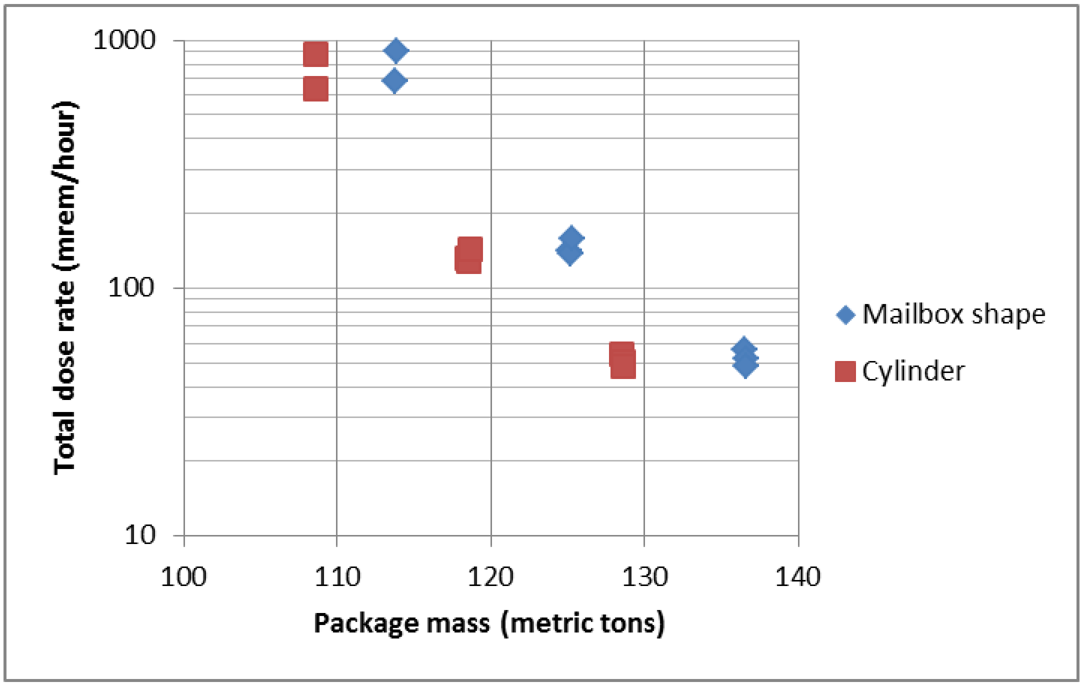


Figure A-8. Total dose rate *vs*. total mass, for mailbox and cylindrical shields of different thicknesses.

Table A-2. Calculated data and statistics for dose rate plots (mailbox shield).

Shape	Gamma shield thickness (cm)	Neutron shield thickness (cm)	Mass (metric tons)	Dose rate (mrem/hour)
Mailbox	1.338751	27.40869	113.8302	680.0636
Mailbox	6.237665	23.14023	136.4988	56.65433
Mailbox	5.618196	26.13408	136.577	51.97965
Mailbox	4.993132	29.06846	136.6541	48.37454
Mailbox	4.150781	23.69494	125.125	141.3815
Mailbox	3.51214	26.75136	125.204	138.2022
Mailbox	2.867445	29.74577	125.2817	158.7921
Mailbox	0.672752	30.46667	113.9086	907.725
Cylinder	1.338751	27.40869	108.571	633.2509
Cylinder	6.237665	23.14023	128.5148	53.80391
Cylinder	5.618196	26.13408	128.5837	50.44705
Cylinder	4.993132	29.06846	128.6514	48.23477
Cylinder	4.150781	23.69494	118.5082	132.2288
Cylinder	3.51214	26.75136	118.5777	128.5721
Cylinder	2.867445	29.74577	118.6461	142.5375
Cylinder	0.672752	30.46667	108.64	882.8309

Tunnel backfill

A supplemental case was also evaluated to examine the dose rate in any headspace remaining during tunnel backfill directly above a package. The analysis is described in Section 6.4. Separate dose rate plots for gamma and neutron fields are presented below.

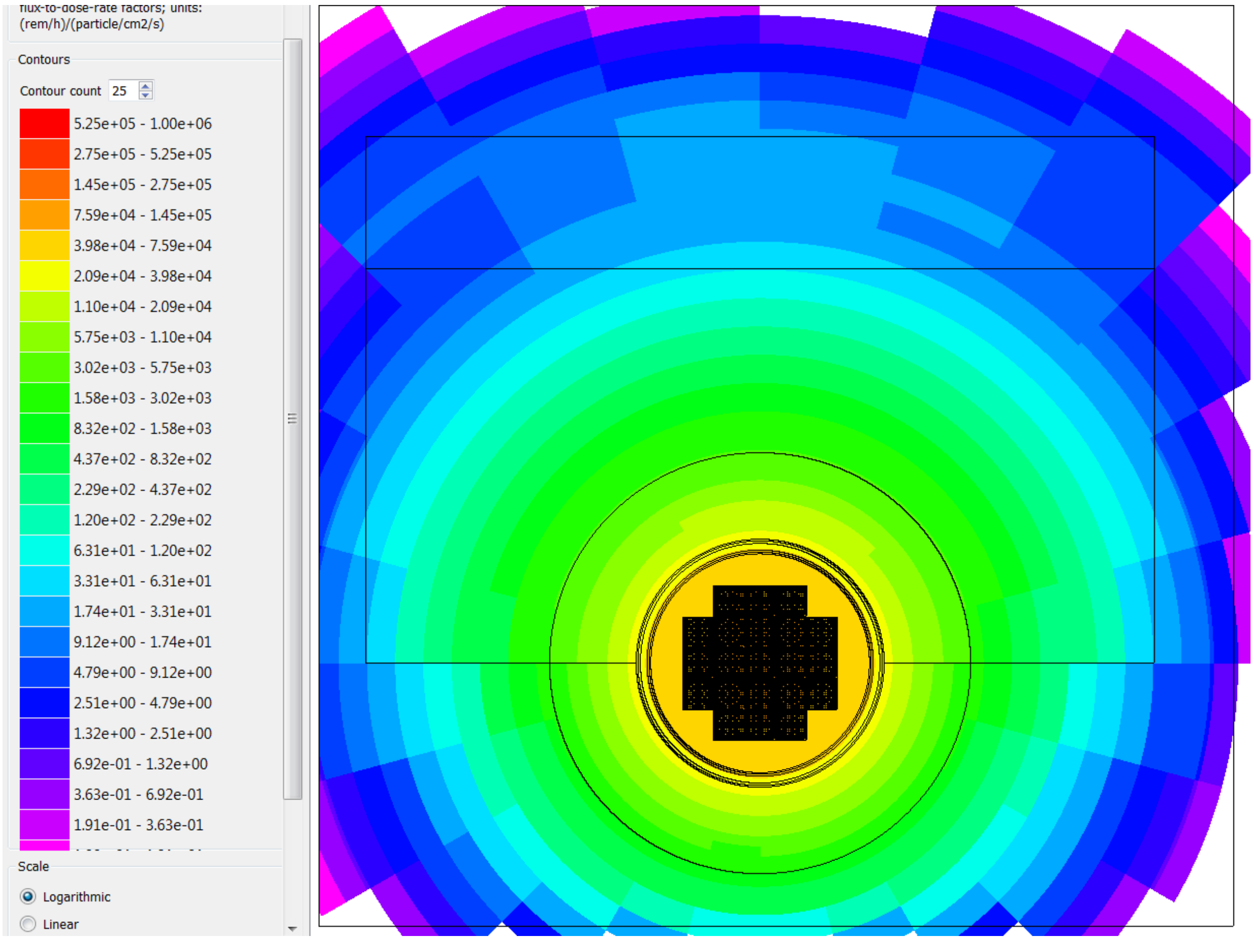


Figure A-9. Neutron dose rate (mrem/hour) over a backfilled package (tunnel and backfill cross-sections shown by black lines).

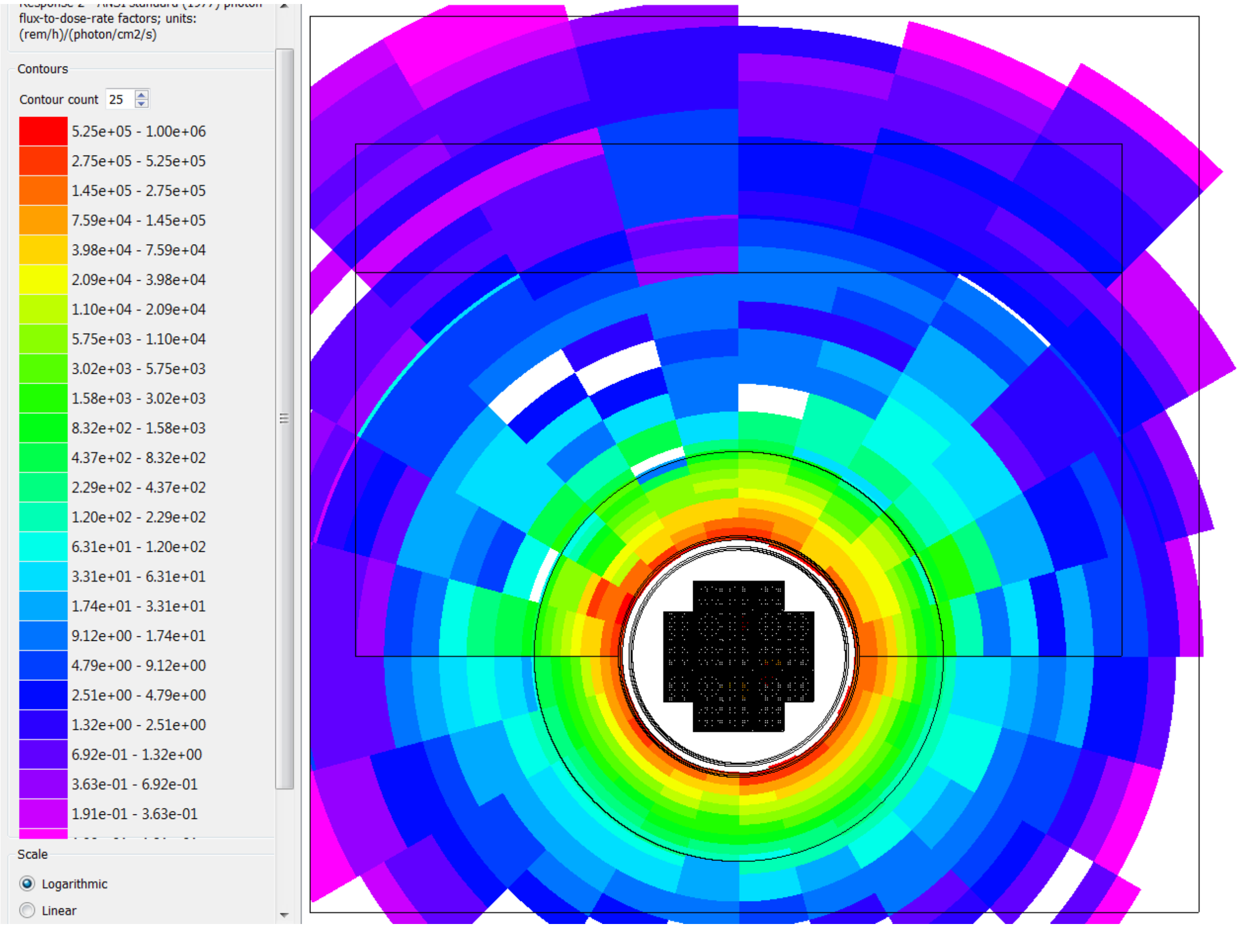


Figure A-10. Gamma dose rate (mrem/hour) over a backfilled package (higher relative uncertainty than neutron dose rate; tunnel and backfill cross-sections shown by black lines).

References for Appendix A:

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.

OCRWM (Office of Civilian Radioactive Waste Management) 2008. *Transportation, Aging and Disposal Canister System Performance Specification*, *Revision 1/ICN 1*. DOE/RW-0585. U.S. Department of Energy. March, 2008.

Appendix B. Waste Package Overpack Loading Analysis

Simulations were performed of a thick-walled steel waste package overpack surrounded by crushed salt backfill, in a disposal tunnel opening in a thick halite formation. Two types of simulations were done: 1) overpack explicitly modeled as bonded to surrounding salt, with and without emplacing each package in a semi-cylindrical floor cavity; and 2) overpack represented by a structural liner using features of the FLAC code, with and without a free-sliding interface at the contact.

Explicit Modeling of Overpack

FLAC V7.00 (Itasca 2011) was used for 2-D plane-strain simulation of a waste package overpack surrounded by crushed salt, within a tunnel opening. The constitutive laws for crushed and intact salt were *cwipp* and *pwipp*, respectively, which are built-in viscoplastic models and are based on an early single-mechanism creep law (Senseny 1985). Property values are the reference values for clean Permian salt. The choice of creep laws is not critical to this study because the timing of creep is of secondary importance. There is uncertainty in creep closure rate predictions which are affected by salt composition, inter-beds, LS-LSR creep (Section 4.6), and so on. The simulations were done for ambient temperature; accelerated creep would occur at elevated temperature. They were run for 100 years of simulation time which gives some allowance for uncertain creep rate.

The tunnel opening dimensions were 4 m high by 6 m wide (half-symmetry model), discretized at 16 cm (host rock and backfill), with nested resolution of 2 cm (overpack and surrounding salt) (Figure B-1). The overpack inner diameter was 1.71 m, corresponding to the outer diameter of the 21-PWR size TAD canister. The overpack outer diameter depended on the wall thickness. The overpack was assigned the elastic properties of steel (without yield properties).

Simulations were repeated with 7-cm, 13-cm and 20-cm overpack wall thicknesses (Figure B-2). Yielding of the overpack was not permitted, rather, the calculated stresses were used to understand the required thickness and yield strength to prevent yielding and associated damage (e.g., damage to spent fuel assemblies from large deformation, followed by overpack rupture). The plots show the maximum principal stress (σ_1) highlighted in areas that could be subject to yielding.

To evaluate the potential for mechanical reinforcement (confinement) of the overpack, the simulations for 7-cm and 13-cm wall thickness were repeated with the waste package embedded halfway into a semi-cylindrical cavity in the floor (Figure B-3). This arrangement was originally proposed to promote heat transfer (Hardin et al. 2013) but it provides mechanical support to the overpack as well.

The results show that with the waste package emplaced directly on the floor, yielding of a steel overpack would likely occur (σ_1 > 800 MPa) even with wall thickness of 20 cm (Figure B-2). However, embedding the overpack into a semi-cylindrical cavity decreased maximum compressive stress for the 13-cm overpack from more than 1,200 MPa to less than 400 MPa (Figure B-3). For the 7-cm overpack maximum compression was reduced to less than 700 MPa (in all parts of the overpack cross-section except the bottom). These scoping results indicate that with confinement and a heavy wall thickness on the order of 13 cm, yield strength of 400 MPa (~60,000 psi) could prevent yielding. To further develop overpack design to resist deformation during the operational period, analyses of this type should be repeated with alternative salt constitutive models, yield properties for steel, heat generating waste with temperature-dependent mechanical and thermal properties, and grid refinement.

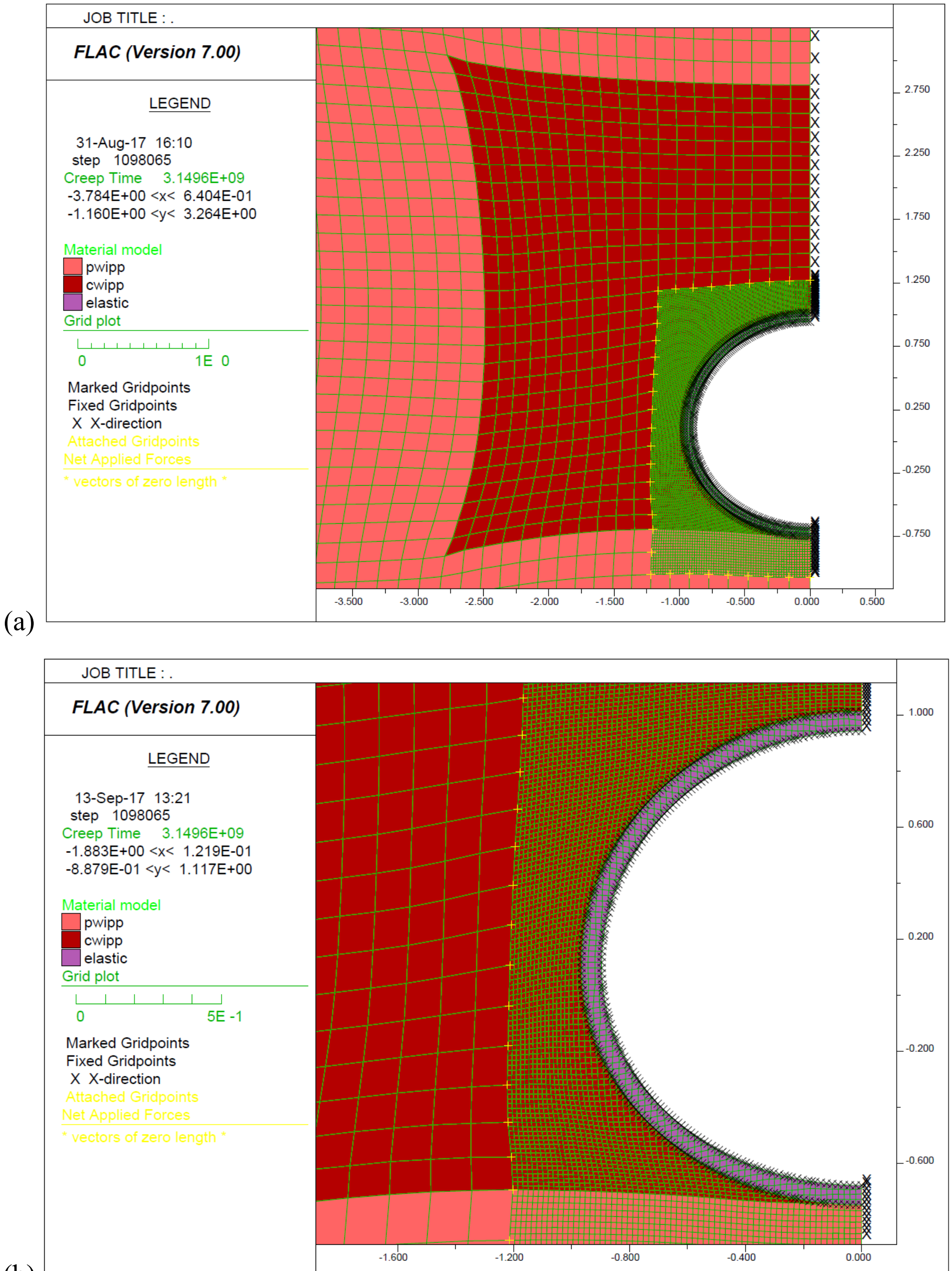


Figure B-1. (upper) Deformed model grid (10^3 yr simulation time) showing tunnel and overpack cross-sections, with nested gridding around the overpack, and 7-cm overpack thickness. (lower) Inset showing 2-cm grid resolution for the overpack and vicinity.

7-cm overpack

13-cm overpack

20-cm overpack

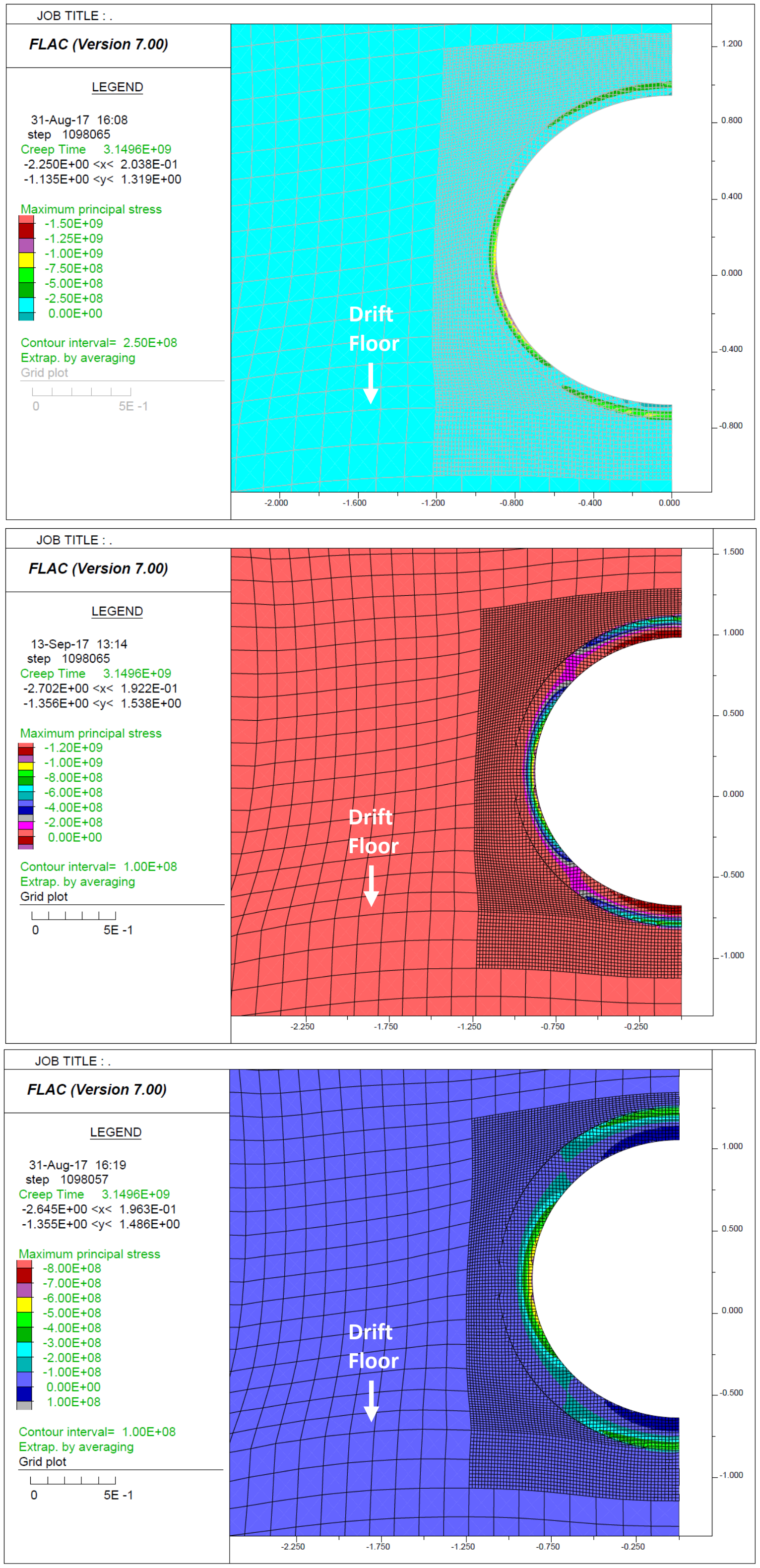
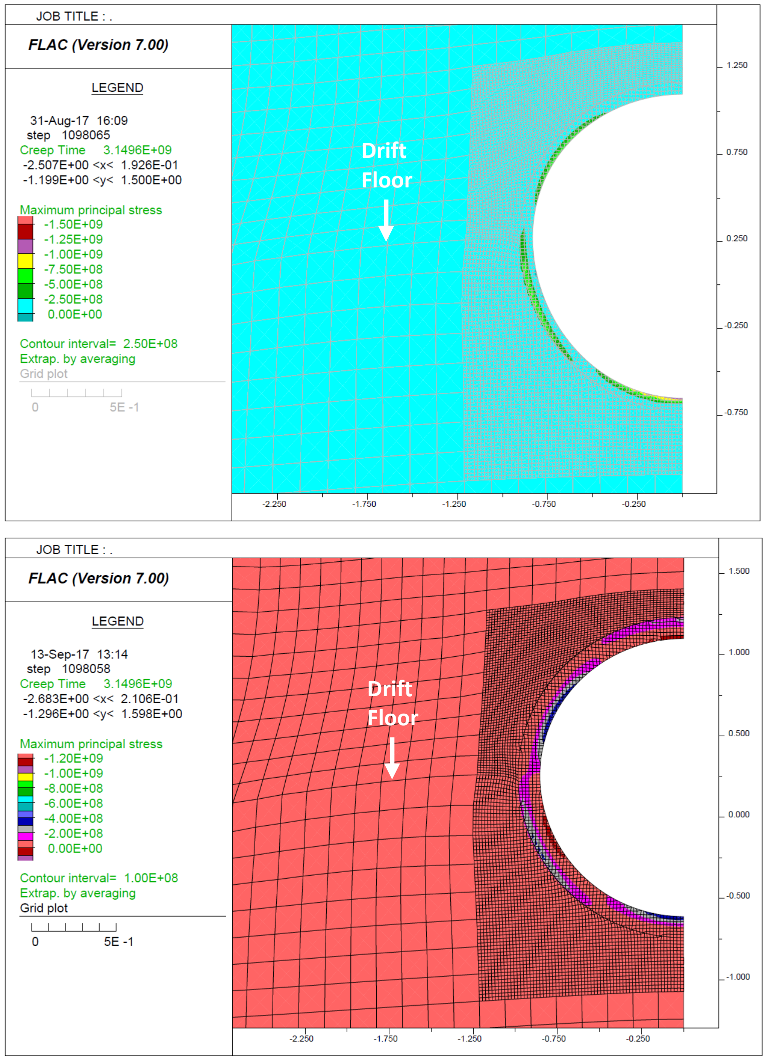


Figure B-2. Maximum principal stress for the liner thicknesses indicated, and packages emplaced directly on the tunnel floor.



7-cm liner

13-cm liner

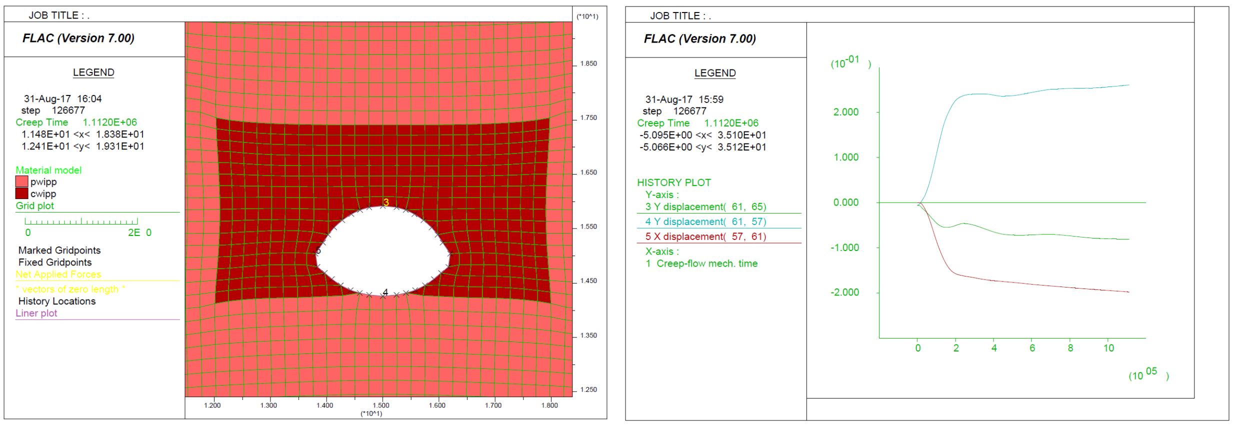
Figure B-3. Maximum principal stress for the liner thicknesses indicated, and packages emplaced into semi-cylindrical cavities in the tunnel floor.

Liner Interface Study

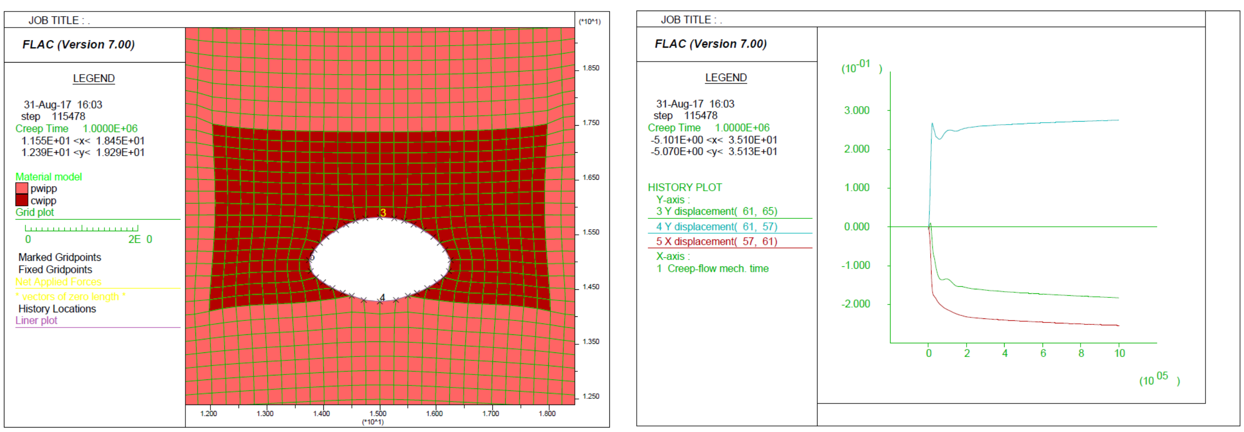
A different set of FLAC calculations was done to examine whether a slip condition at the overpack-salt interface could affect stress buildup and deformation. A somewhat coarser grid was used with the same dimensions and *pwipp-cwipp* constitutive models. Liner elements (normally used to represent a continuous tubular liner as ground support; Itasca 2011) were prescribed along the surface of a model boundary corresponding to the crushed salt-overpack contact. The liner elements were assigned the properties of steel, with dimensions equivalent to a 7-cm thick overpack. Where liner elements connect at grid points, yielding is allowed in response to bending moments, hence the salt-overpack contact is deformed (Figure B-4).

Two sets of liner properties were used: a glued set whereby the backfill and overpack are bonded, and a slip condition whereby the normal and shear stiffnesses of the interface were assigned values of 2×10^6 MPa/m (stiff), friction angle of 8° (minimal effect from normal stress), and a shear bond strength of 0.1 MPa (compliant).

The results are presented as plots of displacements of the top, bottom, and one side of the overpack over a simulation of 100 years (Figure B-4 plots the first ~10^6 seconds). Displacements are similar for the glued and slip conditions, indicating that overpack deformation does not strongly depend on the frictional properties at the interface with crushed salt.



(a) Glued interface showing deformed model grid (left), and overpack displacements (right).



(b) Slip-interface showing deformed model grid (left), and overpack displacements (right).

Figure B-4. Comparison of overpack deformations (displacements 3, 4 and 5 are Y-top, Y-bottom and X-side, respectively) for the glued (upper) and interface slip (bottom) cases.

References for Appendix B

Hardin, E., D. Clayton, R.L. Howard, J.M. Scaglione, E. Pierce, K. Banerjee, M.D. Voegele, H. Greenberg, J. Wen, T. Buscheck, J.T. Carter, T. Severynse and W.M Nutt 2013. *Preliminary*

Report on Dual-Purpose Canister Disposal Alternatives (FY13). FCRD-UFD-2013-000171 Rev. 0. Office of Used Fuel Disposition, U.S. Department of Energy.

Itasca (Itasca Consulting Group) 2011. *FLAC Version 7.00 User's Guide*. Minneapolis, MN.

Senseny, P.E. 1985. "Determination of a Constitutive Law for Salt at Elevated Temperature and Pressure." *Measurement of Rock Properties at Elevated Pressures and Temperatures* (H.J. Pincus and E.R. Hoskins, eds.). American Society for Testing and Materials, Special Technical Publication 869. Philadelphia, PA.

Appendix C. Logistical Analysis

Plots showing detailed results for types of CSNF emplaced in a repository each year, for each of the analyzed scenarios from Section 8, are presented in this Figures D-1 through D-8. For each scenario the repository would begin emplacing waste in 2048. Repository emplacement thermal limits are nominally 10 kW per package at the time of emplacement, except for Scenarios 3 and 5 (8 kW), and Scenarios 4 and 6 (12 kW).

Instead of DPCs, waste would be packaged in disposable 21-PWR/44-BWR TAD-type canisters starting in 2025, for each scenario except for Scenarios 7 and 8, for which the transition would be delayed to 2036 (after a repository would presumably be sited and the particulars of the repository environment known).

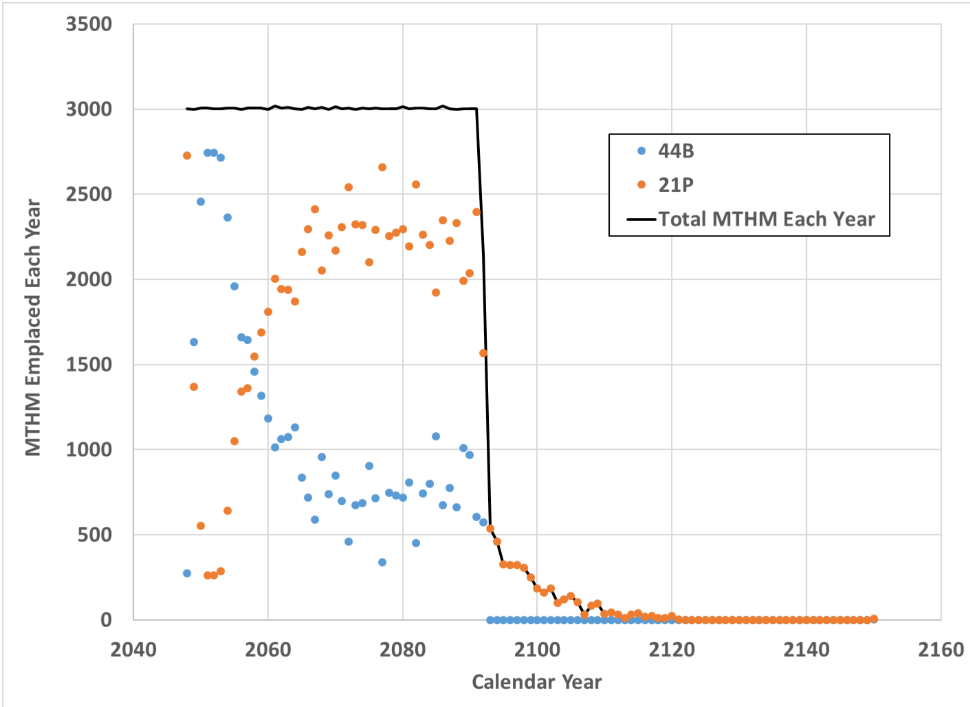


Figure C-1. SNF (in MTHM) emplaced per year for Scenario 1 reported by 44-BWR and 21-PWR packages (10 kW repository heat limit; loading begins in 2025).

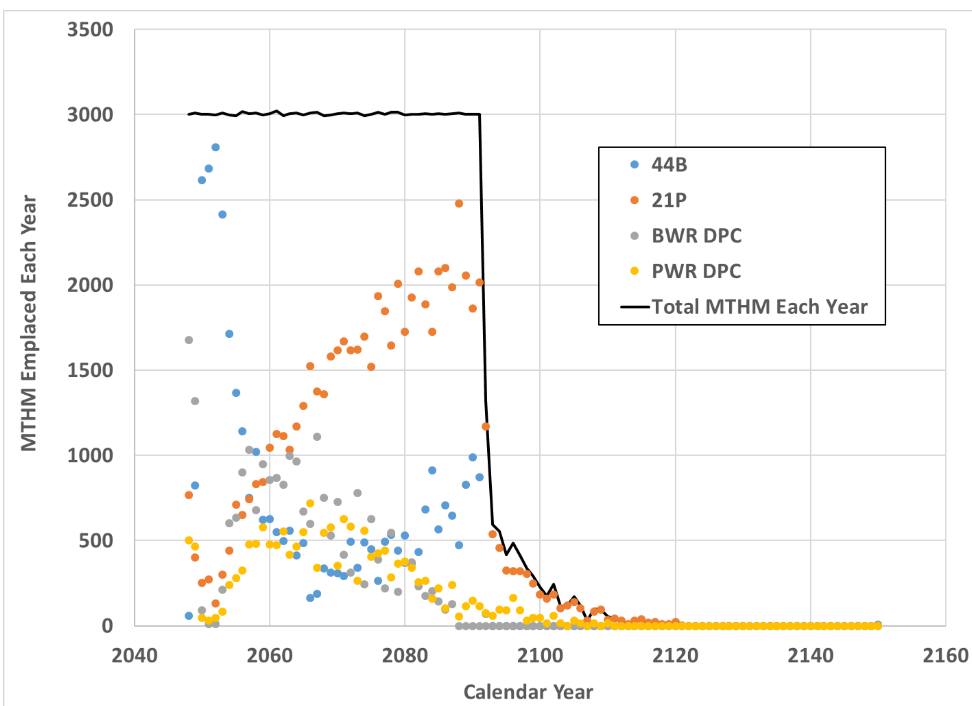


Figure C-2. SNF (in MTHM) emplaced per year for Scenario 2 reported by 44-BWR, 21-PWR, BWR DPC, and PWR DPC packages (10 kW repository heat limit; loading begins in 2025).

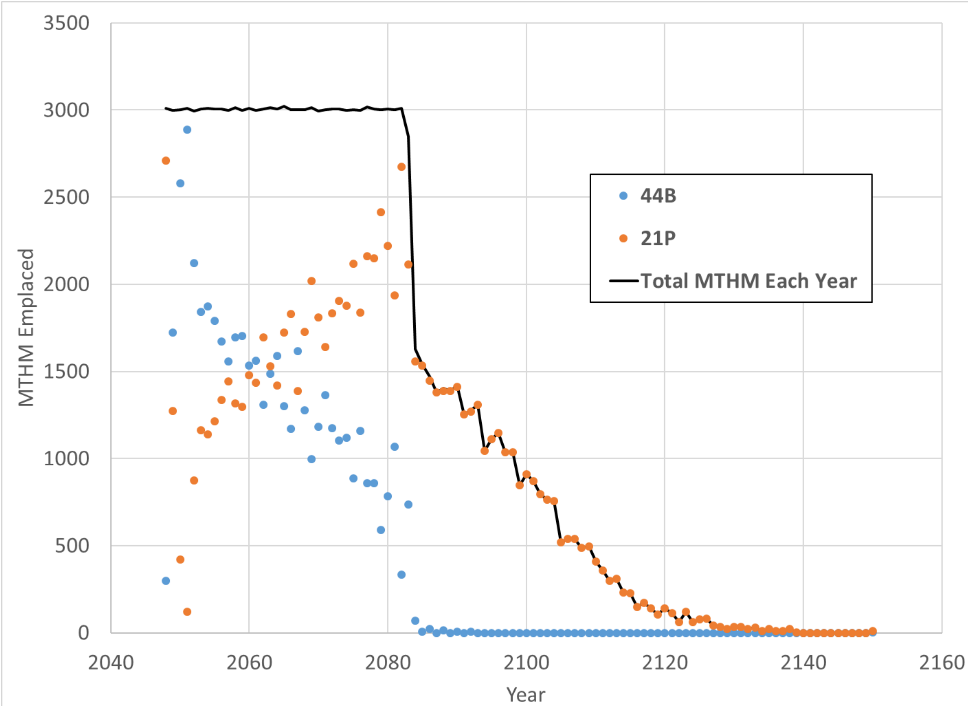


Figure C-3. SNF (in MTHM) emplaced per year for Scenario 3 reported by 44-BWR and 21-PWR packages (8 kW repository heat limit; loading begins in 2025).

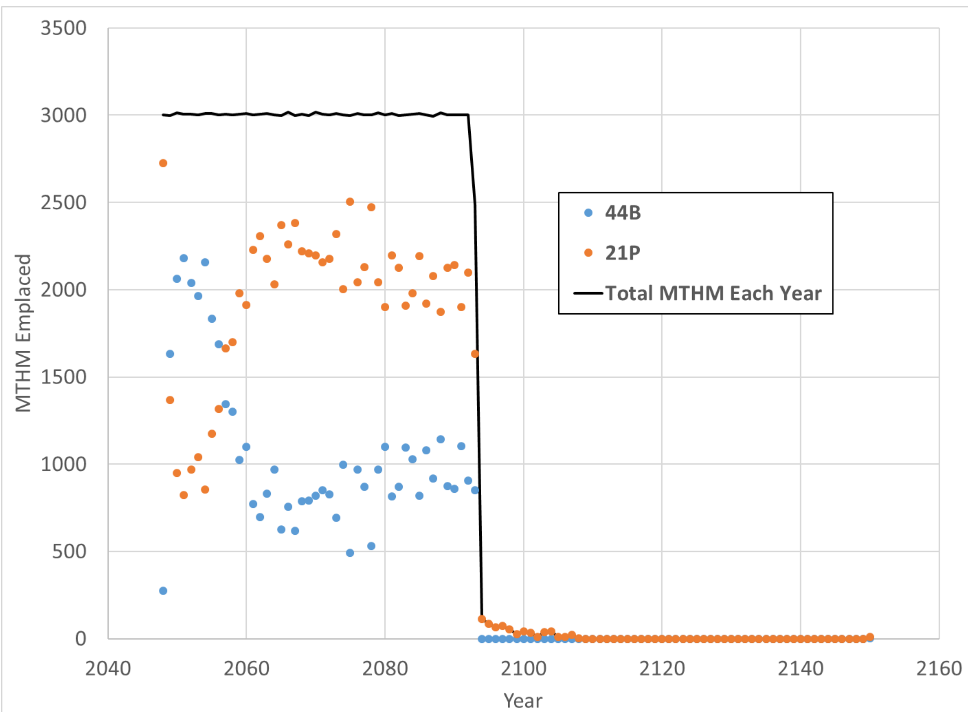


Figure C-4. SNF (in MTHM) emplaced per year for Scenario 4 reported by 44-BWR and 21-PWR packages (12 kW repository heat limit; loading begins in 2025).

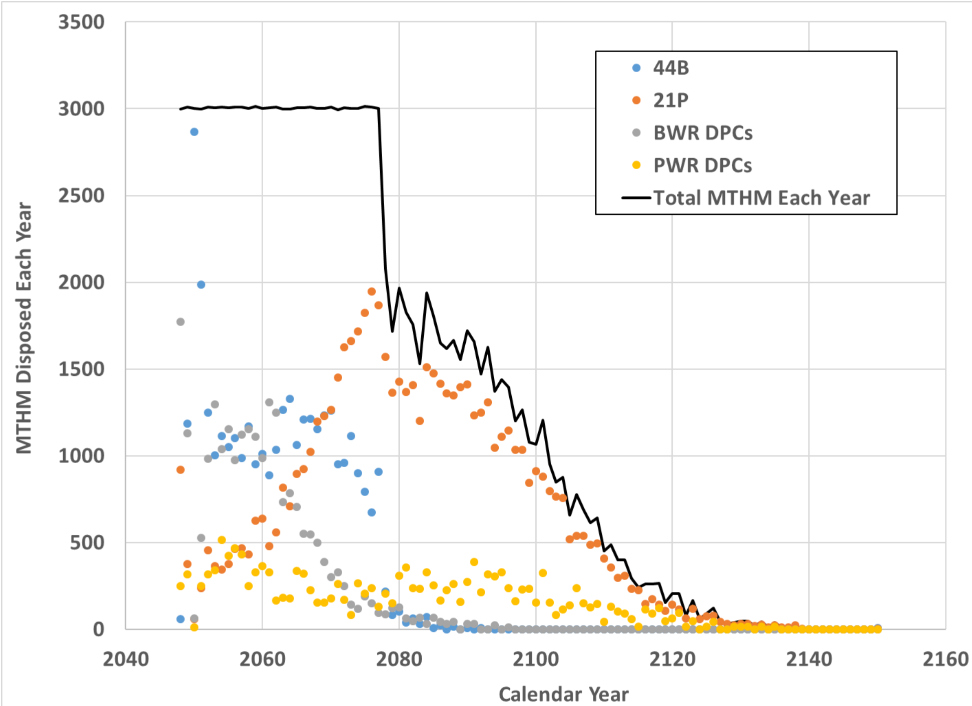


Figure C-5. SNF (in MTHM) emplaced per year for Scenario 5 reported by 44-BWR, 21-PWR, BWR DPC, and PWR DPC packages (8 kW repository heat limit; loading begins in 2025).

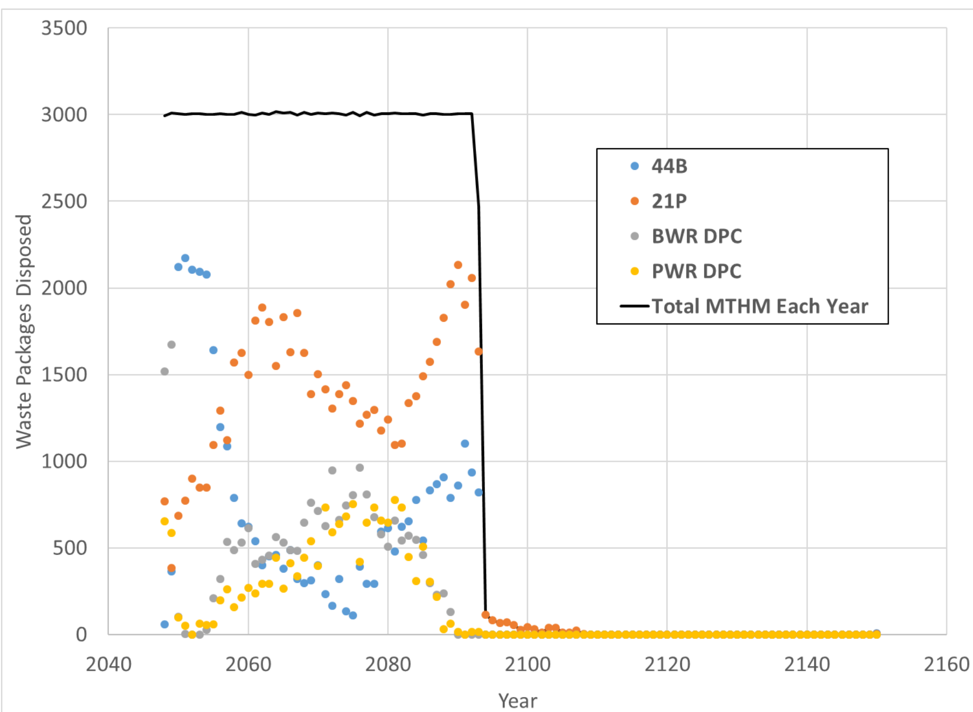


Figure C-6. SNF (in MTHM) emplaced per year for Scenario 6 reported by 44-BWR, 21-PWR, BWR DPC, and PWR DPC packages (12 kW repository heat limit; loading begins in 2025).

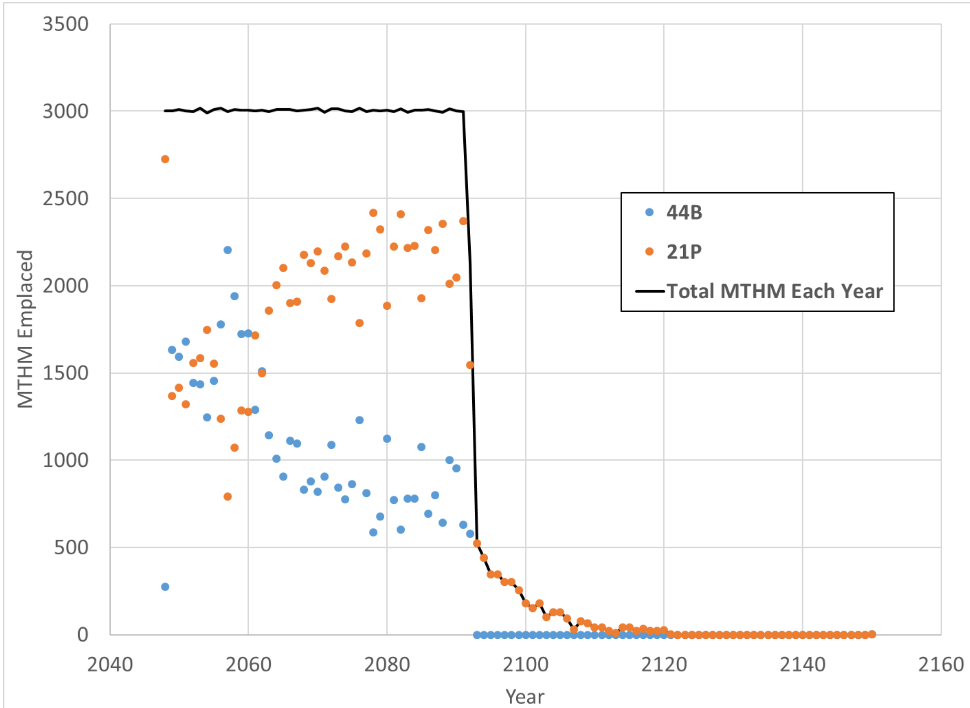


Figure C-7. SNF (in MTHM) emplaced per year for Scenario 7 reported by 44-BWR and 21-PWR packages (10 kW repository heat limit; loading begins in 2036).

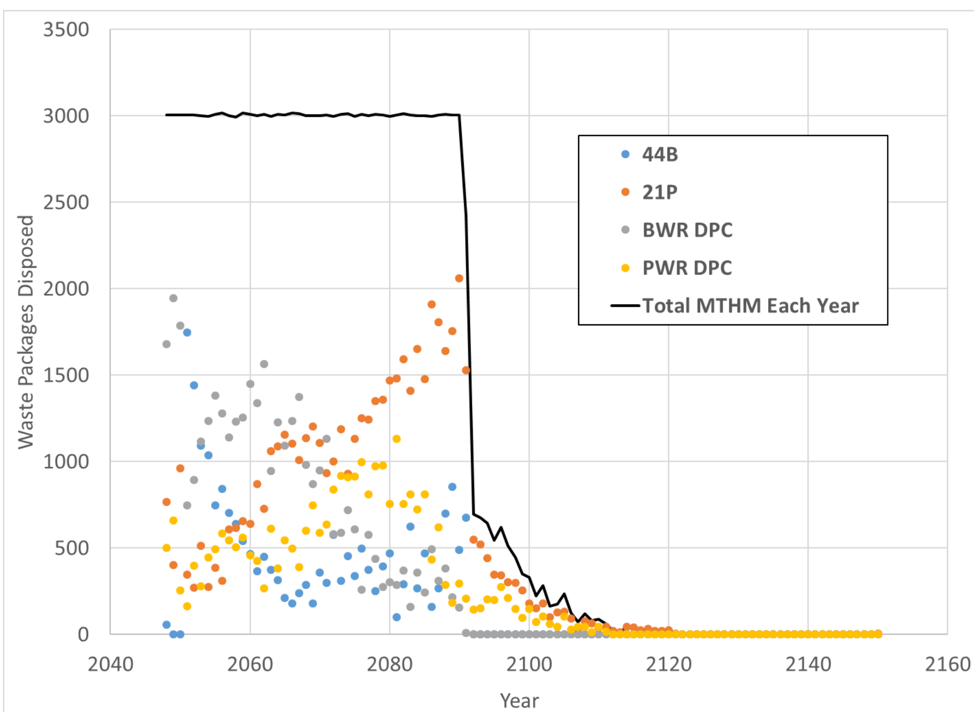


Figure C-8. SNF (in MTHM) emplaced per year for Scenario 8 reported by 44-BWR, 21-PWR, BWR DPC, and PWR DPC packages (10 kW repository heat limit; loading begins in 2036).

Appendix D. Low-Stress Low Strain-Rate Creep Simulations

The following description is summarized from a previously published feasibility study (Hardin et al. 2014).

Weinberg (1993) wrote:

> *...several laboratory studies have indicated that dry salt deforms by dislocation creep and behaves as a power-law fluid at high strain rates" and that "traces of brine in confined salt deforming at slow strain rates cause a change in the deformation mechanism from dislocation creep to solution-transfer creep in relatively fine grained salt (Urai et al. 1986). The salt then becomes weaker and behaves like a Newtonian fluid with a viscosity that is directly proportional to the cube of grain size.*

Early reference creep laws (e.g., Carter and Hansen 1983) do not predict softening at low strain rates possibly because they are conditioned to: 1) higher strain-rate data acquired to represent rock mass response to excavation; 2) test samples that lost moisture during handling or testing; and 3) data acquired by the relaxation test method which increases dislocation density during pre-relaxation loading (van Keken et al. 1993).

Similar conclusions were reported by Urai et al. (1986), who identified a transition mechanism with a small value of power-law stress exponent *n* at strain rates less than 10^{-7} sec^{-1}. The effects of moisture on natural deformation processes were projected to be important for strain rates less than ~10^{-10} sec^{-1}.

Spiers et al. (1986) concluded that rheology of other layers surrounding bedded salt host rock may also be important to control long-term stability, and that stresses around a repository may equilibrate faster than previously thought due to LS-LSR deformation, and that brine migration toward hot canisters may cause softening and increased rates of sinking (which they bounded at 1 m per 10^4 yr). Spiers et al. (1990) introduced a power law for pressure solution

$$\dot{\epsilon} = 6.95 \times \mathrm{V_m} \times 10^{-15}\,\frac{\exp(-24530/RT)}{T}\,\frac{\sigma_1}{d^3} \qquad \text{(D-1)}$$

where

V_m = molar volume (2.693×10^{-5} m^3/mole)
Q = Activation energy (24,530 J/mole)
R = Gas constant (8.314 J/mole-K)
T = Absolute temperature (K)
σ_1 = Differential stress (Pa)
d = Grain size (m)

This power law also incorporates Newtonian behavior (*n*=1), with strong grain size dependence. Spiers et al. (1990) concluded in part that the relative importance of pressure solution diminishes during natural diapirism due to: 1) water loss during progressive shearing; 2) increased deformation rates as diapiric structures evolve; and 3) some other mechanism (possible dislocation related). For the low strain-rate test conditions of Bérest et al. (2005, 2012) and grain size of 1 cm, expression (D-1) yields strain rates on the order of 10^{-15} sec^{-1} at 25°C. For the same conditions it yields viscosity on the order of 10^{19} Pa-sec at 300 K. These results are too slow and viscous to represent the Bérest data discussed below.

The Newtonian form contrasts with power laws conditioned on laboratory data and multi-year observations of room and borehole closure, where strain rates are much greater. For example, the compilation by Weinberg (1993) includes a power law for Permian Salado formation salt:

$$\dot{\epsilon} = 2.42 \cdot 10^{-44}\, \sigma^{4.9} \tag{D-2}$$

using stress units of Pa, and temperature of 300 K.

Simulations Using Bilinear Power-Law Constitutive Model

To explore the modeling implications, expression (D-2) is spliced with a Newtonian power law (n=1) that is constrained by data from Bérest et al. (2005, 2012). The intercept of the segments with different slopes on a log-log plot (Figure 4-3) is constrained by the Bérest et al. data so that the transition stress is σ= 8 MPa. This transition value is model dependent, and does not signify a sharp transition in salt response. The approach is similar to that of Bérest et al. (2012) who analyzed a solution-mined storage cavity using a transition stress of 4 MPa.

The spliced or bilinear constitutive model is used in the FLAC code (Itasca 2011) with a coarse, exploratory grid representing a cross-section through a repository tunnel (Figure D-1). The model domain is a vertical cross-section through a single repository panel, with 30-m spacing between parallel tunnels. The tunnel opening is filled at the time of excavation and emplacement, with crushed salt at initial porosity of 36% and creep-consolidation behavior represented by the *cwipp* constitutive model in FLAC V.7. A 2-m diameter waste package rests on the floor. The package is assigned the density of steel (8×10^3 kg/m^3).

The simulation is run for 1,000 years, and vertical displacements plotted for the tunnel crown and the invert below the waste package (Figure D-2). The plots show the early displacements when the opening is excavated (down at the crown, up at the floor), followed by creep response as the opening closes and the backfill consolidates. The figure compares the bilinear creep law with the original Norton-Hoff law for Salado salt (Weinberg 1993). The Norton-Hoff law produces gradual creep response, then stability as the backfill approaches intact density and the stress state returns to lithostatic. By contrast, the bilinear model exhibits steady-state downward movement of the waste package and the surrounding salt, with velocity approximately 1 meter per 10^4 years.

The maximum shear strain rate (Figure D-3) further represents the differences between the Norton-Hoff law and the bilinear model. The strain rate for the single Norton-Hoff power law shows shear strain localized to the near field, and the deformation field is slowly evolving even at 10^3 years. Much of the strain at this point in the evolution is taking place in the backfill. With the bilinear creep law the maximum shear strain rate occurs in the host rock, and the rate is 2 orders of magnitude greater than the Norton-Hoff power law. Moreover, this deformation field develops rapidly, less than 100 years after emplacement.

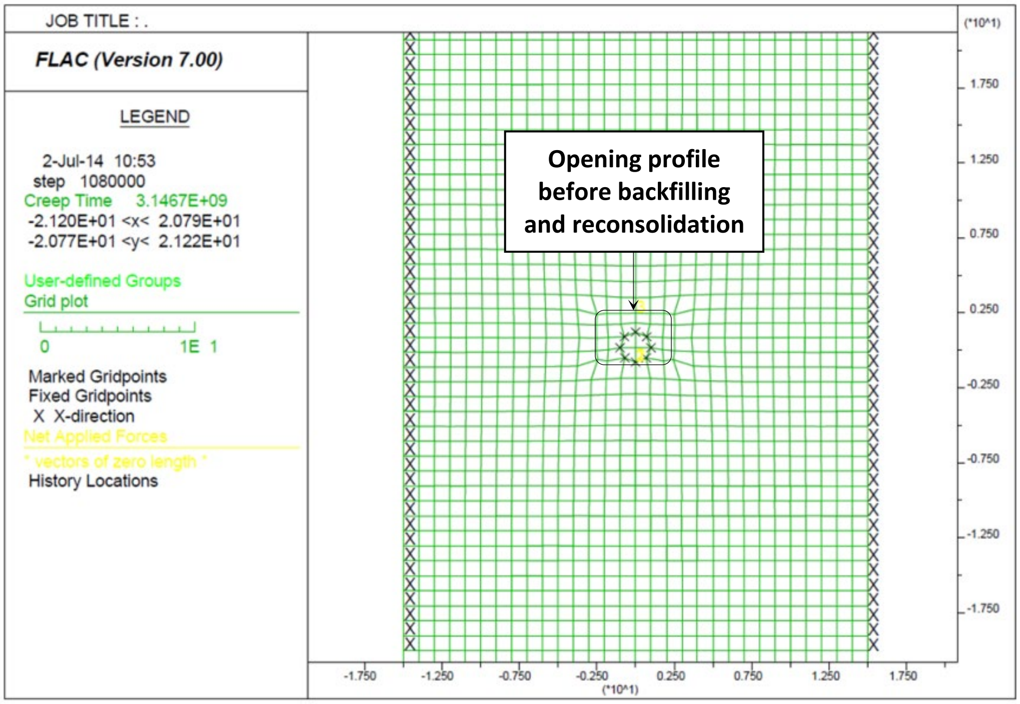
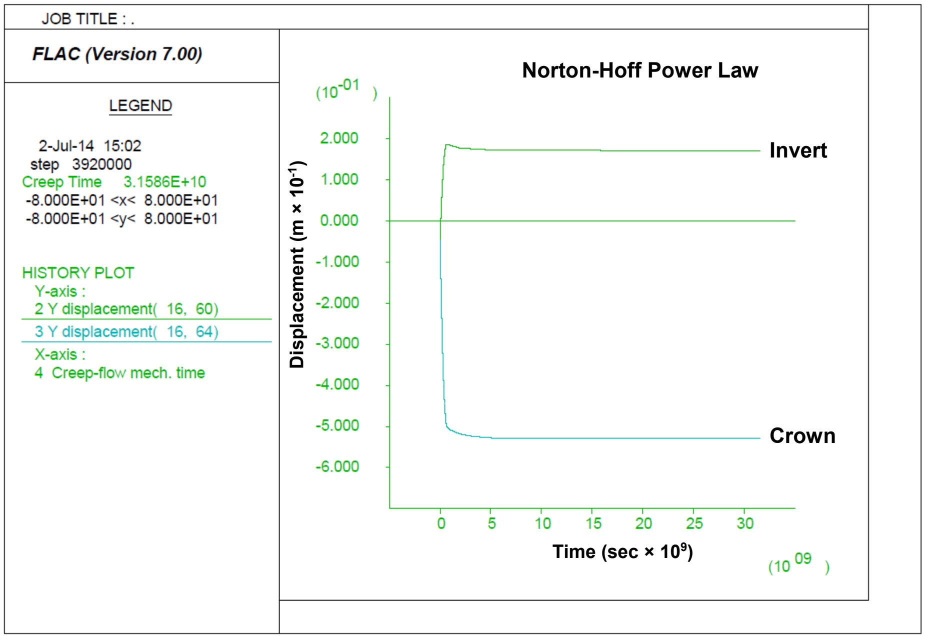
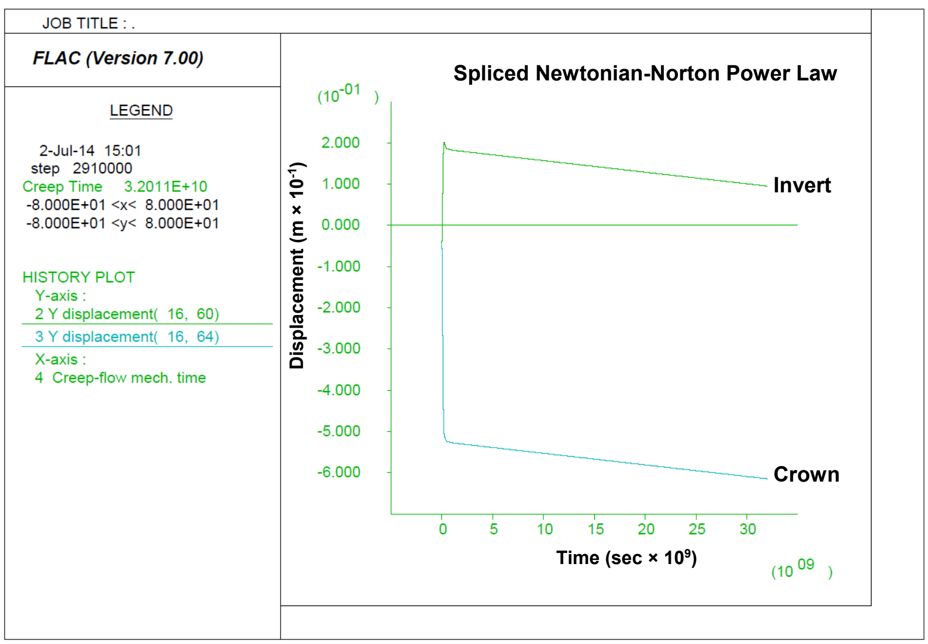


Figure D-1. FLAC model grid used in comparative analysis.



(a)



(b)

Note: In each plot the crown moves downward and the floor upward in response to initial excavation of the tunnel opening, followed by long-term sinking of the package (and its surroundings) in the spliced case. Duration of simulations is approximately 1,000 years.

Figure D-2. Time dependent crown and invert displacements for a rectangular waste emplacement opening in salt, for 100 years, using: (a) a Norton-Hoff power law, and (b) a Norton-Hoff power law spliced to a Newtonian LS-LSR power law at stresses less than 8.1 MPa.

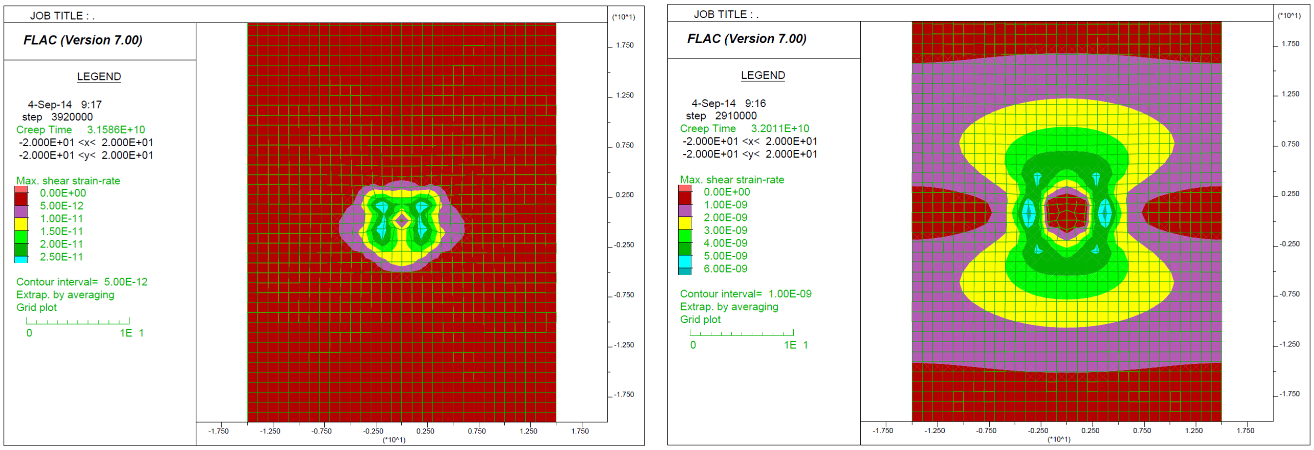


Figure D-3. Maximum shear strain rate at 10^3 years, for a Norton-Hoff power law (left), and the same power law spliced to a Newtonian law for stress differences less than 8.1 MPa (right).

References for Appendix D

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