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MR41C-2712: Numerical Modeling of Thermal-Hydrology in the Near Field of a Generic High-Level Waste Repository

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1) Overview

Disposal in a deep geologic repository is one of the preferred option for long term isolation of high-level nuclear waste. Coupled thermal-hydrologic processes induced by decay heat from the radioactive waste may impact fluid flow and the associated migration of radionuclides. This study looked at the effects of those processes in simulations of thermal-hydrology for the emplacement of U. S. Department of Energy managed high-level waste and spent nuclear fuel. Most of the high-level waste sources have lower thermal output which would reduce the impact of thermal propagation. In order to quantify the thermal limits this study concentrated on the higher thermal output sources and on spent nuclear fuel.

The study assumed a generic nuclear waste repository at 500 m depth. For the modeling a representative domain was selected representing a portion of the repository layout in order to conduct a detailed thermal analysis. A highly refined unstructured mesh was utilized with refinements near heat sources and at intersections of different materials. Simulations looked at different values for properties of components of the engineered barrier system (i.e. buffer, disturbed rock zone and the host rock). The simulations also looked at the effects of different durations of surface aging of the waste to reduce thermal perturbations. The PFLOTRAN code (Hammond et al., 2014) was used for the simulations. Modeling results for the different options are reported and include temperature and fluid flow profiles in the near field at different simulation times.

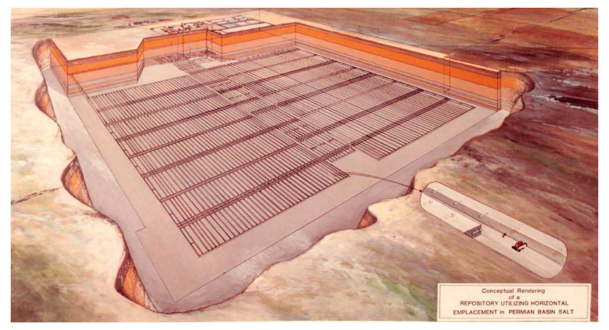

Figure 1. *Nuclear Waste Repository design concept – Bedded Salt Host (DOE 1986).*

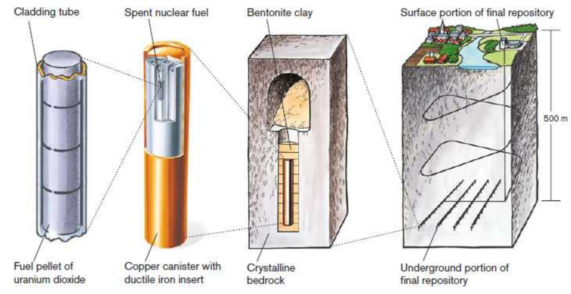

Figure 2. *Nuclear Waste Repository design concept – Crystalline Host Media (*Source: SKB 2011, Figure S-1.)

2) Waste Inventory

Waste inventory included in this analysis looked at both defense high level waste (DHLW) and defense spent nuclear fuel (DSNF) waste forms. The DHLW includes Savannah River Site (SRS) glass, Hanford Site (HS) glass, HS Cs-Sr glass, and Idaho calcine. Thermal data for each waste form was obtained from Wilson (2016). Thermal power per canister as function of projected total DHLW and DSNF number of canisters are shown in Figure 3. The same data are plotted in Figure 4 in the form of thermal power per canister as a function of percentage number of canisters. Figure 4 illustrates that the majority of the DHLW (>70%) canisters have thermal power less than 50 W. A sizable number of DSNF canisters (nearly 50%) are also in this category.

Decay heat curves for DHLW and DSNF waste types with highest range of thermal power are shown in Figures 5. For DSNF, only waste packages with thermal power less than 1 kW were considered. Decay heat curves of DHLW and DSNF waste types with lowest range of thermal power are shown in Figure 6.

In this study, the focus is to investigate the magnitude of thermal extremes. Thus, for both semi-analytical and numerical simulation decay heat curves from the highest thermal range were used (i.e. the decay curves shown in Figure 5).

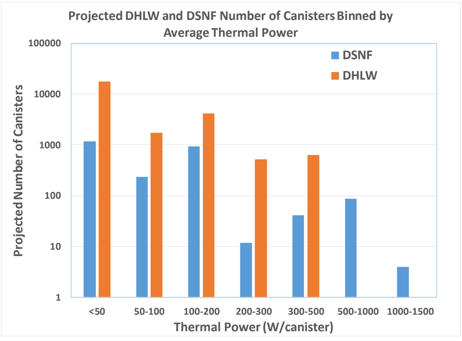

Figure 3. *Thermal bins by total # of canisters for DHLW and DSNF*

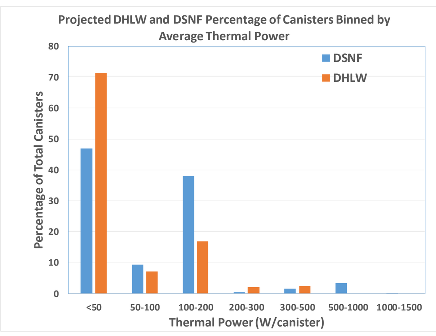

Figure 4. *Thermal bins by % of canisters for DHLW and DSNF.*

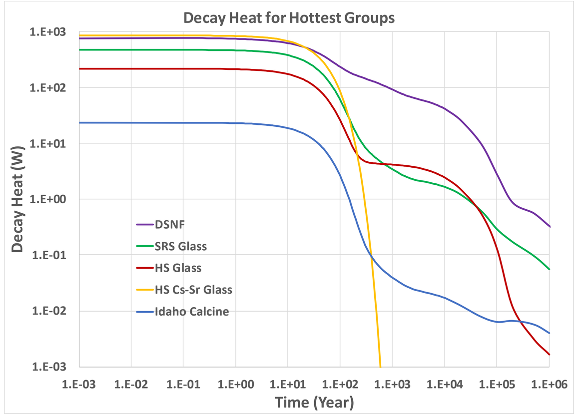

Figure 5. *Thermal decay curves in the highest range of canisters for DHLW and DSNF.*

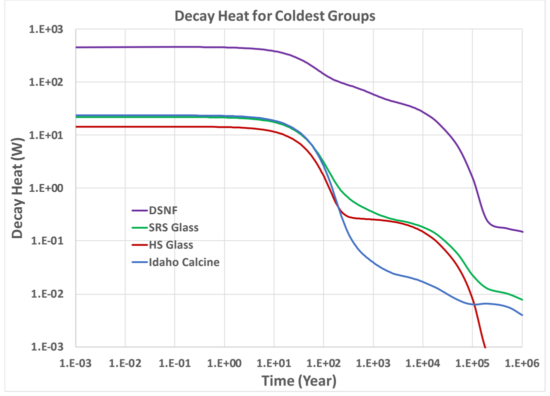

Figure 6. *Thermal decay curves in the lowest range of canisters for DHLW and DSNF.*

3) Waste Packages and Emplacement Details

The disposal setting and dimensions are specific to each disposal concept and waste type. For the crystalline rock concept waste packages are emplaced individually horizontally, encapsulated in swelling clay-based buffer material. The semi-analytical thermal analysis was carried out for single pack (existing canister or waste package) and multi pack (5 glass canisters in a waste package) disposal options. The canister size for each waste type for single pack disposal is given in Table 1 (Carter et al., 2012, Table 3-7). For DSNF, the canister diameter is 0.61m; in this study the total DSNF waste package diameter will equal 0.80 m, as there is additional diameter owing to the use of an overpack. For crystalline rock, the overpack will be a corrosion-resistant material, while the salt design will utilize a steel overpack.

Waste Package Type	Diameter m (in)	Length m (in)
DSNF canister	0.80 (31.5)	4.57 (180)
SRS Glass canister	0.61 (24)	3.05 (120)
Idaho Calcine canister	0.61 (24)	3.05 (120)
HS Glass canister	0.61 (24)	4.57 (180)
HS Cs-Sr canister	0.66 (26)	1.52 (60)

Table 1. *Sizes of single pack canisters for disposal in crystalline and salt repositories.*
DSNF = Defense Spent Nuclear Fuel
SRS = Savannah River Site
HS = Hanford Site

Input for thermal analysis of repository in crystalline rock with single pack canisters
Drift diameter – 1.5 m
Drift spacing – 20 m (base case), 10 m
Waste package spacing – 10 m (base case), 5 m
Buffer thermal conductivity: 0.6 (base case), 1.43 W/m-K
Surface storage time – 10, 50, 100 years

Input for thermal analysis of repository in crystalline rock with multi pack waste packages
Drift diameter – 4.5 m
Drift spacing – 20 m (base case), 10 m
Waste package spacing – 10.31 m (base case), 20 m, 7 m
Buffer thermal conductivity: 0.6 (base case), 1.43 W/m-K
Surface storage time – 10, 50, 100 years

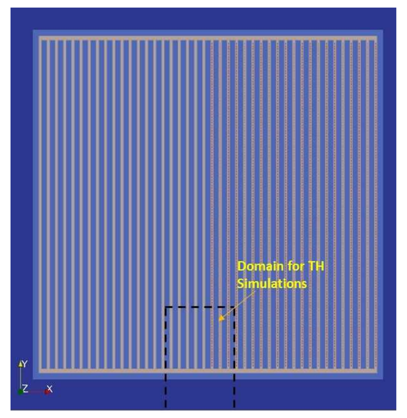

Figure *7. Full repository grid for crystalline case.*

4) Results I – Semi-Analytical Thermal Analysis

Thermal-only, semi-analytical analysis was conducted for the disposal of DOE-managed waste types in crystalline and salt host rocks. The semi-analytical method is based on the approach developed for enclosed emplacement modes by Hardin et al. (2011, 2012). The method was used to calculate the temperature histories for combinations of disposal concept and waste type, assuming a particular emplacement layout for each concept. Thermal responses for DOE-managed waste forms were investigated for disposal concepts in two generic host media (crystalline rock and bedded salt). The output of interest to this work is temperature history at the surface of the waste package and at the drift wall. The general approach for closed systems is based on heat transfer by conduction only, neglecting convection and thermal radiation. These simplifications are reasonable for low permeability media and enclosed emplacement modes (Hardin et al., 2012).

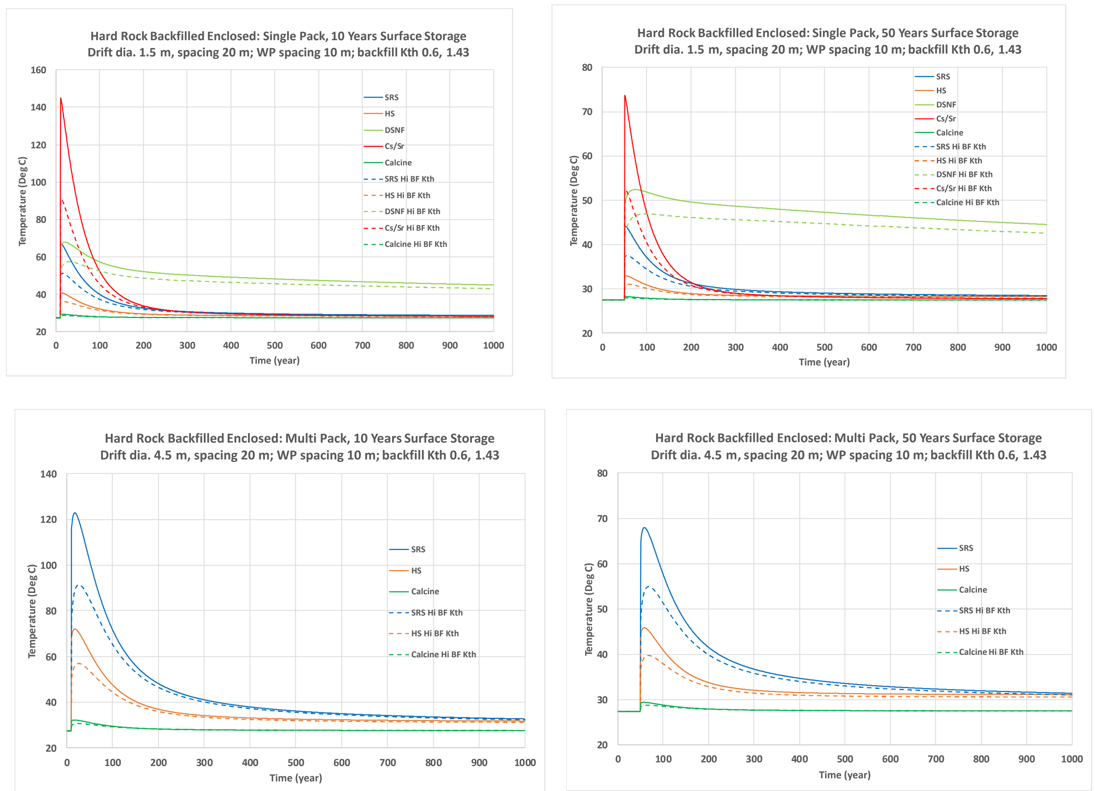

Figure 8. *Temperature histories at waste package surface after surface storage of 10 and 50 years for multi-pack waste packages containing DHLW waste types, for a repository in crystalline medium. The dashed lines represent the case where the high buffer thermal conductivity value of 1.43 W/m-K is used.*

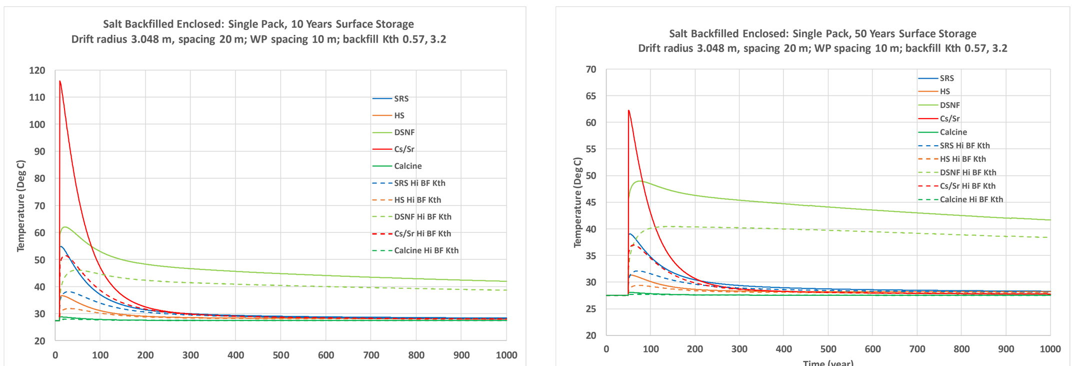

Figure 9. *Temperature histories at waste package surface after surface storage of 10 and 50 years for multi-pack waste packages containing DHLW waste types, for a repository in salt medium. The dashed lines represent the case where the high buffer thermal conductivity value of 3.2 W/m-K is used.*

5) Results II – Thermal-Hydrology Model

Disposal The study looked at thermal conditions in a domain extending over a portion of the repository as shown in Figure 4-13. Selection of the smaller part of the domain allows detailed thermal analysis with a refined mesh. Symmetry conditions on three faces of the domain allow reduced computation burden. The geometry of the domain is 180 m x 1116 m x 1000 m, extending into the host rock in the y-direction and to the surface in the vertical direction. The mesh shown in Figure xx, includes unstructured grid with extensive refinement near drifts and waste packages. The mesh size is 910,585 grid blocks. The selected domain covers 9 drifts with 9 waste packages in each drift. The drift diameter is 4.5 m with 2m DRZ surrounding the drifts. Each waste package is surrounded by buffer material. The domain includes a 10.5 m wide access drift.

Material	Permeability (m^2)	Porosity (-)	Thermal K (W/m-K)	Heat Capacity (J/kg-K)
Granite	1×10^{-18}	0.01	2.5	800.
DRZ	1×10^{-16}	0.01	2.5	800.
Buffer	1×10^{-19}	0.2	0.6/0.85	800.
Waste Package	1×10^{-20}	0.47	46.0	493.

Table 2. *Base case material properties for TH calculations*

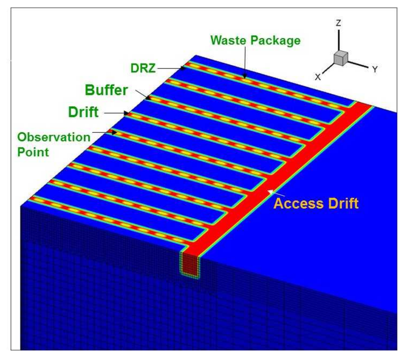

Figure 10*. Plan view of cross-section at repository level showing TH model domain.*

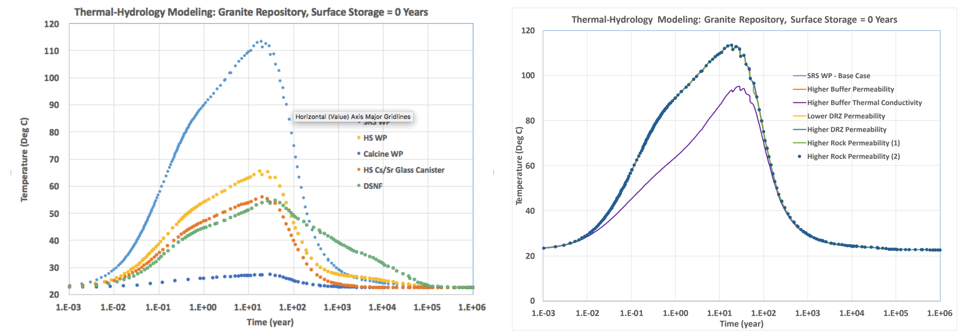

Figure 10*. Temperature history at an observation point for disposal with no surface storage of: left - multi-pack waste packages containing 5 SRS glass canisters, 5 HS Glass canisters, 5 calcine canisters, and single pack HS Cs-Sr glass canisters and DSNF waste packages; and right - multi-pack waste packages containing 5 SRS glass canisters.*

6) Conclusions

- The semi-analytic approach was compared with a TH model, and proven to capture the thermal behavior
- For the crystalline host media, it was shown that unsaturated clay buffer material (of low thermal conductivity) resulted in satisfactory peak temperatures (<100 ºC) for all cases considered (high and low buffer thermal conductivity and10/ 50/100 year surface storage), for all waste types,
- A notable exceptions to the previous bullet being the Cs/Sr waste and SRS HLW glass for 10 years surface storage and low thermal conductivity buffer. These two case were shown to produce satisfactory peak temperatures by either utilizing a longer surface storage time, or a higher buffer thermal conductivity.
- For the salt host media, all waste types and all cases resulted in peak temperatures below design specifications (< 200 ºC).
- Overall, these results suggest that, on the basis of conservative, bounding-case thermal analysis, thermal management of DOE-managed wastes considered (SRS and Hanford HLW glass, DSNF, Calcine waste, and Cs-Sr glass) is achievable, even for the highest thermal output canisters/waste packages for each waste type. For cases where peak temperatures exceed design specifications, thermal management solutions – de-rating for multi-packs (fewer canister), longer surface storage, or use of high thermal conductivity buffer – offer options for effective control peak temperatures.

References

Carter, J. T., Rodwell, P. O, Robinson, B., Kehrman, B. 2012. Costing Study for a Generic Salt Repository: Systems Engineering and Analysis in Support of a Policy Review of Comingling Decision and Related System Design Considerations, FCRD-UFD-2012-000113, Rev. 1., July 2012.

DOE (U.S. Department of Energy) 1986. *Deaf Smith site characterization plan conceptual design report.* U.S. Department of Energy. September, 1986.

Hammond, G. E., P.C. Lichtner, and R.T. Mills, "Evaluating the Performance of Parallel Subsurface Simulators: An Illustrative Example with PFLOTRAN", *Water Resources Research*, 50, doi:10.1002/2012WR013483 (2014).

Hardin, E., J. Blink, H. Greenberg, M. Sutton, M. Fratoni, J. Carter, M. Dupont, and R. Howard 2011. Generic Repository Design Concepts and Thermal Analysis, FCRD-USED-2011-000143, Rev. 0.

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

SKB (Svensk Kämbränslehantering AB [Swedish Nuclear Fuel and Waste Management Company]) 2011. *Long-Term Safety for the Final Repository for Spent Nuclear Fuel at Forsmark*, Technical Report TR-11-01. Three volumes. Stockholm, Sweden: Svensk Kärnbränslehantering AB.

Wilson, J. 2016. Decay Heat of Selected DOE Defense Waste Materials, FCRD-UFD-2016-000636, SRNL-RP-2016-00249.

Acknowledgements

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